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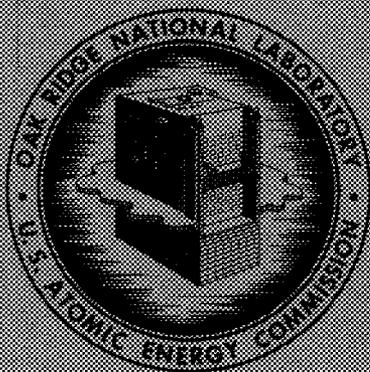
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REVIEW OF RADIOISOTOPES PROGRAM, 1965

J. H. Gillette



**OAK RIDGE NATIONAL LABORATORY**

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ISOTOPIES DIVISION

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J. H. Gillette

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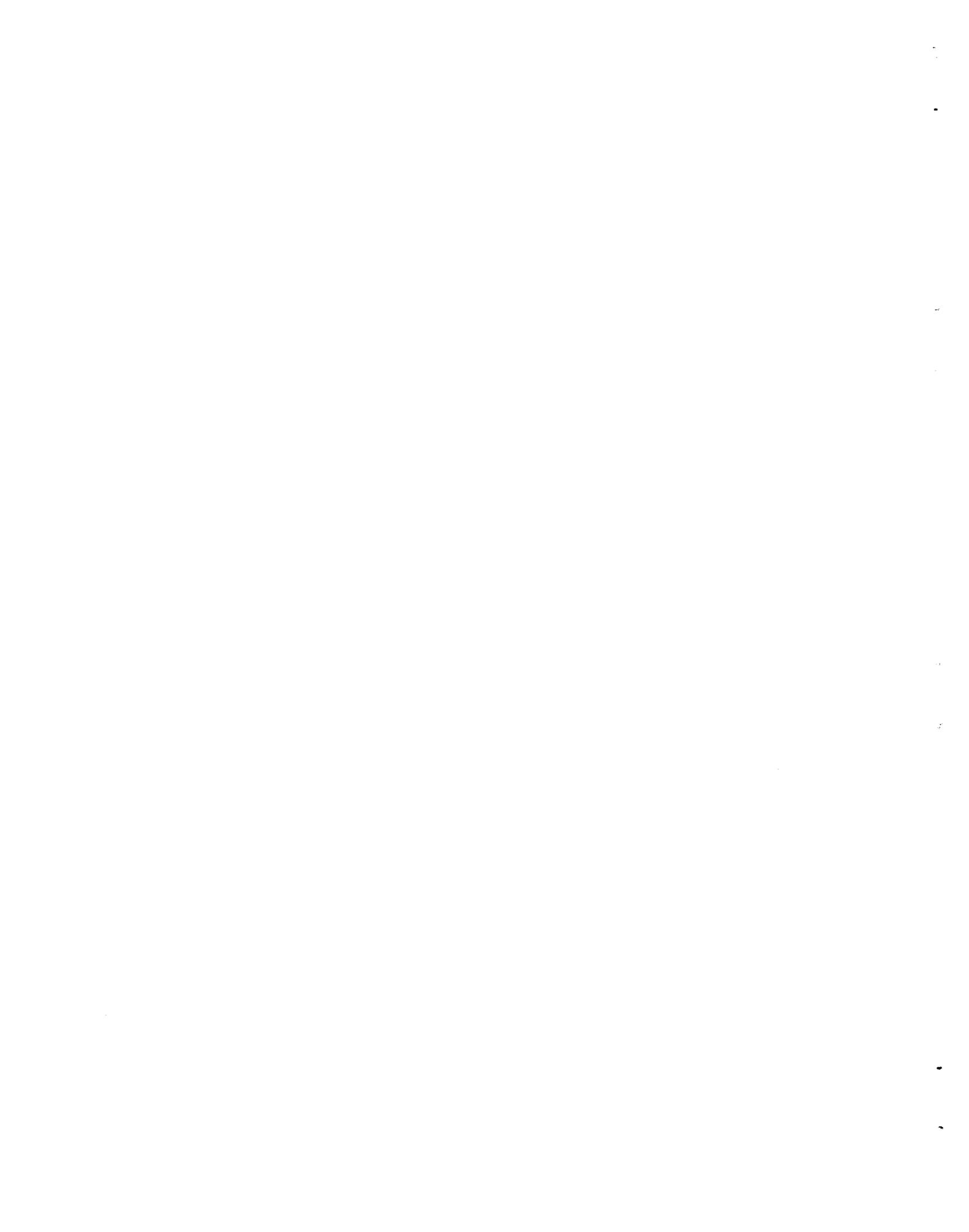
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## REVIEW OF RADIOISOTOPES PROGRAM, 1965

J. H. Gillette

### RADIOISOTOPE APPLIED RESEARCH

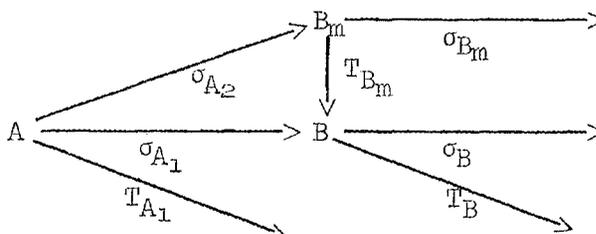
The purposes of the Radioisotope Applied Research Program are to develop and evaluate techniques for producing new radionuclides, to improve methods for preparation of current products, and to characterize the quality of all products. As the uses of radionuclides increase, the specific requirements change, and often new techniques must be developed to meet them. The fields of medicine and biology offer particular challenge because of the stringent requirements for high specific activity and purity and the need for special radiation properties (e.g., half-life). Where nuclear property data are lacking or uncertain, special radionuclide preparations are made from which the needed information may be obtained. The principal radionuclide production facilities are the Oak Ridge Research Reactor (ORR) and the ORNL 86-Inch Cyclotron.

#### ORR and LITR Neutron Flux Measurements

The neutron flux patterns in the ORR and LITR facilities used by the Isotopes Development Center were measured by irradiating monitor specimens in these facilities at full reactor power. Fast neutron flux values were calculated from the  $^{54}\text{Mn}$  yields obtained from the  $^{54}\text{Fe}(n,p)^{54}\text{Mn}$  reaction (fission neutron cross section, 54 mb) in both natural iron and stainless steel wires. The thermal neutron fluxes were calculated from the  $^{60}\text{Co}$  yields obtained from the  $^{59}\text{Co}(n,\gamma)^{60}\text{Co}$  reaction (cross section, 37 barns). The magnitude of the neutron flux and the ratio of the fast-to-thermal neutrons must be known in order to select the optimum position for irradiation and to interpret the results. These data represent the first compilation of fast neutron fluxes for these facilities.

#### Computer Programming for Isotopes Calculations

A computer code, ISOSEARCH, was developed for the CDC 1604-A computer using FORTRAN-63 language. The program is based on the general solutions of the Bateman Equation and utilizes a searching routine which starts with minimum and maximum estimates of the unknown value and arrives at the value that fits the input data. Any unknown value can be found if the other values are known in the generalized reaction scheme:



where

$$\begin{aligned} \sigma_{A_1}, \sigma_{A_2}, \sigma_{B_m}, \text{ and } \sigma_B &= \text{cross sections,} \\ T_{A_1}, T_{B_m}, \text{ and } T_B &= \text{half-lives, and} \\ A, B_m, \text{ and } B &= \text{target isotopes.} \end{aligned}$$

Two other codes of more limited application were developed. One is used to analyze economic factors in radionuclide production and can show, for example, that using an enriched isotope as target material may actually result in a more economical radionuclide preparation even though material costs are higher. The other program is a general routine which is adapted to calculate successive neutron capture chains by numerically integrating differential rate equations. This code can be easily adapted to cyclic changes of flux and can treat alpha decay within a chain without the danger of singularities that can arise in the general Bateman solution. This code will be able to handle the effects of flux variation encountered in the HFIR cycle.

#### Thermal Neutron Reactions

Several radioisotopes of long half-life were prepared with high specific activities by thermal neutron reactions. Parameters affecting the production of long-lived radioisotopes were determined to optimize production yields.

#### Potassium-40

Potassium-40 ( $1.3 \times 10^9$  y) is a long-lived radioisotope useful in instrument calibration and as a tracer. Although small quantities of enriched  $^{40}\text{K}$  (natural abundance, 0.0118%) have been obtained from natural potassium by electromagnetic separation, the isotope is very expensive. In order to reduce costs, continuing efforts are being made to prepare 1-3%  $^{40}\text{K}$  by neutron capture [ $^{39}\text{K}(n,\gamma)^{40}\text{K}$  reaction] from natural potassium and then to isolate the isotope by electromagnetic separation. The best feed material previously available contained 0.19%  $^{40}\text{K}$  (reactor produced). Mass spectrometry of a sample of natural potassium, discharged this year, irradiated for 2.1 yr at a thermal neutron flux of  $1.6 \times 10^{14}$  n/cm<sup>2</sup>.sec showed a  $^{40}\text{K}$  content of 1.33%; a cross section of 1.4 barns was measured for the  $^{39}\text{K}(n,\gamma)^{40}\text{K}$  reaction and <70 barns for the  $^{40}\text{K}(n,\gamma)^{41}\text{K}$  reaction.

#### Chlorine-36

Chlorine-36 ( $3 \times 10^5$  y) is important as a radioactive tracer and as a reactor target material for the production of  $^{33}\text{P}$  by the  $^{36}\text{Cl}(n,\alpha)^{33}\text{P}$  reaction. An irradiation study of six samples of chlorine in the ORR reactor neutron flux showed that the respective cross sections are ~30 and <10 barns for the  $^{35}\text{Cl}(n,\gamma)^{36}\text{Cl}$  and  $^{36}\text{Cl}(n,\gamma)^{37}\text{Cl}$  reactions; therefore, high specific activity  $^{36}\text{Cl}$  can be prepared by long-term irradiation of natural chlorine in a high neutron flux. Several kilograms of natural chlorine as KCl are being irradiated in the Savannah River reactor at a flux of  $>10^{15}$  n/cm<sup>2</sup>.sec.

Cadmium-109

Cadmium-109 can be prepared by the neutron reaction  $^{108}\text{Cd}(n,\gamma)^{109}\text{Cd}$  (most commonly used method) and by the sequence  $^{107}\text{Ag}(n,\gamma)^{108}\text{Ag} \xrightarrow{\beta^-} ^{108}\text{Cd}(n,\gamma)^{109}\text{Cd}$ . The yield of  $^{109}\text{Cd}$  by either reaction is rather low, indicating unfavorable cross sections. The thermal neutron cross section for the  $^{108}\text{Cd}$  reaction was determined to be  $\sim 1.1$  barns by the change in isotopic composition of enriched  $^{108}\text{Cd}$  irradiated in the ORR. Similarly, the thermal neutron cross section of  $^{109}\text{Cd}$  was found to be 650 barns by irradiation of the carrier-free nuclide produced by the  $^{109}\text{Ag}(p,n)^{109}\text{Cd}$  reaction.

Carbon-14

Aluminum nitride is being investigated as a substitute for  $\text{Be}_3\text{N}_2$  in the production of  $^{14}\text{C}$  (5730 y). Test pellets are being irradiated at high thermal flux in a Savannah River reactor and test data should be available by mid-1966. The carbon content of commercially available AlN was found to be too high for it to be used as a  $^{14}\text{C}$  target, and several grams of this compound with a carbon content of  $\sim 50$  ppm were synthesized for these tests. Although the nitrogen density of AlN is not as high as that of  $\text{Be}_3\text{N}_2$ , the fact that the normal carbon content of AlN can be reduced together with its low chemical toxicity make the use of AlN attractive from a processing standpoint.

Production runs during 1965 using  $\text{Be}_3\text{N}_2$  pellets and the processing technique developed in 1964 continued to achieve a  $^{14}\text{C}$  to  $^{13}\text{C}$  ratio of  $\sim 9$  to 1.

Epithermal Neutron Reactions Using Cadmium Filtration

Yields of radioisotopes were determined from neutron reactions which mainly require thermal neutrons but also have a large resonance integral for epithermal neutrons. In many cases, the resonance integral can be effectively used to produce the desired radionuclide while undesirable thermal neutron reactions are reduced by shielding the target with cadmium absorbers. A high resonance integral significantly increases the yield of the desired radioisotope if the sample is irradiated in a reactor neutron flux having a significant amount of epithermal neutrons. Since this type of neutron flux is available in the ORR as well as in the HFIR, these studies are applicable to radioisotope production in both reactors. In the following summaries of experimental work, the thermal neutron flux was monitored with cobalt (thermal cross section, 37 barns) and the epithermal flux monitored with gold (resonance integral, 1570 barns) wrapped in 40-mil-thick cadmium. Targets wrapped in cadmium also used the 40-mil material.

Gold-198

Gold-198 preparations, previously available with activity concentrations of 40-60 curies/g of  $^{198}\text{Au}$  (64.8 h) containing 5%  $^{199}\text{Au}$ , have been improved by the use of cadmium-shielded gold targets to yield 300-500 curies/g of  $^{198}\text{Au}$  with  $< 0.1\%$   $^{199}\text{Au}$ . The increased concentrations have been possible because the  $^{197}\text{Au}(n,\gamma)^{198}\text{Au}$  reaction has a high resonance integral while the  $^{198}\text{Au}(n,\gamma)^{199}\text{Au}$  thermal neutron reaction is inhibited by the use of cadmium filters.

Tantalum-182

In a thermal neutron flux,  $^{182}\text{Ta}$  (112 d) is formed by the  $^{181}\text{Ta}(n,\gamma)^{182}\text{Ta}$  reaction (cross section, 21 barns). Subsequent activation of  $^{182}\text{Ta}$  to  $^{183}\text{Ta}$  (5.2 d) by the  $^{182}\text{Ta}(n,\gamma)^{183}\text{Ta}$  reaction (cross section, 17,000 barns) decreases the specific activity of  $^{182}\text{Ta}$ . The  $^{183}\text{Ta}$  content of  $^{182}\text{Ta}$  was reduced to a low level by filtering out the thermal neutrons and producing  $^{182}\text{Ta}$  by epithermal neutron capture. After a 7-day irradiation of tantalum without a cadmium shield in the ORR at a flux of  $2.0 \times 10^{14}$  n/cm<sup>2</sup>.sec, the amounts of  $^{182}\text{Ta}$  and  $^{183}\text{Ta}$  obtained were 3.3 and 12 mc/mg of tantalum, respectively. Using the cadmium shield, yields of  $^{182}\text{Ta}$  and  $^{183}\text{Ta}$  of 1.8 and  $\leq 0.05$  mc/mg of tantalum, respectively, were obtained.

Cobalt-58

Irradiation of nickel in a fast neutron flux produces  $^{58}\text{Co}$  (71 d) and  $^{58\text{m}}\text{Co}$  (9 h) by the  $^{58}\text{Ni}(n,p)^{58},^{58\text{m}}\text{Co}$  reaction, but the primary yield of  $^{58},^{58\text{m}}\text{Co}$  is reduced by subsequent thermal neutron capture by the  $^{58\text{m}}\text{Co}(n,\gamma)^{59}\text{Co}$  reaction (cross section, 140,000 barns) and by the  $^{58}\text{Co}(n,\gamma)^{59}\text{Co}$  reaction (cross section, 2,500 barns). The effectiveness of cadmium filtration in reducing burnup of  $^{58}\text{Co}$  is demonstrated by the fact that the yield of  $^{58}\text{Co}$  from a cadmium-shielded nickel target was 37 mc/g and from an unshielded sample was 31 mc/g when both targets were irradiated in a flux of  $2.0 \times 10^{14}$  n/cm<sup>2</sup>.sec for 7.5 days.

Gold-199

Carrier-free  $^{199}\text{Au}$  (3.15 d) [formed by the  $^{198}\text{Pt}(n,\gamma)^{199}\text{Pt} \xrightarrow{\beta^-} ^{199}\text{Au}$  reaction] is routinely prepared in the ORR by the irradiation of natural platinum. Normally, trace amounts of  $^{198}\text{Au}$ ,  $\sim 0.1$ - $0.5\%$ , are found in the  $^{199}\text{Au}$  product. Spectrographic analysis of the platinum targets determined that the gold content was  $< 1$  ppb; therefore, the  $^{198}\text{Au}$  could not be produced from gold contamination. The most probable reaction path by which  $^{198}\text{Au}$  could be introduced into the product is by the  $^{196}\text{Pt}(n,\gamma)^{197}\text{Pt} \xrightarrow{\beta^-} ^{197}\text{Au}(n,\gamma)^{198}\text{Au}$  reaction. Experiments were conducted to determine if  $0.1$ - $0.5\%$   $^{198}\text{Au}$  would be produced in the  $^{199}\text{Au}$  product after irradiation for  $\sim 60$  hr in a flux of  $10^{14}$  n/cm<sup>2</sup>.sec, and if so, to find a method of eliminating the  $^{198}\text{Au}$ .

Three 10-mg samples of natural platinum were irradiated in the ORR, with and without a 40-mil cadmium thermal-neutron absorber. Two samples had been purified by solvent extraction to remove traces of gold; the other was unpurified. The yield results are given in Table 1.

Table 1. Gold-198 Content of Gold-199 Product

Target, 10 mg Pt	Yield, mc	
	$^{199}\text{Au}$	$^{198}\text{Au}$
Unpurified and without 40-mil cadmium shield	20.7	0.024
Purified and without 40-mil cadmium shield	20.6	0.024
Purified and with 40-mil cadmium shield	12.0	$< 0.002$



section, 60 mb). Research facilities used in these studies were a partial fuel element and selected fuel core positions of the ORR.

### Partial Fuel Element

The measured yields of the four reactions given in Table 2 were normalized to 1 g of natural element target and to a fission neutron flux of  $2.0 \times 10^{14}$  n/cm<sup>2</sup>.sec for a 7-week irradiation period in the partial fuel element. The iron products were separated from the target materials by extraction with methyl isobutyl ketone from a hydrochloric acid solution of the targets. The cobalt products were recovered by anion exchange with Dowex 1-X10.

Table 2. Radionuclide Yields Obtained After a Seven-Week Irradiation in the Partial Fuel Element

Mode of production	Number of determinations	Average yield, mc per g of target*
$^{59}\text{Co}(n,p)^{59}\text{Fe}$	3	28
$^{62}\text{Ni}(n,\alpha)^{59}\text{Fe}$	3	0.41
$^{58}\text{Ni}(n,\alpha)^{55}\text{Fe}$	1	6.1
$^{58}\text{Ni}(n,p)^{58}\text{Co}$	4	580
$^{60}\text{Ni}(n,p)^{60}\text{Co}$	4	0.59

\* Natural element.

### Other Fuel Core Facilities

Cerium-141. A  $^{141}\text{Ce}$  yield of 0.5 mc per g of praseodymium was obtained from a 50-mg  $\text{Pr}_2\text{O}_3$  target by the  $^{141}\text{Pr}(n,p)^{141}\text{Ce}$  reaction. The target was irradiated for ~19 days in an ORR position with a thermal neutron flux of  $1.9 \times 10^{14}$  n/cm<sup>2</sup>.sec and a fission neutron flux of  $1.6 \times 10^{14}$  n/cm<sup>2</sup>.sec. After dissolving the target in 10 M  $\text{HNO}_3$ , bromate was added to oxidize the cerium to the quadrivalent state and the cerium was extracted with 0.15 M D2EHPA in *n*-heptane. No radiochemical impurities were observed with gamma spectroscopy.

Cesium-136. A 2.2-g target of high-purity  $\text{La}_2\text{O}_3$  was irradiated for ~14 days in a fission neutron flux of  $1.6 \times 10^{14}$  n/cm<sup>2</sup>.sec to yield 31.4  $\mu\text{c}$  of  $^{136}\text{Cs}$  by the  $^{139}\text{La}(n,\alpha)^{136}\text{Cs}$  reaction. Carrier-free cesium was recovered from the target by three successive rare-earth oxalate precipitations using lanthanum carrier.

Rubidium-84. Rubidium-84 was prepared by the  $^{84}\text{Sr}(n,p)^{84}\text{Rb}$  reaction. The  $^{84}\text{Sr}$  and  $^{84}\text{Rb}$  were separated by using Bio-Rex 70, 100-200 mesh ion-exchange resin and eluting with 0.5 M ammonium acetate. The yield of  $^{84}\text{Rb}$  was 79  $\mu\text{c}$  per 13 mg of 83.3%  $^{84}\text{Sr}$  for a 5-day irradiation in the ORR at a fission neutron flux of  $1.6 \times 10^{14}$  n/cm<sup>2</sup>.sec.

Argon-37. A 0.5-g sample of  $^{40}\text{Ca}$  as  $\text{CaCO}_3$  was irradiated for 5 months in a fission neutron flux of  $1.9 \times 10^{14}$  n/cm<sup>2</sup>.sec. The yield of  $^{37}\text{Ar}$  by the  $^{40}\text{Ca}(n,\alpha)^{37}\text{Ar}$  reaction was 243 mc at reactor discharge time. Based on these data, the cross section of the  $^{40}\text{Ca}(n,\alpha)^{37}\text{Ar}$  reaction for ORR fission neutrons is 13.2 mb.

Sulfur-35. High specific activity  $^{35}\text{S}$  is produced in curie quantities by the  $^{35}\text{Cl}(n,p)^{35}\text{S}$  reaction in the ORR. The fission neutron cross section of the  $^{35}\text{Cl}(n,p)^{35}\text{S}$  reaction was determined from numerous irradiations of KCl and NaCl to be  $\sim 810 \pm 40$  mb.

Sulfur-35 was recovered in >99% yield from either the NaCl or KCl target by sorption on 50-100 mesh Bio-Rad HZO-1 (hydrous  $\text{ZrO}_2$ ) followed by elution with 1 N  $\text{NH}_4\text{OH}$ .

Arsenic-74. A sample of 13%  $^{74}\text{Se}$  in a 40-ml cadmium shield was irradiated for 4.45 days in the ORR at a fission neutron flux of  $1.3 \times 10^{14}$  n/cm<sup>2</sup>.sec. The yield of  $^{74}\text{As}$  prepared by the  $^{74}\text{Se}(n,p)^{74}\text{As}$  reaction was 4.2  $\mu\text{c}$  of  $^{74}\text{As}$  per mg of 13% selenium target. From these data, the fission neutron cross section of the reaction was determined to be  $\sim 6.6$  mb.

Phosphorus-33. Phosphorus-33 (25 d) was prepared in millicurie quantities by two methods. One method consisted of irradiating  $^{35,36}\text{Cl}$  in a fast neutron flux to yield  $^{32,33}\text{P}$  by the  $^{35}\text{Cl}(n,\alpha)^{32}\text{P}$  and  $^{36}\text{Cl}(n,\alpha)^{33}\text{P}$  reactions and depleting the 14.3-d  $^{32}\text{P}$  by decay.

Several developmental runs were made using  $\text{K}^{35,36}\text{Cl}$  targets to produce research quantities of  $^{33}\text{P}$  by the  $^{36}\text{Cl}(n,\alpha)^{33}\text{P}$  reaction to obtain production data needed to evaluate the preparation of >95%  $^{33}\text{P}$  in millicurie quantities. Three samples of KCl with an isotopic composition of 37%  $^{35}\text{Cl}$ , 38%  $^{36}\text{Cl}$ , and 25%  $^{37}\text{Cl}$  were irradiated in a fission neutron flux of  $\sim 1.2 \times 10^{14}$  n/cm<sup>2</sup>.sec in the ORR. These samples contained the highest enrichment of  $^{36}\text{Cl}$  available (produced from long-term irradiation of natural chlorine). The experimental conditions and results of these irradiations are given in Table 3.

Table 3. Production of Phosphorus-33 by the  $^{36}\text{Cl}(n,\alpha)^{33}\text{P}$  Reaction

KCl target, g of Cl	Time irradiated, months	Time aged, months	$^{33}\text{P}$ yield,* mc
0.25	3	4.0	$\sim 2$
0.25	4	$\sim 3.5$	$\sim 3$
0.55	5	3.4	7.5

\*5%  $^{32}\text{P}$  impurity.

The preliminary runs demonstrated that 95%  $^{33}\text{P}$  containing 5%  $^{32}\text{P}$  could be produced in millicurie amounts. The 95%  $^{33}\text{P}$  purity was arbitrarily set, and if larger amounts of  $^{32}\text{P}$  were acceptable to the user, less aging would be required and yields of acceptable quality  $^{33}\text{P}$  would be higher. If highly

enriched  $^{36}\text{Cl}$  (>90%) were available, little aging would be required and a larger amount of 95%  $^{33}\text{P}$  could be obtained than with the present material.

The  $^{32}\text{S}(n,p)^{32}\text{P}$  and  $^{33}\text{S}(n,p)^{33}\text{P}$  reactions were also studied in a fission neutron flux; however, the  $^{32}\text{S}(n,p)^{32}\text{P}$  reaction has a larger cross section than the  $^{33}\text{S}(n,p)^{33}\text{P}$  reaction and unless >70% enriched  $^{33}\text{S}$  is available, the production of suitable  $^{33}\text{P}$  (low in  $^{32}\text{P}$ ) by this method is only marginally satisfactory.

The  $^{33}\text{S}(n,p)^{33}\text{P}$  reaction also has a thermal neutron cross section, whereas the  $^{32}\text{S}(n,p)^{32}\text{P}$  reaction does not. Irradiation of  $^{32,33}\text{S}$  in a very pure thermal flux yielded >99%  $^{33}\text{P}$ . Two samples of enriched  $^{33}\text{S}$  were irradiated in highly thermalized neutron positions of the D<sub>2</sub>O column of the CP-5 reactor at Argonne National Laboratory at a thermal neutron flux of  $2.1 \times 10^{13}$  n/cm<sup>2</sup>.sec and a fission-to-thermal neutron flux ratio of 1:200. The thermal neutron reaction,  $^{33}\text{S}(n,p)^{33}\text{P}$ , has the advantage that >99%  $^{33}\text{P}$  is obtained with no aging period required; however, only milligram quantities of sulfur with >60%  $^{33}\text{S}$  are available. The experimental conditions and yields of two irradiations are given in Table 4. A cross section of 2.4 mb for the thermal neutron  $^{33}\text{S}(n,p)^{33}\text{P}$  reaction was obtained in both cases.

Table 4. Production of Phosphorus-33 by the  $^{33}\text{S}(n,p)^{33}\text{P}$  Reaction

Target, mg of S	Irradiation time, days	$^{33}\text{P}$ yield, mc
50 (68.1% $^{33}\text{S}$ )	47	0.6
275 (61.1% $^{33}\text{S}$ )	23	~2

Potassium-43. Three samples of calcium enriched in  $^{43}\text{Ca}$  were irradiated in the ORR to determine the yield and radiochemical purity of  $^{43}\text{K}$  produced by the  $^{43}\text{Ca}(n,p)^{43}\text{K}$  reaction. The three ~11-mg samples (as CaO) had an isotopic composition of 23.5%  $^{40}\text{Ca}$ , 2.5%  $^{42}\text{Ca}$ , 49.3%  $^{43}\text{Ca}$ , and 15.7%  $^{44}\text{Ca}$ . After a 66-hr irradiation in a fission neutron flux of  $1.1 \times 10^{14}$  n/cm<sup>2</sup>.sec, the average yield was 5.98  $\mu\text{c}$  of  $^{43}\text{K}$  per mg of enriched target. These data show that the cross section for the  $^{43}\text{Ca}(n,p)^{43}\text{K}$  reaction for fission neutrons is 0.3 mb. The  $^{42}\text{K}$  content of  $^{43}\text{K}$  was <5% at the time of reactor discharge.

#### Radioisotope Generators

Radioisotope generators or "cows" are particularly useful for supplying short-lived radionuclides for medical purposes. A short-lived daughter is eluted from an adsorption column in which a long-lived parent is fixed to provide a ready source of the radionuclide throughout the practical life of the parent.

#### Cesium-131

Cesium-131 which decays entirely by electron capture with a single 0.030-Mev  $^{131}\text{Xe}$  K X ray is used in diagnostic heart function studies. A  $^{131}\text{Cs}$  (9.7 d) generator was developed by placing  $^{131}\text{Ba}$  (11.6 d) on a Bio-Rex 70 ion-exchange column and eluting the cesium with 0.5 M ammonium acetate. The

parent activity is prepared in the ORR by the  $^{130}\text{Ba}(n,\gamma)^{131}\text{Ba}$  reaction (production cross section, 7.6 barns) from either natural barium carbonate or barium nitrate enriched in  $^{130}\text{Ba}$ . There was <0.009% (limit of detection)  $^{131}\text{Ba}$  in the eluted  $^{131}\text{Cs}$ .

### Rhenium-188

Tungsten-188 (69 d) decays to  $^{188}\text{Re}$  (17 h) which, in turn, decays to stable  $^{188}\text{Os}$ . Since rhenium behaves chemically like technetium,  $^{188}\text{Re}$  may be useful for medical applications in a manner similar to  $^{99\text{m}}\text{Tc}$ . A major advantage of  $^{188}\text{W} \rightarrow ^{188}\text{Re}$  over  $^{99}\text{Mo} \rightarrow ^{99\text{m}}\text{Tc}$  is the longer half-life of the parent (69 d compared with 66 h for  $^{99}\text{Mo}$ ). Although the 0.159-Mev gamma of  $^{188}\text{Re}$  is similar to the 0.142-Mev gamma of  $^{99\text{m}}\text{Tc}$ ,  $^{188}\text{Re}$  has the disadvantage of having a 2.0-Mev beta and several high energy gammas of low abundance.

An experimental isotope generator was prepared by sorbing the  $^{188}\text{WO}_4^{2+}$  on 5 ml of HZO-1 (hydrous zirconium oxide). Eighty-five percent of the  $^{188}\text{Re}$  containing  $<1 \times 10^{-2}\%$   $^{188}\text{W}$  was eluted from the column with four volumes (~20 ml) of normal saline.

### Cyclotron-Produced Iodine-123

Iodine-123 (13 h) decays by electron capture with the emission of 0.159-Mev gamma rays. The short half-life and low-energy gamma emission make this nuclide particularly suitable for diagnostic scanning applications in medicine.

The production parameters of the  $^{123}\text{Te}(p,n)^{123}\text{I}$  reaction in the ORNL 86-Inch Cyclotron are given in Table 5. Enriched  $^{123}\text{Te}$  was used as target material in order to reduce or eliminate the quantity of other iodine isotopes which would also be produced by the (p,n) reaction. The aluminum capsule and the tellurium were dissolved in sulfuric acid, and the iodine was distilled without the addition of carrier. The irradiation and processing were completed for each run in 8 hr. More than 95% of the first 79.2% enriched  $^{123}\text{Te}$  target was recovered by precipitation from acid solution using sulfur dioxide and was reused for the second bombardment.

Table 5. Production of Iodine-123 by the  $^{123}\text{Te}(p,n)^{123}\text{I}$  Reaction

Target composition, %			Irradiation conditions	$^{123}\text{I}$ yield, mc	Other radioactive iodines, %		
$^{123}\text{Te}$	$^{124}\text{Te}$	$^{126}\text{Te}$			$^{124}\text{I}$	$^{125}\text{I}$	$^{126}\text{I}$
48.6	14.1	11.5	3 h,* 75 $\mu\text{a}$	7	3.04	--	1.05
79.2	5.1	4.9	2 h,** 184 $\mu\text{a}$	250	0.82	0.08	0.19
79.2	5.1	4.9	2 h,** 187 $\mu\text{a}$	169	0.87	0.07	0.25

\* Bombarded in external beam.

\*\* Bombarded in internal beam.

## RADIOISOTOPE SEPARATIONS AND PURIFICATION DEVELOPMENT

Recovery of Xenon-133 and Iodine-131

The former  $^{131}\text{I}$ - $^{133}\text{Xe}$  process facility was abandoned during the year, and a new process for recovery of these isotopes was placed in operation. In the new process,\*  $^{131}\text{I}$  having a very high radiochemical purity and high specific activity is obtained along with  $^{133}\text{Xe}$ . The new equipment (Fig. 1) can be operated in a manipulator cell in contrast with the fixed equipment system formerly used.

Xenon-133 product purity relative to nonradioactive gas contamination has been improved since the new process approaches a sealed system and results in less atmospheric gas contamination of the separated  $^{133}\text{Xe}$ . A molecular sieve purification system, shown in Fig. 2, has been placed in operation for final purification of the  $^{133}\text{Xe}$  obtained during dissolution of the fission-product target.

Recovery of By-Product Molybdenum-99

After removal of  $^{133}\text{I}$ ,  $^{99}\text{Mo}$  is recovered from the waste effluent by sulfide precipitation using mercury as a carrier. High radiochemically pure  $^{99}\text{Mo}$  is obtained by adsorption on an alumina ion-exchange column. Since this product is carrier free, it is preferred over  $^{99}\text{Mo}$  produced by irradiation of normal molybdenum for  $^{99\text{m}}\text{Tc}$  generators. Separation of the short half-life, rare-earth fission products has been automated by use of a modified sample-bottle changer which reduces the time requirements and places the process on a continuous elution cycle. Additional development work is scheduled to obtain sharper bands and to reduce cross contamination of the isotopes as they are eluted from the ion-exchange column.

Separation of Sulfur-35 and Chlorine-36

Greater product purity and a reduction in time requirements have resulted from changes in the process to separate  $^{35}\text{S}$  and  $^{36}\text{Cl}$  produced by neutron irradiation of  $\text{NaCl}$ . The previous system was based on fractional distillation of the isotopes to obtain separation; the new process uses  $\text{ZrO}_2$  in an ion-exchange column to retain the  $^{35}\text{S}$ . The  $^{36}\text{Cl}$  passes through the column and is recovered from this effluent by distillation after the solution has been acidified with  $\text{H}_2\text{SO}_4$ . The  $^{35}\text{S}$  is then eluted from the  $\text{ZrO}_2$  with dilute  $\text{NH}_4\text{OH}$  and acidified with  $\text{HCl}$  to yield a carrier-free product. The new process also reduces the ion-exchanger capacity requirements since the sodium or potassium component of the irradiation target is not retained on the column as it had been previously.

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\*F. N. Case and E. H. Acree, Large Scale Preparation of High Purity Iodine-131 and Xenon-133 by Sorption Techniques, USAEC Rpt. ORNL-3840, Oak Ridge National Laboratory (January 1966).

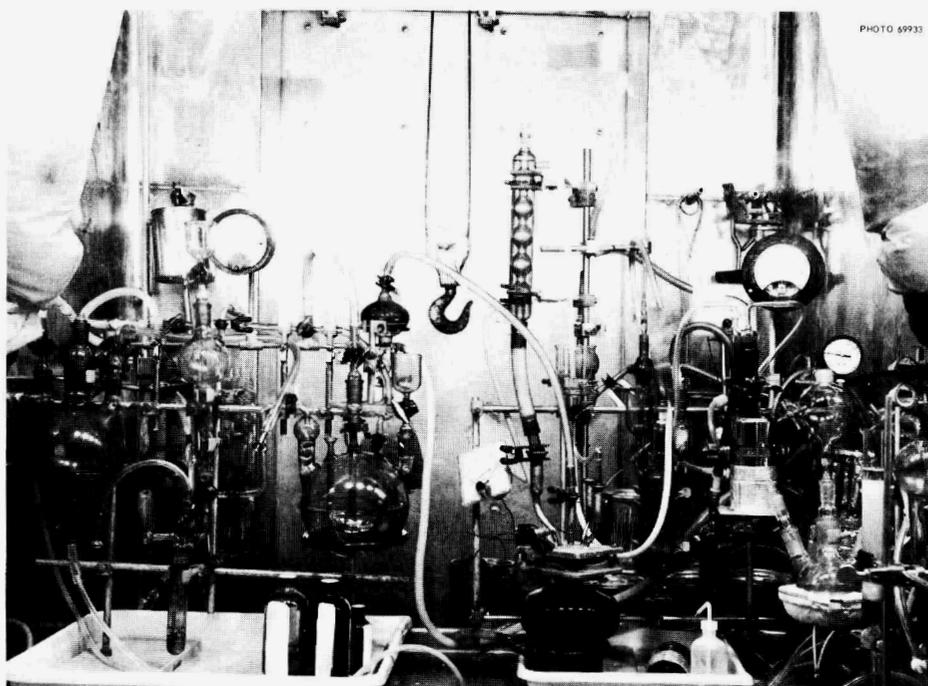


Fig. 1. Equipment for Recovery of  $^{133}\text{Xe}$ - $^{131}\text{I}$  in a Manipulator Cell.

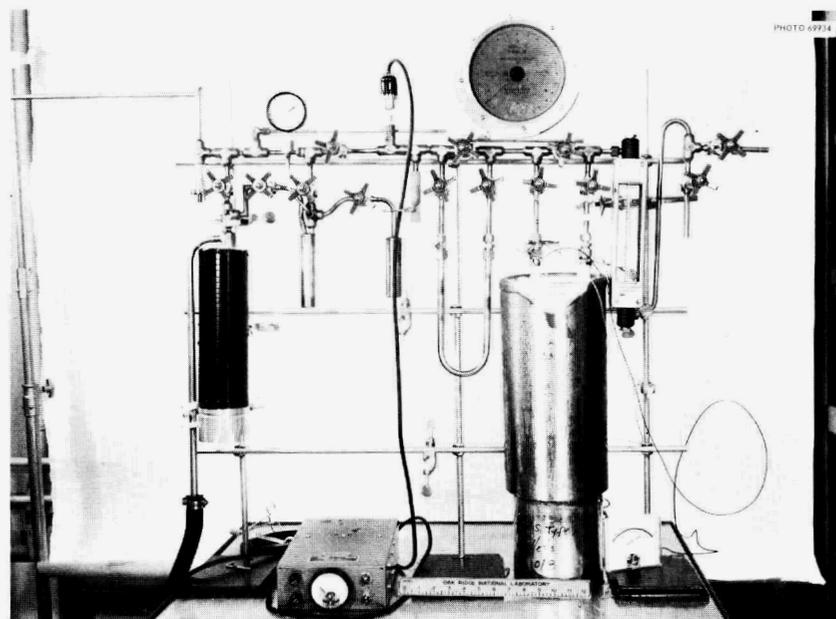


Fig. 2. Xenon-133 Purification System Utilizing Molecular Sieve.

### Enrichment of Krypton-85

Normal fission-product  $^{85}\text{Kr}$  is widely used in self-illuminating signs and for impregnating metals for wear tests. The effectiveness of these techniques would be greatly increased if higher enrichments of  $^{85}\text{Kr}$  could be made available. Increasing the  $^{85}\text{Kr}$  content of available materials would also stimulate development of other applications for this radionuclide (e.g., as a heat source for power generators).

To meet current and anticipated demands, a system was designed to enrich the  $^{85}\text{Kr}$  content of fission-product krypton from the present 4.7% (60 curies/liter) to ~30% (400 curies/liter) and, eventually, to 50% (640 curies/liter). The basic unit of this system is a thermal diffusion column 250 ft long which contains 36 water-cooled tubes. Each tube is 3/4 in. in dia and 7 ft long, and contains a coaxial tubular heater 0.44 in. in dia. The krypton occupies the annular space between the heater and the outside tube. This unit of 36 tubes is enclosed in a single water jacket (~10-in. o.d. by 8-ft length). The total system (effective length, 500 ft) employs two 36-tube units to provide partially enriched material to a third unit which makes the final enrichment to product level. Two systems have been installed in adjoining cells.

In July, three of the units were loaded with nonradioactive krypton for operational testing; the other three were loaded in August. By November, the first three units appeared to be in good operating condition, and testing was discontinued. Since there was a demand for concentrated nonradioactive  $^{78}\text{Kr}$ , several liters of this isotope (enriched from its natural abundance of 0.35% to concentrations varying between 6 and 12%) were recovered from these operational tests. One month after the first three units were reloaded, this time with 2100 curies of fission-product krypton, mass spectrometer analysis of a product sample withdrawn from one unit showed a concentration of 6.5%  $^{85}\text{Kr}$ . Operation of these three units will be continued until equilibrium is reached. In the future, a radiation monitoring system will be installed to indicate the peak enrichment in each column, and this unit will eliminate the need for continuous sampling. A system of storage tanks will also be installed in an adjoining cell to hold intermediate cuts of gas for reprocessing.

## RADIOISOTOPE SOURCE DEVELOPMENT

### Iridium-192 Sources

Targets of iridium sponge were found to be effective in the preparation of high specific-activity  $^{192}\text{Ir}$ , and a carbon arc for fusing the sponge into spheres for use as radiographic sources was found to be efficient and to reduce the time required for source preparation. Iridium-192 spheres containing a concentration of 1000 curies/g have been prepared. The burnup of  $^{192}\text{Ir}$  during target irradiation is controlled by optimizing the irradiation time.

Beta Source for Simulating Space Radiation

A 20- by 40-in.  $^{90}\text{Sr}$  source (Fig. 3) was prepared for NASA to be used in test irradiators of Pegasus micrometeoroid detection panels. The source utilizes 1/8-in.-dia stainless steel tubes with a 10-mil wall thickness, and the design permits the preparation of plane sources having unlimited size (depending only on the size of the shielded chamber where the source is assembled and used).

Strontium-90 was sorbed by molecular sieve pellets  $\sim 1/16$  in. in dia, and each tube was loaded with  $\sim 10$  curies. The tubes were Heliarc welded to provide a leakproof seal, the seal being leak tested in vacuum at 300- $\mu$  pressure. A Faraday cup measurement on the fully loaded source (in a vacuum chamber) showed  $5 \times 10^8$  electrons/cm $^2$  two inches from the tube. A calculated output for this source was derived from data obtained using sources of identical geometry but with microcurie loadings. The calculated output and the Faraday cup measurement agreed within 1%. Radiation from the source was uniform over the surface, and the beta component was  $>93\%$  of the total radiation observed. The energy spectrum for this  $^{90}\text{Sr}$  source is given in Fig. 4.

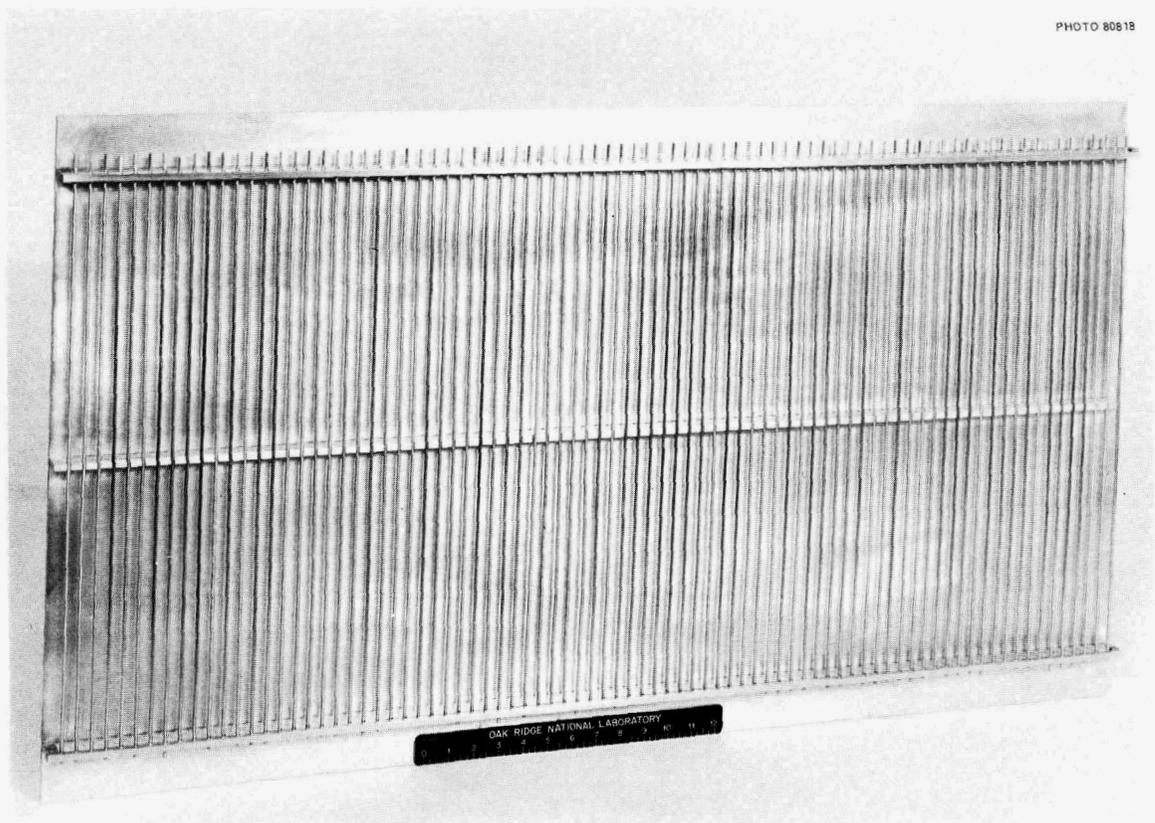


Fig. 3. Strontium-90 Source (20 by 40 in.) for Space Radiation Simulation.

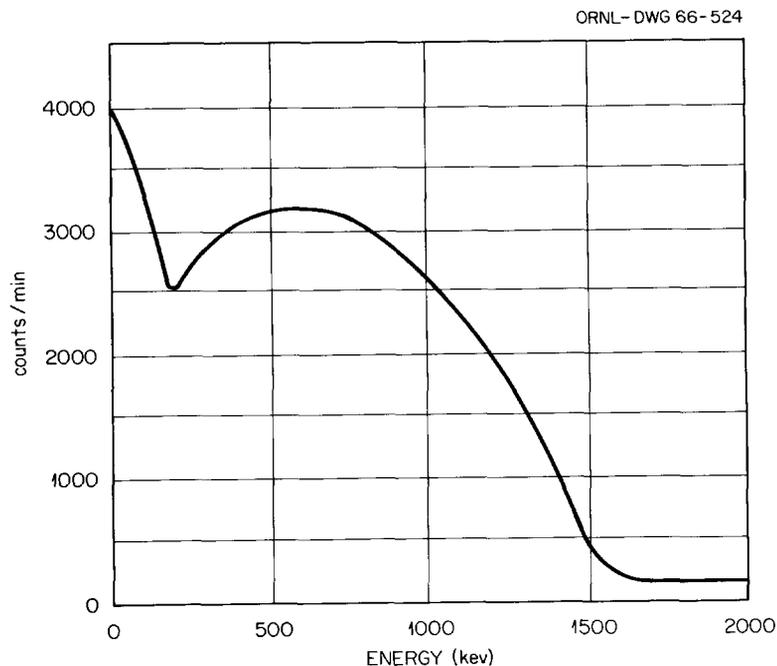


Fig. 4. Energy Spectrum of Strontium-90 Source.

#### Beta Sources for Process Irradiators

Cermets of  $^{90}\text{Sr}$  and aluminum are being investigated in an effort to achieve a high output source with maximum integrity for use in process irradiators. When mixtures of aluminum and strontium titanate or aluminum and strontium silicate are compacted and heated above the melting point of the aluminum, the resulting pellet has less tendency to "dust" than when either strontium compound is used alone. It is necessary to maintain an aluminum-to-strontium compound ratio of  $>0.5$  to yield a pellet having metallic properties, since below this concentration of aluminum the pellets have glass-like properties.

#### Beta-Excited Secondary X-Ray Sources

A  $^{147}\text{Pm}$  oxide mixture containing antimony powder was fabricated into a pellet and encapsulated in aluminum to provide a source for the U. S. Geological Survey. This source produces the antimony  $K_{\alpha}$  X ray and will be used as a prospecting device in the detection of silver.

An investigation of low energy photons for measuring egg shell thickness was carried out in cooperation with the U. S. Department of Agriculture. Outputs from  $^{55}\text{Fe}$ ,  $^{109}\text{Cd}$ , and  $^{99}\text{Tc}$  sources were found to be too low to be useful in making transmission measurements. Backscatter measurements using  $^{90}\text{Y}$  were useful for shell thicknesses equivalent to  $\sim 12 \text{ mg/cm}^2$  but inadequate for thicker shells. A  $10\text{-}\mu\text{c}$  source of  $^{106}\text{Ru}$ - $^{106}\text{Rh}$  is currently being evaluated by the U. S. Department of Agriculture.

Krypton-85 Source for Helicopter Formation-Hold System

A secondary X-ray source utilizing  $^{85}\text{Kr}$  radiation to excite the characteristic 0.098-Mev X ray in 4-mil uranium foil was fabricated and evaluated for output at various distances. The unit was developed for a helicopter formation-hold system which uses a radioactive source positioned near the tail section of the aircraft and a detection system located near the front so each of four helicopters in a formation will be able to locate the other three within a specific range. The conversion efficiency of  $^{85}\text{Kr}$  beta to the 0.098-Mev X ray using depleted uranium foil was 0.5%. The count rate from a 30-curie source could be distinguished from background up to 300 ft (as shown in Fig. 5); however, the background interference was high beyond this distance. The source has been delivered to Industrial Nucleonics for actual system testing.

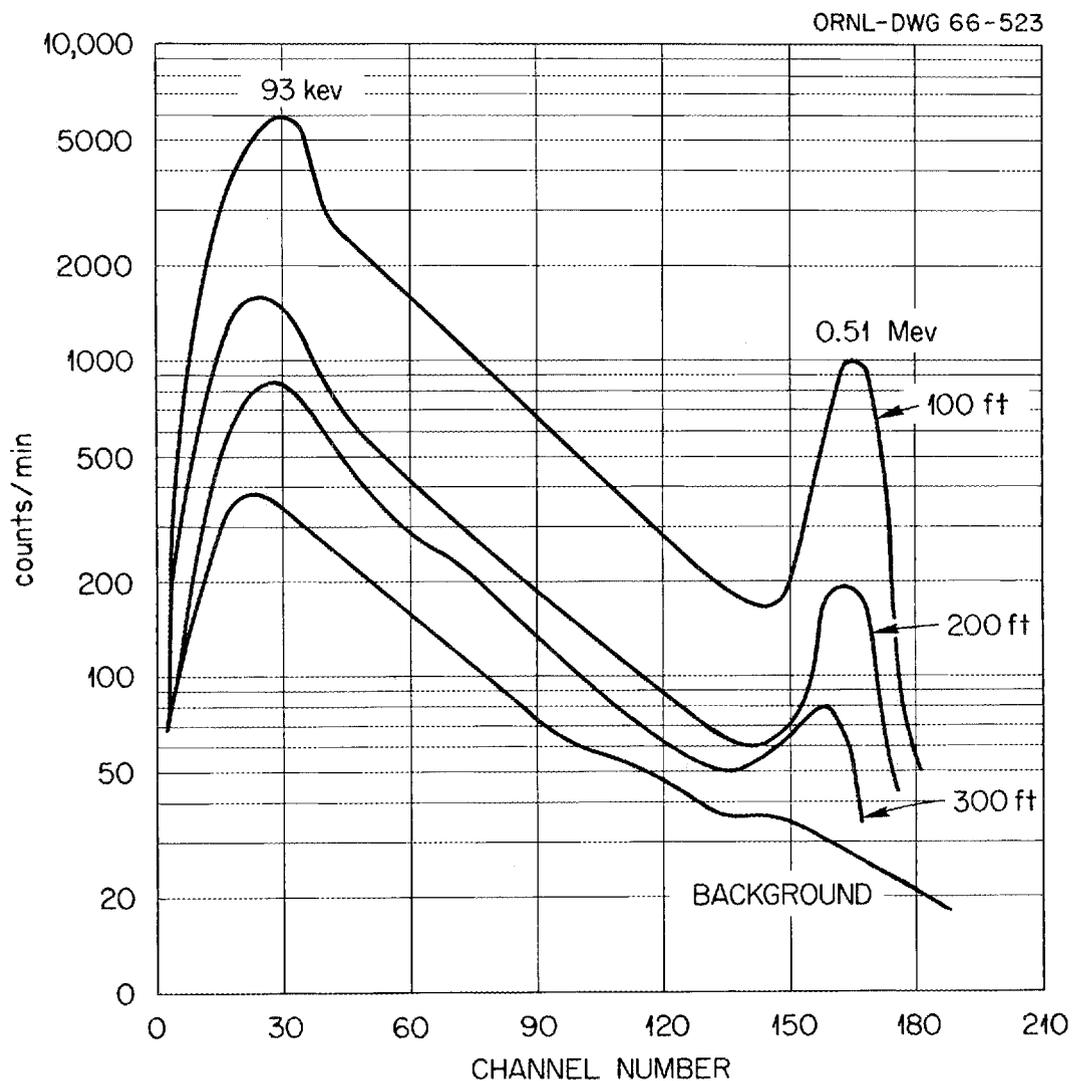


Fig. 5. Detection of Output at Various Distances From a 30-Curie Krypton-85 Source for a Helicopter Formation-Hold System.

### Heat Source for Underseas Experiments

An underwater warm-up station for divers engaged in U. S. Navy Sealab experiments was designed and built using an 83,500-curie  $^{90}\text{SrTiO}_3$  source with a thermal output of 526 watts to warm the water. The source is shielded with a cast steel shell (16 in. dia, 1.5 in. thick) and has a uranium insert (3.1 in. thick) which reduces the radiation dose to <10 mr/hr. Forty fins, 16 x 3 x 1/4 in., are provided for heat transfer to air during transport; temperatures of the container in 77°F ambient air were 110°F in the center of the top and 113°F at midside. The container has a welded closure to prevent water leakage.

The warm-up station consists of a 24-in.-dia concrete pipe (3-1/2 in. thick, 8 ft long) painted with an epoxy sealant. In use, the top of the pipe is open for entry by the diver and the bottom is sealed and insulated. The entire station is filled with water and placed inside a cylindrical steel housing which is open at the bottom for entry and has air trapped in the top for breathing. Figure 6 shows an isometric view of the station.

Calculations indicated that the heat source would raise the temperature of the water inside the concrete pipe by ~40°F; however, tests conducted for 199 hr in a concrete pipe with a 250-watt heater indicated that ~20°F was a more accurate estimate. Navy personnel had indicated that a 10°F rise in temperature would be sufficient with any additional increase being desirable.

### Cerium-144 Source for Radiant Energy Conversion Device

A source containing 30,500 curies of  $^{144}\text{CeO}_2$  was prepared for the Allison Division of General Motors. The crushed sintered  $^{144}\text{CeO}_2$  powder was loaded into an Inconel capsule which was welded, leak tested, and cleaned to smear tolerance. The completed source was placed in a shielded test device containing thermophotovoltaic cells to determine the radiation degradation properties of the Allison cells. For the duration of the six-month experiment, the surface temperature of the  $^{144}\text{Ce}$  source receptacle was maintained at ~56°F. Data on photocell performance were accumulated and transmitted to Allison personnel at weekly intervals for evaluation. Due to the encouraging results of this experiment, a more elaborate test involving 90,000 curies of  $^{144}\text{Ce}$  is being planned.

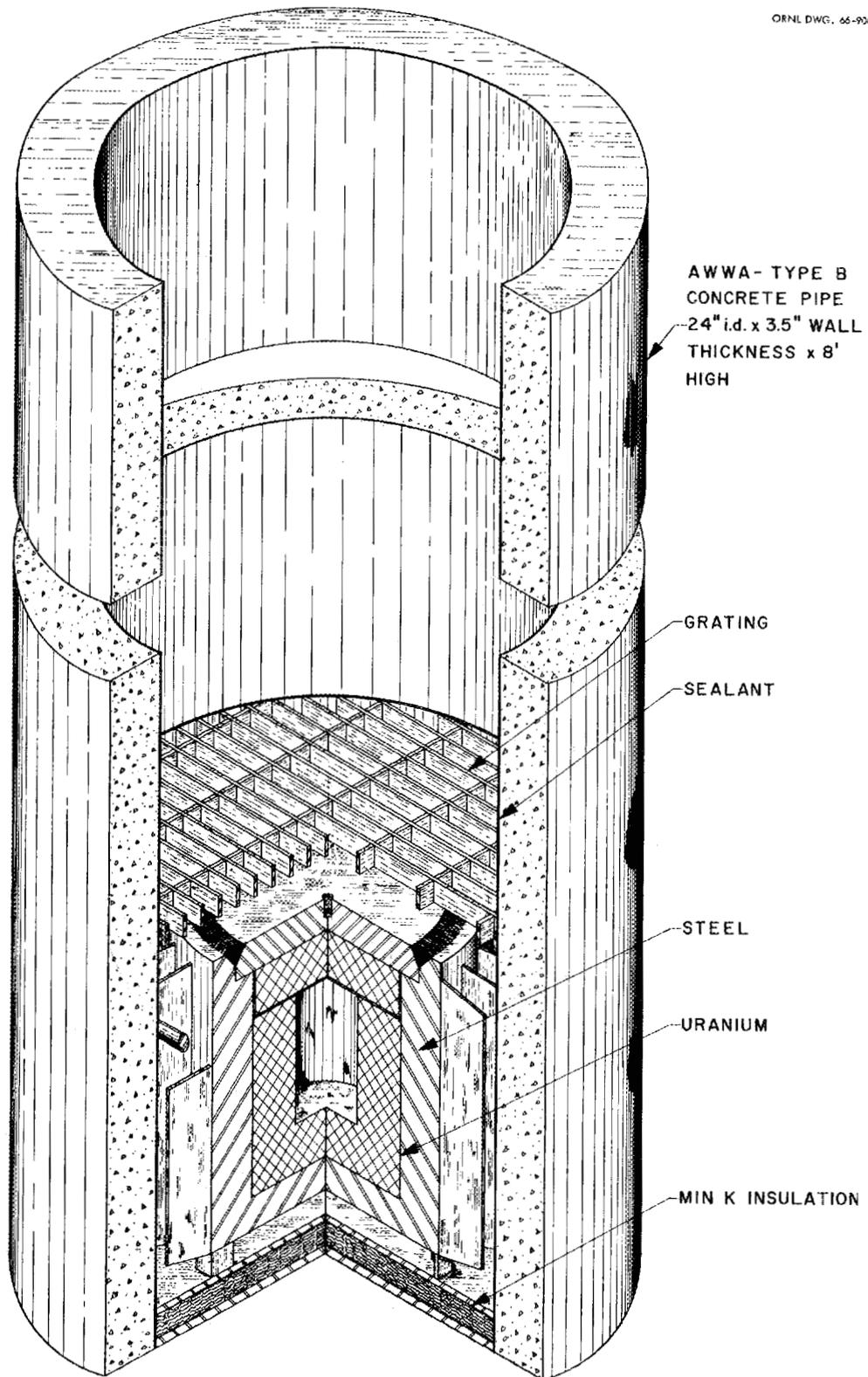


Fig. 6. Sealab-II Warm-Up Station.

## RADIOISOTOPE SOURCE SAFETY TESTING

The steadily increasing use of radioisotope sources in research and industry has made it desirable to define those safety characteristics of source capsules which are important in insuring that radioactive material does not leak to the surrounding environment during normal use or in the event of accidents. Several hundred source capsules of various types have been subjected to a variety of mechanical and temperature tests to determine their containment integrity. A simple method of classifying sealed sources according to their structural characteristics is being developed.

Source Capsule Classification System

The Source Capsule Classification System has been reviewed and a few minor changes were made. The Class D maximum temperature resistance test was reduced from 2550°F for 1 hr to 2400°F for 1 hr because there were a few stainless steel source capsules which met all of the Class D requirements except this one. This minor decrease in temperature allows these capsules to fall into this class.

The Class E maximum temperature resistance test was changed from 4500°F for 1 hr to 4500°F for 30 min because there are no probable instances when a sealed source might be exposed to this temperature for more than 30 min. This test is performed in a vacuum or inert gas atmosphere and is designed to check the integrity of protective coatings. An additional test was added to the Class E maximum temperature resistance class in order to check the source for oxidation resistance after the 4500°F test. This test is conducted in air at 2400°F for 1 hr.

The internal pressure test was changed to an external pressure test in order to simplify the test procedure. The abrasion test requirement was dropped from the classification system because abrasive tests conducted on aluminum, brass, and stainless steel capsules indicated that the amounts of material removed by abrasion would be so small that source capsule failure from this cause would be unlikely.

The Source Capsule Classification Test Procedures were completely revised to describe in greater detail the method of performing each test and the sequence in which steps are executed and to specify the general type of test equipment to be used. Packages containing the Source Capsule Classification System, the Source Capsule Classification Test Procedures, and the Rationale Behind the Selection of Individual Values in the ORNL Source Capsule Classification System were mailed to 35 interested industrial firms and federal agencies for comments.

The second program of trial classification of inactive, commercial, sealed sources according to the requirements of the Source Capsule Classification System was completed. The sources, which were purchased from commercial suppliers, were selected from identical designs used in the first trial classification. Comparison of results showed that these sources received the same classification in the second trial test as they had received in the first trial classification test; therefore, the ORNL system is capable of giving reproducible classifications over a one-year time interval.

### Source Applications Guide

A Source Applications Guide has been under study. It is designed to establish the classification requirements of a source capsule by evaluating each of the logical qualities in a particular source use and in a use environment which relate to safety. Numerical ratings are assigned which, in turn, can be related to the classification test requirements. Some of the qualities under consideration include quantity of isotope, relative biological hazard of isotope, form of activity, location of use, training of personnel, operating environment (corrosiveness, pressure, and temperature), fire and explosion potential at place of use, source capsule housing or device, and field experience.

### Source Testing Film

An information film on radioisotope source testing is in preparation. The film will illustrate the use of the ORNL Source Capsule Classification System as a means of assuring safe usage of these sources. Scenes will be shown illustrating the methods used in testing sources for classification under the ORNL system.

### Source Capsule Leak Definition

A water-leach test on a doubly encapsulated teletherapy source containing an inactive cesium chloride pellet has been in progress for ten months. Both the inner and outer capsules have 0.002-in.-dia holes drilled through the end windows. Periodic sampling of the leach water has shown no cesium leakage. This test will be continued.

### Tritium Release From Tritiated Phosphor Paints

The amount and the chemical form of tritium released from five tritiated phosphor paints (supplied by three manufacturers) during the 4.5-hr drying period immediately following application were determined. Tritium losses in the tests varied from 0.15 to 2.56% among the five paints tested. The results of the tests show that tritium was released throughout the drying period. Although the amount of tritium released decreased slightly with time, a significant amount of the total tritium release occurred during the last hour of the tests. The results indicated that 37-100% of the total tritium released was in the form of tritiated water.

These same five paints were also tested to determine the amount and the chemical form of tritium released at elevated temperatures. Known aliquots of tritiated phosphor, mixed and cured according to the manufacturers' instructions, were heated for 30-min intervals at temperature increments of 50° from 0-200°C while air was flowed past them. Tritium losses in the tests varied from 5 to 25% among the five paints tested; the amounts released increased steadily with increasing temperature. The results showed that 75 to 95% of the total tritium released was in the form of tritiated water.

A study of water-leach rates over a ten-year period using the same five tritiated phosphor paints was initiated. The paint samples were prepared according to the manufacturers' instructions and were applied uniformly over a 0.5-in.-dia area on an aluminum backing. Additional samples were prepared to study the effects of paint thickness and exposed surface area on the water-leach rates. The test procedure consists of immersion of the paint sample in distilled water for 24 hr at room temperature without agitation. The tritium content of the leach water is determined by liquid scintillation counting.

The first set of leach tests was completed, and the tritium losses from the five paint samples tested ranged from 0.096 to 0.169% for paint with activities from 200 to 1050 mc/g. The sample weight and the exposed surface area were held constant so that differences in paints and variations resulting from various activities could be observed. In other tests, samples of the same paints with varied exposed surface areas and varied paint thicknesses were observed.

#### Testing Special Sources

Several special sources were tested during the year to determine their temperature and structural characteristics. Some were new designs such as stainless steel tubular sources or stainless steel capsules with beryllium windows. Several sources were tested for the Division of Isotopes Development including the Brookhaven National Laboratory  $^{60}\text{Co}$  irradiator source and the  $^{90}\text{Sr}$  ice detector probes.

#### Nickel-Plated Uranium Tests

A group of nickel-plated uranium samples was tested to determine if this type of material is a suitable substitute for stainless steel clad uranium for use in shipping casks. The samples failed temperature tests in air in the range of 350 to 465°C and tests in vacuum in the range of 514 to 720°C. Failures were observed in the form of blisters or wrinkles in the plating. Results of these tests are shown in Table 6.

Table 6. Nickel-Plated Uranium Sample Tests

Type of nickel	Atmosphere	Failure temperature, °C
Watts*	Air	465
Watts	Air	350
Sulfonate*	Vacuum	720
Watts, barrel plated*	Vacuum	514
Watts	Vacuum	681

\*Subjected to salt fog test.

## ISOTOPES INFORMATION AND QUARTERLY REVIEW

The responsibilities of the Isotopes Information Center include (1) producing the quarterly Technical Progress Review, Isotopes and Radiation Technology; (2) establishing a library of accessioned material, relevant to the Division of Isotopes Development interests (isotope applications and production), suitable for retrieving documents and information for publishing the Technical Progress Review and for providing critical state-of-the-art reviews covering various subjects within the AEC-DID areas of interest; (3) writing brochures dealing with production and uses of isotopes; and (4) carrying out miscellaneous assignments appropriate to the functions of the Isotopes Division, the Isotopes Development Center, and the AEC Division of Isotopes Development.

### Isotopes and Radiation Technology

During 1965, Vol. 2 of Isotopes and Radiation Technology was completed, one issue of Vol. 3 was published, the finished draft of the second issue was submitted to DTIE, and the rough draft of the third issue was partially completed. Two feature articles were published, one by Artandi and Van Winkle of Ethicon, Inc., dealing with "Comparison of Electron-Beam and Gamma-Irradiation Plants" and the other by Marshall Brucer, describing "118 Medical Radioisotope Cows." Contributions by authors other than Information Center staff increased somewhat over 1964, in keeping with the plan to have most of the articles written by authorities in the field.

### Information Center Library

During the year the number of articles accessioned increased to ~4000, with another 1500 articles scanned and indexed but not put into the system. Approximately 2500 accessions have been "addressed" on the Termatrix. As part of our effort to reduce the document storage problem, ~2000 microfiche copies of our accessions have replaced full-sized copies; it is planned to eventually have a complete microfiche file of all accessions except books and bibliographies. The Thesaurus for the Center was formalized. A cooperative scanning agreement with AEC-DTIE has resulted in routine cover-to-cover scanning of several hundred journals with a minimum of effort. Contacts have been made with Euratom, John Crerar Library, Dix Associates, the Institute for Scientific Information, Armed Materials Research Agency, and Scientific Information Exchange in connection with additional scanning programs. Participation in the Nuclear Science Abstracts computer-retrieval study has shown considerable promise and search questions are now being developed. The Center initiated a program aimed at establishing interest profiles of Isotopes Development Center personnel to permit selective dissemination of information (SDI) on a routine basis.

### Miscellaneous Activities

About 20 Division talks were edited and ~15 Division reports were edited and published or submitted for publication. Considerable time was spent on the FY 1965 AEC Radioisotope Customer List. Cooperative effort was expended on an "Isotopes" brochure and on a monograph, "Biophysical Separations Laboratory." More than 30 translations of foreign technical in-scope

articles were prepared by staff members of the Information Center. Packets of isotopes applications reprints were assembled for mailing to ~500 industrial organizations as a part of the Information Center's attempts to publicize the advantages of radioisotopes. A display panel describing the Isotopes Information Center was designed and built. A course in technical report writing was conducted by members of the Center. Staff members participated in the following conferences and meetings during the year: three speakers at Southern Interstate Nuclear Board briefings; an observer at the 7th Annual Contractors' Meeting of the DID in Washington; lecturer at the ORINS Health Physics Course, "Industrial Uses of Radiation"; member of Program Committee for the Battelle Ad Hoc Forum on Information Centers; chairman of technical session at Gatlinburg ANS meeting; participant at Gordon Research conference on "Scientific Information Problems in Research"; speaker at AEC meeting on "Specialized Information and Data Centers"; chairman and arranger for Isotopes Development Center Program Review Meeting; speaker for TV talk on radioisotope uses; and participant in Traveling Lecture program as speaker at Knoxville College, Bethany Nazarene College (Oklahoma), and University of Mississippi. An Activation Analysis Exhibit was displayed and manned at International Activation Analysis Conference at Texas A&M University. Forty-one submissions to the Intern. J. Appl. Rad. Isotopes were reviewed. Approximately 200 letters were written in response to technical inquiries and numerous other responses were made by telephone and in person.

## ISOTOPES PRODUCTION AND SALES

### Fission-Products Development Laboratory Operations

During 1965, 1,200,000 curies of  $^{90}\text{SrCO}_3$  was received and stored in the FPDL solution storage system. Production of  $^{90}\text{SrTiO}_3$  was curtailed during the year but 100,000 curies of  $^{90}\text{SrO}$  was produced for two heat sources for NUMEC.

One shipment (450,000 curies) of  $^{144}\text{Ce}$  was received as a mixed rare-earth sulfate. This material was eluted from the cask and stored for future heat source requirements. Ten 2000-curie  $^{144}\text{CeO}_2$  powder sources were prepared for compatibility studies with Inconel, 3162C stainless steel, Haynes 25, Hastelloy C, and Inor 8.

A total of 510,000 curies of  $^{137}\text{Cs}$  was received in twelve shielded transfer tank (STT) shipments. Two additional STT's were placed in service to increase the flow of  $^{137}\text{Cs}$  to the FPDL. Twenty doubly encapsulated  $^{137}\text{CsCl}$  sources containing 173,500 curies were fabricated for St. Gobain Nuclaire for a gamma process irradiator. A total of 183,000 curies of  $^{137}\text{CsCl}$  powder was singly encapsulated in stainless steel containers (5000 curies each) in preparation for storage in the  $^{137}\text{Cs}$  Gamma Irradiator. All future powder prepared at FPDL will be stored in this manner. Miscellaneous sources (116) and powder shipments accounted for 113,000 curies of  $^{137}\text{CsCl}$  delivered to customers.

Cobalt-60 bulk shipments amounted to 676,000 curies during the year. The increased demand for this service will require new facilities in the coming year. Construction of  $^{60}\text{Co}$  cells in the FPDL will be started early in 1966 to handle an estimated 2.0 megacuries of bulk  $^{60}\text{Co}$  per year.

Sales Report - Calendar Years 1964 and 1965

Account No.	Isotope	Calendar Year 1964					Calendar Year 1965				
		Sales proceeds	Fund cost of sales	No. of shipments	Millicuries shipped	Fund cost of production	Sales proceeds	Fund cost of sales	No. of shipments	Millicuries shipped	Fund cost of production
<u>Inventory Items</u>											
3651	Carbon-14	\$213,913	\$124,228	155	53,981	\$ 106,800	\$ 343,456	\$ 176,577	170	83,448	\$ 143,032
3652	Cesium-137	326,846 <sup>a</sup>	361,739	195	298,921,101	409,427	190,531	481,553	161	336,429,637	594,898
3653	Cobalt-60	110,113	148,396	18	207,442,017	53,738	303,740	149,803	31	606,790,140	237,259
3654-104	Promethium-147	27,420	59,533	138	4,602,306	36,342	29,895	49,007	230	6,590,724	12,198
3654-106	Technetium-99	51,927	46,587	45	1,225,223 mg	48,501	5,697	3,363	36	104,284 mg	420
3654-107	Krypton-85	80,136	78,937	168	5,455,505	13,041	113,170	108,688	193	6,761,849	20,149
3661-288	Chlorine-36	23,954	18,611	127	77	11,402	17,275	9,644	102	53	21,549
3661-327	Nickel-63	10,708	4,821	69	1,599	7,667	17,292	10,346	38	2,818	15,028
3661-365	Thallium-204	2,940	1,251	57	2,852	3,574	1,642	1,497	40	1,829	1,826
3662	Strontium-90	17,725	30,849	119	8,255,493	1,312,314	20,740	107,876	103	38,603,632	1,181,845
3663	Tritium	111,165	91,197	182	99,954,689	90,541	240,846 <sup>b</sup>	167,975	205	284,594,792	175,532
<u>Total Inventory Items</u>		<u>\$976,847</u>	<u>\$966,149</u>	<u>1,273</u>	<u>1,225,223 mg</u> <u>624,589,620</u>	<u>\$2,095,247</u>	<u>\$1,284,284</u>	<u>\$1,266,329</u>	<u>1,309</u>	<u>1,278,658,922</u>	<u>\$2,403,736</u>
<u>Major Product Items</u>											
3654-117	Xenon-133	\$ 14,854	\$ 24,899	166	711,522		\$ 16,850	\$ 19,811	167	339,949	
3655	Iodine-131	33,606	40,796	1,254	146,907		18,849	35,784	856	59,317	
3660	Phosphorus-32	51,337	52,148	2,001	53,526		61,634	40,382	1,598	46,879	
3661-266	Barium-133	13,720	c	68	136		20,562	4,723 <sup>c</sup>	51	202	
3661-281	Calcium-45 (P-3)	18,989	11,462	152	424		4,869	8,948	35	108	
3661-284	Calcium-47	44,641	36,078	220	220		52,630	54,578	210	226	
3661-323	Mercury-203	28,521	d	165	27,371		5,977	5,876	98	5,236	
3661-333	Potassium-42	9,680	8,652	567	4,224		10,834	11,079	487	3,898	
3661-360-2	Sulfur-35	17,645	11,195	462	14,008		26,318	31,196	359	14,425	
3661-386	Phosphorus-33			2				447	1	i	
<u>Total Major Product Items</u>		<u>\$232,993</u>	<u>\$185,230</u>	<u>5,057</u>	<u>958,338</u>		<u>\$ 218,523</u>	<u>\$ 212,824</u>	<u>3,862</u>	<u>470,241</u>	
<u>Development Fission Products</u>											
3654-105	Europium-155	\$ 25		1	1		\$ 325		3	16	
3654-106	Iodine-129	21,877		41	27,133 mg		4,176		18	4,741 mg	
<u>Total Development Fission Products</u>		<u>\$ 21,902</u>	<u>\$ 13,331</u>	<u>42</u>	<u>27,133 mg</u> <u>1</u>		<u>\$ 4,501</u>	<u>\$ 7,160</u>	<u>21</u>	<u>4,741 mg</u> <u>16</u>	

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Account No.	Isotope	Calendar Year 1964				Calendar Year 1965			
		Sales proceeds	Fund cost of sales	No. of shipments	Millicuries shipped	Sales proceeds	Fund cost of sales	No. of shipments	Millicuries shipped
<u>Minor Fission Products</u>									
3654-100	Barium-140	\$ 4,387		101	1,674	\$ 5,093		83	941
3654-101	Strontium-89	9,006		144	3,312	16,625		111	2,402
3654-103	Curium-144 - Processed	3,000		66	73,253	2,760		73	23,531
3654-109	Gross Fission Products	702		24	256	695		23	157
3654-110	Zirconium-95-Niobium-95	6,673		133	3,863	6,908		95	2,420
3654-111	Niobium-95	2,054		40	306	2,075		36	119
3654-112	Praseodymium-143	900		7	18	100		2	2
3654-113	Neodymium-147	2,154		21	134	1,500		14	31
3654-114	Ruthenium-103	1,584		31	141	3,349		35	176
3654-115	Ruthenium-106	4,724		58	580	6,880		41	418
3654-116	Yttrium-91	2,111		36	726	2,491		38	538
<u>Total Minor Fission Products</u>		\$ 37,295	\$ 60,943	661	84,273	\$ 48,476	\$ 52,306	551	30,735
<u>Miscellaneous Development Items - Processed Materials</u>									
3661-250	Antimony-122	\$ 80		3	14	\$ 61		2	9
3661-256	Argon-37	270		9	14	636		15	28
3661-260	Arsenic-76	440		26	209	685		38	250
3661-262	Arsenic-77	100		4	8	30		1	2
3661-264	Barium-131	420		7	7	810		10	10
3661-270	Bismuth-210	275		6	25	100		2	11
3661-273	Cadmium-109	4,598		35	64	1,777		22	18
3661-274	Cadmium-115	562		13	164	363		9	73
3661-300	Gallium-72	235		8	56	611		13	101
3661-304	Gold-199	345		9	64	1,425		26	237
3661-305	Iodine-130	1,212		41	960	253		12	137
3661-317	Lanthanum-140	2,368		44	693	651		31	221
3661-321	Mercury-197	5,535		60	12,491	21,057		59	19,145
3661-325	Molybdenum-99	602		52	602	5,262		62	21,040
3661-331	Palladium-109	225		15	174	395		26	266
3661-335	Praseodymium-142	115		3	35	156		1	52
3661-337	Rhenium-186	459		18	144	411		16	139
3661-343	Samarium-153	425		13	157	323		9	103
3661-350	Silver-111	225		8	24	700		18	93
3661-367	Thulium-170	1,500		14	4,905	2,291		18	11,398
3661-370	Tungsten-187	50		2	2	789		19	91
3661-382	Argon-38								
<u>Total Miscellaneous Development Items - Processed Materials</u>		\$ 20,341	\$ 73,050	390	20,812	\$ 38,786	\$ 46,160	409	53,434

Account No.	Isotope	Calendar Year 1964				Calendar Year 1965			
		Sales proceeds	Fund cost of sales	No. of shipments	Millicuries shipped	Sales proceeds	Fund cost of sales	No. of shipments	Millicuries shipped
<u>Minor Processed Material Items</u>									
3661-252	Antimony-124	\$ 1,411		45	260	\$ 927		32	157
3661-272	Bromine-82	2,851		80	713	2,199		64	654
3661-276	Cadmium-115	2,619		26	103	1,419		21	42
3661-295	Cobalt-60	3,692		108	1,425	3,720		73	2,145
3661-296	Copper-64	4,392		147	3,383	10,358		106	5,466
3661-298	Europium-152-154	970		21	87	755		18	68
3661-302	Gold-198	1,627		108	20,537	1,855		111	4,528
3661-306	Hafnium-181	611		24	221	915		30	435
3661-308	Indium-114	1,125		34	187	1,275		37	225
3661-311	Iron-55-59	385		8	11	665		6	19
3661-329	Osmium-191	80		3	8	291		10	24
3661-339	Ruthenium-86	6,511		269	9,349	5,479		260	9,896
3661-345	Scandium-46	1,183		49	3,394	1,088		47	1,051
3661-347	Selenium-75	844		29	389	219		8	104
3661-348	Silver-110	1,349		55	415	1,636		56	892
3661-356	Sodium-24	8,044		427	1,177	8,130		332	978
3661-363	Tantalum-182	850		27	138	880		19	191
3661-368	Tungsten-185	710		20	149	628		17	138
3661-372	Yttrium-90	973		12	284	3,780		27	1,260
<u>Total Minor Processed Materials</u>		<u>\$ 40,227</u>	<u>\$ 71,557</u>	<u>1,492</u>	<u>42,250</u>	<u>\$ 46,199</u>	<u>\$ 63,879</u>	<u>1,274</u>	<u>28,273</u>
<u>Special AEC-Approved Production Services<sup>e</sup></u>									
3661-102	Cerium-141	\$ 707		29	168	\$ 149		6	44
3661-160-1	Iridium-192	10,157		53	1,369,297	4,268		23	461,582
3661-200	Activation Analyses	1,378		2	2 Anal.	1,225		1	1 Anal.
3661-254	Antimony-125	3,000		23	29	1,505		6	16
3661-278-9	Calcium-45 (P-1, P-2)	22,065		301	2,591	5,487		73	549
3661-285	Cesium-134	743		27	312	212		10	88
3661-293	Cobalt-58	1,330		14	69	40		1	2
3661-303	Iodine-125	15,075		8	15,015				
3661-314	Iron-55	9,274		39	315	625		4	12
3661-315	Iron-59	49,209		467	1,640	21,934		184	822
3661-358	Strontium-85	40,207		178	804	4,675		25	93
3661-366	Tin-113	2,288		46	64	904		16	26
3661-374	Zinc-65	8,922		151	4,410	1,714		48	597
3661-379	Chromium-51	3,996		85	2,115	282		10	143
3661-381	Selenium-75	34,887		98	1,398	16,338		48	654
<u>Total Special AEC-Approved Production Services</u>		<u>\$ 203,258</u>	<u>\$ 83,637</u>	<u>1,521</u>	<u>1,398,027</u>	<u>\$ 59,358</u>	<u>\$ 44,291</u>	<u>453</u>	<u>464,628</u>

Account No.	Isotope	Calendar Year 1964				Calendar Year 1965			
		Sales proceeds	Fund cost of sales	No. of shipments	Millicuries shipped	Sales proceeds	Fund cost of sales	No. of shipments	Millicuries shipped
<b>Total Radioisotopes</b>		\$1,532,863	\$1,453,897	10,436	627,093,321	\$1,700,127	\$1,699,949	7,879	1,280,706,249
<b>Radioisotope Services</b>									
3658-190	86-Inch Cyclotron	\$ 173,381	\$ 180,937	168	168 S.I.	\$ 263,235	\$ 187,414	163	188 S.I.
3658-193	Reactor Service Irradiations	51,782	40,857	428	414 S.I.	61,371	44,148	404	400 S.I.
3658-194	Cobalt-60 Gamma	4,724	3,497	7	6 S.I.	1,434	836	3	3 S.I.
3658-196	Naval Research Laboratory <sup>a</sup>	79,011	87,187			76,522	70,025		
3659-203	Special Services	87,391	69,937			93,986	75,358	3	
3659-204	Target Preparation - Radioisotopes	17,400	34,318			495	(19,826) <sup>g</sup>		
3659-205	Cobalt-60 Source Selection	18,949	10,451			15,166	13,481		
3659-206	Cesium-137 Source Fabrication	49,115	38,991			57,275	29,373		
3659-209	Incoming Transportation Charges - Customer Service	1,237	1,719			797	766		
3663-521	Tritium Target Fabrication	19,520	14,714	120		4,741	4,284	27	
<b>Total Radioisotope Services</b>		\$ 502,510	\$ 482,608	723	588 S.I.	\$ 575,024	\$ 405,859	605	591 S.I.
<b>Packing and Shipping</b>		\$ 246,054	\$ 221,451			\$ 216,663	\$ 157,240		
<b>Isotopes Sales Department</b>			\$ 199,098				\$ 175,337		
<b>TOTAL</b>		\$2,281,427	\$2,357,054	11,159	627,093,321	\$2,491,814	\$2,438,385	8,484	1,280,706,249

Anal. - Analyses  
S.I. - Service Insertions

<sup>a</sup>Includes fund cost transfer of \$225,843 to Brookhaven National Laboratory for 214,994 curies of <sup>137</sup>Cs.

<sup>b</sup>Includes \$49,569 for 93,904 curies of tritium transferred to Project KENRE.

<sup>c</sup>Reclassified from Miscellaneous Developmental Items - Processed Materials, effective July 1, 1965. The costs of sales for calendar year 1964 and the first six months of calendar year 1965 are included in the total Miscellaneous Developmental Items.

<sup>d</sup>Reclassified from Minor Processed Material Items. The cost of sales for calendar year 1964 was included in the total for Minor Processed Material Items.

<sup>e</sup>Represents materials and services from which ORNL has been withdrawn from routine production and distribution. These materials and services are furnished only when materials of customers' specifications are not available commercially.

<sup>f</sup>Fund costs only are billed.

<sup>g</sup>Includes a credit adjustment of \$22,715 to costs incurred during the last half of FY 1965.

## PUBLICATIONS

Articles

- E. H. Acree and R. S. Pressly, Preparation of Xenon-131m, Xenon-129m, and Krypton-79, Trans. Am. Nucl. Soc. 8: 76 (1965).
- J. F. Allen and J. J. Pinajian, A Strontium-87m Generator for Medical Applications, Intern. J. Appl. Radiation Isotopes 16: 319-25 (1965).
- J. F. Allen, An Improved Technetium-99m Generator for Medical Applications, Intern. J. Appl. Radiation Isotopes 16: 332-34 (1965).
- P. S. Baker and A. F. Rupp, Eds., Isotopes and Radiation Technology 2(3), 2(4), and 3(1), (1965).
- P. S. Baker, D. A. Fuccillo, M. W. Gerrard, and R. H. Lafferty, Jr., Radioisotopes in Industry, one of AEC "Understanding the Atom" series (November 1965).
- W. D. Box, Electrodeposition of Technetium-99 Metal, Nucl. Appl. 1: 155-57 (1965).
- W. D. Box, Vacuum Distillation of Magnesium Metal, Nucl. Appl. 1: 158-59 (1965).
- J. L. Burnett, Melting Points of Curium and Americium Trifluorides, Trans. Am. Nucl. Soc. 8: 335 (1965).
- F. N. Case and E. H. Acree, Purification of Curie Quantities of Fission-Produced Iodine-131 by Sorption on Platinum, Trans. Am. Nucl. Soc. 8: 171 (1965).
- C. W. Friend and J. R. Knight, Computer Code Abstracts, Nucl. Sci. Eng. 23: 206-07 (1965).
- C. W. Friend, A. R. Jenkins, R. E. Lewis, and J. R. Knight, Computer Codes for Isotopes Calculations, Isotopes and Radiation Technology 3(1): 18-21 (1965).
- K. W. Haff, Radioisotope Shipping Container Development, Chem. Eng. Progr. Symp. Ser. 61: 74-77 (1965).
- K. W. Haff and L. B. Shappert, International Symposium on Packaging and Transportation of Radioactive Materials, Nucl. Safety 6: 371-74 (1965).
- E. Lamb, Transportation of Irradiated Fuel Elements, Nucl. Safety 6: 249-54 (1965).
- R. E. Lewis, R. E. McHenry, and T. A. Butler, Half-Life of Cesium-137, Trans. Am. Nucl. Soc. 8: 79-80 (1965).

R. E. Lewis and T. A. Butler, Reactor Production and Chemical Separation of Carrier-Free Phosphorus-33 From Chlorine-36 and Enriched Sulfur-33 in the Oak Ridge Research Reactor, Trans. Am. Nucl. Soc. 8: 173-74 (1965).

R. E. McHenry, Melting Points of Curium and Americium Oxides, Trans. Am. Nucl. Soc. 8: 75 (1965).

R. E. McHenry, Leach Rate of Strontium-90 Titanate Radioisotopic Power Fuel, Trans. Am. Nucl. Soc. 8: 172-73 (1965).

H. A. O'Brien, Jr., Reactor Production of Carrier-Free Manganese-54 from Natural Iron, Intern. J. Appl. Radiation Isotopes 16: 747-49 (1965).

P. B. Orr, Separation of Curie Quantities of Yttrium-90 from Strontium-90 by an Elution Process, Trans. Am. Nucl. Soc. 8: 334-35 (1965).

#### Reports

P. S. Baker, Comp., Research Materials Coordination and Planning Meeting, Nov. 14-15, 1963, USAEC Rpt. ORNL-TM-1047, Oak Ridge National Laboratory (April 1965).

P. S. Baker, Old Armor Plate and Lead, in Research Materials Coordination and Planning Meeting, Nov. 14-15, 1963, USAEC Rpt. ORNL-TM-1047, pp. 106-108, Oak Ridge National Laboratory (April 1965).

P. S. Baker and M. W. Gerrard, Proceedings of Symposium on Low Energy X- and Gamma Sources and Applications, Chicago, Oct. 20-21, 1964, USAEC Rpt. ORNL-IIC-5, Oak Ridge National Laboratory (November 1965).

K. R. Balasubramanian and J. J. Pinajian, The Extraction of Indium From Hydrochloric Acid Solutions with n-Amyl and n-Octyl Alcohol and Its Possible Use for Separating Tin and Indium, USAEC Rpt. ORNL-TM-1059, Oak Ridge National Laboratory (April 1965).

T. A. Butler and R. E. McHenry, Comps., Curium-244 Status Report, USAEC Rpt. ORNL-3842 (classified), Oak Ridge National Laboratory (August 1965).

T. A. Butler, W. R. Rathkamp, and H. B. Greene, Review of ORNL Thermal Diffusion Program, January-December 1964, USAEC Rpt. ORNL-3815, Oak Ridge National Laboratory (June 1965).

L. M. Carter, Solid Radioactive Waste Disposal from Fission-Products Development Laboratory, USAEC Rpt. ORNL-TM-1178, Oak Ridge National Laboratory (August 1965).

F. N. Case, Research Pools, in Research Materials Coordination and Planning Meeting, Nov. 14-15, 1963, USAEC Rpt. ORNL-TM-1047, pp. 87-92, Oak Ridge National Laboratory (April 1965).

C. W. Friend and J. R. Knight, ISOCRUNCH -- Modifications to the Crunch Program for the IBM 7090, USAEC Rpt. ORNL-3689, Oak Ridge National Laboratory (January 1965).

- C. W. Friend and A. R. Jenkins, ISOTOPES - A Program for Neutron Product Yields and Decay Calculations Using a Control Data 1604-A Computer, USAEC Rpt. ORNL-3673, Oak Ridge National Laboratory (January 1965).
- C. W. Friend and J. R. Knight, ISOPOWER - Computer Program for Calculating Power Output of Single Radioisotope or Mixed Radionuclide Power Sources, USAEC Rpt. ORNL-3826, Oak Ridge National Laboratory (December 1965).
- M. W. Gerrard, P. S. Baker, and R. H. Lafferty, Jr., Selected Abstracts of Non-U.S. Literature on Production and Industrial Uses of Radioisotopes, USAEC Rpt. ORNL-TM-1356, Oak Ridge National Laboratory (December 1965).
- J. H. Gillette, Review of Radioisotopes Program, 1964, USAEC Rpt. ORNL-3802, Oak Ridge National Laboratory (May 1965).
- K. W. Haff, Radioisotope Shipping Container Development, in Proc. Packaging and Transportation of Radioactive Materials Intern. Symp., Albuquerque, N. M., January 1965, USAEC Rpt. SC-RR-65-98, pp. 413-22, Sandia Corporation (June 1965).
- C. R. King, Vacuum Leak Testing of Radioactive Source Capsules, USAEC Rpt. ORNL-3664, Oak Ridge National Laboratory (January 1965).
- E. Lamb, The Oak Ridge National Laboratory Curium Program, in Research Materials Coordination and Planning Meeting, Nov. 14-15, 1963, USAEC Rpt. ORNL-TM-1047, pp. 62-68, Oak Ridge National Laboratory (April 1965).
- R. E. Lewis, T. A. Butler, and E. Lamb, Recovery of Cesium-137 From Fission-Product Wastes and Transport by an Aluminosilicate Ion Exchanger, USAEC Rpt. ORNL-3765, Oak Ridge National Laboratory (May 1965).
- R. E. Lewis and T. A. Butler, Reactor Cross Sections for Production of Phosphorus-33 from Chlorine-36 and Enriched Sulfur-33 in Oak Ridge Research Reactor, USAEC Rpt. ORNL-TM-1065, Oak Ridge National Laboratory (October 1965).
- R. E. McHenry, Separation of Cerium-144 from Other Rare Earths by Electrolytic Oxidation and Differential Extraction, USAEC Rpt. ORNL-TM-657, Oak Ridge National Laboratory (February 1965).
- H. A. O'Brien, Jr., and W. D. Box, Promethium-147: Radiation Measurements of a One-Watt, Prototype Heat Source, USAEC Rpt. ORNL-TM-1177, Oak Ridge National Laboratory (November 1965).
- P. B. Orr, R. S. Pressly, and E. J. Spitzer, Evidence of the Absence of Long-Lived Isotopes of Promethium from Fission of Uranium, and the Purification of Promethium for the Establishment of a Primary Spectrographic Standard, USAEC Rpt. ORNL-3631, Oak Ridge National Laboratory (January 1965).
- P. B. Orr, Ion Exchange Purification of Gram Quantities of Fission-Produced Samarium with Diethylenetriaminepentaacetic Acid as the Eluant, USAEC Rpt. ORNL-TM-1062, Oak Ridge National Laboratory (September 1965).

C. L. Ottinger and R. W. Schaich, Hazards Report for Building 3517 -- Fission Products Development Laboratory, USAEC Rpt. ORNL-TM-753 Rev., Oak Ridge National Laboratory (February 1965).

C. L. Ottinger, E. E. Pierce, and R. W. Schaich, Preparation of a Cesium ( $^{137}\text{Cs}$ ) Borosilicate Glass Power Source, USAEC Rpt. ORNL-3669, Oak Ridge National Laboratory (April 1965).

J. J. Pinajian, The Effect of an Inert Diluent (Hexane) on the Distribution Coefficients of  $^{99m}\text{Tc}$  and  $^{131}\text{I}$  in Methyl Ethyl Ketone/Water and Methyl Ethyl Ketone/2 M  $\text{NH}_4\text{OH}$ , USAEC Rpt. ORNL-TM-1058, Oak Ridge National Laboratory (April 1965).

J. J. Pinajian, Accelerator-Produced Isotopes, in Research Materials Coordination and Planning Meeting, Nov. 14-15, 1963, USAEC Rpt. ORNL-TM-1047, pp. 72-76, Oak Ridge National Laboratory (April 1965).

W. R. Rathkamp, Thermal Diffusion, in Research Materials Coordination and Planning Meeting, Nov. 14-15, 1963, USAEC Rpt. ORNL-TM-1047, pp. 15-22, Oak Ridge National Laboratory (April 1965).

A. F. Rupp, J. A. Cox, and F. T. Binford, Radioisotope Production in Power Reactors, USAEC Rpt. ORNL-3792, Oak Ridge National Laboratory (May 1965).

G. S. Wamanacharya, Electroplating of Cadmium-109 on Copper Wires, USAEC Rpt. ORNL-TM-1174, Oak Ridge National Laboratory (September 1965).

#### PAPERS PRESENTED AT SCIENTIFIC AND TECHNICAL MEETINGS

E. H. Acree and R. S. Pressly, Preparation of Xenon-131m, Xenon-129m, and Krypton-79, presented at Am. Nucl. Soc., Gatlinburg, Tenn., June 21-24, 1965.

P. S. Baker, Reactor-Produced Radionuclides, presented at Symp. Med. No. 9, Radioactive Pharmaceuticals, Oak Ridge, Tenn., November 1-4, 1965.

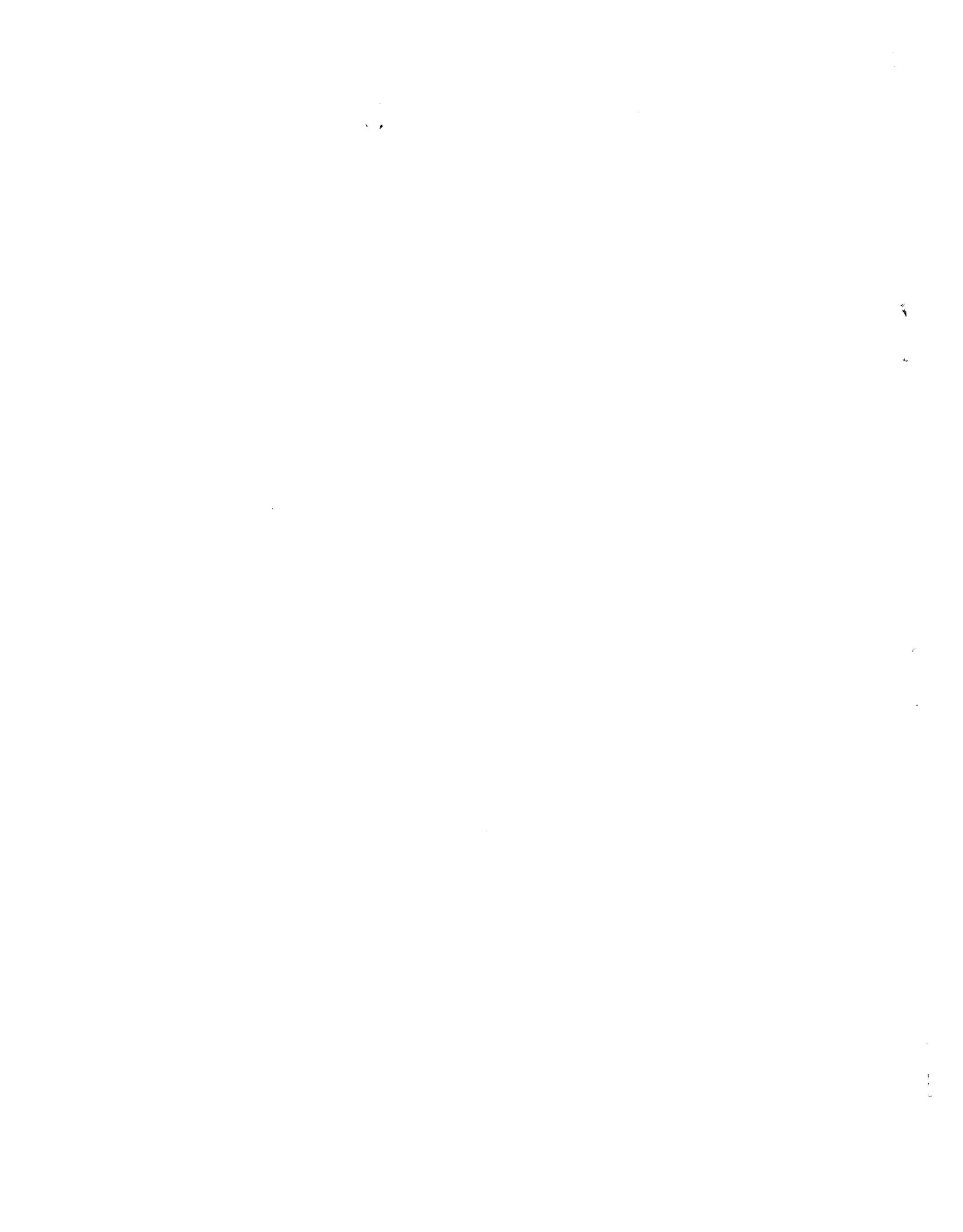
J. L. Burnett, Melting Points of Curium Trifluoride and Americium Trifluoride, presented at Am. Nucl. Soc., Washington, November 15-18, 1965.

F. N. Case and E. H. Acree, Purification of Curie Quantities of Fission-Produced Iodine-131 by Sorption on Platinum, presented at Am. Nucl. Soc., Gatlinburg, Tenn., June 21-24, 1965.

F. N. Case and T. A. Butler, Status of Methods for Recovery of Isotopes From Irradiated Reactor and Cyclotron Targets, presented at Am. Chem. Soc., Atlantic City, N. J., September 12-17, 1965.

K. W. Hafl, Radioisotope Shipping Container Development, presented at Intern. Symp. Packaging Transport. Radioactive Mater., Albuquerque, N. M., January 12-15, 1965.

- J. H. Hamilton, S. R. Amtey, B. van Nooijen, A. V. Ramayya, and J. J. Pinajian, Internal Conversion Coefficient Measurements for Use as Standards in the Energy Range 800-1200 Kev, presented at Bases Nucl. Spin-Parity Assign., APS, Gatlinburg, Tenn., November 11-13, 1965.
- H. B. Hupf, J. S. Eldridge, and J. E. Beaver, Production and Characterization of Iodine-123 for Medical Investigators, presented at Soc. Nucl. Med., Iowa City, Iowa, November 7, 1965.
- R. E. Lewis and T. A. Butler, Reactor Production and Chemical Separation of Carrier-Free Phosphorus-33 From Chlorine-36 and Enriched Sulfur-33 in the Oak Ridge Research Reactor, presented at Am. Nucl. Soc., Gatlinburg, Tenn., June 21-24, 1965.
- R. E. Lewis, R. E. McHenry, and T. A. Butler, Half-Life of Cesium-137, presented at Am. Nucl. Soc., Gatlinburg, Tenn., June 21-24, 1965.
- R. E. McHenry, Leach Rate of Strontium-90 Titanate Radioisotopic Power Fuel, presented at Am. Nucl. Soc., Gatlinburg, Tenn., June 21-24, 1965.
- R. E. McHenry and P. R. Kuehn, Melting Points of Curium and Americium Oxides, presented at Am. Nucl. Soc., Gatlinburg, Tenn., June 21-24, 1965.
- P. B. Orr, Separation of Curie Quantities of Yttrium-90 from Strontium-90 by Elution with Diethylenetriaminepentaacetic Acid and n-Hydroxyethyl-ethylenediaminetriacetic Acid Using Various Concentrations and pH, presented at Am. Nucl. Soc., Washington, November 15-18, 1965.
- J. C. Posey, P. R. Kuehn, and R. E. McHenry, The Influence of Temperature and Oxygen Pressure on the Composition of Curium Oxide, presented at Am. Chem. Soc., Atlantic City, N. J., September 12-17, 1965.
- T. C. Quinby, E. E. Pierce, and R. E. McHenry, Hot Presses for Glove Box and Manipulator Cell Use, presented at Am. Nucl. Soc., Washington, November 15-18, 1965.



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