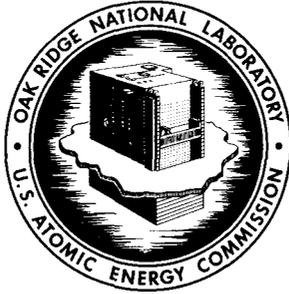




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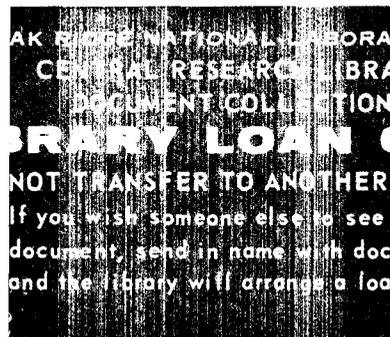


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RADIOISOTOPE PROGRAM (8000) PROGRESS REPORT FOR OCTOBER 1970

A. F. Rupp



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ORNL-TM-3228

Contract No. W-7405-eng-26

Isotopes Development Center

RADIOISOTOPE PROGRAM (8000) PROGRESS REPORT
FOR OCTOBER 1970

A. F. Rupp

Work Sponsored by
AEC Division of Isotopes Development

DECEMBER 1970

OAK RIDGE NATIONAL LABORATORY
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RADIOISOTOPE PROGRAM (8000) PROGRESS REPORT
FOR OCTOBER 1970

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 $\leq 5\%$ phosphorus-32. 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, long-term ecological and agricultural studies, synthesis of tagged complex organophosphorus compounds, and double labeling experiments.

Two methods of producing phosphorus-33 are being evaluated. Both methods are based on the irradiation of highly enriched targets, either sulfur-33 (>92 at. %) or chlorine-36 (approximately 63 at. %), in a fast neutron flux to produce phosphorus-33 by either the $^{33}\text{S}(n,p)^{33}\text{P}$ or $^{36}\text{Cl}(n,\alpha)^{33}\text{P}$ reaction, respectively. Highly enriched targets are required in order to reduce the phosphorus-32 content of the product and to meet the product specification of $\leq 5\%$ phosphorus-32.

In the September report¹ it was reported that the product $\text{H}_3^{33}\text{PO}_4$ contained only 175 mCi, whereas a yield of approximately 1 Ci had been expected. The quartz ampule was leached with 6 M HCl and the leach was found to contain a large quantity of soft beta activity. This leach solution was processed by the normal procedure. After the fourth iron hydroxide precipitation the solution was checked for gamma activities because the quartz was known to contain the following activities: ^{181}Hf , ^{124}Sb , ^{95}Zr , ^{95}Nb , ^{65}Zn , ^{46}Sc , $^{134,136}\text{Cs}$, $^{154,156}\text{Eu}$, ^{59}Fe , ^{60}Co , and traces of ^{141}Ce , ^{233}Pa , and ^{103}Ru . These activities may arise as fission products or by activation of casual contaminants perhaps from low-level uranium impurities in the quartz ampule. Table 1 shows the gamma activity after the fourth $\text{Fe}(\text{OH})_3$ precipitation (mid-process) corrected to an equivalent volume basis with the final product. A Bio-Rad AG 50W-X8 cation column was used finally to remove the iron and other cationic species, and the final product gamma activity and mid-process gamma activity are compared in Table 1.

¹A. F. Rupp, Radioisotope Program (8000) Progress Report for September 1970, ORNL-TM-3183, Oak Ridge National Laboratory.

Table 1. Gamma Activity at Mid-Process and in Final Product

Radioisotope	Activity ($\mu\text{Ci/ml}$)	
	Mid-Process	Final Product
Cerium-141	0.16	-
Hafnium-181	0.06	-
Chromium-51	2.72	2.1
Antimony-124	0.15	0.20
Europium-156	1.0	-
Scandium-46	1.03	-
Cobalt-60	0.1	-
Niobium-95	-	~0.03

The cleanup column did not remove chromium-51, antimony-124, and niobium-95 which are probably present in anionic complexes. If chromium-51 is present as the CrO_4^{2-} ion, it can be removed by reducing it to chromium(III) with iron(II) before the final cation column. This method will be tried if this species appears in future products.

The product $\text{H}_3^{33}\text{PO}_4$ was well within our specifications and contained 17.5 mCi/ml phosphorus-33 and 0.28 mCi/ml phosphorus-32. The yield was 875 mCi phosphorus-33 which, when combined with the 175 mCi previously recovered, gives the expected total yield of 1.05 Ci. All outstanding orders for phosphorus-33 have now been filled. Work is continuing on processing the remaining targets and recovering the K^{36}Cl target material.

A typical phosphorus-33 product was analyzed by both the Cerenkov-liquid scintillation analytical method currently being developed^{2,3} and by the end-window counter-liquid scintillation analytical method currently being used for product assay. The results agreed exceedingly well — the phosphorus-32 values were identical, and the phosphorus-33 values were within 6%; this illustrates the way these independent methods complement each other.

A general program was developed for a Wang Model 700A programmable electronic calculator to calculate both single and double neutron capture yields. This program will be utilized to estimate the amount of low enrichment chlorine-36 that can be added to the present chlorine-36 target material and still yield a product within the $\leq 5\%$ phosphorus-32 specified for this product. The present product prepared from the current target material contains $\leq 1.3\%$ phosphorus-32 at time of shipment. The addition of lower enrichment chlorine-36, and consequent increase in phosphorus-32 content of the product, is deemed necessary to provide additional target material and to compensate for reactor burnup and processing losses.

²A. F. Rupp, Radioisotope Program (8000) Progress Report for August 1970, ORNL-TM-3133, Oak Ridge National Laboratory, p. 1.

³A. F. Rupp, ORNL-TM-3183, p. 1.

2. Potassium-43

The objectives of this project are: to prepare potassium-43 by the $^{43}\text{Ca}(n,p)^{43}\text{K}$ 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-42 has been used, primarily in animals, to locate tumors, to tag red blood cells, and to study blood flow, but it has very limited use in humans because of its high beta energy (3.53 MeV), its high-energy (1.52 MeV) gamma ray, and its relatively short half-life (12.4 hr). Potassium-43, with a half-life of 22.4 hr and gamma-ray emissions of 0.373 and 0.617 MeV, is more suitable for metabolic and clinical studies because the lower dose rate would permit multiple doses. It has been suggested as a possible tool for studying blood flow through the heart and for diagnosing myocardial infarctions or immune rejection of transplanted organs. It has also proved capable of diffusing through a kidney with greatly reduced renal function, providing a good image of this organ.

Two targets of ^{43}CaO were irradiated 65 hr and processed on October 12, 1970. One target contained 90.2 mg of recovered ^{43}CaO and the other contained 90.1 mg of new 61.6% ^{43}CaO . The two targets were dissolved separately and the solutions were assayed. Table 2 compares the observed total target activities corrected to discharge.

Table 2. Analyses of Samples of Dissolved ^{43}CaO Targets

Nuclide	Half-Life	Activity at Discharge (mCi)	
		Recovered Target	New Target
Potassium-43	22.4 h	14.3	13.2
Potassium-42	12.4 h	3.42	3.09
Sodium-24	15.0 h	0.94	0.85
Manganese-56	2.58 h	7.64	0.42
Strontium-87m	2.83 h	18.5	25.2
Calcium-47	4.53 d	0.217	0.13
Scandium-46	83.8 d	1.55	-
Scandium-47	3.40 d	0.21	0.03
Lanthanum-140	40.2 h	0.25	-
Samarium-153	46.8 h	0.54	0.10
Europium-152m	9.3 h	1.3	-
Bromine-82	35.4 h	0.054	-
Tungsten-187	23.9 h	-	0.22

The product yield was 80% of the potassium-43 in the target. It was noted that the potassium-43 elution was not as sharp as previous ones, perhaps due to the fact the column was not equilibrated as long as usual.

Shipments were made under our medical cooperative programs to Johns Hopkins, University Hospitals of Cleveland, University of Mississippi Medical Center, and Peter Bent Brigham Hospital.

B. Exploratory Development of Products and Techniques

1. Silicon-32

The formation paths for reactor production of silicon-32 depend upon the production of and subsequent neutron reaction of an intermediate. Silicon-32 exhibits many desirable physical decay characteristics that make it an interesting and attractive radionuclide for a variety of uses: (1) the only long-lived radionuclide of silicon for use in agriculture, geology, solid-state physics, etc., (2) as a parent for a ^{32}Si - ^{32}P generator system which would offer a constant supply of phosphorus-32 for medical and scientific application, and (3) as a long-lived Cerenkov counting standard for phosphorus-32 determinations.

The three neutron capture reaction paths for the production of silicon-32 are shown in Fig. 1. The first reaction path requires only thermal neutrons, whereas the other two require fast neutrons for the (n,p) and (n, α) reactions. The production rate, A_{32} , for each of these paths will be directly proportional to the number of atoms of the intermediate formed and to the magnitudes of both the neutron fluxes and cross sections required for neutron capture by the intermediate to produce silicon-32, i.e., (approximately)

$$A_{32} \propto (\sigma_{\text{th}})_a (T_{1/2})_b (\sigma)_b (\Phi)_{b \rightarrow c} \quad (1)$$

where

A_{32} = production rate of silicon-32 per gram-atom of target isotope (a) in the same flux for all three reaction paths,

a = the target isotope,

b = the radioactive intermediate, and

c = the final product, silicon-32.

Substituting the data shown in Fig. 1 into Eq. (1), an estimate for the various production rates can be obtained:

$$\begin{aligned} A_{32} (1) &\propto (0.1 \text{ b})(2.62 \text{ h})(1.1 \text{ b})(\Phi_{\text{th}}) , \\ A_{32} (2) &\propto (0.19 \text{ b})(14.3 \text{ d})(\sigma_{\text{n,p}} = ?)(\Phi_{\text{th}})(\Phi_{\text{f}}/\Phi_{\text{th}}) , \\ A_{32} (3) &\propto (0.22 \text{ b})(87.2 \text{ d})(\sigma_{\text{n,\alpha}} = ?)(\Phi_{\text{th}})(\Phi_{\text{f}}/\Phi_{\text{th}}) , \end{aligned}$$

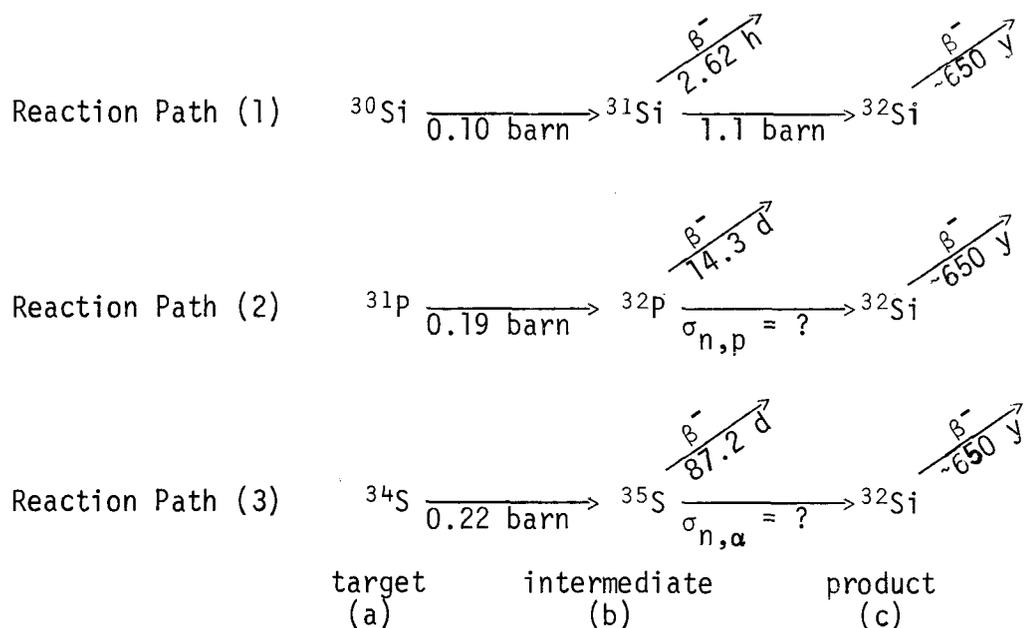


Fig. 1. Silicon-32 Production Paths.

where ϕ_f/ϕ_{th} represents the fast to thermal neutron flux ratio for the reactor of interest [$(\phi_f/\phi_{th}) \simeq 0.33$ (ref. 4) for the removable beryllium positions in the HFIR]. Further, the relative production rates for the three paths may be expressed as:

$$A_{32}(1):A_{32}(2):A_{32}(3) = 0.012:0.90 \times \sigma_{n,p} \text{ (in barns)}:6.3 \times \sigma_{n,\alpha} \text{ (in barns)}.$$

The magnitude of the $^{32}\text{P}(n,p)^{32}\text{Si}$ and $^{35}\text{S}(n,\alpha)^{32}\text{Si}$ cross sections must be estimated to determine the relative merit of the reactions of interest. To do this the empirical approach of Roy and Hawton⁵ was employed and yielded estimates for the (n,p) reaction of 65-460 mb, and for the (n, α) reaction of approximately 6 mb. These estimated values were rounded off ($\sigma_{n,p} \approx 0.1$ barn, $\sigma_{n,\alpha} \approx 0.006$ barn) to permit estimates of the relative production rates:

$$A_{32}(1):A_{32}(2):A_{32}(3) = 0.012:\sim 0.09:\sim 0.04 .$$

As can be readily seen, paths (2) and (3) would seem to offer larger production rates than direct irradiation of silicon-30. The fact that both paths (2) and (3) would produce carrier-free silicon-32 also make them more attractive than path (1). Even though the $\sigma_{n,p}$ and $\sigma_{n,\alpha}$ cross sections may be vastly misestimated, an exploratory experiment seems warranted.

⁴R. E. Lewis, J. K. Poggenburg, and H. B. Hupf, Neutron Flux Monitoring of Irradiation Facilities in the High Flux Isotope Reactor (HFIR) (to be published).

⁵J. C. Roy and J. J. Hawton, Table of Estimated Cross Sections for (n,p), (n, α), and (n,2n) Reactions in a Fission Neutron Spectrum, Rpt. CRC-1003 (December 1960).

Path (2) has been selected, since it seems to offer the largest production rate and the target material is naturally occurring and readily available. Assuming the smaller $\sigma_{n,p}$ estimate (65 mb) obtained using Roy and Hawton's method, a silicon-32 production of approximately 0.7 $\mu\text{Ci/g}$ of phosphorus-31 has been estimated for six cycles in a removable beryllium position in the HFIR by using the program described in the phosphorus-33 section of this report. Work will begin next month on selection of a suitable phosphorus target for multicycle irradiation in the HFIR.

2. Gadolinium-153

Gadolinium-153 (241 days) is an important radionuclide because it decays by electron capture yielding predominantly 100-keV photons. This energy range is useful in backscatter and transmission gaging, as well as having applications in fluorescent X-ray production; medical applications for bone-density gages and narrow-beam transmission for body scanning to supplement stethoscopic findings appear to be of importance also. An especially important part of our work is preparation of highly enriched gadolinium-152 synthetically by extracting it chemically from exhausted europium control plates from HFIR. The greater than 90% enriched gadolinium-152 is not available from calutron separations, and in any case, it would be prohibitively expensive if it could not be obtained as a by-product.

Totally enclosed equipment to process HFIR control plates for recovery of gadolinium-152/153 has been fabricated and placed in cell for operation. The enclosed recovery system is necessary because the electrolytic cell used to separate gadolinium and europium produces an aerosol of electrolyte solution causing a considerable contamination problem in the hot cell.

One section of the control plates has been dissolved to separate the aluminum matrix from the europium oxide. The rare-earth residue from this separation has been prepared as feed for the electrolytic separation step. Approximately 10 g of europium and 1 to 2 g of gadolinium are involved in this separation. A second batch containing a similar quantity of europium/gadolinium is also in process.

Interest in gadolinium-153 continues to grow and we now have need for 10 to 25 Ci to meet existing orders, plus an additional 200 Ci to complete our commitment to NASA.

Of special current interest is the program at Vanderbilt Medical Center to use gadolinium-153 as a source of photons to locate stable gallium in brain tissue by means of the secondary X ray. This involves a highly collimated beam of 100-keV gamma rays from a gadolinium-153 source with a detector in a fixed position relative to the source. By rotating the gadolinium-153 source it may be possible to precisely locate regions of high gallium uptake. These regions would indicate tumors.

C. Isotopic Power

1. Thulium-170

The objectives of this research are to provide sufficient data on thulium-170, which has been proposed as an isotopic power source for short-duration missions, to permit reasonable assessment of its potential application, and to permit preliminary engineering design of power sources. The economics for reactor production of thulium-170 (128.5 day; $E_{\beta_{av}} = 0.32$ MeV) from natural thulium-169 are favorable and the sesquioxide offers a promising fuel compound, having a practical specific power of approximately 2 W/g and a power density of approximately 16 W/cm³. The study will include measurements of thermal conductivity and thermal diffusivity, leach rates for safety analysis, high-temperature compatibility with suitable containment metals, vapor pressure, and fuel form densification.

Loading of 18 ¹⁷⁰Tm₂O₃ pellets together with the test metal disks into the inner capsules has been completed, and welding (electron-beam) of the inner capsules is in progress. Photomicrographs of one fractured ¹⁷⁰Tm₂O₃ pellet show some hairline cracks mainly in the radial direction along the grain boundaries. This observation and the fact that one of the compatibility pellets suddenly fractured several minutes after it was placed on a cold (room temperature) porcelain plate indicate that the thermal shock was probably the major cause of fracture.

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

<u>Processed Units</u>		<u>Service Irradiations</u>	
<u>Radioisotope</u>	<u>Total (mCi)</u>	<u>Type</u>	<u>Number</u>
Calcium-47	16	Gold	1
Copper-67	<u>15</u>	Yttrium Oxide	2
		Chromium	1
		Platinum	<u>1</u>
Total	31	Total	5

E. Source Fabrication

A 20-Ci ytterbium-169 source was prepared.

F. Miscellaneous

1. HFIR Target Testing

One removable beryllium stringer was tested and found to meet specifications. Twenty hydraulic tube rabbits were subjected to leak and pressure tests; seven of the rabbits contained stable material and thirteen contained radioactive material. Two zinc-67 targets did not meet specifications.

2. Krypton-85 Light Sources

Two quartz ampules were constructed and will be coated with a phosphor and filled with krypton-85 to be used as light sources. The light from each central source will be conveyed by fiber optic light pipes to other locations to evaluate the practicality of this system for providing instrument lights. Four fiber optic light conductors in various diameters and lengths have been obtained for this project.

Two 1-in.-diam quartz bulbs were coated internally with phosphors, one with phosphor only and one with a mixture of phosphor and silica gel, to see if the light intensity from the source can be increased by adding an absorber for krypton which may increase the amount of krypton in contact with the phosphor. These bulbs will be filled with krypton-85 and the light output will be measured.

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 hr) production in acceptable purity and quantity and to provide gallium-67 for clinical applications research and development. Interest in this isotope has been spurred by evidence, obtained by the Medical Division of Oak Ridge Associated Universities (ORAU), of a high uptake of carrier-free gallium-67 by lymphoid tumors in both animals and humans.

Gallium-67 decays by electron capture with the emission of four main gamma rays of 93, 184, 296, and 388 keV with intensities of 40, 23, 20, and 8%, respectively. Carrier-free gallium-67 is produced by the $^{68}\text{Zn}(p,2n)^{67}\text{Ga}$ reaction by bombarding natural, high purity zinc tubing targets (0.375-in. OD by 0.020-in. wall thickness) with 22-MeV protons in the ORNL 86-Inch Cyclotron. Irradiation of natural zinc also produces appreciable amounts of gallium-66 (9.5 hr) that must be allowed to decay before human use.

One gallium-67 preparation was shipped to ORAU during this period, and eight smaller orders were filled for clinicians interested in the applications of this radionuclide. The operating performance (i.e., high current beam resistance) of a new batch of high purity zinc target extrusions was tested during this irradiation and found to be comparable to the previous batch. A random sample of this target material was irradiated at a beam current of approximately 300 μA (actual average beam current = 294 μA) until failure, which occurred after 4.12 hr under these conditions (actual integrated current = 1.212 mA-hr). These operating limits and previous production experience indicate that a maximum production limit of approximately 550 mCi per irradiation is characteristic of the present target configuration.

B. Exploratory Development of Products and Techniques

1. Rubidium-84

The objectives of this program are to improve the technology for the production of higher purity rubidium-84. Rubidium-84 is regularly produced by the ORNL Isotopes Division by proton bombardment of natural krypton targets, but this product is unsuitable for positron camera scanning due to appreciable amounts of rubidium-83 produced simultaneously. The activity of the rubidium product obtained from proton bombardment of natural krypton in the ORNL 86-Inch Cyclotron is comprised of 9.7% rubidium-83 (83 days), 68% rubidium-84 (33 days), and 22% rubidium-86 (18.7 days) at end of bombardment. The longer half-life of rubidium-83 reduces the $^{84}\text{Rb}/^{83}\text{Rb}$ ratio even further with time. Appreciable amounts of rubidium-83 are undesirable for positron scintigraphy, because its principal gamma emissions [521 keV (46%), 530 keV (31%), and 553 keV (16%)] are similar enough in energy to the 511-keV annihilation radiation of rubidium-84 (intensity = 42%) to prevent energy resolution by the positron camera, thereby leading to increased background and loss of spatial resolution. Interest in using rubidium-84 for myocardial scanning with a positron camera has increased the demand for a higher purity product, which is being made from an enriched krypton-84 target.

One rubidium-84 production run with an enriched krypton-84 target (4.2 at. % krypton-83, 88.6 at. % krypton-84, and 6.9 at. % krypton-86) was performed during this period and the product was supplied to customers. This product is shipped as an unprocessed product without further purification. An unexpected cyclotron target holder failure resulted in the loss of a portion (approximately 25%) of the enriched krypton-84 target. The feasibility of a complete gas target holder assembly redesign will be undertaken in the next quarter in order to determine if the occurrence of similar future losses can be avoided.

C. Accelerator Pilot Production (Production and Inventory Accounts)

Table 3 gives the October 1970 accelerator irradiations and runs for ORNL and non-ORNL customers.

Table 3. Cyclotron Irradiations and Runs for October 1970

Product	No. of Runs	Time (hr:min)			Total Charges
		Beam	Misc.	Total	
<u>ORNL Programs</u>					
Cobalt-56	1	4:00	1:15	5:15	\$ 631.46 ^a
Gallium-67	2	7:17	2:15	9:32	858.11
Iodine-123	1	<u>1:10</u>	<u>1:15</u>	<u>2:25</u>	<u>383.43</u>
Total		12:27	4:45	17:12	\$ 1,873.00

Table 3. continued

Product	No. of Runs	Time (hr:min)			Total Charges
		Beam	Misc.	Total	
<u>Non-ORNL Programs</u>					
Silver-105	1	1:00	1:15	2:15	\$ 333.45
Cobalt-57	1	50:00	1:15	51:15	8,388.64 ^a
Molybdenum-99	1	0:20	1:06	1:26	200.66
Radiation Damage	1	33:45	7:45	41:30	6,180.00
Rubidium-84	1	8:00	1:15	9:15	1,794.99
Technetium-95m	1	0:12	1:15	1:27	208.00
Technetium-99m	2	0:12	2:00	2:12	308.00
Yttrium-87	3	8:45 ^b	3:45	12:30	1,970.30
Total		102:14	19:36	121:50	\$19,384.04

^aPurchased but not shipped.

^bAn extra 15 min included due to high pressure.

FISSION PRODUCTS - 08-01-03

A. Krypton-85 Enrichment

No gas transfers were made in any of the five units (see Fig. 2) enriching krypton-85. The present status of all units is shown below. The count rates indicate the relative levels of enrichment.

Unit	Activity in Unit (Ci)	Time Since Last Product Removal (days)	Count Rate in Product Section (counts/min)	
			Sept 1970	Oct 1970
A	-	164	-	-
AB	1537	40	10,000	8,500
B	1406	40	7,850	6,800
C	1566	40	7,600	6,400
CD	2281	213	9,650	14,100
D	1360	40	5,900	5,400

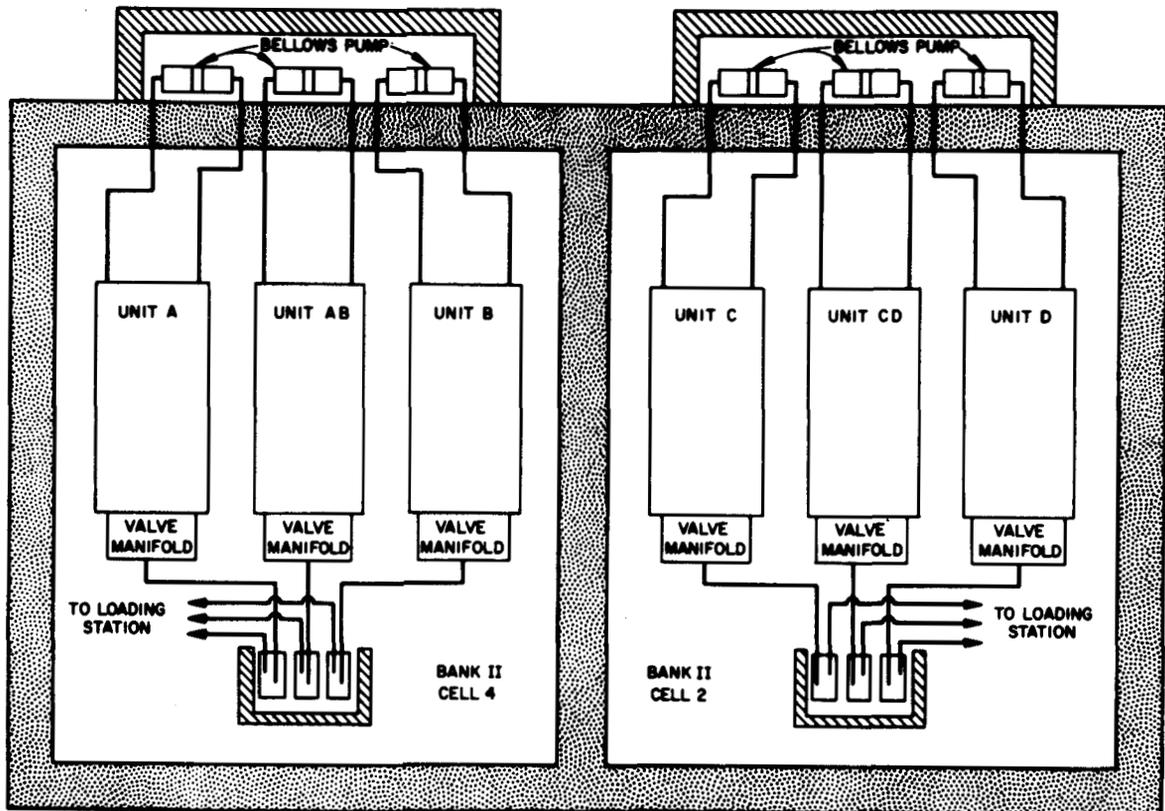


Fig. 2. Schematic Arrangement of Krypton-85 Columns.

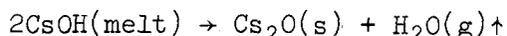
B. Cesium-137 Gamma Source Development

1. Cesium Source Form Development

Cesium-137 chloride has been the compound of choice for radiation sources in applications of moderate temperature conditions. In many respects cesium chloride is the ideal compound; the cesium weight per unit volume is high, the radiation resistance of the CsCl is excellent, the compatibility with stainless steel at ambient temperatures is excellent, and its preparation is straightforward. However, the projected conditions of use of cesium-137 gamma sources are increasingly severe with respect to temperature, and some applications indicate the need for a cesium source material which has low solubility. In view of these projected conditions of use, the testing of $^{137}\text{CsCl}$ at elevated temperatures is being done and the development of a low solubility, temperature-resistant source form of cesium is being studied.

a. Cesium Niobate

A few experimental runs have been made to produce cesium niobate using Method 1 (CsOH melt-Nb₂O₅ reaction) described previously.⁶ The reaction between CsOH and Nb₂O₅ was found to be very slow in the low temperature range (270-500°C). This inference is based on the X-ray diffraction patterns and chemical analysis of the product which consists mainly of Cs₂O and Nb₂O₅. Upon dissolution of the product in water, a very fine precipitate (probably hydrous niobium pentoxides) formed, most of which went through the filter paper and remained in suspension for a number of hours. The major reaction in the low temperature range is presumably dehydration of CsOH according to:



When the reaction temperature was increased to approximately 1000°C, a product having fibrous crystalline structure with bluish-white color formed. The major losses (approximately 10%) of material occurred during the reaction up to approximately 500°C, part of which is attributable to evolution of water produced in the reaction. In the 500-1000°C range the losses were slight (less than 2%).

To study stability of the material in aqueous solutions, samples were placed in the solutions of HNO₃ (concentrated and 3 N), HCl (concentrated), and NaOH (3 N) and the solutions were heated to approximately 100°C under reflux for approximately 2 hr. The amounts of the material dissolved in various solvents are shown below:

<u>Solvent</u>	<u>Concentrated HNO₃</u>	<u>3 <u>N</u> HNO₃</u>	<u>Concentrated HCl</u>	<u>3 <u>N</u> NaOH</u>
Wt % of material dissolved	50.6	52.3	42.7	44.1

According to the X-ray diffraction data, the material is a compound or mixture of compounds of cesium and niobium and contains very little of the reactants (CsOH and Nb₂O₅).

2. Cesium-137 Source Form Characterization

The purpose of this task is to characterize cesium-137 source forms for operational and safety requirements. The objective of the current study is to find a means of eliminating the solid-solid phase transition which occurs at 469°C with an accompanying volume increase.

Installation of the heating stage in the X-ray unit and alignment of the apparatus have been completed. Fabrication of a special sample holder (tantalum) for the heating stage will be finished shortly. Adjustment and calibration of the dilatometer have been carried out using the fused quartz and tungsten standards. Both apparatuses are to be used in correlating the structural transformation with the volumetric expansion

⁶A. F. Rupp, ORNL-TM-3183.

for selected CsCl-KCl mixtures. The study of the relation between the structural transformation and the volumetric expansion of selected CsCl-KCl mixtures will be continued.

C. Cesium-137 Process and Product Study

1. Cesium-137 Process Study

The purpose of this task is the study and development of economical processes for recovery, purification, and conversion of cesium-137.

To conduct an economic analysis of the three modified processes under consideration for possible use in the FPDL operation, the fixed capital costs (F) for individual processes (Processes 1, 2, and 3) were calculated based on the generalized semiempirical equation discussed earlier.⁷ The results summarized in Table 4 also list the data for the two processes (Processes 4 and 5) previously considered. The data used in the equation have been estimated from the conceptual process flowsheets as well as the assumed processing conditions.

Table 4. Comparison of Fixed Capital Costs^a for Various Conceptual ¹³⁷CsCl Processes^a

Process	Eluant	N	f _a	f _t	f _c	F ^b
1	NH ₄ Cl	4	0.2	0.07	3.73	14.9 K
2	(NH ₄) ₂ CO ₃ -NH ₄ OH	6	0.27	0.07	4.37	26.2 K
3	(NH ₄) ₂ SO ₄	8	0.25	0.07	4.17	33.4 K
4	HCl	4	0.4	0.07	5.92	23.7 K
5	HNO ₃	6	0.3	0.11	5.14	30.9 K

^aN = number of functional units, f_a = process alloy factor, f_t = process temperature factor, f_c = complexity factor.

^bK = CPⁿ(I/ \bar{I}), where C is a constant, P is the production capacity (Ci/year), n is a factor derived from a statistical correlation between F and P, and I and \bar{I} are construction cost indices. K is assumed to be common to all the processes.

According to Table 4, Process 1 (NH₄Cl eluant) seems to be economically the most attractive from the standpoint of the fixed capital cost. Although the fixed capital cost for Process 4 (HCl eluant) is the second lowest, the preliminary experimental results indicate that the elution efficiency of the acid-type eluant is considerably lower than that of the ammonium-salt type. A similar statement is applicable to Process 5 (HNO₃ eluant). For these reasons, Processes 1, 2, and 3 are preferable to Processes 4 and 5.

⁷A. F. Rupp, Radioisotope Program (8000) Progress Report for March 1970, ORNL-TM-2695, Oak Ridge National Laboratory.

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

1. Processing and Process Status

<u>Item</u>	<u>Cesium-137 (Ci)</u>
In-process material	173,000
Cesium-137 chloride products	103,000
Sources in fabrication	0
Completed sources awaiting shipment	47,000

2. Operational Summary

<u>Cesium-137</u>	<u>October 1970</u>		<u>FY 1971</u>	
	<u>Number</u>	<u>Amount (Ci)</u>	<u>Number</u>	<u>Amount (Ci)</u>
HAPO shipments received	0	0	0	0
Product batches prepared	0	0	0	0
Sources fabricated	0	0	39	9,100
Special form cans loaded	0	0	7	77,000
Sources shipped to customers	0	0	260	129,000
Special form cans shipped to customers	1	25	8	77,000

One powder shipment containing 25 Ci of cesium-137 was made to General Nuclear.

3. Current Orders

A list of current orders for cesium-137 as sources or bulk powder is shown below:

<u>Customer</u>	<u>Amount (Ci)</u>	<u>Estimated Shipping Date</u>
Brookhaven National Laboratory	~203,000	March 1971
Atomic Energy of Canada Ltd.	24,200	December 1970
Radiochemical Centre, England	20,000	December 1970
C.E.A., France	200,001	April 1971
Minnesota Mining & Mfg. Company	500	November 1970
Technical Operations, Inc.	100	December 1970
American Hoechst Corporation	<u>5</u>	April 1971
Total	447,806	

Orders on hand for bulk powder or sources to be scheduled when released by the customer include Atomic Energy of Canada, approximately 155,800 Ci, and Radiation Resources, Inc., 200,001 Ci. Cesium sources containing 40,000 Ci are on hand awaiting receipt of shipping instructions and payment in advance.

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

1. Processing and Process Status

A batch of 42,000 Ci of $^{90}\text{SrCO}_3$ which had been in storage for several months was dissolved and reprecipitated to provide fresh fuel for a ^{90}SrO pellet. The dried $^{90}\text{SrCO}_3$ was calcined in the resistance-heated hot press; the resulting ^{90}SrO assayed 50.3 Ci/g. A fraction of this fuel (31,000 Ci) was used to prepare a pellet; the remainder was reserved. Decanning, blending, and assay of stored $^{90}\text{SrTiO}_3$ products received from Quehanna operations continued to the point of opening the SNAP-7F sources. On opening the first two of these units (out of a total of 17 containing 240,000 Ci), it was found that the pellets were contained in an inner sleeve (presumably stainless steel) and that the pellets were very tightly fitted inside the sleeve. Metal disk spacers between pellets were also tightly fitted. The only method for recovery of the $^{90}\text{SrTiO}_3$ from these sources will be to split the liners lengthwise. Since equipment for this operation was not immediately available, the campaign was stopped at this point. The additional $^{90}\text{SrTiO}_3$ needed to fill current orders, above that already recovered from Quehanna powder cans, will be taken from FPDL-produced material.

One 210-W ^{90}SrO pellet was pressed on the resistance-heated hot press. This pellet was made in the same fashion as are $^{90}\text{SrTiO}_3$ pellets except that the fuel was enclosed in a thin platinum sheath. The power density of this 7.9-cm-diam pellet was 1.47 W/cm³ including the platinum. The pellet was loaded into a stainless steel can for welding and shipment to the customer, Sanders Nuclear Corporation. A 67-W, 5.1-cm-diam pellet of $^{90}\text{SrTiO}_3$ was prepared for encapsulation as a source for Mitsubishi. Two small tamped-powder units (117 Ci each) of $^{90}\text{SrTiO}_3$ were also loaded.

Several problems were encountered in the operation of the resistance-heated hot press. These can be generally considered as two types — die failures and hydraulic system failures — although each of these can contribute to the other. One of the causes of die breakage is degradation of the graphite die bodies due to oxidation. This furnace is not a sealed unit, and protection from oxidation depends on a flow of argon to the cavity. During heating, the furnace expands sufficiently to create an opening at the point at which the two halves of the clamshell meet. This allows cell atmosphere to be introduced into the cavity. Although the cell is in turn blanketed with flowing argon, there is still enough air leakage to cause oxidation at the high temperatures (up to 1300°C) at which the furnace is operated. The die assembly used in the past for pellets above 5-cm diameter has been an outer jacket of high strength spun graphite around a thin graphite die sleeve. The use of the spun graphite jacket is dictated by the need for strength at the pressures needed for good compaction; below 5-cm diameter, solid graphite dies can be used. Since the spun graphite jackets are expensive and require long-term delivery, the oxidation degradation is a serious problem. One alternative is to use a ceramic outer jacket, and this is presently being tested. A drawback to this method is that the ceramic material is subject to thermal shock. Another possibility is to use solid graphite dies at lower pressure,

thus sacrificing some power density. This method may be usable in some cases where high density is not required. It may also be possible to use a suitably protected refractory metal die; this will be tested in the near future. Hydraulic system failures have occurred on both rams and hydraulic hoses. Most of this is caused by softening of the seals under the high temperatures in the system, and new equipment which is more heat resistant is being obtained.

The current strontium-90 process status is as follows:

<u>Item</u>	<u>Strontium-90 (Ci)</u>
In-process material	1,390,000
⁹⁰ SrTiO ₃ products	495,000
Sources in fabrication	450,000
Completed sources awaiting shipment	123,000 ^a
Returned SNAP sources	<u>325,000</u>
Total	2,783,000

^aIncludes 19,700 Ci in special form containers.

2. Operational Summary

<u>Strontium-90</u>	<u>October 1970</u>		<u>FY 1971</u>	
	<u>Number</u>	<u>Amount (Ci)</u>	<u>Number</u>	<u>Amount (Ci)</u>
HAP0 shipments received	0	0	0	0
Product batches prepared	0	0	0	0
Sources completed	6	143,300	6	143,300
Special form containers loaded	4	50,700	4	50,700
Shipments to customers	5	146,100	6	146,200

3. Current Orders

A listing of current orders for strontium-90 as sources or bulk powder is shown below:

<u>Customer</u>	<u>Amount (Ci)</u>	<u>Estimated Shipping Date</u>
U. S. Navy	557,000	November 1970
U. S. Navy	208,000	February 1971
C.E.A., France	300	November 1970
Buchler and Company, Germany	-1,000	November 1970
Radiochemical Centre, England	225	November 1970

Orders on hand for bulk powders or sources to be scheduled and shipped as released by the customer include approximately 105,000 Ci and approximately 20,000 Ci for Isotopes, Inc.

4. Source Fabrication

Six 2.82-in.-diam $^{90}\text{SrTiO}_3$ pellets prepared in September were loaded into liners and encapsulated in three Sentinel-8 capsules for Isotopes, Inc. The preparation of these units offered an opportunity for comparing the calculated thermal outputs, which are based on calorimetric assay of aliquots of fuel, to actual measurements of the thermal outputs of the completed units. In most cases, the appropriately sized calorimeter has not been available for measurements on completed sources, and source outputs have been quoted on the calculated basis. The results of these assays and the calculated values are shown below.

<u>Source</u>	<u>Source Calorimetry (W)</u>	<u>Calculated Output (W)</u>
Sn-202	302 ± 1	303
Sn-203	302 ± 1	308
Sn-204	302 ± 1	304

This good agreement, in all cases well within the quoted $\pm 5\%$ tolerance, provides support for the reliability of the incremental fuel assay method of determining thermal outputs of large sources. Test weld inspection revealed weld penetrations ranging from 0.075 to 0.110 in. with an average of 0.0912 in. Calorimetry of the completed sources indicated that each source contained 302 ± 1 W (thermal). Prior to loading into Sentinel-8 generators, each source was leak tested using krypton-85. All sources had leak rates of less than 5×10^{-6} cm³/sec. Testing of the generators at ORNL was performed by the customer representatives with the assistance of IDC personnel.

Strontium-90 sources and special form containers produced during October are shown below:

<u>Customer</u>	<u>No. of Sources</u>	<u>Amount (Ci)</u>
Isotopes, Inc.	3	135,000
Mitsubishi International Corporation	1	9,800
Sanders Associates	1	30,000
AEC (Kilowatt Program)	2	<u>234</u>
Total		175,034

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

<u>Isotope</u>	<u>Number of Batches</u>	<u>Amount (Ci)</u>
Tellurium-132	1	2
Xenon-133	2	~700
Ruthenium-103	1	<u>13</u>
Total		715

G. Strontium-90 Silicate Beta Sources

1. Matrix Studies

Laboratory tests are being conducted to determine the feasibility of using graphitic carbon as a matrix for SrSiO_3 beads in the preparation of strontium-90 beta sources. Several mock-up source samples were prepared to study the minibead loading and the strength after baking. The samples were prepared by extruding a mixture of minibeads, graphite flour, and 8251 Varcum containing maleic anhydride as the hardening agent. Density measurements indicate that essentially theoretical loading can be achieved. A sample density of 2.8 g/cm^3 resulted from the extrusion of a mixture containing 83% by weight minibeads and 17% graphite. The strength of the sample after baking for 1 hr at 200°C was adequate for ordinary handling purposes. Additional strength if required can be achieved by prolonged baking at higher temperatures; however, a temperature limitation of approximately 900°C because of crystal structure changes in the SrSiO_3 beads precludes attainment of the temperature (approximately 2000°C) needed for graphitization of the binder. Measurements of compression and tensile strength versus composition and baking temperatures will be made to completely determine the best procedure for final source fabrication.

An evaluation is being made of the results of tests designed to determine the compatibility of the aluminum matrix and strontium silicate minibeads at 600 , 700 , 800 , and 900°C . No additional laboratory work has been done, but tests involving prolonged source exposure at lower temperatures are planned.

An alternative method for making the final closure in the source tube is being investigated. This method consists of fusing a $1/16$ -in.-diam stainless steel bead in the fill hole by a heavy direct current surge. Preliminary tests performed on empty source capsules by the welding machine manufacturer indicated considerable promise, but this work is being discontinued because of lack of funds.

Beta source output measurements are planned on the completion of the electronic equipment required for operation of an extrapolation ionization chamber.

H. Source Fabrication

A 20-Ci promethium-147 source was prepared for Lawrence Radiation Laboratory.

APPLICATIONS AND TECHNOLOGY SUPPORT - 08-01-04

A. Radioisotope Characterization, Quality Control, and Standards

1. Radioisotope Special Analysis and Quality Control

Work was continued on standards furnished by various suppliers. By means of independently calibrated gamma ionization chambers in three laboratories, measurements were made of an iron-59 preparation by the National Bureau of

Standards (NBS). Results varied from -1.3% to +1.2% with respect to NBS, averaging -0.4%. Three standards of silver-110m have been received from commercial suppliers. Our preliminary values were 4% and 14% below those of two of the companies, and the third preparation was found to contain predominantly 127-year silver-108m, subsequently acknowledged by the company. An absolute calibration will be made of the standard for which our result is closest to that of the manufacturer.

A survey was made of techniques for measurement of traces of krypton-85 in inactive gases. The present routine method consists of gamma-spectrometric determination of the 0.51-MeV gamma radiation (0.42%) from a sample of approximately 5 ml. The sensitivity, as background-equivalent concentration, is 4 $\mu\text{Ci/liter}$. If a 3-in.-diam by 2-in.-high cylindrical container were used, the sensitivity would be 0.2 $\mu\text{Ci/liter}$. Beta counting of a 10-cm³ thin-window capsule would lead to a sensitivity of 0.05 $\mu\text{Ci/liter}$, while internal beta techniques, liquid scintillation, and gas proportional counting would have even better sensitivities. To avoid the necessity of developing special methods and equipment, it was decided to continue to use the present technique, unless new requirements necessitate a more sensitive one.

B. Radioisotope Safety

1. Surveillance of Regulations and Cask Evaluations

Two papers entitled "Containers for Shipment of Radioactive Material" and "Accident Resistant Shipping Containers for Radioactive Materials" were delivered at the Second Annual Conference on Transportation of Radioactive Material, University of Virginia, Charlottesville, Virginia, October 25-27, 1970.

a. Review of Regulations

Comments were submitted to the Oak Ridge Operations Office on the Proposed Revisions to the IAEA Regulations for the Safe Transport of Radioactive Materials. A major problem area that may develop with these proposed revisions is the requirement of a shipper to demonstrate a large number of package leak rates. These rates depend on the nuclide being shipped and on whether it is being shipped as a solid, liquid, or gas. Some of the leak rates are beyond present day capability of measurement.

b. Cask Testing

The outdoor test of the heat dissipation and neutron shielding capabilities of the Lithium Hydroxide Shielded Curium Cask is continuing. Loaded with a 10⁹-n/sec curium-244 source (thermal output of 287 W), it has now been subjected to ambient temperatures ranging from 40 to 95°F in various types of summer and autumn weather and no adverse test data were found.

A stainless steel right cylindrical package 36 in. in diameter and 35.5 in. in height, with an annular space for a 4-1/4-in.-thick layer of lithium hydroxide monohydrate as fire shielding, has been fabricated.

The inner cavity is 27 in. in diameter and 26 in. in height; the total weight is 1350 lb. This package is designed to replace the ORNL Sugarman Cask Fire and Impact Shield, offering a heat dissipation capability of 500 W. This shield will be used as an outer container for any USA-DOT Specification 55 package, or equivalent, weighing 2500 lb or less. A 30-min fire test will be carried out after the shield cavity has been loaded with lithium hydroxide.

2. Tests of Promethium-147 Luminous Light Sources

The radiation output of eight unshielded commercial promethium-147 light sources is being measured using a number of radiation survey instruments. The response of an ORNL cutie-pie detector and that of a commercial 3.5-in.-diam air ionization chamber having a 1-mg/cm² Mylar window is given in Table 4. The response of these instruments to a 10.1- μ Ci ORNL standard promethium-147 source is also given.

Table 4. Response of Detectors to Radiation from Light Sources

Distance (cm)	Detector Reading (mR/hr)								
	From 0.25-in.-diam Source			From 0.50-in.-diam Source			100 μ Ci	2 mCi	10.1 μ Ci
	100 μ Ci	2 mCi	5 mCi	100 μ Ci	2 mCi	5 mCi	Watch Dial	Compass Dial	Source ^a
<u>Cutie-Pie</u>									
0	700	>10,000	>10,000	540	7,700	>10,000	240	6,400	-
1	500	9,100	>10,000	390	5,500	>10,000	160	4,500	290
2	340	6,200	>10,000	260	3,800	8,100	95	3,000	170
3	220	4,100	7,000	170	2,500	5,500	60	2,000	100
4	140	2,700	4,700	110	1,600	3,700	39	1,300	48
5	90	1,800	3,100	65	1,100	2,500	26	780	32
6	60	1,200	2,100	44	700	1,700	18	530	21
7	41	790	1,500	29	480	1,200	12	370	15
8	30	550	1,000	21	330	750	9	260	11
9	21	390	690	14	240	530	7	190	8
10	15	280	490	10	170	380	5	130	6
15	4	54	94	3	32	72	2	26	2
20	2	13	22	1	8	17	1	7	1
25	0	4	5	0	2	4	0	2	<1
<u>Ionization Chamber</u>									
0	>300	>300	>300	>300	>300	>300	155	>300	265
1	>300	>300	>300	250	>300	>300	120	>300	190
2	245	>300	>300	195	>300	>300	89	>300	125
3	180	>300	>300	140	>300	>300	64	>300	82
4	130	>300	>300	102	>300	>300	46.5	>300	56
5	95	>300	>300	74	>300	>300	31.5	>300	39
6	69	>300	>300	51.5	>300	>300	19.5	>300	26
7	50	>300	>300	37.5	>300	>300	14.5	>300	18
8	36.5	>300	>300	24.5	>300	>300	10.5	300	12.1
9	23.5	>300	>300	18.0	295	>300	6.9	235	7.9
10	17.5	>300	>300	13.0	220	>300	5.1	175	6.0
15	3.8	87	150	1.8	54	110	1.8	41.5	1.1
20	0.7	18	36	0.5	9.3	23	0.2	7.9	0.3
25	0.2	3.7	6	0.1	1.3	4.4	0.1	1.1	0.1

^aPromethium-147 standard source - not light source.

TECHNOLOGY UTILIZATION - 08-01-05

A. Information Center

The Isotopes Information Center has passed another milestone, with 20,000 documents indexed and retrievable on the Termatrix system, the most recent 11,000 of which can also be retrieved by computer. In October, requests were filled for 69 documents prepared by the Center, for 56 prepared elsewhere, and for 64 specific searches on the IIC system. Ten sales letters were translated. A list of reviews in progress is shown below.

<u>Title</u>	<u>Author(s)</u>	<u>Status, % Complete</u>
DID Research and Developments - 1970	Isotopes Information Center staff	5
Indium-113m: Preparation and Uses	Martha Gerrard	10
Flow Measurement	R. H. Lafferty, Jr.	92
Iodine-125	P. S. Baker and Martha Gerrard	25
Isotopic Methods of Examination and Authentication in Art and Archaeology	F. J. Miller, E. V. Sayre, and B. Keisch	In reproduction
List of AEC Radioisotope Customers with Summary of Radioisotope Shipments, FY 1970	Ruth Curl, compiler	25
Oceanography	Isotopes Information Center Staff	a
Patent Review - Process Radiation Development	R. E. Greene, P. S. Baker and Helen Warren	99
Potato Sprout Inhibition by Radiation	F. E. McKinney	40
Radiation vs Glass ^b	D. N. Hess	20
Radioisotopes in the Pharmaceutical Industry	P. S. Baker and Martha Gerrard	5
Radioisotopes in the Steel Industry	Republic Steel Corporation	20
Radioisotopes in X-Ray Fluorescence Analysis	R. H. Lafferty, Jr.	5
Selected Abstracts of World Literature on Production and Industrial Uses of Radioisotopes (ORNL-IIC-30, Part 3)	Martha Gerrard and P. S. Baker	30
Self-Diffusion in Liquids	F. J. Miller	75
Strontium-90	Roberta Shor, R. H. Lafferty, Jr., and P. S. Baker	90
Technetium-99m: Preparation and Uses	Martha Gerrard and P. S. Baker	40
Radioisotopes in the Textile Industry	F. J. Miller	90
Use of Radioisotopes in Sedimentology	V. Romanovsky, translated and edited by Martha Gerrard	10
Grain Disinfestation - A Worldwide Review	F. E. McKinney	20
Wood Plastics	R. E. Greene	99
Yttrium-90	Martha Gerrard	10
<u>Permuted Isotopes and Radiation Technology Indexes</u>	Helen Raaen	98
Presowing Irradiation of Seeds	Machine translation of Russian book, edited by Martha Gerrard and P. S. Baker	1st draft complete
Design and Application of X-Ray Emission Analyzers Using Radioisotopic X- or Gamma-Ray Sources	J. R. Rhodes (Martha Gerrard and R. E. Greene, eds.)	90

^aDraft completed; being held at DID.

^bIndefinite.

B. Isotopes and Radiation Technology

Galley proofing of Isotopes and Radiation Technology 8(3) was started and work on 8(4) manuscript was continued.

C. Publication

Martha Gerrard and P. S. Baker, Selected Abstracts of World Literature on Production and Industrial Uses of Radioisotopes (From NSA April-June 1970), ORNL IIC-30 (Pt. 2) (October 1970).

RADIOISOTOPE APPLICATIONS DEVELOPMENT

BASIC TECHNOLOGY DEVELOPMENT - 08-03-01

A. Analytical Application of Cerenkov Radiation

The objective of this project is to develop and evaluate new applications of Cerenkov radiation. Areas of study will be drawn from problems in fission-product analysis, environmental radioactivity, biomedical assay, and special Cerenkov counting systems.

When two or more beta-emitting nuclides are present in a Cerenkov counting sample, the observed pulse-height distribution retains the energy features of the individual nuclides. We have shown previously that mixtures of nuclide in the sample can be resolved easily with a multichannel analyzer (MCA). However, for environmental monitoring systems, the use of an MCA is probably not justified. We are studying the possibility of isotope resolution in Cerenkov systems with simple monitoring equipment. The purpose of this study is to determine not only the gross activity in a sample but to assign some value to the energy distributions of the nuclides present.

Our preliminary work with chlorine-36, yttrium-91, and phosphorus-32 is encouraging. Mixtures of chlorine-36 and yttrium-91 in aqueous media have been determined with errors ranging from 0.5 to 5%. Other mixtures with phosphorus-32 have yielded similar results. It is interesting to note that the ratio of end-point energies of chlorine-36 and yttrium-91 is only 2.2. It is known that the resolution of nuclides this close is difficult or impossible in simple, homogeneous detection systems. However, the Cerenkov threshold at 263 keV effectively changes the ratio to about 3.5 thus making the resolution of these nuclides somewhat easier. (The carbon-14:tritium ratio is 8.6.)

SYSTEMS ENGINEERING APPLICATIONS - 08-03-02

A. Oceanographic Systems 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.

Preparations for the November experiments at Point Mugu have been completed. The PDP-8/I computer and associated hardware have been checked and will be used during the November experiment to plot data on-site at Point Mugu. It will not be mounted on board the tow vessel, since it is necessary to modify and remove some equipment from the instrument shelter. These modifications will be completed before the next experiment which is scheduled to be on the East coast.

The design of the Point Mugu experiment has been completed and centers on two major efforts: (1) determination of sand transport patterns around the completed groin, and (2) volume measurements seaward of the breaker zone.

An attempt to correlate sand transport measurements from plug injection with linear transport rates from a line surface injection will be made. If a direct correlation does exist, it will greatly simplify the problem of determining volume transport. A complete description of the individual experiments and accomplishments will be reported in the December Newsletter.

RADIOISOTOPE SALES

Requests for quotations were received from Donald W. Douglas Laboratory for 800,000 Ci of promethium-147 and for tritium in quantities greater than 25,000 Ci. An order was received from Aerojet Nuclear Systems Company to cover the technical service charges for encapsulation of 208,000 Ci of strontium-90 ordered by the U. S. Navy into stainless steel liners.

A listing of current orders for promethium-147 as sources or bulk powder is shown below:

<u>Customer</u>	<u>Amount (Ci)</u>	<u>Estimated Shipping Date</u>
McDonnell Douglas Astronautics Co.	100,000	November 1970 and February 1971 ^a
Minnesota Mining & Mfg. Co.	1,000	November 1970
C.E.A., France	10,000	December 1970

^aTo be shipped from Richland, Washington.

Shipments made during October include 216 Ci of xenon-133, 15,772 Ci of tritium, 320 Ci of krypton-85, 100 Ci of tritium-helium mixture containing 10 volumes of tritium and 40 volumes of helium-4, and 40 Ci of 22.8% enriched krypton-85.

The radioisotope sales proceeds and shipments for the first three months of FY 1970 and FY 1971 are given in Table 5. It will be noted that sales have dropped approximately 30% compared to the same period last year.

Table 5. Radioisotope Sales and Shipments

Item	July thru September 1969	July thru September 1970
Inventory items	\$136,943	\$ 79,585
Major products	32,068	31,170
Radioisotope services	53,787	44,177
Cyclotron irradiations	35,806	27,376
Miscellaneous processed materials	25,489	11,665
Packing and shipping	<u>18,885</u>	<u>20,735</u>
Total Radioisotope Sales	\$302,978	\$214,708
Total Radioisotope Shipments	627	794

ADMINISTRATIVE

Travel of IDC personnel and visitors to the IDC are given in Tables 6 and 7, respectively.

Table 6. Travel of IDC Personnel

Site Visited	Purpose of Visit
Salzburg, Austria	Present paper at IAEA symposium
Philadelphia, Pennsylvania	Attend American Society for Information Sciences meeting
Cincinnati, Ohio	Attend Southeastern Chapter of the Society of Nuclear Medicine
New York, New York	Attend ANSI Committee meeting
Houston, Texas	Attend 1970 Electron Microscopy Society of America meeting
Gatlinburg, Tennessee	Attend the Analytical Chemistry Division Conference in Nuclear Technology
Decatur, Alabama	Orientation on boiling water power reactors and discussion of reactor operating problems

Table 6. continued

Site Visited	Purpose of Visit
Aiken, South Carolina	Attend Research Materials Conference at Savannah River Laboratory
Washington, D. C.	Participate in conference sponsored by American Chemical Society to discuss plans regarding publication of "Annual Reviews in Chemical Engineering"
Washington, D. C.	Consult with AEC-DID and National Heart and Lung Institute
Louisiana State University Baton Rouge, Louisiana	Give lecture and discuss cooperative program with LSU
University of Virginia Charlottesville, Virginia	Deliver two papers at Second Conference on Transportation of Radioactive Materials

Table 7. Visitors to IDC

Visitor (affiliation)	Purpose of Visit
Sanders Nuclear Corporation	Review of thulium-170 program
Lockheed Nuclear	Discuss cesium-137 source fabrication
Vanderbilt University Medical Center, Nashville	Discuss gadolinium-153 and americium-241 sources
Consultant, Los Angeles	Discuss radioisotope applications
AEC-DTIE, Oak Ridge	Tour Information Center
Library of Congress, Washington, D. C.	Tour Information Center
Administrative Systems and Data Processing, Princeton, N. J.	Tour Information Center
American Society for Information Science, Washington, D. C.	Tour Information Center
National Technical Information Service, Springfield, Virginia	Tour Information Center
Human Relations Area Files, Inc. New Haven, Connecticut	Tour Information Center

Table 7. continued

Visitor (affiliation)	Purpose of Visit
Idaho AEC	Discuss isotope applications
Grenoble, France	Tour Information Center
Fontenay-aux-Roses, France	Tour Information Center
Chemical Abstracts Service, Columbus, Ohio	Tour Information Center

PUBLICATIONS

REPORTS

A. F. Rupp, Radioisotope Program (8000) Progress Report for September 1970, ORNL-TM-3183, Oak Ridge National Laboratory.

E. Lamb, Isotopic Power Fuels Monthly Status Report for September 1970, ORNL-CF-70-10-21, Oak Ridge National Laboratory.

PAPERS PRESENTED AT SCIENTIFIC MEETINGS

D. C. Camp, J. C. Manthuruthil, A. V. Ramayya, J. H. Hamilton, and J. J. Pinajian, The Decay Scheme of ^{206}Bi , presented at Nuclear Physics Divisional Meeting of the American Physical Society, Houston, Texas, October 15-17, 1970.

F. N. Case, D. L. Kau, D. E. Smiley, and A. W. Garrison, Radiation-Induced High Pressure Oxidation of Process Effluents, presented at Symposium on the Use of Nuclear Techniques in the Measurement and Control of Environmental Pollution, Salzburg, Austria, October 26-30, 1970.

F. N. Case, E. H. Acree, and H. R. Brashear, A Survey System for Use in Tracing Radionuclide-Tagged Sediment in the Marine Environment, presented at Symposium on the Use of Nuclear Techniques in the Measurement and Control of Environmental Pollution, Salzburg, Austria, October 26-30, 1970.

R. D. Seagren, Containers for Shipment of Radioactive Material, presented at Second Conference on Transportation of Radioactive Materials, University of Virginia, October 25-27, 1970.

R. D. Seagren, Accident Resistant Shipping Containers for Radioactive Material, presented at Second Conference on Transportation of Radioactive Materials, University of Virginia, October 25-27, 1970.

Robert O. Smith, Patrick H. Lehan, Harper K. Hellems, and J. K. Poggenburg, Bolus Versus Continuously Decreasing Isotope Infusion for Myocardial Scanning, presented at Southeastern Chapter of the Society of Nuclear Medicine, Cincinnati, Ohio, October 28-31, 1970.

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