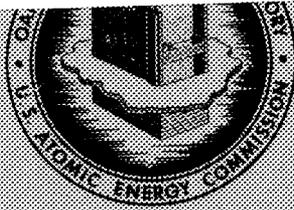


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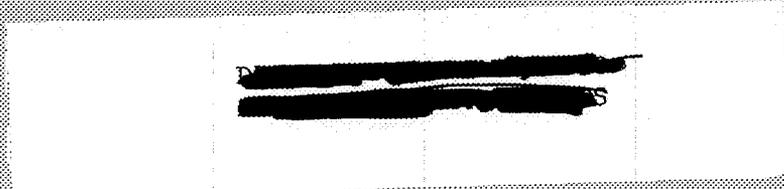
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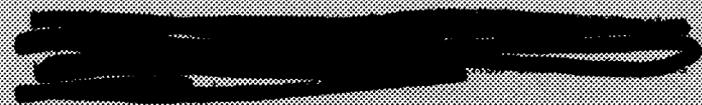
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RADIOISOTOPE PROGRAM (8000) PROGRESS REPORT
FOR DECEMBER 1971

A. F. Rupp

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ORNL-TM-3677

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Isotopes Development Center

RADIOISOTOPE PROGRAM (8000) PROGRESS REPORT
FOR DECEMBER 1971

A. F. Rupp

Work Sponsored by
AEC Division of Applied Technology

JANUARY 1972

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RADIOISOTOPE PROGRAM (8000) PROGRESS REPORT
FOR DECEMBER 1971

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

A target containing 1.06 g of 69% enriched $K^{36}Cl$ was processed after irradiation for four cycles in the HFIR. Since previous low yields were suspected to occur because of formation of polyphosphates in neutral solution, the iron hydroxide precipitation was made immediately following dissolution of the KCl target. The yield of phosphorus-33 product one week after discharge was greater than 1 Ci as was expected, thus supporting the need for expeditious completion of the first $Fe(OH)_3$ precipitation. In a previous run the dissolved target was allowed to stand for a few days with a resultant yield of about half that expected. The phosphorus-32 content which was 6.5% at processing will reach 5% on January 2 at which time there will be 735 mCi of phosphorus-33. No gamma activities were detected in the product. The phosphorus-32 content should have been at the 1% level, but a spark source analysis of the target showed the presence of natural sulfur which would introduce phosphorus-32 via the $^{32}S(n,p)^{32}P$ reaction.

The chlorine-35 abundance has decreased to a few percent and at this level the chlorine-36 will be burned out faster than it is replaced and its abundance with respect to chlorine-37 will gradually decrease. For this reason the target will be extended by blending in 53% enriched chlorine-36 to raise the abundance of chlorine-35 to the level of 5-10%.

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-43, with a half-life of 22.5 hr and gamma-ray emissions of 0.373 and 0.617 MeV, is potentially useful for metabolic and clinical studies of blood flow, rejection of transplanted organs, and kidney function because the dose rate would permit multiple doses.

Three batches of potassium-43 were shipped. Recipients included National Institutes of Health, Johns Hopkins Medical Institutes, University Hospitals of Cleveland, University of Texas Medical Center of Galveston, and V. A. Center, Wood, Wisconsin. Data for the runs are summarized in Tables 1 and 2.

Table 1. Yield of Potassium-43 from Calcium-43 Targets

	Targets 434-5	Targets 418,437	Targets 445-6
Weight ^{43}CaO (mg)	273 ^a	253 ^b	261 ^c
Irradiation time (hr)	65.95	65.5	66.0
Product delivery date	11/30/71	12/8/71	12/28/71
Total product at 8:00 AM on delivery date (mCi)			
Potassium-43	14.9	14.0	19.1
Potassium-42	1.9 (13.0%)	1.7 (12.1%)	2.3 (12.2%)

^a49% calcium-43 recovered 10/28/71.

^b61.6% calcium recovered 10/13/71 and 11/11/71.

^c49% calcium-43 recovered 12/15/71 from targets 434-5.

The batches consisted of two targets totaling over 250 mg ^{43}CaO . The objective of 20 mCi of product per batch was approached in the batch containing targets 445-6. No problem of separation of sodium and potassium was encountered even though the ion-exchanger loading resulting from processing two targets was about 50% of the column capacity. Because of this satisfactory separation, a procedure to remove most of the calcium by metathesis of the oxide to the carbonate and filtering has been abandoned because of poor yield of potassium-43. Only complete dissolution of the target assures good potassium-43 recovery, and the subsequent addition of a precipitating

Table 2. Radionuclides Found in Dissolver Sample by Gamma Scanning

Radio-nuclide	Half-Life	Activity at Discharge ^a			Activity Concentration at Discharge ($\mu\text{Ci}/\text{mg}$ of target)		
		(mCi)			Targets		
		Targets 434-5	Targets 418,437	Targets 445-6	Targets 434-5	Targets 418,437	Targets 445-6
⁴³ K	22.5 h	37.5	39.3	48	137	155	183
⁴² K	12.4 h	9.5	9.8	12	36	39	45
²⁴ Na	15.0 h	7.4	1.9	4.6	27	7.5	17
⁵⁶ Mn	2.58 h	35.5	—	44	130	—	167
^{87m} Sr	2.83 h	18.4	47.4	30	67	187	116
⁴⁷ Ca	4.53 d	0.94	0.47	1.0	3.4	1.86	3.4
⁴⁶ Sc	83.8 d	10.4	2.4	3.9	38	9.5	15
⁴⁷ Sc	3.4 d	1.1	0.56	1.0	4.0	2.2	3.9
⁶⁴ Cu	12.7 h	0.44	3.6	7.4	1.6	14	28
¹⁴⁰ La	40.2 h	—	0.35	0.04	—	1.4	0.17
¹⁵³ Sm	46.7 h	—	0.35	0.04	—	0.95	0.68
^{69m} Zn	13.8 h	—	0.18	0.4	—	0.71	1.5
¹⁹⁹ Au	3.15 d	—	—	1.1	—	—	4.5
⁵¹ Cr	27.7 d	—	0.04	0.05	—	0.16	0.19

^aTotal target.

agent involves the same amount of column loading. The target ⁴³CaO can be dissolved in a near stoichiometric amount of HCl, thus eliminating the need for boiling to dryness to remove excess HCl.

The elution of the sodium-24 and potassium-43 peaks is monitored by a shielded in-cell gamma probe. The calcium is recovered by stripping the column with 50 ml of 5 M HCl. Most of the scandium impurity is left on the column while the calcium is completely removed.

The presence of neptunium-239 (2.35 d) was also detected in the column strip solution following the 50-ml portion containing the calcium. Neptunium-239 has been a suspected contaminant in other samples, but its presence is now confirmed. The origin is believed to be from natural uranium in the latest batch of Supersil quartz used for ampuls. It had been previously observed that the broken quartz ampuls exposed to a high fluence in the HFIR would yield fission products when leached with strong HCl. The amount of neptunium-239 was only a few microcuries, but caution should be exercised in dissolving targets so as to avoid leaching the ampul fragments as much as possible. Careful checks for long-lived contaminants have been made on all batches of potassium-43, and the usual level of such contaminants is <10 nCi/ml at the reference time. Typical contaminants have been ⁵⁹Fe, ⁶⁰Co, ⁴⁷Sc, ⁵¹Cr, and ⁸⁴Rb.

B. Gadolinium-153

Gadolinium-153 (241 days) is an important radionuclide because it decays by electron capture to yield predominantly 100-keV photons. This energy range is useful in backscatter and transmission gaging, as well as having applications in other fields. Highly enriched transmutation product gadolinium-152 is prepared by chemically separating it from neutron burned-out europium control plates from HFIR. The >90% abundant gadolinium-152 is not available from calutron separations and can only be obtained at reasonable cost as a by-product of the HFIR reactor operations at the present time.

A 3-Ci batch of gadolinium-153 was purified to obtain a product with a radionuclide purity level of >99.98%. The only identifiable contaminant was 72.4-day terbium-160, which is produced from gadolinium-158. This material was encapsulated and shipped to a customer who is using it to determine traces of lead in metals.

A 5-Ci batch is currently being processed to fill several orders for gadolinium-153 sources.

C. Tellurium-123m

Several sources containing tellurium-123m have been fabricated for experiments to develop a new method for obtaining information concerning the electron distribution in matter by Compton scattering measurements. The isotope decays by isomeric transition with a half-life of 120 days and emits a 159-keV gamma with an abundance of 84%. Tellurium-123m is produced by neutron irradiation of enriched tellurium-122. Published literature¹ lists the cross section of tellurium-122 to produce tellurium-123m as approximately 1 barn.

Four irradiations, one in the HFIR for 3 days and three in the ORR for 21 days, have produced approximately 1 mCi/mg of target. This yield indicates an effective burnup cross section for tellurium-123m of approximately 8,000 barns. To test a method to increase the tellurium-123m yield, two 3-mg samples were placed in the ORR, one shielded with cadmium and one unshielded, and irradiated for 7 days. The cadmium-shielded sample produced 0.48 mCi/mg of target, and the unshielded sample produced 0.68 mCi/mg. Additional cross-section measurements are needed before the production parameters can be fully understood. Any increase in yield helps, since the enriched target is expensive and in short supply; however, the target material is being recycled to make full use of available material.

¹*Nuclear Data Sheets*, Part 5, 1959-1965.

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

<u>Processed Units</u>		<u>Service Irradiations</u>	
<u>Radioisotope</u>	<u>Amount (mCi)</u>	<u>Type</u>	<u>Number</u>
Calcium-47	4	Tellurium-122	2
Copper-67	12	Platinum-196	2

1. Copper-67

A review of the copper-67-zinc separation procedure routinely used for production of carrier-free copper-67 (61.6 h) was made. This procedure, based on solvent extraction of copper into 0.02 *M* thenoyltrifluoroacetone (TTA) in benzene from pH 3.5 aqueous solution and anion-exchange chromatography to remove traces of zinc not separated by the solvent extraction, was found to be unduly lengthy, unreliable for routine use, and inefficient. Since the final products from this earlier procedure had been shown to be physiologically suitable for human use, it was decided to stay with the basic TTA-benzene extraction system, but with greatly altered conditions and procedures.

A new procedure has been devised, based on the data of Poskanzer and Foreman,² and Scruggs, Kim, and Li,³ and tested by both tracer-level experiments and a full-scale production run. The new procedure is based on the extraction of copper into 0.2 *M* TTA in benzene from 0.01 *M* HCl solution, followed by extensive washing of the organic with 0.01 *M* HCl to remove traces of zinc-65 rather than the anion-exchange column used previously. High-resolution gamma pulse height analysis of an "aged" sample (approximately 10 days old) of the final product from the first production-scale run using this new procedure was performed to characterize the product. A product purity (exclusive of other copper isotopes) of $\geq 99.80\%$ copper-67 was determined. The major radiocontaminant was neptunium-239 (2.35 d) thought to be leached from the quartz irradiation ampul. The only other radiocontaminants detected were scandium-46 (83.8 d) and iron-59 (45.0 d), both present at a level of $< 0.01\%$. No zinc isotopes were detected in the final product, whereas the initial dissolver solution contained (as of 0800 on the day of shipment) approximately 45 mCi of zinc-65 (243.7 d) and approximately 640 mCi of zinc-69m (13.7 h). A chemical yield of $> 90\%$ was achieved for this first production run. Additional process development will be undertaken in an attempt to reduce the amount of neptunium-239 contaminant in the final product. Future production runs will be followed to determine the operating reliability of this new process and to better characterize the final products obtained.

²A. M. Poskanzer and B. M. Foreman, Jr., *J. Inorg. Nucl. Chem.* 16, 323 (1961).

³R. L. Scruggs, T. Kim, and N. C. Li, *J. Phys. Chem.* 67, 2194 (1963).

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 $^{68}\text{Zn}(p,2n)^{67}\text{Ga}$ reaction in acceptable purity and quantity and to provide gallium-67 for clinical applications research and development. Interest in this isotope has been spurred by evidence, obtained by the Medical Division of Oak Ridge Associated Universities (ORAU), of a high uptake of carrier-free gallium-67 by lymphoid tumors in both animals and humans.

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

Due to the interruption of the regular production schedule by the Christmas holidays, only three weekly shipments of 100-mCi batches of gallium-67 citrate were supplied to ORAU during this period as part of the Cooperative Group to Study Localization of Radiopharmaceuticals (CGSLR) project, with minor fractions of each batch being packaged as the chloride and supplied to others.

Development work has been continued on an electrodeposited zinc-on-copper target design during this period. Irradiation of a 10.2-mil-thick zinc plate target under the conditions described earlier⁴ yielded essentially the same production rate for the $^{68}\text{Zn}(p,2n)^{67}\text{Ga}$ reaction path as an 8.5-mil-thick plate target. These data indicate that, under the irradiation conditions and geometry used, an 8.5-mil zinc plate is "infinitely thick" for the (p,2n) reaction. The effect of target thickness will be investigated further in order to optimize the (p,2n) production rate and target cooling. Once the optimal target design has been determined, the design will be irradiated at increasingly higher proton beam currents in order to determine its failure point. Values for these parameters and their reliability will be determined before routine production with enriched zinc-68 is undertaken.

Experiments using natural zinc targets indicate that a maximum thick target production rate of approximately 1.4 Ci/mA-hr may be expected for a 98.5% enriched zinc-68 target. Such an enriched target would offer a cyclotron yield of at least 280 mCi/hr and a product with $\leq 5\%$ gallium-66 at end of bombardment.

⁴A. F. Rupp, *Radioisotope Program (8000) Progress Report for November 1971*, ORNL-TM-3653, Oak Ridge National Laboratory (December 1971).

2. Indium-111

The objectives of this program have been to define and optimize the pertinent production parameters for the economical production of indium-111 of satisfactory quality for radiopharmaceutical manufacture. The current objectives are to obtain practical operating experience of the procedures developed by pilot production and to provide this radionuclide to interested customers as a special research material -- in batch lots -- for their evaluation. Indium-111 has been suggested for such studies as spinal-cerebral cisternography, aerosol lung studies, delayed brain scanning, visualization of the lymphatic system, metabolic studies of indium-labeled macroaggregates and colloids, and tumor localization. Indium-111 has gamma emissions of 173 keV (89%) and 247 keV (94%) ideally suited for external detection and an optimal half-life (2.83 days) for labeling and distribution studies which must be carried out over 24 hr or longer.

Four batches of indium-111 were prepared and portions supplied to interested customers as a special research material. The new Friday shipping schedule, with a decay allowance as of 0800 (EST) of the next Monday, has been enthusiastically accepted by users who feel it will allow them greater flexibility in patient scheduling and radioisotope utilization.

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

Table 3 gives the December 1971 accelerator runs for ORNL and non-ORNL programs.

Table 3. Accelerator Irradiations and Runs for December 1971

Product	No. of Runs	Time (hr:min)			Total Charges
		Beam	Misc.	Total	
<u>ORNL Programs</u>					
Bismuth-206	1	1:30	1:00	2:30	\$ 245
Gallium-67	5	11:55	6:05	18:00	1,859
Indium-111	3	13:45	3:05	16:50	<u>1,693</u>
Total					\$ 3,797
<u>Non-ORNL Programs</u>					
Cobalt-57	2	100:00	2:30	102:30	\$17,433

FISSION PRODUCTS - 08-01-03

A. Krypton-85 Enrichment

A bad motor was replaced on the bellows pump of unit B (see Fig. 1). One liter of product was removed from each of units C and D and transferred into the high-level unit CD, increasing the load by 213 Ci total. The product loss was replaced in units C and D by adding 4.7% feed material. Units A and B, which had been operating at reduced pressure, were brought up to normal operating pressure (1 psig) by adding feed material.

Unit	Activity in Unit (Ci)	Time Since Last Product Removal (days)	Count Rate in Product Section (counts/min)	
			Nov. 1971	Dec. 1971
A	264	160	5,100	4,650
AB	Xenon	120	—	—
B	264	160	1,600	3,950
C	1737	15	5,800	4,100
CD	2269	120	14,600	14,850
D	1615	15	6,200	4,250

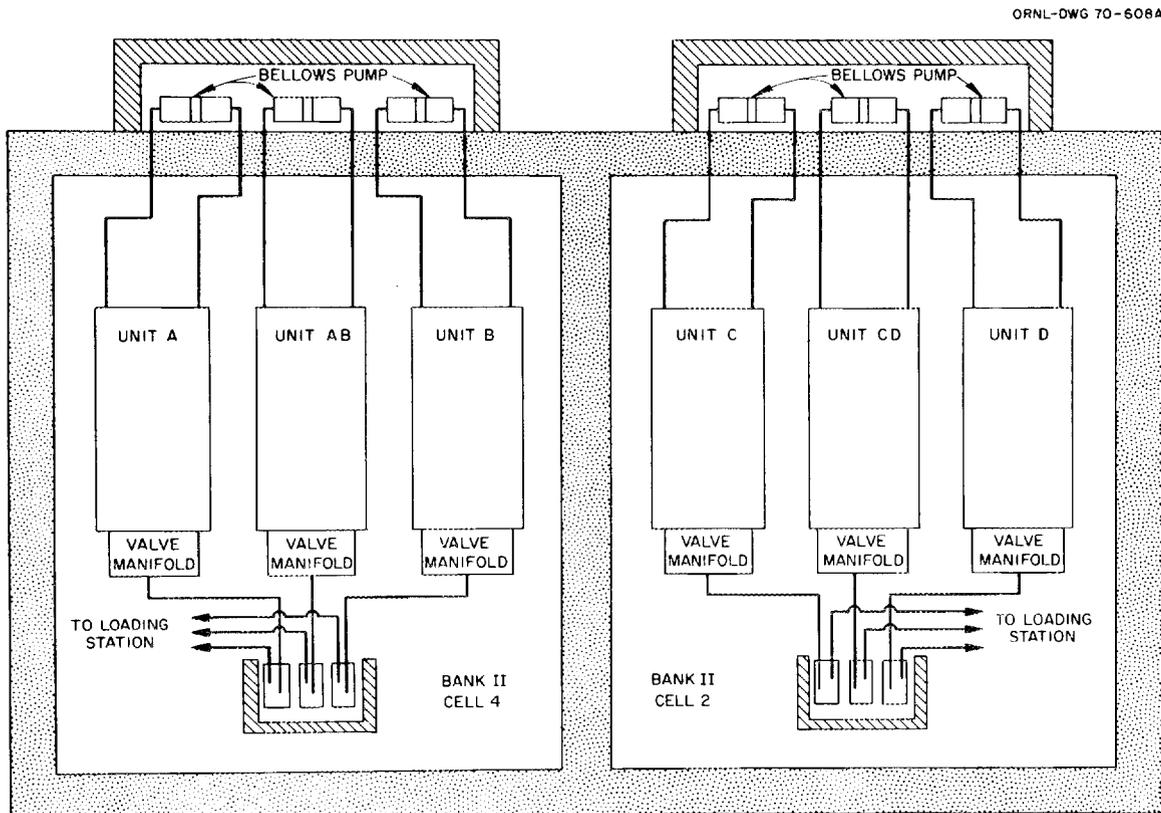


Fig. 1. Schematic Arrangement of Krypton-85 Columns.

B. Cesium-137 Gamma Source 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.

The exploratory phase of the work on the development of alternative cesium-137 source compounds has been concluded. The progress made to date is reviewed and the results are consolidated to assess significant accomplishments in this project. Summarized below are highlights of the work accomplished to date.

1. Criteria and Fundamental Consideration of Potential Source Compounds

At the outset of this project, a number of criteria were established to provide a guideline in the evaluation and selection of potential cesium-137 source compounds. Of these, the high degree of stability at elevated temperatures and the low solubility in common aqueous solutions are the two key factors that have been used in the preliminary evaluation of the cesium compounds prepared.

Fundamental consideration of the nature of various crystalline compounds revealed that compounds with high degree of thermal stability, low solubility, and good mechanical properties are generally characterized by the combination of: (a) strong ionic bond, (b) crystal structure with large lattice energy, (c) small ionic sizes, and (d) large ionic charges. These characteristics were employed as first approximations in the search for potential cesium-137 source compounds.

2. Selection of Compounds and Development of Preparation Methods

Based on the above considerations, several complex oxides, including cesium niobates, cesium tantalates, cesium vanadates, and ceramic oxides of the type $\text{Cs}_2\text{O}\cdot\text{Al}_2\text{O}_3\cdot n\text{SiO}_2$ and $\text{Cs}_2\text{O}\cdot\text{MgO}\cdot n\text{SiO}_2$ were selected as the possible candidate compounds for the cesium-137 sources. The methods of preparation previously proposed for such types of compounds were generally either

impractical or not economical for a larger-than-bench scale production. Accordingly, development of simpler and more efficient preparation methods was undertaken. The methods outlined below appear to be economically most attractive from the viewpoint of large-scale production.

a. Cesium Niobates, Tantalates, and Vanadates

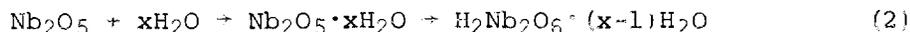
The preparation methods developed are applicable to all of these compounds, the major differences being the temperature and time in the reaction and calcination steps.

(1) Molten salt reaction: This method involves a reaction of molten CsOH (mp = 272°C) with Nb₂O₅, Ta₂O₅, or V₂O₅, for example:



A relatively high reaction rate can be maintained at elevated temperatures (~500°C), but the rate of undesirable side reaction (e.g., decomposition of CsOH and losses by vaporization) also increases with the temperature. Such difficulty can be partially overcome by carrying out the reaction under reflux so as to control release of H₂O produced in reaction (1).

(2) Slurry reaction: The basis for this method is the reaction between the aqueous solution of either CsOH or Cs₂CO₃ and suspended particles of Nb₂O₅, Ta₂O₅, or V₂O₅. The reaction can proceed by, for example:



The products from both methods (1) and (2) were calcined at temperatures ranging from 700 to 1000°C.

b. Ceramic Oxides, Cs₂O·Al₂O₃·nSiO₂ and Cs₂O·MgO·nSiO₂

The methods developed for preparation of this type of compound were considerably simpler than those that have been proposed previously by others. It requires no pressure vessel and only a relatively short time for reaction and calcination.

The method consisted of a two-step reaction by first forming a water-glass-type compound Cs₂O·nSiO₂ that is subsequently reacted with Al₂O₃, Al(OH)₃, MgO, or Mg(OH)₂. The resulting products were then calcined at 1100-1400°C.

3. Assessment of Selected Products as Potential Cesium-137 Gamma Sources

Tables 4 and 5 summarize characteristics of various selected products from the present study and indicate that tantalates, niobates, and aluminosilicates have desirable properties that make them potentially

Table 4. Characteristics of Selected Products of Cesium Tantalates, Niobates, and Vanadates

Sample No.	Atomic M/Cs Ratio ^a	Estimated Density (g/cm ³)	Seawater Insolubles (wt %) ^b	Melting Temperature ^c (°C)		Volumetric Sp Activity (Ci/cm ³) ^d
				Incongruent	Complete	
<u>Cesium Tantalates</u>						
M-11	1.28	6.7	93.7	—	1445	68
M-3	1.10	6.6	88.3	—	1385	74
M-13	1.10	6.6	90.0	—	1405	74
M-15	1.55	6.9	100.0	—	1405	60
<u>Cesium Niobates</u>						
JE-6	1.40	5.0	92.0	400;650	880	66
JE-8	1.49	5.0	90.0	875;1005	1325	64
<u>Cesium Vanadates</u>						
JY-2	1.19	—	30.7	—	<700 ^e	74
JY-4	0.88	—	5.5	—	<700 ^e	83

^aM stands for Ta, Nb, and V for tantalates, niobates, and vanadates, respectively.

^bPowder samples.

^cBased on observation under the high-temperature microscope.

^dEstimated — assume an isotopic purity of 36.5% cesium-137. The volumetric specific activity of ¹³⁷CsCl gamma source is approximately 100 Ci/cm³.

^eEstimated value.

Table 5. Characteristics of Selected Cesium Aluminosilicate Products Cs₂O·Al₂O₃·nSiO₂

Sample No.	Atomic Ratio		Estimated Density (g/cm ³)	Distilled Water Insolubles (wt %) ^a	Melting ^b Temperature (°C)	Volumetric Sp Activity (Ci/cm ³) ^c
	Si/Cs	Al/Cs				
OC-2	1.05	1.14	3.2	85.9	1305	59
OC-4	1.02	1.14	3.2	63.0	1355	60
OC-10	0.67	1.15	3.2	77.8	1695	62
OC-11	0.89	1.16	3.2	84.1	1665	60

^aPowder samples.

^bBased on observation under the high-temperature microscope.

^cEstimated — assume an isotopic purity of 36.5% cesium-137. The volumetric specific activity of ¹³⁷CsCl gamma source is approximately 100 Ci/cm³.

useful cesium-137 gamma sources. Most of these products have melting temperatures in excess of 1300°C (1305-1695°C) and are highly insoluble in distilled water or seawater (85-100% insoluble). Such results confirm the prediction from fundamental considerations of the nature of chemical bonding of various elements involved in the formation of these compounds. Their volumetric specific activities (60-74 Ci/cm³) compare favorably with that of ¹³⁷CsCl gamma source (~100 Ci/cm³).

Cesium tantalates seem to have an overall advantage with respect to volumetric specific activity (up to 74 Ci/cm³), melting temperature (~1400°C), and solubility (~90% insoluble). Although cesium niobates are also characterized by relatively high insolubility (~90%) and specific activity (64 Ci/cm³), the phenomenon of incongruent melting (e.g., at 875 and 1005°C) is an unfavorable condition that might cause some compatibility problem with the source container.

On the other hand, consideration of the density indicates that the aluminosilicates (~3.2 g/cm³) have an advantage in that the mass absorption coefficient for the gamma radiation would be appreciably smaller than either cesium tantalates (~6.6 g/cm³) or niobates (~5 g/cm³). However, the problem of relatively high mass absorption coefficient for cesium niobates might be partially overcome by a proper irradiator source design.

Cesium vanadates do not appear to be a desirable cesium-137 gamma source in view of their high solubility and low melting temperature. They do not have any advantage over the ¹³⁷CsCl gamma source.

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

1. Processing and Process Status

The cesium-137 carbonate intermediate product prepared in November was converted to cesium-137 chloride, yielding an estimated 118,000 Ci in 12 batches. A "cleanup" batch is in progress, and after its completion the product preparation equipment will be put in standby. Analysis of the products prepared this month is not complete, but preliminary analysis (four batches) indicates that the material is acceptable.

After considerable decontamination effort, the cesium-137 alum purification cell was decontaminated sufficiently to allow replacement of the agitators in both crystallizers. The planned replacement of a failed solids-addition valve in one of the vessels could not be done due to the high radiation field in its immediate vicinity. However, a minor process modification was made to permit continued operation without using that valve. Crystallization processing was resumed in mid-December. The initial decontamination solutions used for tank cleaning had been reserved; these were processed first to recover the relatively low cesium-137 in them. Then the recycled cesium-137 solutions from the previous oxalate cycle were scavenged and feed processing was begun. A fraction of approximately 200,000 Ci of purified cesium-137 alum was accumulated. The current cesium-137 process status is as follows (estimated values):

<u>Item</u>	<u>Cesium-137 (Ci)</u>
In-process material	870,000
¹³⁷ CsCl products	164,000
Sources in fabrication	0
Completed sources awaiting shipment	70,900
Total	1,104,900

2. Operational Summary

	<u>December 1971</u>		<u>FY 1972</u>	
	<u>No.</u>	<u>Curies</u>	<u>No.</u>	<u>Curies</u>
HAPO shipments received	0	0	1	414,200
Product batches prepared	12	118,000	18	188,000
Sources fabricated	0	0	32	50,000
Special form containers loaded	0	0	13	600
Sources shipped	0	0	16	18,620
Special form containers shipped	0	0	13	1,500

3. Current Orders

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

<u>Customer</u>	<u>Amount (Ci)</u>	<u>Estimated Shipping Date</u>
Brookhaven National Laboratory	203,000	FY 1972
Atomic Energy of Canada, Ltd.	49,700 ^a	February 1972
Lockheed-Georgia Company	<u>~35,000</u>	b
Total	287,700	

^aAn order is on hand for an additional ~104,600 Ci as bulk powder to be scheduled and shipped as released by customer.

^bSources are in storage awaiting receipt of customer's containers.

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

1. Processing and Process Status

A shipment of strontium-90 was received from the Atlantic Richfield Hanford Company in a HAPO-IB cask. The quoted loading on the cask is 630,000 Ci, which would make this the largest shipment received to date. The high loading was definitely indicated by the filter temperature (i.e., inside cask) which was 118°C. The highest filter temperature previously recorded was 101°C on the last shipment containing 530,000 Ci of strontium-90. Air cooling (15 cfm at 60 psi) was not sufficient to lower the filter temperature enough for direct chilled water cooling, so the cask was cooled with 80°C water to 90°C, then with 10°C water to 35°C at which temperature it stabilized. The cask was vented on December 20; no pressure was detected.

Analysis of the strontium-90 carbonate product from the test run made in November showed a rare-earth separation considerably better than that achieved in the first test run. The rare-earth-to-strontium weight ratio was reduced by a factor of 10. Two additional test runs (Nos. 3 and 4) of ~40,000 Ci each were made. Run 3 was made in the same manner as the

first two tests, first separating the rare earths by oxalate precipitation, then precipitating the strontium-90 as carbonate. On run 4 the oxalate separation was done as in the other runs, but a calculated amount of TiO_2 was added during the carbonate precipitation (as is done in the "standard" strontium titanate procedure). After analyses are completed, the carbonates from runs 2 and 3 will be dry-blended with TiO_2 and the blends calcined to Sr_2TiO_4 product material. Test pellets (10-cm diameter, ~250 W each) will then be pressed from the products of runs 2, 3, and 4. The results of the test pressings should point up differences, if any, in the pelletization characteristics of fuels made by the two methods. The dry-blend method is thought to be the better in the sense of more reproducible products, but it is considerably more time-consuming and costly than the slurry method.

The high content of inert rare earths in one batch of strontium-90 feed will make it necessary to process all of this batch by both oxalate precipitation and carbonate precipitation. The actual amount of rare earths present is not known accurately, but it is approximately 10-15 wt %.

Rare-earth analysis by two methods, emission spectrography and spark source analysis, gave results varying by nearly 70% with the spark source analysis being routinely higher. Since it was thought that the high strontium content in the samples might cause interference with the rare-earth analysis, some relatively large samples (about one gram of strontium each) were separated by crystallizing the strontium as the nitrate from concentrated HNO_3 . The resulting filtrates, which contained essentially all of the rare earths and a large fraction of the calcium present in the feed, were then analyzed by both methods. No significant differences were found between the rare-earth contents shown by direct analysis of the feed and by analysis of the separated rare earths. Again the results obtained by the two analysis methods did not agree, but for a given method the reproducibility was good. Since a given method will give reproducible results, separation efficiency can be determined; but the absolute values of rare-earth contents in finished products are questionable. Investigation of this discrepancy will continue.

The current strontium-90 process status is as follows (estimated values):

<u>Item</u>	<u>Strontium-90 (Ci)</u>
In-process material	1,465,000
Strontium-90 products	534,000
Sources in fabrication	0
Returned SNAP sources	341,000
Completed sources awaiting shipment	<u>0</u>
Total	2,340,000

2. Operational Summary

<u>Item</u>	<u>December 1971</u>		<u>FY 1972</u>	
	<u>No.</u>	<u>Curies</u>	<u>No.</u>	<u>Curies</u>
HAP0 shipments received	1	630,000	1	630,000
Product batches prepared ^a	2	75,000	3	116,000
Sources fabricated	0	0	1	49,600
Special form containers loaded	0	0	0	0
Sources shipped	2	65,000	7	160,500
Special form containers shipped	0	0	1	100
Shipments to customers	0	0	12	45,900

^aDoes not include test batch 4.

3. Current Orders

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

<u>Customer</u>	<u>Amount (Ci)</u>	<u>Estimated Shipping Date</u>
U. S. Navy (Gulf)	31,000	February 1972
U. S. Navy	324,000	a
U. S. Navy	<u>208,000</u>	a
Total	563,000	

^aFurther work and subsequent shipment pending receipt of instructions from customer.

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

<u>Isotope</u>	<u>Number of Batches</u>	<u>Amount (Ci)</u>
Xenon-133	2	~700
Iodine-131	1	50
Ruthenium-103	1	31

SOURCE DEVELOPMENT - 08-01-04

A. Radioisotope Characterization, Quality Control, and Standards

1. Radioisotope Characterization

ASTM Committee E-10 on radioisotopes and radiation effects met December 6-7. Among the items discussed were radioactivity standards, particularly cesium-137, silver-110m, and barium-140, reliable nuclear data, and

measurement techniques. The NRC report, *Nuclear Data Compilations -- The Lifeblood of the Nuclear Sciences and Their Applications (1971)*, and resulting action by the National Science Foundation were called to the attention of the Committee. The American Nuclear Society Groups, ICONS and ANS-16, were mentioned as possible agencies for coordinating efforts on methods and standards. Members described the increasing program at the National Bureau of Standards on neutron calibrations.

A table of properties of 73 radionuclides of possible medical usefulness was reviewed to assure correctness of data, and a limited survey of biomedical applications of stable isotopes was made. An inquiry from NASA on the cross section for production of selenium-75 resulted in explanation of an inconsistency in their data.

2. Radioisotope Special Analysis and Quality Control

It has been decided to proceed actively to maintain calibration capability for a variety of radionuclides, especially with respect to gamma ionization chambers. In particular, the one located in Building 4501 will be designated a reference instrument for measurement of appropriate nuclides. Committee N44 of the American National Standards Institute (ANSI) has recommended revision of NBS Handbook 80, *A Manual of Radioactivity Procedures (1961)*. Thus in due time an excellent source of information on calibrations and measurements will become available. For determination of parts-per-million concentrations of sodium in lithium-7, neutron activation analysis was recommended. The radioactivity of a Health Physics smear sample was identified as promethium-147, in collaboration with members of the Radiochemical Analyses Group. Information was given on analysis for uranium-232 in uranium-234 and for natural uranium in an ore sample, and a standard was supplied for the latter. An inquiry from the Environmental Protection Agency on methods for zirconium-95 determination was answered by reference to a monograph and two manuals.

B. Dose Rate Measurements with Extrapolation Chamber

Source measurement studies using the extrapolation chamber were continued with the first phase completed. Measurements of a large uniform source were made using three different collecting electrode sizes (3, 10, and 30 mm) as the source was gradually decreased in size. A gradual decrease in extrapolation chamber measurements was observed for each electrode as the source size was decreased. Comparison measurements were made using thermoluminescent dosimeter rods. Analysis of the data will be necessary before the true surface dose rate and the correction factors necessary to provide precise dose rate values can be determined.

C. Douglas Aircraft Company Source Rack

A source rack constructed by the Douglas Aircraft Company of Long Beach, California, was loaded with 71 strontium-90 beta sources. This device will be used in experiments to remove the static charge on jet fuel as

it is pumped into aircraft. These sources were originally fabricated for use in a beta irradiator, and after the irradiator was taken out of service the sources were removed and stored for approximately 9 months in the Graphite Reactor storage canal. The sources are 3-13/16 in. long by 0.415 in. wide by 0.105 in. thick, with a 0.010-in. wall of 347 stainless steel. Each source contains approximately 120 Ci in the form of strontium silicate microspheres imbedded in aluminum.

The sources were scrubbed until the smears read less than 100 counts/min; then the sources were placed in separate tubes containing water for 96 hr. A 1-ml sample of each test tube was monitored. Those showing more than 100 counts/min were recleaned. The sources were then smeared just prior to loading in the rack. After the rack was loaded, the entire assembly was scrubbed and smeared. No activity was detected on these smears.

D. Radiation Processing (Not Supported by O8 Program; WFO Account)

1. Radiolytic Oxidation of Dye Waste Solution

Pollution of rivers by dye waste effluent from textile mills and dye chemical manufacturers is a serious problem. Past experimental work has shown that radiolytic oxidation of the waste solutions may be a practical method of destroying certain dyes and chemicals. The purpose of this investigation is to study and develop a method of radiolytic oxidation. The method developed must be economically feasible. A cobalt-60 gamma radiation source is being used to irradiate samples under a high pressure oxygen atmosphere. The solutions include both individual dyes and mixtures of dyes. This work is being supported by the American Association of Textile Chemists and Colorists (AATCC) and the Environmental Protection Agency.

The following materials were processed through the newly installed dynamic irradiator: a 55-gal batch of standard dye solution used in earlier studies, a 40-gal batch of plant effluent from a silk screen textile plant, and a 55-gal batch from a weaving, dyeing, and finishing plant. Each effluent is typical for silk screen printing and synthetic fiber weaving, dyeing, and finishing plants. Saturation of the charcoal was calculated to occur after 20 gal of the standard dye passed through the column. The same charcoal was used for all solutions processed. The irradiation conditions were:

Flow rate	3 gal/hr
Column length	30 ft
Granular charcoal	11 lb
Dose rate to irradiator	5×10^5 R/hr ^a
Void volume in irradiator	~1 gal
Dose to solution [flowing at 3 gal/hr]	1.7×10^5 R
Column operating pressure	500 psi
Gas feed to irradiator	Oxygen

^aDose measurements to be verified by additional measurements.

Both of the authentic effluents were collected over a 1-hr period from a process stream that feeds the mill effluent treatment plant. The silk screen plant waste contained an emulsion of varsol, resins, and dispersed dyes used as silk screen printing paste for polyester fabric, plus sanitary sewage and miscellaneous detergents and surfactants. The dyeing and finishing plant waste consisted of detergents, textile finishing agents, dye bath additives, and dispersed dyes, plus untreated sanitary waste from approximately 4,000 employees. This stream was sampled as feed to the waste treatment plant.

Both wastes were filtered through a sintered polyethylene filter prior to treatment in the irradiator. This filtration removed particulates but colloidal material passed, yielding a feed that was quite cloudy going into the activated charcoal irradiator. Analyses have not been completed; however, preliminary information is shown in the tables.

The silk screen effluent was treated with charcoal alone and was found to have poor color adsorption characteristics. Only 51% of the color was removed by a combination of charcoal adsorption and Millipore filtration. This determination was made by shaking the effluent with excess granular charcoal for a period of time equivalent to its residence time in the irradiator at 3-gal/hr flow rate (20 min); filtering through 80- μ m Millipore filter media; and determining light absorbance in the 350-, 450-, and 550-m μ wavelengths to calculate a color factor. This color factor determination (see Table 6) was used in the calculation of color removal (Table 7) for all samples measured.

Table 6. Color Loss from Silk Screen Mill Effluent
by Charcoal and Millipore Filtration

Sample	Color Factor	COD (mg/liter)
Original sample after filtration through sintered polyethylene	1.555	1065
Original sample after exposure to granular charcoal and ultra filtration	0.765	876

The color factor as determined by adsorbance is sensitive to the presence of any colloidal solids present in the sample. They are observed to be white and, while they do not contribute to color as usually considered (color in solution), their contribution to absorbance of light is illustrated in Table 8.

Table 7. Color and COD Removal from Silk Screen Textile Plant Effluent by Gamma Radiation Treatment on Charcoal Bed

Sample No.	Amount Thru Irradiator (gal)	Time From Start (hr)	Flow Rate (gal/hr)	Color Factor	% Color Removal	COD (mg/liter)
Feed ^a	—	—	—	1.555		1065
1	10	5.5	2	0.159	90	
2	18	10.5	2	0.489	69	
3	23	12.5	2	0.300	81	262
4	28	14.5	2	0.174	89	
5	33	17.0	2	0.332	79	293 ^b
6	38	20.0	2	0.188	88	

^aFiltered through sintered polyethylene.

^bCOD after removal of colloids with clay was determined to be 92 mg/liter.

Table 8. The Effect of Colloids on Light Absorbance Determination

Sample No.	Color Factor Before Removal of Colloids	% Color Removal From Original	Color Factor After Colloid Removal	% Color Removal From Original
Standard ^a	1.555	—	—	—
1	0.300	81	0.089	99.4
2	0.174	89	0.059	99.6
3	0.332	79	0.029	99.9
4	0.188	88	0.032	99.8

^aUntreated.

It is apparent from these experiments that color present as dissolved color is effectively removed by irradiation in the presence of charcoal. Also, the colloids altered during the irradiation process can be easily removed by the addition of a flocculating agent producing a marked decrease in the light absorbance used as a measure of color.

Charcoal is not effective in removing the colloids. Samples that had been processed through the irradiator to an 82% color removal (not corrected for colloid effect) did not show a reduced color factor when exposed to an excess of new granular charcoal for 20 hr.

The silk screen textile effluent is especially difficult to treat due to the dispersed dyes present and the solvents and resins present in the printing inks. The odor of the effluent as received was a very strong solvent odor — varsol/pine oil combination. This odor was completely removed in the activated carbon-irradiation processing.

Waste from the dyeing and finishing plant was processed in the same way as the silk screen plant effluent. The irradiator column feed solution was pink and contained much colloidal material. Data from this run are shown in Table 9.

Table 9. Dyeing and Finishing Plant Waste Color and COD Removal by Gamma Radiation with Granular Charcoal Present

Sample No.	Amount Thru Irradiator (gal)	Time From Start (hr)	Flow Rate (gal/hr)	Color Factor	% Color Removal from Original	COD (mg/liter)
Feed	—	—	—	2.255		1894 ^a 1710 ^b
1	3	1	3	0.018	99.5	46
2	10.5	3	3	0.049	98.0	116
3	15.5	6	3	0.145	95.6	123
4	18.0	8.5	1	0.210	91.0	200
5	31.5	13.0	3	0.253	88.8	—
6	42.0	16.5	3	0.185	92.0	154
7	54.0	20.5	3	0.575	74.5	160

^a Before Millipore filtration.

^b After Millipore filtration.

The problem of colloids in the treated solution was very severe using dye plant waste; however, the colloid in the treated solution coagulated rapidly within 6 hr after removal from the irradiator from samples 1 through 4 and less slowly thereafter as the run progressed. This can be seen in the effect on the color factor shown in Table 9. As was observed with the silk screen waste, when the colloids in the irradiator column product were removed by the addition of clay, the color removal (measured by light absorbance) reached 99% (Table 10).

Table 10. Dyeing and Finishing Plant Waste. Correction of Color Absorbance Value by Removal of Colloid from Sample

Sample No.	Color Factor Before Removal of Colloids	% Color Removal From Original	Color Factor After Colloid Removal	% Color Removal From Original	COD (mg/liter)
Feed	1.126	—	—	—	—
1	0.253	88.8	0.077	99.7	77
2	0.365	84.0	0.017	99.9	77
3	0.488	78.6	0.018	99.9	a
4	0.550	76	0.031	99.8	58

^a Not determined.

The best color removal obtained by lime treatment and charcoal combination using polyethylene filtered dye plant waste without prior treatment by irradiation was 83%; by lime without charcoal, 75%.

The dyeing and finishing plant waste was examined for coliform bacteria. The results are shown in Table 11.

Table 11. Reduction in Bacteria Count
by Irradiation Processes

Sample No.	Flow Rate (gal/hr)	Colonies Coliform Bacteria per 100-ml Sample
Feed	—	1,000
1	3	90
2	2	22

TECHNOLOGY UTILIZATION - 08-01-05

A. Information Center

In December, 77 requests for information were filled by dispatching 144 items. Seven reports and articles, as well as five sales letters, were translated. Pertinent abstracts were selected from NSA and distributed to AEC's Division of Applied Technology and to Isotopes Division personnel for current awareness. Copy of the 600-word Project Summaries was sent to Reproduction, and work continued on the industry briefings. Accessioned documents in the Center now total 23,420. A list of reviews is shown on page 22.

B. Isotopes and Radiation Technology

Manuscript for *Isotopes and Radiation Technology* 9(4) was 90% completed.

C. Translations

The following reports on work done at the All-Union Scientific Research Institute of Radiation Technology, Atomizdat, Moscow, published in *Radiation Technology*, Vol. 6, 1971, were translated by Martha Gerrard:

I. N. Ivanov, N. Yu. Markun, and A. S. Shtan', Neutron Activation Analysis of Ores for Fluorite on a Conveyor, ORNL-tr-2556

Yu. N. Kazachenkov and A. S. Shtan', Use of Isotopic Neutron Sources in Resonance-Spectrum Analysis for Elements, ORNL-tr-2557

<u>Title</u>	<u>Author(s)</u>	<u>Status (% Complete)</u>
<u>Radioisotope Briefings for Industry</u>		
Aerospace	Helen P. Raaen	55
Building	F. E. McKinney	5
Coal	R. H. Lafferty	10
Food	F. E. McKinney	30
Minerals	R. H. Lafferty	80
Paint	R. E. Greene	5
Petrochemicals	H. P. Raaen	10
Plastics	R. E. Greene	40
Pulp and Paper	Martha Gerrard	50
Stone, Clay, and Glass	R. H. Lafferty	5
Textiles	Martha Gerrard	10
<u>State-of-the-Art Reviews</u>		
Iodine-125	Martha Gerrard and P. S. Baker	In Reproduction
Technetium-99m: Preparation and Uses	Martha Gerrard and P. S. Baker	65
Radioisotopes in the Textile Industry	F. J. Miller and P. S. Baker	DID Reviewing
Self-Diffusion in Liquids	F. J. Miller	75
Grain Disinfestation	F. E. McKinney	20
Snow Gaging and Water Management	U. S. Department of Agriculture	Not Received
Radioelectrochemistry	Helen P. Raaen	90
The Environment and Isotopes	Helen P. Raaen	98.5
Semiconductors	Helen P. Raaen	20
Patent Literature on Process Radiation and Irradiator Design, Part 2. British and Canadian Patents, 1950 through 1970	R. E. Greene, Helen S. Warren and P. S. Baker	57
<u>State-of-the-Art Reviews for Isotopes and Radiation Technology</u>		
Radioisotopes in the Steel Industry, Part 3	Republic Steel	Not Received
Potato Sprout Inhibition by Radiation, Part 2	F. E. McKinney	50
<u>Other Reports</u>		
Selected Abstracts of World Literature on Production and Industrial Uses of Radioisotopes, Part 3 of 1971	Martha Gerrard and P. S. Baker	80
Use of Isotopes in Monitoring and Control of Environmental Pollution	W. E. Mott and Martha Gerrard	90
Fruit and Vegetable Bibliography (revision)	F. E. McKinney	In Reproduction
Special Sources (supplement)	F. E. McKinney	80
Bibliography of DID Contractor Publications	P. S. Baker	90
Isotope Education Experiments (Elementary)	Roberta Shor, R. H. Lafferty, and P. S. Baker	In Reproduction
Isotope Education Experiments (Advanced)	Roberta Shor, R. H. Lafferty, and P. S. Baker	50
List of AEC Radioisotope Customers, FY 1971	Ruth Curl (compiler)	90
<u>Translations</u>		
Effect of Gamma Radiation on Nonalcoholic Drinks and Wine-Brandy Products (Russian book)	Martha Gerrard and P. S. Baker	95
Presowing Irradiation of Seed (Russian book)	Martha Gerrard	50
Russian Food-Irradiation Conference	Martha Gerrard and P. S. Baker	20

A. Ya. Malyshev, A. I. Ragozinskii, N. P. Korotkov, A. I. Dmietriev, and B. P. Sapelkin, Correlation of Output Parameters of Radioisotopic Thermoelectric Generators (IZOTEGs) with Load, ORNL-tr-2558

N. P. Korotkov, A. I. Dmietriev, A. Ya. Malyshev, and B. P. Sapelkin, Dependability of Radioisotopic Thermoelectric Generators, ORNL-tr-2559

V. N. Chubukov, N. F. Miroshin, and T. I. Ermakova, Modification of Polyurethane Compounds Based on Castor Oil and Diisocyanates for Use in High-Voltage Terminals in Radiation Technology, ORNL-tr-2560

V. N. Antonov, V. I. Gorgonov, R. V. Dzhagatspanyan, N. G. Kon'kov, O. N. Krutetskaya, Yu. G. Lyaskin, V. B. Osipov, and Yu. A. Panin, Initiation of Chemical Reactions with Accelerator Electrons, ORNL-tr-2561

R. Constant, Development of Iodine-131 Production Techniques, Belgian Rpt. BLG-442, May 1970, translated from French by Martha Gerrard, ORNL-tr-2430.

RADIOISOTOPE SALES

A request for quotation was received from Radiation International Corporation for 100,000-200,000 Ci of cesium-137 as BNL-type strips. Orders were received for enriched krypton-85 from Measurex Corporation for 100 Ci, American Atomics Corporation for 50 Ci, and International Chemical and Nuclear Corporation for 40 Ci. Orders were received from Radiochemical Centre, England, for 200 Ci of normal krypton-85 and from Western Electric Company, Allentown, Pennsylvania, for 150 Ci of normal krypton-85.

Shipments made during the month include 2,000 Ci of tritium to Canrad Precision Industries, 3,000 Ci to New England Nuclear Corporation, and 10,000 Ci to Radium Chemie A. Zeller and Company. Three curies of gadolinium-153 was shipped to United Aircraft Corporation.

The radioisotope sales proceeds and shipments for the first five months of FY 1971 and FY 1972 are given in Table 12.

Table 12. Radioisotope Sales and Shipments

Item	July thru November 1970	July thru November 1971
Inventory items	\$175,532	\$148,101
Major products	52,015	24,247
Radioisotope services	139,995	125,449
Cyclotron irradiations	56,078	58,514
Miscellaneous processed materials	30,710	30,281
Packing and shipping	35,145	30,705
Total	\$489,475	\$417,297
Number of radioisotope shipments	1,334	1,081

ADMINISTRATIVE

Travel by IDC personnel and visitors to the IDC are given in Tables 13 and 14.

Table 13. Travel of IDC Personnel

Site Visited	Purpose of Visit
Tipp City, Ohio	Consult with light gas gun vendor
Vanderbilt University	Conduct holmium-166m experiment

Table 14. IDC Visitors

Visitors (affiliation)	Subject Discussed
General Electric	Isotopic power

PUBLICATIONS

JOURNALS

L. C. Brown, Chemical Processing of Cyclotron-Produced ^{67}Ga , *Int. J. Appl. Radiat. Isotop.* 22, 710-13 (1971).

REPORTS

E. Lamb, *Isotopic Power Fuels Monthly Status Report for November 1971*, ORNL CF-71-12-10, Oak Ridge National Laboratory.

R. G. Niemeyer, *Measurements of Tritium Release From Gas Chromatograph Detector Foils*, ORNL-TM-3633, Oak Ridge National Laboratory (December 1971).

T. C. Quinby, *Jacketing of Source Compound During Hot Pressing*, ORNL CF-71-12-4, Oak Ridge National Laboratory (Dec. 6, 1971).

A. F. Rupp, *Radioisotope Program (8000) Progress Report for November 1971*, ORNL-TM-3653, Oak Ridge National Laboratory.

INTERNAL DISTRIBUTION

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EXTERNAL DISTRIBUTION

53. G. A. Andrews, ORAU, Medical Division, Oak Ridge, Tennessee
54. Hal Atkins, Brookhaven National Laboratory, Upton, New York
55. D. S. Ballantine, AEC, Washington, D. C.
- 56-59. R. F. Barker, AEC, Washington, D. C.
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