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PACIFIC NORTHWEST LABORATORY
DIVISION OF ISOTOPE
DEVELOPMENT PROGRAMS
QUARTERLY REPORT: FEBRUARY-APRIL 1969

June 1969

AEC RESEARCH &
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UC-23, Radioisotope and
Radiation Applications

PACIFIC NORTHWEST LABORATORY
DIVISION OF ISOTOPE DEVELOPMENT PROGRAMS
QUARTERLY REPORT: FEBRUARY - APRIL 1969

Edited by W. F. Sheely

June 1969

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ABSTRACT

Pm₂O₃ and Sm₂O₃ CHARACTERISTICS

Phase Studies

X-ray diffraction analysis of Pm₂O₃ after various heat treatments and with various fractions of Sm₂O₃ is continuing. Lattice parameters have been calculated and the effect of Sm₂O₃ solution on the lattice parameters noted.

Property Measurements

A computer program has been developed to calculate the surface tension of pendant liquid drops by analysis of their profiles. The drop-weight technique, which relates the weight of a falling drop to the radius of the rod from which it falls by the surface tension of the drop liquid, is being applied to Sm₂O₃ liquid.

An effort to derive the heat capacity of ¹⁴⁷Pm₂O₃ has commenced. A laser heat-pulse method, using small samples to avoid shielding and self-heating problems, is being investigated. The thermal properties of the sesquioxides of neighboring lanthanides are being analyzed to obtain an estimate of the Pm₂O₃ heat capacity.

PROMETHIUM METAL PROPERTIES

Phase Studies

The lattice parameter of the double hexagonal unit cell of promethium metal has remained unchanged during six months' storage. The c/2a ratio corresponds to neodymium and samarium.

PROMETHIUM PURIFICATION

Process Improvement

Laboratory tests demonstrated that by use of a higher cross linked resin in the feed absorption column, batch size could be

significantly increased. However, because the capacity of the present system of columns was exceeded by use of the higher cross-linked resin, high promethium losses were experienced with successive runs. Therefore, the lower cross-linked resin will be used at the next required resin change.

FISSION PRODUCT UTILIZATION

Radiation Characteristics of Fission Product Rhodium and Palladium

Preliminary analysis of the rhodium isotopic content of a sample of expended Yankee Reactor fuel has been made. The palladium isotopic content has not yet been determined. However, as was expected, no measurable gamma output of the palladium has been found.

Low Cost Cesium Strontium Sources

A program for evaluation of the usefulness as sources of encapsulated ^{137}Cs and ^{90}Sr separated reactor wastes has been started.

SOURCE DEVELOPMENT

Nondestructive Testing

Acoustic emission studies of the submerged arc welding process indicate that slag inclusions can be detected during improper bead formation. Radiographic studies of weld beads laid on various metals and alloys confirmed that acoustic emissions can indicate the presence of cracks, porosity, and inclusions in many materials but not in some. Acoustic emission appeared also to be sensitive to lack of penetration in electron beam weld. Efforts are under way to evaluate a technique of locating the flaw that causes detected acoustical emission.

A program to establish nondestructive test procedures to evaluate the cladding and core characteristics of stainless-steel-clad $^{90}\text{SrSiO}_3$ sources has been initiated. Based on

communication with major AEC laboratories performing source development, a testing program and objectives have been established.

A program aimed at developing ^{252}Cf as a transportable neutron source has begun. Computer calculations of the differential neutron spectra and activation foil reaction rates have been made for various detector foils exposed to a bare ^{252}Cf spectrum. Similar calculations will be made for moderated ^{252}Cf neutron spectra, now being calculated.

Source Fabrication Technology

Reports relating to promethium source fabrication technology that have been issued include "Promethium-147 Radioisotope Application Program AMSA Heat Source - Final Report" (BNWL-994) and "Fabrication of $^{147}\text{Pm}_2\text{O}_3$ by High Energy Rate Forming," (BNWL-1041). Another report presenting a general summary of $^{147}\text{Pm}_2\text{O}_3$ Fabrication Technology is in the preparation stage and will emphasize the favored fabrication process for Pm_2O_3 : cold pressing and sintering.

As part of a continuing review of commercial metal and alloy source cladding candidates, physical and mechanical property data for superalloys and refractory metals and alloys are reviewed and PNL fabrication experience discussed.

TECHNOLOGY UTILIZATION

Evaluation of Isotope Production in Power Reactors

Target irradiation yield data covering a wide range of irradiation conditions have been calculated for ^{237}Np , ^{241}Am , a 50/50 mixture of $^{241}\text{Am}/^{243}\text{Am}$ and ^{169}Tm target materials. Variables considered were neutron flux, spectral hardness, and target density. The manner in which these variables affect production of ^{241}Am , ^{242}Cm and ^{238}Pu is shown.

The TAIRCOST program has been used to calculate the cost of ^{238}Pu produced by ^{241}Am target irradiations (considering two types of reactor spectra). The optimum irradiation time and cost were calculated over a range of cost assumptions and operating parameters.

CIRCULATORY SUPPORT SYSTEMS

Radiation Characteristics

Studies of the shielding and in-phantom dose prediction and in-phantom dosimetry have been completed for promethium systems.

Gamma interaction coefficients in QAD have been updated. Neutron transport calculations have been made on lithium hydride and lithium halide moderator compounds, and neutron removal cross-sections are being determined.

Neutron and gamma dose rates from ^{252}Cf in tissues have been determined, assuming a ^{252}Cf source the size of the SRL californium needle and compared with experimental values.

The calibration of dosimeters designed for the in-phantom plutonium dosimetry measurements has been nearly completed. Photon spectral measurements have begun on the LASL-1 source.

In-phantom dose rate measurements from ^{252}Cf sources have begun.

Promethium Isotopic Characteristics

The $^{146}\text{Pm}/^{147}\text{Pm}$ ratios in all ^{147}Pm targets irradiated in the Hanford K reactor have been analyzed and used to calculate a ^{146}Pm cross-section. The two results obtained, 3700 and 3300 b, are much larger than that determined in the Oak Ridge Research Reactor, probably because of the higher incidence of resonance capture in the ORR.

The Savannah River promethium irradiations have been completed, and $^{146}\text{Pm}/^{147}\text{Pm}$ ratios will be determined.

TABLE OF CONTENTS

LIST OF FIGURES	ix
LIST OF TABLES	xi
INTRODUCTION	1
Pm ₂ O ₃ AND Sm ₂ O ₃ CHARACTERISTICS	3
Phase Studies	3
Stability Studies	4
Property Measurements	4
Compatibility Studies	12
PROMETHIUM METAL PROPERTIES	13
Phase Studies	13
PROMETHIUM PURIFICATION	15
Process Improvement	15
PROMETHIUM TECHNOLOGY BOOK	17
FISSION PRODUCT UTILIZATION	19
Radiation Characteristics of Fission Product Rh and Pd.	19
Low Cost Cesium and Strontium Sources	20
SOURCE DEVELOPMENT	21
Nondestructive Testing.	21
Source Fabrication Technology	33
TECHNOLOGY UTILIZATION	41
Evaluation of Isotope Production in Power Reactors.	41
CIRCULATORY SUPPORT SYSTEMS	51
Radiation Characteristics	51
Promethium Isotopic Characterization	52
DISTRIBUTION	55

LIST OF FIGURES

1	Radiographic System for Liquid Expansion, Viscosity, and Surface Tension Measurements	5
2	Parameters for Determination of Surface Tension from the Shape of a Pendant Drop	6
3	Radiograph of a Pendant Drop Formed by Liquid Sm_2O_3 in a Molybdenum Capsule	8
4	The Heat Capacities of Sesquioxides of Rare Earths Adjacent to Promethium on the Periodic Table	10
5	Acoustic Emission During Submerged Arc-Welding Compared to Weld Bead Conformation	22
6	SrSiO_3 Irradiation Source Description	25
7	Bare ^{252}Cf Differential Neutron Spectrum from Time-of-Flight Measurements	29
8	^{252}Cf Neutron Spectrum in Lethargy Space	30

LIST OF TABLES

1	Crystal Lattice Parameters of Lanthanide Sesquioxides	3
2	Isotopic Composition of Rhodium from Yankee Reactor Irradiated Fuel	19
3	Calculated Activity of Detector Foils Exposed to a Bare ^{252}Cf Neutron Spectrum	31
4	Foil Activation Cross Sections in a Bare ^{252}Cf Neutron Spectrum	32
5	$^{147}\text{Pm}_2\text{O}_3$ Pellet Pressing and Sintering Data	35
6	Properties of Superalloys	37
7	Properties of Refractory Metals and Alloys	38
8	Composition of ^{237}Np Targets After Irradiation, at.%	42
9	Composition of ^{241}Am Targets After Irradiation, at.%	44
10	Composition of 50/50 $^{241}\text{Am}/^{243}\text{Am}$ Targets After Irradiation, at.%	45
11	Composition of ^{169}Tm Targets After Irradiation, at.%	47
12	Base Cost Sets	48
13	Variation in Price of ^{238}Pu Produced in a Power Reactor	48
14	Variation in Price of ^{238}Pu , Produced in a Thermal Reactor	49

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INTRODUCTION

This is the tenth report of a series initiated in 1966 to provide detailed and timely information on the results of the isotopic fuels program and related research being conducted at the Pacific Northwest Laboratory by Battelle-Northwest for the U.S. Atomic Energy Commission. The previous reports of the series are BNWL-385, 466, 570, 680, 681, 781, 877, 924, and 1010.

This report covers the three-month period, February 1969 through April 1969.

Pm₂O₃ AND Sm₂O₃ CHARACTERISTICSPHASE STUDIES

P. G. Pallmer

Analysis of the X-ray diffraction results of Pm₂O₃ after various heat treatments and with various Sm₂O₃ fractions has continued. Particular attention has been paid to the indexing of observed reflections and the calculation of a best fit of parameters using all available observations or indexed reflections. The results for the monoclinic (B) and cubic (C) modifications are consistent with those of Nd₂O₃ and Sm₂O₃ (Table 1). The data of Weigel and Scherer⁽¹⁾ were recalculated with the assignment of the observed reflection at $\theta = 15.86$ to $\bar{1}12$ rather than 112, with a significant change that brings the parameter β into agreement with our results.

TABLE 1. *Crystal Lattice Parameters of Lanthanide Sesquioxides*

<u>Parameter</u>	<u>B-Phase, Å</u>				<u>C-Phase, Å</u>
	<u>a</u>	<u>b</u>	<u>c</u>	<u>β</u>	<u>a</u>
Nd ₂ O ₃	14.35	3.666	8.99	100.34	11.08
Pm ₂ O ₃	14.19	3.64	8.93	100.1	10.99
Sm ₂ O ₃	14.16	3.621	8.84	100.05	10.93
Ref. 1	14.22	3.656	8.94	100.0	

When these parameters are compared to those of the other rare earths for linearity in proportion to ionic radii, a uniform interpolation is not obtained. The slight improvement in parameters obtained by a least squares type fit to all the indexed reflections requires a re-examination of the parameter change with solution of Sm₂O₃ in Pm₂O₃. A linear variation is still expected because of the small difference between the promethium and samarium ionic radii.

STABILITY STUDIES

(No effort was applied to this area during the quarter.)

PROPERTY MEASUREMENTS

J. J. Rasmussen and R. L. Gibby

The surface tension of liquid Sm_2O_3 is being determined from the shape of pendant drops radiographed in closed refractory metal capsules, and by a drop-weight technique. The experimental system for observing the formation of the pendant drops is illustrated in Figure 1. A computer program has been written to determine the surface tension from the profile of the drop. The program has been written to obtain a solution to the differential equation describing the shape of a pendant or sessile drop.

$$bf''(Z) = \left(2 + \frac{\beta Z}{b}\right) \left\{1 + [f'(Z)]^2\right\}^{3/2} - b/X \left\{1 + [f'(Z)]^2\right\} f'(Z) \quad (1)$$

where $f'(Z) = dZ/dX$

$$f''(Z) = d^2Z/dX^2$$

$$\beta = gb^2\rho/\gamma \quad (2)$$

and b is the radius of curvature at the base of the drop, g is the gravitational acceleration, γ is the surface tension, ρ is the density of the liquid Sm_2O_3 and X and Z are the drop profile coordinates illustrated in Figure 2. The program is set up by measuring X, Z pairs from the drop profile along the arc segment from $\phi = 0^\circ$ to $\phi = 90^\circ$, where ϕ is the angle formed by the tangent to the arc at the point X, Z and the X -axis. The largest value of X and the associated Z value are found and are assumed to be X_{90} and Z_{90} , which are the drop coordinates at the equator. A table of X_{90}/Z_{90} as a function of β , taken from Bashforth and Adams,⁽²⁾ is searched to obtain the initial value of β ; the

FIGURE 1. Radiographic System for Liquid Expansion, Viscosity, and Surface Tension Measurements

FIGURE 2. Parameters for Determination of Surface Tension from the Shape of a Pendant Drop

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FIGURE 3. *Radiograph of a Pendant Drop Formed by Liquid Sm_2O_3 in a Molybdenum Capsule*

function $Z(X, b, \beta)$ is generated from Equation (1). The parameters b and β are then adjusted to obtain the best fit between the experimental values, and the calculated values and the surface tension is calculated by using Equation (2).

The drop-weight technique for measuring the surface tension involves catching the pendant drops which fall from the suspended Sm_2O_3 rod into a crucible placed in the bottom of the capsule. The weight of the drops are then determined, and the surface tension is calculated from Equation (3).

$$\gamma = mgF/r \quad (3)$$

where γ is the surface tension; mg is the weight of the drop; F is a correction factor to adjust for the drop volume, which remains attached to the rod; and r is the radius of the rod.

The Sm_2O_3 rods were formed by hydrostatic pressing at 30,000 psi without binder. These rods were sintered at 1730 °C for 4 hr in air to 83% TD, and a rod 0.207 in. diam by 1.987 in. long was machined. The rod was suspended from the top of a molybdenum capsule which was filled with 0.1 atm argon. A tungsten crucible of known weight was placed in the bottom of the capsule to catch the drops of Sm_2O_3 falling from the rod (Figure 3). Five such drops were formed and four of these were allowed to fall into the tungsten crucible. Analyses of these drops to allow surface tension calculation is underway.

During this quarter an investigation was initiated to derive the high temperature (100-1600 °C) heat capacity of $^{147}\text{Pm}_2\text{O}_3$, which is important to isotopic fuel design and safety analysis. A pulse method of measuring the heat capacity of small samples was investigated, and a literature survey was instigated to determine if the heat capacity of $^{147}\text{Pm}_2\text{O}_3$ could be estimated to sufficient accuracy from the heat capacity data reported for the neighboring lanthanide sesquioxides.

The radioactivity of $^{147}\text{Pm}_2\text{O}_3$ poses several problems in measuring heat capacity. Although in small quantities (<1 g), only soft beta (which can be easily shielded) is emitted, in the quantities necessary for drop calorimetry studies (~ 20 - 30 g), bremsstrahlung radiation becomes a problem and heavy shielding is required. Naturally, with this quantity of material, the heat output also increases the sample temperature requiring adjustment of heat content to 298°K , as well as application of correction factors to the experimental measurements. In view of these difficulties, direct heat capacity measurements utilizing a small sample (~ 200 mg) should be a more promising approach to the problem. Presently, calculations have indicated that the laser heat-pulse method, used to determine thermal diffusivity, might be modified to also yield heat capacity data; however, experimental measurements have not been made to verify these calculations.

The elements of the 4f series exhibit many similar properties. In most instances they combine with oxygen to form sesquioxides, which may crystallize in the cubic (C-type), hexagonal (A-type), or monoclinic (B-type) forms. Although many of the rare earth sesquioxides are polymorphic, only Pm_2O_3 exists at a fairly low temperature in all three crystalline forms. In the case of Pm_2O_3 , the cubic form transforms irreversibly between 750 to 1000°C to the monoclinic form, which transforms reversibly at higher temperatures to the hexagonal form; after ceramic processing to form a dense body Pm_2O_3 generally exists in the monoclinic form.

The heat capacity data reported for the sesquioxides of the three elements adjacent to promethium in the lanthanide series are shown in Figure 4. Heat capacities for Eu_2O_3 and Sm_2O_3 were reported by Pankratz, King, and Kelley⁽³⁾ while the

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FIGURE 4. The Heat Capacities of Sesquioxides of Rare Earths Adjacent to Promethium on the Periodic Table

heat capacity for Nd_2O_3 was also reported by Pankratz et al.⁽³⁾ and by Blomeke and Ziegler.⁽⁴⁾ The data shown for Eu_2O_3 and Sm_2O_3 are for the material in the monoclinic (B-type) structure while the Nd_2O_3 data are for a hexagonal (A-type) structure. The discontinuity exhibited at 1200 °K in the Sm_2O_3 data presumably corresponds to the C → B transition. However, the appearance of this discontinuity is not understood at this time, since this transition is reported to be irreversible. Furthermore, it is thought to be second order which would not result in a discontinuous change in the heat capacity. The discontinuity shown for Nd_2O_3 is also not understood because it does not correspond to any reported structural transformation. On the other hand, the data for Eu_2O_3 exhibit the expected variation of heat capacity with temperature. Further analysis of heat capacity data, which will include the data available for other rare earth sesquioxides, is necessary to explain these results. An estimate of the heat capacity of $^{147}\text{Pm}_2\text{O}_3$ from available data for the neighboring sesquioxides will then be made.

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2. T. Bashforth and J. C. Adams. An Attempt to Test The Theories of Capillary Action, University Press, Cambridge, England, 1883.
3. L. B. Pankratz, E. G. King, and K. K. Kelley. U.S. Bureau of Mines, Rept. Invest. No. 60633, pp. 1-18. 1962.
4. J. O. Blomeke and W. T. Ziegler. J. Am. Chem. Soc., vol. 73, p. 5099. 1951.

COMPATIBILITY STUDIES

H. T. Fullam and F. P. Roberts

The hydrolysis studies of promethium sesquioxide have been completed, and a final report is now being written. (Authors: F. P. Roberts and H. T. Fullam.)

A paper on promethium technology, authored by H. T. Fullam and H. H. Van Tuyl, has been submitted to Isotopes and Radiation Technology for publication.

Three formal reports* have been written, two of which are now at printing. The third will be ready for publication shortly.

* H. T. Fullam. The Use of Diffusion Barriers in Reducing Metal-Metal Interactions, at 1100 °C, BNWL-1015.

H. T. Fullam. Cold Pressing of Rare Earth Oxide-Aluminum Composites, BNWL-1014.

H. T. Fullam. The Production of ^{147}Pm By the Neutron Irradiation of ^{146}Nd .

PROMETHIUM METAL PROPERTIESPHASE STUDIES

P. G. Pallmer

After six months of observation, no significant change in the lattice parameter of the double hexagonal unit cell of promethium metal has been found. The preliminary lattice parameters, previously reported, were calculated from individual a- and c-type spacings. A least-squares fit to the observed reflections for the films obtained so far yielded the lattice parameters: $a = 3.66 \pm 0.01 \text{ \AA}$ and $c = 11.80 \pm 0.04 \text{ \AA}$. The axial ratio $c/2a = 1.61$ calculated from these values corresponds to the same ratios for neodymium and samarium. The density calculated from the better X-ray diffraction data now available is $7.13 \pm 0.07 \text{ g/cm}^3$, which may be compared to the measured density of $7.22 \pm 0.03 \text{ g/cm}^3$ obtained from a bulk sample.

PROMETHIUM PURIFICATIONPROCESS IMPROVEMENT

J. A. Partridge

Laboratory tests during the preceding quarter had shown that the batch size in the promethium recovery and purification process could be increased by use of higher cross-linked resin in the feed absorption column. This was demonstrated on full scale tests during this quarter. In the first run, with 12% cross-linked resin in the feed absorption column, 399,000 Ci of promethium were recovered, an increase from the 350,000 Ci recovered with 8% cross-linked resin.

However, high promethium losses were experienced with subsequent runs; apparently, because the capacity for the present system of columns had been exceeded. Because of flow-rate requirements, when the europium breakthrough occurs on the bottom of the fourth column, the feed absorption column and the first elution column are taken out of the system. Apparently, the rare-earth bands ahead of the promethium are of such length that the promethium is not completely off of the first elution column when that column must be removed from the system.

Promethium losses per run were reduced when approximately 10% of the resin was removed from the feed absorption column.

Thus, while it was demonstrated that the batch size could be increased significantly by going to the 12x resin, the higher promethium losses in the present system of columns suggest that we return to the use of the 8x resin at the time of the next required resin change.

Prior to the last resin change, it was demonstrated that the resin was good for 14 purification runs.

Promethium Recovery

A total of 11 promethium purification runs were completed during the quarter in which 3.70 MCi were processed.

PROMETHIUM TECHNOLOGY BOOK

An expanded outline of the book "Promethium Technology" has been completed and responsibilities for chapter preparation assigned.

FISSION PRODUCT UTILIZATION

RADIATION CHARACTERISTICS OF FISSION PRODUCT Rh AND Pd

F. P. Roberts

This study has the purpose of determining the nature and amounts of radiation from the rhodium and palladium formed in power reactors. The results from this work will allow an evaluation of usefulness of these fission products as industrial materials.

In an earlier study it was found that 11 year old rhodium from Hanford reactors contained $\sim 3\mu\text{Ci}$ of $^{102\text{m}}\text{Rh}$ (2.9 year half-life). This well-aged material contained no measureable ^{102}Rh (205 day half-life), but a short-cooled fission products is likely to contain this isomer. Both isomers of ^{102}Rh decay by emission of penetrating gamma rays and if present in large quantities, they would give rise to serious dose rate.

Fission product palladium is known to contain ~ 16 to 20% ^{107}Pd (7×10^6 year half-life). This isotope decays by emission of a 0.035 MeV beta and no gamma. Because of the low energy of decay and the long half-life, it is believed that problems due to radioactivity will not restrict its use; however, this study will verify this.

During the period covered by this report, a method for isolating and purifying both rhodium and palladium from spent power reactor fuels has been developed. Two rhodium and palladium samples have been isolated from Yankee Reactor fuel, and preliminary evaluation of the rhodium isotopic composition has been made (Table 2).

TABLE 2. *Isotopic Composition of Rhodium from Yankee Reactor Irradiated Fuel*

Sample No.	$^{102\text{m}}\text{Rh}$ (2.9 year half-life)	^{102}Rh (205 day half-life)
1	$100 \pm 40 \mu\text{Ci/g}$	$55 \pm 20 \mu\text{Ci/g}$
2	$110 \pm 45 \mu\text{Ci/g}$	$69 \pm 28 \mu\text{Ci/g}$

The measured burnup of the fuel from which the samples were taken was 33000 MWd/tonne, and the cooling time was 3 1/2 years.

As was expected, evaluation of the palladium showed no measurable gammas. The isotopic composition has not been measured yet.

Further work is in progress to improve accuracy of the rhodium data and to make dose-rate calculations for some typical industrial applications.

LOW COST CESIUM AND STRONTIUM SOURCES

H. H. Van Tuyl

A short-term feasibility study of utilizing Cs-137 and Sr-90 reactor wastes (encapsulated in the Hanford Waste Management) as heat or radiation sources has been begun. The Division of Production has authorized ARHCO to make information available on the design features of the Encapsulation Facility as they become available from current studies and the project design effort. Programs of coordination and cooperation with BNL and ORNL are being established.

SOURCE DEVELOPMENTNONDESTRUCTIVE TESTINGMaterials Characterization - W. D. Jolly

Ultrasonic characterization of radioisotope encapsulation materials has been deferred to FY-71. The primary reason for this is that the development of an acoustic-emission nondestructive test for welds has been given priority over the materials characterization study.

Advanced NDT Methods and Concepts - W. D. Jolly

Recent publicity concerning the development of the acoustic-emission technique for nondestructively testing welds during welding has stimulated considerable interest from industry. Almost 200 inquiries from manufacturers and government laboratories reflect the need for improved nondestructive methods for testing welds. The initial development of this process (sponsored by the DID) is finding wide industrial application in fields far removed from isotopes.

The welding techniques and materials evaluated during this period include submerged arc welding in A212 and A302 steel, gas-tungsten arc welding in 6061 aluminum, Inconel 718, and 4130 steel, and electron-beam welding in titanium. Of the applications investigated, the submerged arc welding process, which is used extensively in the manufacture of pressure vessels for nuclear power plants and other steam generators, appears to be the most promising of the applications of the acoustic-emission technique to on-line weld inspection. Test results indicate that the acoustic-emission technique can predict slag inclusion in submerged arc welds. This is possible because the acoustic emission instrument is sensitive to variations in bead conformation. Figure 5 demonstrates the effect of changes

FIGURE 5. Acoustic Emission During Submerged Arc-Welding Compared to Weld Bead Conformation

in bead conformation on the acoustic-emission rate recorded during a submerged arc welding test. Sensitivity to bead conformation stems from the fact that the flux, used in submerged arc welding, forms a brittle glassy cover over the weld bead. On a smooth weld bead, this glassy cover lifts easily as it cools. However, if the bead conformation becomes rough or rolled in such a manner that the slag can be trapped, the slag fractures extensively as it attempts to lift away from the weld. Thus, an increase in the acoustic-emission rate during welding one pass indicates the probable occurrence of a slag inclusion on the following pass. In addition to the ability to predict slag inclusions, the acoustic-emission response also is sensitive to the occurrence of cracking and porosity during welding on submerged arc welding.

Using the gas-tungsten arc welding technique, the acoustic emission system was evaluated on several materials, including 6061 aluminum, Inconel 718, 4130 steel, and titanium. Radiographic data from welds in these materials verify that the acoustic-emission system was sensitive to:

- Cracks, porosity, and inclusions in the 6061 aluminum
- Porosity and inclusions in Inconel 718
- Cracking in titanium and 4130 steel.

The radiographic data did not indicate cracking in the Inconel 718, or porosity and inclusions in titanium and 4130 steel. The acoustic-emission system was also evaluated on butt welds in titanium made by the electron-beam welding process. The acoustic-emission response appeared to be sensitive to lack of penetration in the electron-beam welds, probably because of the porosity accompanying the lack of penetration in electron-beam welds. The acoustic-emission response to cracking was not determined in this test since radiographic data did not verify the presence of cracks.

The results of this evaluation program and the results of other experiments previously reported indicate the versatility of the acoustic-emission technique for in-process weld inspection. For example, the same instrument was used to monitor acoustic emission from welds in 1/4 in. diam SS tubing with 16 mil wall thickness and a weld in 6 in. thick steel plate, 4 1/2 ft long.

The acoustic-emission instrument presently used has a major shortcoming in its inability to locate the defects to which it responds. We have just completed the fabrication of a new instrument (to be capitalized under the DID program) that should correct this shortcoming. This instrument incorporates two acoustic-emission receiver channels and the signal-processing electronics necessary to present on a strip-chart recorder the location of a weld defect producing acoustic emission. This new instrument also presents the acoustic-emission rate information on a second channel of the same strip-chart recorder. Evaluation of this instrument is now underway to determine the accuracy with which an emission source can be located and the useful range of the location technique. The addition of the defect location features to this new instrument will relieve the difficulty encountered in correlating the acoustic emission response with radiographic or destructive-test analysis of weld coupons.

SrSiO₃ Irradiation Sources - R. S. Steffens and J. C. Crowe

A research program to establish basic nondestructive test methods for monitoring the physical parameters of the SrSiO₃ irradiation sources (Figure 6) was initiated. The parameters to be monitored are clad anomalies, core homogeneity, and a closure weld integrity. The clad is rectangular 347 SS tubing which is 0.006 in. thick. The core consists of SrSiO₃ mini-spheres embedded in an aluminum matrix. Peripheral closure

FIGURE 6. *SrSiO₃ Irradiation Source Description*

welds seal each end of the capsules. The thin-clad and composite type core presents unique problems to the nondestructive monitoring of the source characteristics.

The clad inspections can be classified into the three following categories:

- Inspection of the cylindrical tubing as obtained from the vendor;
- Inspection of the tubing after fabrication into the rectangular shape;
- Inspection after fueling and subsequent source use.

Standard procedures exist for the ultrasonic and/or eddy current inspection of the cylindrical, vendor-fabricated tubing. The prime areas of concern after the tubing has been fabricated into the rectangular tubing are the corners which complicate the inspection and which thus require refined signal analysis. Nondestructive inspection of the clad under fueled conditions is complicated by the varied nondestructive test characteristics of the stainless steel, aluminum and SrSiO_3 spheres, and caution must be exercised in the test application. Eddy currents ultrasonic, and thermal test parameters will be developed or defined to provide the basis for nondestructively testing the clad as required by the source fabrication organization.

Potential core characteristics to be nondestructively monitored are: (1) distribution of the mini-spheres in the area adjacent to the inner clad surface; and (2) gross voids. Nondestructive test method guidelines will be established to assure core quality.

The peripheral closure welds used to seal the SrSiO_3 sources are difficult to inspect with standard nondestructive test techniques. Since this type weld is being increasingly used in the AEC complex and in private industry, guidelines for weld joint design to permit weld testability should be established.

The SrSiO_3 irradiation source nondestructive test development objectives described above resulted from communications with the major AEC agencies performing source development.

^{60}Co Irradiation Sources - R. S. Steffens

Development of high-speed ultrasonic test techniques and methods for inspecting the cobalt-source type closure welds was resumed. Second generation multi-element receiver-brush transmit ultrasonic transducer pairs have been fabricated to facilitate development and evaluation of high speed ultrasonic test systems. The closure weld on these samples is being used to establish nondestructive techniques for inspecting that type peripheral closure welds.

Neutron Radiography

J. L. Cason

The scope of work projected for FY 1969-1970 is developing ^{252}Cf as a transportable neutron source camera or apparatus for neutron radiography. This program can be classified into four major areas of emphasis:

1. Analytical analysis of optimum moderator material, imaging techniques and spectrum tailoring.
2. Acquiring a 300 μg source (will parallel Phase I)
3. Providing experimental evidence to confirm the analytical investigation and design work for a prototype apparatus.
4. Fabricating a prototype design followed by extensive testing.

Spectrum tailoring emphasis will be placed on neutron energy ranges corresponding to resonance peaks in the cross sections for various direct and indirect imaging foils for (n,α) and (n,γ) reactions.

Neutron spectrum data have been obtained from time of flight measurements. The spectra data were used as input to SAND II for obtaining the differential neutron spectra and spectral average cross sections (Figure 7). This calculation utilized the first 27 groups of GAM I group structures. A plot of $\phi(u)$ versus u was prepared and is given in Figure 8.

Various foil reaction rates were programmed to determine response to the bare ^{252}Cf spectrum. These results are shown in Figure 7, and the data are given in Table 3. Other foils are being programmed for reaction rate determination to help determine which foils should be experimentally examined as choices for imaging fast neutrons. Emphasis was placed on known imaging foils such as gold, gadolinium, indium and possibly copper, with results summarized and given in Table 4.

Similar foil response calculations will be prepared after moderated spectrum data have been obtained for the cases of BeH_2 , ZrH_2 , and TiH_2 as a function of moderator thickness.

Preliminary calculations based on one-dimensional transport theory for a point source with spherical shell moderators have been prepared for BeH_2 , ZrH_2 , and TiH_2 . Results indicate that less than 10 cm of moderating material will be required to tailor the neutron spectra.

Initial computer calculations with the ANSEN Code were unsuccessful due to the improper operation of the subroutine (Tape Maker) preliminary pass. The purpose of this auxiliary code was to obtain the neutron cross section needed for calculations in ANSEN. Present computer calculations are in progress by using the HRG Code in place of the subroutine in ANSEN, along with the ANSEN Code to provide the data needed to obtain plots of neutron spectra as a function of moderator thicknesses.

FIGURE 7. Input Differential Spectrum [Bare ^{252}Cf Spectrum (Smith). (Time of Flight)]

FIGURE 8. ^{252}Cf Neutron Spectrum in Lethargy Space

TABLE 3. *Calculated Activity of Detector Foils Exposed to a Bare ²⁵²Cf Neutron Spectrum*

ACTIVITY CALCULATION RESULTS						
(NORMALIZED TO 1.00-10 MFV)						
Foil Reaction	SATURATED CALCULATED (DPS/NUCLEUS)	NOMINAL			ACTUAL OUTSIDE LIMITS (PERCENT)	ACTIVITY LIMITS (PERCENT)
		5.00 PERCENT ACTIVITY LIMITS (MFV),				
		LOWER	UPPER			
ZFE54 (N.P)MN54	9.591-26	2.500+00	8.500+00	5.461	5.057	
W-DNI58(N.A)FE55	6.283-26	3.00+00	9.500+00	4.775	5.014	
TI47(N.P)SC472	2.340-26	2.100+00	7.700+00	4.256	4.846	
TI48(N.P)SC487	6.316-28	6.700+00	1.260+01	4.559	4.891	
FE58(N.G)FF597	1.421-27	5.500-02	3.300+00	5.140	4.825	
SC45(N.G)SC467	6.736-27	6.300-02	4.000+00	4.940	5.087	
CO59(N.G)CO607	6.285-27	1.275-01	3.000+00	4.875	5.285	
NI58(N.P)CO58Y	1.395-25	2.300+00	8.100+00	5.359	5.028	
NI60(N.P)CO60	8.243-27	2.800+00	1.010+01	5.082	4.901	
CU63(N.A)CO60	7.359-28	6.200+00	1.160+01	5.573	4.931	
GD158G(N.G)GD159	8.402-26	4.000-02	3.100+00	4.995	5.121	
U238(N.G)U239M	6.923-26	1.100-01	3.300+00	5.048	4.802	
LI6(N.A)H3	3.218-25	1.500-01	3.800+00	4.974	4.813	
PU239(N.F)FPM	1.788-24	3.200-01	6.200+00	4.855	4.965	
GB10(N.A)LI7	3.423-25	5.500-02	4.400+00	4.903	5.057	
TH232(N.G)TH233	8.818-26	9.200-02	3.500+00	5.007	5.086	
AU197(N.G)AU198	8.335-26	6.000-02	3.500+00	4.900	4.980	
IN115(N.G)IN116M	1.313-25	1.275-01	3.500+00	4.956	5.076	
CU63(N.G)CU64	9.813-27	3.400-02	4.600+00	5.003	4.891	
TH232(N.F)FP	8.641-26	1.600+00	7.900+00	6.279	4.947	
IN115(N.N)IN115M	1.946-25	1.300+00	6.000+00	5.797	5.101	
ZN64(N.P)CU64	5.278-26	2.400+00	8.500+00	4.509	4.857	
HFE54(N.P)MN54	1.153-25	2.500+00	8.000+00	4.445	5.097	
AL27(N.P)MG27	6.748-27	3.400+00	9.800+00	4.423	5.005	
SI28(N.P)AL28	1.872-26	5.500+00	1.050+01	4.597	4.860	
TI27(N.2N)TI26	1.759-27	1.010+01	1.490+01	5.796	5.033	
ZR90(N.2N)7R89	2.383-28	1.260+01	1.690+01	5.331	4.914	
TI46(N.P)SC46R	2.205-26	3.500+00	9.700+00	4.519	4.865	
NP237(NF)FPM	1.362-24	7.600-01	6.700+00	5.039	4.966	
U238(N.F)FPM	3.433-25	1.500+00	7.600+00	3.783	4.908	
U235(NF)FPM	1.245-24	2.400-01	6.300+00	5.109	5.080	
IN115(N.N)MM	2.117-25	1.300+00	6,500+00	5.328	5.020	

TABLE 4. *Foil Activation Cross Sections in a Bare ^{252}Cf Neutron Spectrum*

Foil Reaction	Average Cross Section, barns $\times 10^{-3}$	Nominal Limits, MeV	
		Lower	Upper
G (Gamma)			
A (Alpha)			
Cu 63 (N,A) CO 60	0.7359	6.200	1.160
Gd 158 (N,G) Gd 159	84.02	4.00×10^{-2}	3.10
Au 197 (N,G) Au 198	83.35	6.00×10^{-2}	3.50
In 115 (N,G) In 116	131.3	1.275×10^{-1}	3.50
Cu 63 (N,G) Cu 64	9.81	3.40×10^{-2}	4.60

It is expected that computer calculations will be completed by April 25, 1969, which will provide a complete analytical analysis of the problem. Actual experimentation with the ^{252}Cf will begin as soon as the source is obtained from ORNL; however, unexpected delay is being experienced in obtaining the source.

Results will be assimilated and presented in a manner to provide the general public with charts, tables, etc., in an effort to develop new applications for ^{252}Cf . Exploration applications for portable neutron radiography include Aerospace materials and components, Biological applications, Public Utilities Department, Civilian Power Reactor surveillance and testing and reactor fuels examination. Other areas such as petroleum, mining, and oceanographic research could adapt the source for activation analysis studies as well as neutron radiography.

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SOURCE FABRICATION TECHNOLOGY

N. C. Davis

Promethium Source Fabrication

A recently published report⁽¹⁾ on promethium-147 fabrication technology details the PNL contribution to a DID-sponsored Air Force Program. The abstract of this report is as follows:

A promethium-147 fueled 60 W_t heat source was fabricated for the Air Force for use in heating a navigation device for the Advanced Manned Strategic Aircraft (AMSA) Program. A high strength impact resistant capsule was designed, and a capsule fabrication process was developed. A series of dummy (Sm₂O₃ stand-in for ¹⁴⁷Pm₂O₃) capsules was fabricated, impact tested, and destructively evaluated to demonstrate process quality and to determine impact strength. Process and quality control inspection procedures were established and the ¹⁴⁷Pm₂O₃ fueled capsule was produced and delivered to the Air Force for laboratory testing and flight demonstration of the AMSA navigational equipment.

Another recent addition to the promethium fabrication technology documentation is the "Fabrication of ¹⁴⁷Pm₂O₃ by High Energy Rate Forming."* The abstract is presented below:

High energy rate forming (HERF) has been used as a fabrication method for the densification of encapsulated promethium sesquioxide. The development and characteristics of the HERF

* Report number BNWL-1041 is presently being published.

process are presented. Simulated radioisotope materials have been used to develop the capability for controlled dimensional integrity, and $^{147}\text{Pm}_2\text{O}_3$ has been densified in various metal claddings for compatibility testing.

A general summary report on $^{147}\text{Pm}_2\text{O}_3$ Fabrication Technology is in the preparation stage. It will summarize the various fabrication procedures and techniques that have been used at PNL during the preparation of approximately fifty $^{147}\text{Pm}_2\text{O}_3$ heat sources. These range in size from the 60 W_t AMSA capsule to a miniature 0.001 W_t source fabricated with 0.0005 g of the oxide.

The conventional cold-pressing and sintering process, which has been used extensively and which presently, is felt to be the best process for most applications, will be emphasized in this report. The promethium sesquioxide pellet-pressing and sintering data presented in Table 5 are a collection of the experience gained at PNL working with sources for specific applications. Since the pellets fabricated were used for compatibility studies or specific thermal uses, the many variables that could be evaluated to obtain the optimum pressed and sintered product have not been fully explored. Development work with the samarium stand-in has been beneficial, but not at all inclusive when working with the radioactive and thermally-hot promethium powder.

Cladding and Encapsulation Development

A select group of materials, including superalloys and refractory metals, has received consideration for use as primary and/or secondary claddings for $^{147}\text{Pm}_2\text{O}_3$ heat sources. Material compatibility, welding characteristics, and measurements of ultrasonic properties for nondestructive testing have

TABLE 5. $^{147}\text{Pm}_2\text{O}_3$ Pellet Pressing and Sintering Data

Pellet Run Number	Applied Pressure, TSI	Pressed Density, g/cm ³	Percent of TD	Sintering Temp, °C	Time, hr	Final Density, g/cm ³	Final Percent of TD
1	63	5.13	69.0	1200	10	6.07	81.7
1-A	Sintering Rerun			1250	16	6.35	85.4
2	63	5.16	69.5	1200	10	6.10	82.0
2-A	Sintering Rerun			1250	10	6.21	83.6
3	63	5.25	70.6	1300	20	6.26	84.2
3-A	Sintering Rerun			1400	11	6.55	88.1
3-B	Second Rerun			1400	34	6.70	90.2
4	60	5.17	69.6	1400	11	6.20	83.4
4-A	Sintering Rerun			1400	34	6.60	88.9
5	63	5.24	70.6	1400	34	6.63	89.3
6	58	5.04	67.8	1400	34	6.51	87.6
7 (Avg)	63	5.50	74.0	1400	45	6.67	89.7
7-A	First Rerun			1400	20	----	
7-B	Second Rerun			1460	26	----	
7-C	Third Rerun			1495	24	6.75	90.9
8 (Avg)	63	5.78	77.8	1150	16	Very Slight Change	
8-A	Sintering Rerun			1250	16	6.13	82.5
9	50	5.30	68.5	1450	16	7.01	94.3
10 (Est)	63	----	70-75	1450	30	----	90-95
11 (Est)	63	----	70-75	1450	30	----	90-95
11-A	Sintering Rerun			1500	24	No Apparent Change	
12 (Est)	50	----	68-72	1450	30	----	90-94

NOTES: Pellet run no. represent singles unless noted as average, in which case variation was ~1%.

Runs 1, 2, and 8 were ~0.7 in. diam x 0.250 in. long

Runs 3 through 7 were ~1 in. diam x 0.350 in. long

Run 9 was ~1.5 in. diam x 0.350 in. long

Runs 10-12 were ~1.5 in. diam x 1.5 in. long with domed ends.

been presented at various times in this series of reports. To add to this collection of data pertinent to heat source components, a review has been made of physical and mechanical properties of this select group of materials. The data are presented in Tables 6 and 7. It may be used as an initial guide in comparing the material properties. This information is typical and available from standard sources. The fabrication experience obtained at PNL is summarized below. In a particular design application, additional evaluations and testing would be necessary.

Superalloys

The major emphasis in this area of high-temperature materials has been with Hastelloy-X (a nickel-base alloy) and Haynes-25 (a cobalt-base material). Hastelloy-X was used as a secondary cladding in the microthruster program,⁽²⁾ and Haynes-25 was selected as a high impact strength material for the AMSA applications.⁽¹⁾

Machining

Both alloys are machinable by using tungsten carbide tools, moderate cutting speeds, and light depth of cut. Although experience at PNL indicates that the cobalt alloy is more difficult to machine, literally dozens of the special internal and external radius cladding components for the AMSA test capsules have been successfully made. A special round carbide tool insert was used for cutting, with particular emphasis on tool sharpness and coolant flow to prevent work hardening. Final finishes were obtained by increasing the speed and decreasing the depth of cut. A tracer attachment was used on the lathe to produce the radii.

TABLE 6. *Properties of Superalloys*⁽³⁾

	Type	
	Haynes-25*	Hastelloy-X*
<u>Composition, %</u>	C: 0.05-0.15; Mn: 1.0-2.0; Cr: 19.0-21.0; Cr: 19.0-21.0; Ni: 9.0-11.0; W: 14.0-16.0; Fe: 3.0; Si: 1.0; Co: bal.	Co: 0.50-2.50 Mo: 8.00-10.00 Fe: 17.00-20.00 Cr: 20.50-23.00 W: 0.20-1.00 C: 0.50-0.15 Si: 1.00, Mn: 1.0, Ni: bal.
<u>Physical Properties</u>		
Density, lb/in. ³	0.33	0.297 lb/in. ³
Melting Temp, °F	2425 - 2570	2300 - 2470
Ther Cond (1300 °F), Btu/hr/ ft ² /°F/ft	13.1	12 (1100 °F)
Coef of Ther Exp (70- 1800 °F), per °F	9.4 x 10 ⁻⁶	9.0 x 10 ⁻⁶ (70-1600 °F)
Spec Ht, Btu/lb/°F	0.09 (80-212 °F)	0.10 (70-212 °F)
Elec Res, microhm-cm	188.7	118.3
<u>Mechanical Properties</u>		
Mod of Elast in Tension, psi		
Room Temp	34.2 x 10 ⁶	26.6 x 10 ⁶
1200 °F	27.4 x 10 ⁶	24.3 x 10 ⁶ (1000 °F)
1500 °F	26.3 x 10 ⁶ (1400 °F)	--
Ten Str, 1000 psi		
Room Temp	146	114
1200 °F	--	94 (1000 °F)
1500 °F	47 (1600 °F)	52
Yld Str (0.2% offset), 1000 psi		
Room Temp	67	52
1200 °F	--	41.5 (1000 °F)
1500 °F	35 (1600 °F)	37
Elong (in 2 in.), %		
Room Temp	64	43
1200 °F	--	45 (1000 °F)
1500 °F	30 (1600 °F)	33.5
Rupture Str (1500 °F), 1000 psi		
10 hr	30	22
100 hr	22	14.5
1000 hr	18	--

* Trade Mark: Union Carbide Corporation

TABLE 7. Properties of Refractory Metals and Alloys (3)

	Metal or Alloy					
	Tantalum	Tantalum-10 W	Molybdenum	(0.5 Ti; 0.1 Zr)	Tungsten	Rhenium
Physical Properties						
Density, lb/in. ³	0.60	0.608	0.37	0.37-	0.70	0.76
Melting Point, °F	5425	5516	4730	4730	6170	5760
Ther Cond (212 °F, Btu/hr/sq ft/°F/ft)	31.5	--	84.5	84.5	99.6	43.7
Coef of Ther Exp (70 °F), per °F	3.6 x 10 ⁻⁶	--	2.7 x 10 ⁻⁶	2.7 x 10 ⁻⁶	2.5 x 10 ⁻⁶	3.7 x 10 ⁻⁶
Specific Heat (70 °F), Btu/lb/°F	0.036	--	0.065	0.065	0.034	0.035
Elec Res (70 °F), mΩ-cm	12.5	20.0	5.2	5.2	5.48	19.0
Mechanical Properties						
Mod of Elast in Tension, psi						
Room Temp	27 x 10 ⁶	21 x 10 ⁶	47 x 10 ⁶	46 x 10 ⁶	59 x 10 ⁶	68 x 10 ⁶
1600 °F	26 x 10 ⁶	10 x 10 ⁶	33 x 10 ⁶	34 x 10 ⁶	55 x 10 ⁶	54.5 x 10 ⁶
2400 °F	--	9 x 10 ⁶	21 x 10 ⁶	20 x 10 ⁶	--	--
Tensile Stress, 1000 psi						
Room Temp	60	160	95	125	220	280
1600 °F	30	95	50	83	66	134
2400 °F	13	23	16	54	45	66
Yield Stress, 1000 psi						
Room Temp	48	158	82	105	220	270
1600 °F	15	90	30	53	15	--
2400 °F	--	20	10	40	--	--
Red of Area, %						
Room Temp	95+	85+	60	55	0	35
1600 °F	95+	--	85	74	--	--
2400 °F	95+	--	--	96	90	--
Hardness (VHN)						
Cold Worked	155	--	250	325	480	600
Stress Relieved	150	250	290	310	450	570
Recrystallized	80	216	190	200	350	450

Refractory Metals

The metals of particular interest in this group include molybdenum, tantalum, tungsten, rhenium, and their alloys. These metals are of primary interest as claddings because of compatibility and radiation shielding considerations. Exceptions would include secondary, structural cladding such as TZM (0.5 Ti, 0.1 Zr, Bal Mo) and Ta-10 W which have been considered for their impact resistance at high temperature. Rhenium was used as a primary cladding in the microthruster application; tantalum was used in the AMSA program.

Fabrication

The general term, fabrication, is used rather than machining to describe the wide range of techniques used in fabricating these refractory metals. There is little similarity between the metals in consideration of fabrication methods. At PNL all of the metals and many of the alloys have been fabricated. Reasonable results can be expected by using common machining techniques with tantalum, molybdenum and their alloys. However, tungsten and rhenium require grinding, abrasive cutting, and extremely difficult cutting with carbide tools. Tungsten and tungsten-rhenium alloys are more machinable than pure rhenium, which is usually ground and machined by electric discharge machining (EDM).

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TECHNOLOGY UTILIZATIONEVALUATION OF ISOTOPE PRODUCTION IN POWER REACTORS

D. E. Deonigi, R. W. McKee, E. T. Merrill, and S. E. Nunn

Target Irradiation Yields

Target irradiation yield data covering a wide range of irradiation conditions have been calculated for ^{237}Np , ^{241}Am , a 50/50 mixture of $^{241}\text{Am}/^{243}\text{Am}$, and ^{169}Tm target materials. The calculations were made by using the computer code BURN. Calculations were carried out for three different neutron fluxes and at six different target densities for each target material so that by cross plotting the data, the required yield data for virtually any intermediate condition can be defined. In addition, calculations were made by using two different values for the spectral index, r (the spectral index is the Westcott factor defining relative hardness of the spectrum). An r value of 0.2 is representative of current light-water power-reactor designs, and using an r value of 0.05 provides comparative data for a well-thermalized reactor. The data presented here are all based on an average thermal neutron temperature of 400 °C. Higher neutron temperatures would result in significantly higher yields of ^{238}Pu from ^{237}Np or ^{241}Am irradiations.

The data for ^{237}Np target irradiations are shown in Table 8. The 2.1 day ^{238}Np present at discharge would decay rapidly to ^{238}Pu and for practical purposes can be considered as ^{238}Pu . The irradiation times selected for each flux are adjusted to result in the same set of integrated fluxes. For example, 90 days irradiation at 5.0×10^{13} n/cm²/sec results in the same integrated flux as 180 days irradiation at 2.5×10^{13} n/cm²/sec.

When the ^{238}Np (Table 8) is added to the ^{238}Pu , there are only small differences between ^{238}Pu yields at 2.5×10^{13} and 1×10^{14} n/cm²/sec for the same integrated flux (or total exposure). The ^{238}Pu yields are slightly lower at the higher flux

TABLE 8. Composition of ^{237}Np Targets After Irradiation, at. %

Irradiation Time, Days	Target Density, g/cm ³	Well-Thermalized Reactor			Light Water Reactor		
		Spectral Index, r = 0.05			Spectral Index, r = 0.2		
		²³⁷ Np	²³⁸ Pu	²³⁸ Np	²³⁷ Np	²³⁸ Pu	²³⁸ Np
Flux = 2.5 x 10 ¹³ n/cm ² /sec							
180	1.6	92.6	6.6	0.1	91.1	7.9	0.1
	0.8	92.3	6.8	0.1	90.0	8.9	0.2
	0.4	92.0	7.1	0.1	88.7	10.0	0.2
	0.2	91.7	7.4	0.1	87.6	11.0	0.2
	0.02	91.1	7.9	0.1	85.4	13.0	0.2
	dilute	91.1	7.9	0.1	85.3	13.0	0.2
360	1.6	85.7	11.6	0.1	83.0	13.9	0.1
	0.8	85.1	12.1	0.1	80.9	15.6	0.1
	0.4	84.6	12.5	0.1	78.8	17.3	0.2
	0.2	84.0	13.0	0.1	76.8	19.0	0.2
	0.02	82.9	13.8	0.1	72.9	22.1	0.2
	dilute	82.9	13.8	0.1	72.8	22.1	0.2
720	1.6	73.4	18.0	0.1	68.8	21.4	0.1
	0.8	72.5	18.6	0.1	65.5	23.6	0.1
	0.4	71.5	19.2	0.1	62.0	25.9	0.1
	0.2	70.6	19.8	0.1	58.9	28.0	0.1
	0.02	68.8	21.0	0.1	53.1	31.7	0.1
	dilute	68.8	21.0	0.1	53.1	31.8	0.1
Flux = 5.0 x 10 ¹³ n/cm ² /sec							
90	1.6	92.6	6.4	0.2	91.1	7.7	0.3
	0.8	92.3	6.6	0.2	90.0	8.7	0.3
	0.4	92.0	6.9	0.2	88.7	9.7	0.3
	0.2	91.7	7.2	0.3	87.6	10.7	0.4
	0.02	91.1	7.7	0.3	85.4	12.6	0.4
	dilute	91.1	7.7	0.3	85.3	12.7	0.4
180	1.6	85.7	11.4	0.2	83.0	13.7	0.3
	0.8	85.1	11.9	0.2	80.9	15.3	0.3
	0.4	84.6	12.3	0.2	78.8	17.1	0.3
	0.2	84.0	12.7	0.2	76.8	18.7	0.3
	0.02	82.9	13.6	0.3	72.9	21.7	0.4
	dilute	82.9	13.6	0.3	72.8	21.8	0.4
360	1.6	73.4	17.8	0.2	68.8	21.2	0.2
	0.8	72.5	18.4	0.2	65.5	23.4	0.2
	0.4	71.5	19.0	0.2	62.0	25.6	0.2
	0.2	70.6	19.6	0.2	58.9	27.7	0.3
	0.02	68.8	20.8	0.2	53.1	31.4	0.3
	dilute	68.8	20.8	0.2	53.1	31.4	0.3
Flux = 1.0 x 10 ¹⁴ n/cm ² /sec							
45	1.6	92.6	6.1	0.5	91.1	7.3	0.5
	0.8	92.3	6.3	0.5	90.0	8.2	0.6
	0.4	92.0	6.6	0.5	88.7	9.2	0.7
	0.2	91.7	6.8	0.5	87.6	10.1	0.7
	0.02	91.1	7.3	0.5	85.4	12.0	0.9
	dilute	91.1	7.3	0.5	85.3	12.0	0.9
90	1.6	85.7	11.0	0.4	83.0	13.2	0.5
	0.8	85.1	11.4	0.4	80.9	14.8	0.5
	0.4	84.6	11.9	0.5	78.8	16.0	0.6
	0.2	84.0	12.3	0.5	76.8	18.0	0.7
	0.02	82.9	13.1	0.5	72.9	21.0	0.7
	dilute	82.9	13.1	0.5	72.8	21.0	0.7
180	1.6	73.4	17.3	0.4	68.8	20.6	0.4
	0.8	72.5	17.9	0.4	65.5	22.8	0.4
	0.4	71.5	18.5	0.4	62.0	25.0	0.5
	0.2	70.6	19.1	0.4	58.9	26.9	0.5
	0.02	68.8	20.2	0.4	53.1	30.6	0.5
	dilute	68.8	20.2	0.4	53.1	30.6	0.5

due to ^{238}Pu fissioning, and this effect would become more pronounced at higher fluxes. The importance of the resonance absorption is shown by the increased ^{238}Pu yields with more dilute targets and in the harder spectrum. As the target dilution is increased, more of the target material is exposed to the epithermal neutrons (less self-shielding); in the harder spectrum, there are simply more epithermal neutrons.

The data for ^{241}Am target irradiations are shown in Table 9. The ^{241}Am for this type of target irradiation could be recovered separately from ^{243}Am by processing a plutonium stockpile to recover the ^{241}Am growing in from ^{241}Pu decay. In addition, this data provide the information necessary to adjust the 50/50 $^{241}\text{Am}/^{243}\text{Am}$ irradiation to other starting mixtures.

The ^{242}Cm , shown in Table 9, decays with a 163 day half-life to ^{238}Pu . If this ^{242}Cm can be separated from the other components following irradiation and allowed to decay separately, it would furnish an exceptionally pure source of ^{238}Pu . Irradiation at higher fluxes increases the proportion (at a given exposure) of ^{238}Pu potentially available at high purity through ^{242}Cm separation and decay. Target dilution and a harder spectrum increase the ^{238}Pu yield, but not proportionally as much as in the case of ^{237}Np irradiation.

The results for target irradiation of the 50/50 mixture of $^{241}\text{Am}/^{243}\text{Am}$ are shown in Table 10. Early separation of ^{242}Cm for recovery of very high purity ^{238}Pu is a possibility here also. The ^{244}Cm yields are shown to be quite sensitive to target dilution or density and to spectral hardness, with significantly higher yields at the lower densities and in the harder neutron spectrum. For a fixed exposure, the ^{244}Cm yields are not very sensitive to the magnitude of the neutron flux.

TABLE 9. Composition of ^{241}Am Targets After Irradiation, at. %

Irradiation Time, Days	Target Density, g/cm ³	Well-Thermalized Reactor			Light Water Reactor		
		Spectral	Index, r = 0.05		Spectral	Index, r = 0.2	
		²⁴¹ Am	²⁴² Cm	²³⁸ Pu	²⁴¹ Am	²⁴² Cm	²³⁸ Pu
Flux = 2.5 x 10 ¹³ n/cm ² /sec							
180	1.6	68.2	16.7	7.3	68.5	16.6	7.3
	0.8	67.7	17.0	7.4	66.3	16.7	7.8
	0.4	67.0	17.4	7.6	63.6	19.1	8.5
	0.2	66.1	17.8	7.8	60.5	20.6	9.2
	0.02	63.8	19.0	8.39	52.3	24.7	11.37
	dilute	63.3	19.3	8.52	50.6	25.5	11.83
360	1.6	46.5	19.1	19.0	46.9	19.0	19.1
	0.8	45.8	19.4	19.3	43.9	19.9	20.3
	0.4	46.4	19.6	19.7	40.4	20.9	21.7
	0.2	43.8	20.0	20.1	36.6	22.0	23.3
	0.02	40.7	20.9	21.4	27.4	24.3	27.4
	dilute	40.0	21.1	21.7	25.6	24.7	28.2
720	1.6	21.7	13.0	33.7	22.0	13.0	34.3
	0.8	21.0	13.0	34.1	19.3	13.0	35.6
	0.4	20.1	13.0	34.5	16.3	12.9	37.1
	0.2	19.2	13.0	35.0	13.4	12.7	28.6
	0.02	16.5	12.9	36.3	7.5	11.8	41.9
	dilute	16.0	12.9	36.5	6.6	11.6	42.4
Flux = 5.0 x 10 ¹³ n/cm ² /sec							
90	1.6	68.3	20.1	4.1	68.5	20.0	4.1
	0.8	67.7	20.5	4.2	66.3	21.3	4.4
	0.4	67.0	20.9	4.3	63.6	23.0	4.8
	0.2	66.2	21.4	4.4	60.5	24.9	5.2
	0.02	63.8	22.9	4.7	52.3	30.0	6.4
	dilute	63.3	23.2	4.8	50.6	31.0	6.7
180	1.6	46.6	27.4	12.0	46.9	27.2	12.0
	0.8	45.8	27.7	12.2	43.9	28.6	12.8
	0.4	44.9	28.2	12.4	40.4	30.2	13.7
	0.2	43.8	28.7	12.7	36.6	31.9	14.7
	0.02	40.7	30.1	13.5	27.4	36.0	17.4
	dilute	40.0	30.4	13.7	25.6	36.7	18.0
360	1.6	21.7	25.3	25.7	22.0	25.2	26.1
	0.8	21.0	25.4	26.0	19.3	25.7	27.2
	0.4	20.2	25.5	26.3	16.3	26.1	26.7
	0.2	19.2	25.7	28.4	13.4	26.3	29.7
	0.02	16.6	26.0	27.8	7.5	26.3	32.8
	dilute	16.0	26.1	28.1	6.6	26.2	33.3
Flux = 1.0 x 10 ¹⁴ n/cm ² /sec							
45	1.6	68.3	22.2	2.2	68.5	22.0	2.2
	0.8	67.7	22.6	2.2	66.3	23.5	2.4
	0.4	67.0	23.1	2.3	63.6	25.4	2.6
	0.2	66.2	23.6	2.4	60.6	27.5	2.8
	0.02	63.8	25.3	2.5	52.3	33.2	3.4
	dilute	63.3	25.6	2.6	50.6	34.4	3.6
90	1.6	46.6	33.3	6.8	46.9	33.1	6.8
	0.8	45.9	33.8	6.9	43.9	34.9	7.3
	0.4	44.9	34.3	7.1	40.4	37.0	7.8
	0.2	43.8	35.0	7.2	36.7	39.2	8.4
	0.02	40.7	36.8	7.7	27.4	44.5	9.9
	dilute	40.1	37.2	7.8	25.6	45.5	10.3
180	1.6	21.7	37.9	16.6	22.0	37.8	16.8
	0.8	21.0	38.2	16.7	19.3	38.8	17.5
	0.4	20.2	38.5	17.0	16.3	39.8	18.4
	0.2	19.2	38.8	17.3	13.4	40.7	19.3
	0.02	16.6	39.7	18.0	7.5	42.1	21.5
	dilute	16.0	39.9	18.2	6.6	42.2	21.8

TABLE 10. Composition of 50/50 $^{241}\text{Am}/^{243}\text{Am}$ Targets
After Irradiation, at. %

Irradiation Time, Days	Target Density, g/cm ³	Well-Thermalized Reactor					Light Water Reactor				
		Spectral Index, r = 0.05					Spectral Index, r = 0.2				
		²⁴¹ Am	²⁴³ Am	²⁴² Cm	²⁴⁴ Cm	²³⁸ Pu	²⁴¹ Am	²⁴³ Am	²⁴² Cm	²⁴⁴ Cm	²³⁸ Pu
		Flux = 2.5 x 10 ¹³ n/cm ² /sec									
180	1.6	34.1	47.9	8.4	2.2	3.6	34.2	47.1	8.3	3.0	3.6
	0.8	33.8	47.7	8.5	2.4	3.7	33.1	46.5	8.9	3.5	3.9
	0.4	33.5	47.5	8.7	2.6	3.8	31.8	45.8	9.5	4.3	4.2
	0.2	33.1	47.3	8.9	2.8	3.9	30.3	44.8	10.3	5.1	4.6
	0.02	31.9	46.3	9.5	3.8	4.2	26.2	41.2	12.3	8.7	5.7
	dilute	31.6	46.0	9.6	4.1	4.3	25.3	40.1	12.8	9.8	5.9
360	1.6	23.3	45.9	9.6	4.3	9.5	23.4	44.3	9.5	5.7	9.5
	0.8	22.9	45.6	9.7	4.6	9.7	21.9	43.3	10.0	6.6	10.1
	0.4	22.4	45.3	9.8	4.9	9.9	20.2	42.0	10.5	7.9	10.9
	0.2	21.9	44.8	10.0	5.4	10.1	18.3	40.3	11.0	9.3	11.6
	0.02	20.3	42.9	10.4	7.2	10.7	13.7	34.1	12.2	15.1	13.7
	dilute	20.0	42.4	10.5	7.7	10.8	12.8	32.6	12.4	16.6	14.1
720	1.6	10.8	42.1	6.5	8.0	16.9	11.0	39.4	6.5	10.2	17.2
	0.8	10.5	41.6	6.5	8.5	17.0	9.6	37.6	6.5	11.6	17.8
	0.4	10.1	40.9	6.5	9.1	17.2	8.2	35.3	6.5	13.4	18.6
	0.2	9.6	40.1	6.5	9.8	17.5	6.7	32.7	6.4	15.5	19.3
	0.02	8.3	36.9	6.5	12.8	18.1	3.7	23.4	5.9	23.0	21.0
	dilute	8.0	36.1	6.5	13.7	18.3	3.3	21.8	5.8	24.4	21.2
		Flux = 5.0 x 10 ¹³ n/cm ² /sec									
90	1.6	34.1	47.9	10.1	2.2	2.1	34.2	47.1	10.0	3.0	2.1
	0.8	33.9	47.7	10.2	2.4	2.1	33.1	46.5	10.7	3.6	2.2
	0.4	33.5	47.5	10.5	2.6	2.1	31.8	45.8	11.5	4.3	2.4
	0.2	33.1	47.3	10.7	2.8	2.2	30.3	44.8	12.5	5.2	2.6
	0.02	31.9	46.3	11.5	3.8	2.4	26.2	41.2	15.0	8.7	3.2
	dilute	31.6	46.0	11.6	4.1	2.4	25.3	40.1	15.5	9.8	3.4
180	1.6	23.3	45.9	13.7	4.4	6.0	23.4	44.4	13.6	5.7	6.0
	0.8	22.9	45.6	13.9	4.6	6.1	22.0	43.3	14.3	6.7	6.4
	0.4	22.4	45.3	14.1	5.0	6.2	20.2	42.0	15.1	7.9	6.9
	0.2	21.9	44.8	14.3	5.4	6.4	18.3	40.3	16.0	9.4	7.4
	0.02	20.4	42.9	15.1	7.2	6.8	13.7	34.1	18.0	15.2	8.7
	dilute	20.0	42.4	15.2	7.8	6.9	12.8	32.6	18.3	16.8	9.0
360	1.6	10.9	42.1	12.6	8.2	12.8	11.0	39.4	12.6	10.4	13.0
	0.8	10.5	41.6	12.7	8.6	13.0	9.6	37.6	12.8	11.9	13.6
	0.4	10.8	40.9	12.8	9.2	13.2	8.2	35.3	13.0	13.7	14.2
	0.2	9.6	40.1	12.8	10.0	13.4	6.7	32.7	13.2	15.8	14.9
	0.02	8.3	36.9	13.0	13.0	13.9	3.7	23.4	13.2	23.5	16.4
	dilute	8.0	36.1	13.1	13.9	14.0	3.3	21.8	13.1	24.9	16.6
		Flux = 1.0 x 10 ¹⁴ n/cm ² /sec									
45	1.6	34.1	47.9	11.1	2.2	1.1	34.2	47.1	11.0	3.0	1.1
	0.8	33.9	47.7	11.3	2.4	1.1	33.1	46.5	11.8	3.6	1.2
	0.4	33.5	47.5	11.5	2.6	1.1	31.8	45.8	12.7	4.3	1.3
	0.2	33.1	47.3	11.8	2.8	1.2	30.3	44.8	13.8	5.2	1.4
	0.02	31.9	46.3	12.6	3.8	1.3	26.2	41.2	16.6	8.7	1.7
	dilute	31.6	46.0	12.8	4.2	1.3	25.3	40.1	17.2	9.8	1.8
90	1.6	23.3	45.9	16.7	4.4	3.4	23.5	44.4	16.6	5.8	3.4
	0.8	22.9	45.6	16.9	4.6	3.5	22.0	43.3	17.4	6.7	3.6
	0.4	22.5	45.3	17.2	5.0	3.5	20.2	42.0	18.5	8.0	3.9
	0.2	21.9	44.8	17.5	5.4	3.6	18.3	40.3	19.6	9.5	4.2
	0.02	20.4	42.9	18.4	7.3	3.8	13.7	34.1	22.3	15.3	5.0
	dilute	20.0	42.4	18.6	7.9	3.9	12.8	32.6	22.8	16.9	5.1
180	1.6	10.9	42.1	19.0	8.3	8.3	11.0	39.4	18.9	10.5	8.4
	0.8	10.5	41.6	19.1	8.7	8.4	9.7	37.6	19.4	12.0	8.8
	0.4	10.1	40.9	19.2	9.3	8.5	8.2	35.3	19.9	13.8	9.2
	0.2	9.6	40.1	19.4	10.1	8.6	6.7	32.7	20.3	15.9	9.7
	0.02	8.3	36.9	19.9	13.2	9.0	3.8	23.4	21.1	23.7	10.7
	dilute	8.0	36.1	20.0	14.1	9.1	3.3	21.8	21.1	25.1	10.9

The ^{169}Tm target irradiation data are shown in Table 11. Because of the relatively short 127 day half-life for ^{170}Tm , the ^{170}Tm yields are very sensitive to neutron flux. Target dilution also improves the yield as does spectral hardness.

^{238}Pu Cost Studies

Irradiation of ^{237}Np targets has been generally assumed to be the source of future ^{238}Pu requirements. However, ^{238}Pu can also be produced by target irradiation of ^{241}Am , produced by decay of ^{241}Pu , and recovered by processing plutonium stockpiles.

The production costs for ^{238}Pu from ^{241}Am target irradiation have been calculated with the TAIRCOST program for the two types of reactors as described in the discussion of target irradiation yields. The first is an LWR, having a spectral index $r = 0.2$ and a flux of $2.5 \times 10^{13} \text{ n/cm}^2/\text{sec}$. The second is a well-thermalized reactor, having a very soft spectrum with an $r = 0.05$ and a flux of $1.0 \times 10^{14} \text{ n/cm}^2/\text{sec}$.

The yield data for irradiation of ^{241}Am under these conditions were previously discussed, and the cost calculations were based on an assumed target density of 0.8 gm/cm^3 . This information was then fed into TAIRCOST to obtain the optimum irradiation time and cost for ^{238}Pu production over a wide range of input cost assumptions. The cost assumptions for high- and low-cost base cases are shown in Table 12. In addition to these base cases, a number of parameters were varied from the base cases to determine their effects on the costs. The results of these calculations are summarized in Table 13 for the LWR, and in Table 14 for the well-thermalized reactor.

The inventory charge, reprocessing time, and cost of ^{241}Am were all changed separately to determine their effect. The cost of the ^{241}Am target is found to have the largest effect of any of the variables in the ranges considered. In comparing

TABLE 11. *Composition of ^{169}Tm Targets After Irradiation, at.%*

Irradiation Time, Days	Target Density, g/cm ³	Well-Thermalized Reactor Spectral Index, r = 0.05			Typical Power Reactor Spectral Index, r = 0.2		
		¹⁶⁹ Tm	¹⁷⁰ Tm	¹⁷¹ Tm	¹⁶⁹ Tm	¹⁷⁰ Tm	¹⁷¹ Tm
		Flux = 2.5 x 10 ¹³ n/cm ² /sec					
180	1.1	94.7	3.2	0.1	93.6	3.9	0.1
	0.56	94.5	3.4	0.1	92.7	4.5	0.1
	0.28	94.2	3.6	0.1	91.6	5.2	0.2
	0.14	93.8	3.8	0.1	90.0	6.1	0.2
	0.01	90.7	5.7	0.2	78.8	12.8	0.4
	dilute	90.6	5.8	0.2	78.2	13.2	0.4
360	1.1	89.7	4.2	0.3	87.6	5.1	0.4
	0.56	89.3	4.4	0.3	86.0	5.7	0.4
	0.28	88.8	4.6	0.3	83.9	6.5	0.5
	0.14	88.0	4.9	0.3	81.1	7.6	0.5
	0.01	82.3	7.1	0.5	62.0	14.6	1.1
	dilute	82.0	7.2	0.5	61.1	14.9	1.2
720	1.1	80.5	4.3	0.6	76.7	5.0	0.7
	0.56	79.8	4.4	0.7	73.9	5.6	0.8
	0.28	78.8	4.6	0.7	70.3	6.3	1.0
	0.14	77.5	4.9	0.7	65.7	7.1	1.1
	0.01	67.8	6.7	1.0	38.5	10.8	2.0
	dilute	67.3	6.8	1.1	37.3	10.9	2.1
Flux = 5.0 x 10 ¹³ n/cm ² /sec							
90	1.1	94.7	4.0	0.1	93.6	4.9	0.2
	0.56	94.5	4.2	0.1	92.7	5.6	0.2
	0.28	94.2	4.4	0.1	91.6	6.4	0.2
	0.14	93.8	4.7	0.1	90.0	7.6	0.2
	0.01	90.7	7.1	0.2	78.8	16.1	0.5
	dilute	90.6	7.2	0.2	78.2	16.6	0.5
180	1.1	89.7	6.1	0.4	87.6	7.4	0.5
	0.56	89.3	6.4	0.4	86.0	8.3	0.6
	0.28	88.8	6.7	0.4	83.9	9.6	0.6
	0.14	88.0	7.1	0.5	81.1	11.2	0.8
	0.01	82.3	10.5	0.7	62.0	21.9	1.5
	dilute	82.0	10.6	0.7	61.1	22.4	1.6
360	1.1	80.5	7.5	1.0	76.7	8.9	1.3
	0.56	79.8	7.8	1.1	73.9	9.9	1.5
	0.28	78.8	8.1	1.2	70.3	11.2	1.7
	0.14	77.5	8.6	1.3	65.7	12.7	2.0
	0.01	67.8	12.0	1.8	38.5	20.8	3.7
	dilute	67.3	12.2	1.9	37.3	21.1	3.8
Flux = 1.0 x 10 ¹⁴ n/cm ² /sec							
45	1.1	94.7	4.5	0.1	93.6	5.5	0.2
	0.56	94.5	4.7	0.1	92.7	6.3	0.2
	0.28	94.2	5.0	0.2	91.6	7.2	0.2
	0.14	93.8	5.3	0.2	90.0	8.5	0.3
	0.01	90.7	8.0	0.2	78.8	18.2	0.6
	dilute	90.6	8.1	0.2	78.2	18.7	0.6
90	1.1	89.7	7.6	0.5	87.6	9.2	0.6
	0.56	89.3	7.9	0.5	86.0	10.4	0.7
	0.28	88.8	8.3	0.5	83.9	12.0	0.8
	0.14	88.0	8.9	0.6	81.1	14.0	0.9
	0.01	82.3	13.1	0.8	62.0	27.7	1.9
	dilute	82.0	13.3	0.9	61.1	28.3	1.9
180	1.1	80.5	11.0	1.5	76.7	13.1	1.8
	0.56	79.8	11.4	1.6	73.9	14.5	2.0
	0.28	78.8	11.9	1.6	70.3	16.5	2.3
	0.14	77.5	12.6	1.8	65.7	18.9	2.7
	0.01	67.8	17.8	2.5	38.5	32.1	5.2
	dilute	67.3	18.1	2.6	37.3	32.6	5.3
360	1.1	64.8	12.0	3.8	58.8	13.8	4.5
	0.56	63.6	12.4	3.9	54.6	15.0	5.0
	0.28	62.1	12.9	4.1	49.4	16.4	5.6
	0.14	60.0	13.5	4.3	43.2	17.9	6.3

TABLE 12. Base Cost Sets

	<u>Low Base Costs</u>	<u>High Base Costs</u>
Inventory Charge, %/yr	10	10
Encapsulation and Reprocessing Losses, %	2	2
Encapsulation Cost, \$/kg	700	4,000
Reprocessing Cost, \$/kg	2,000	15,000
Irradiation Costs		
Handling, \$/cm ³	0.1	0.3
Neutrons, \$/g	2,700	2,700
Revenue Loss, \$/cm ³ -yr	0.0	3.50
Reprocessing Time, months	24	24

TABLE 13. Variation in Price of ²³⁸Pu Produced
in a Power Reactor

(r = 0.2, flux = 2.5 x 10¹³)

²⁴¹ Am Value, \$/g	Base Cost Set	Cost Component Changed from Base Set	New Value	Optimum Irradiation Time, months	Total Cost of ²³⁸ Pu, \$/g
100	Low			12.0	256
		Inventory Charge	5%	11.6	218
		Inventory Charge	15%	12.0	295
		Reprocessing Time	12 mo.	10.9	230
	High			12.2	311
		Inventory Charge	5%	12.2	272
		Inventory Charge	15%	12.2	350
		Reprocessing Time	12 mo.	12.2	285
200	Low			12.1	484
		Inventory Charge	5%	11.6	407
		Inventory Charge	15%	12.0	561
		Reprocessing Time	12 mo.	10.9	432
	High			12.2	539
		Inventory Charge	5%	11.6	462
		Inventory Charge	15%	12.2	616
		Reprocessing Time	12 mo.	11.7	488

TABLE 14. Variation in Price of ^{238}Pu , Produced
in a Thermal Reactor

($r = 0.05$, flux = 1×10^{14})

^{241}Am Value, \$/g	Base Cost Set	Cost Component Changed from Base Set	New Value	Optimum Irradiation Time, Months	Total Cost of ^{238}Pu , \$/g
100	Low			4.4	218
		Inventory Charge	5%	4.2	194
		Inventory Charge	15%	4.5	243
		Reprocessing Time	12 mo.	4.1	198
	High			4.6	256
		Inventory Charge	5%	4.5	232
		Inventory Charge	15%	4.6	280
		Reprocessing Time	12 mo.	4.5	236
200	Low			4.4	411
		Inventory Charge	5%	4.1	363
		Inventory Charge	15%	4.5	459
		Reprocessing Time	12 mo.	4.1	370
	High			4.5	449
		Inventory Charge	5%	4.4	400
		Inventory Charge	15%	4.5	497
		Reprocessing Time	12 mo.	4.3	408

the low base case with 100 and \$200/g ^{241}Am , the cost is increased from 256 to \$484. The inventory charge causes a significant change in cost, but not nearly as great as the cost of target. In reducing this inventory charge rate from 10 to 5%, cost is reduced from 256 to \$218/g for the \$100/g ^{241}Am low-base case. The reprocessing time affects the cost even less; a reduction of this time from 24 to 12 months reduces the cost from 256 to \$230.

In making a comparison of the two types of reactors as a means of producing ^{238}Pu , the advantage of the high flux reactor is not nearly as great as one might expect from the flux difference. In addition, the optimum irradiation time is found to be approximately one year in each LWR case, thus allowing this type of reactor to discharge ^{241}Am with its normal refueling cycle. For the \$200/g ^{241}Am low-base cost set, the cost of ^{238}Pu is reduced from \$484 for the LWR to \$411 for the high-flux thermal reactor. Even this cost advantage may be offset by the necessity to operate in the high-flux reactor with a lower target density to maintain high flux, which, in turn, would increase fabrication costs proportionally.

Further cost studies are planned using the TAIRCOST code. These will include a comparison of ^{238}Pu costs produced from ^{237}Np , with ^{238}Pu produced from ^{241}Am , an examination of the interrelated effects of density, flux depression and fabrication costs on target irradiation costs, and an evaluation of mixed $^{241}\text{Am}/^{243}\text{Am}$ target irradiation. The later two-component cases will require some extensive changes in the TAIRCOST program.

CIRCULATORY SUPPORT SYSTEMSRADIATION CHARACTERISTICS

F. T. Cross and J. C. Sheppard

Promethium Systems

The final report on the "In-Phantom Dosimetry of $^{147}\text{Pm}_2\text{O}_3$ Heat Sources in Relation to Circulatory Support Systems" was completed. This terminates the studies of shielding, dose prediction and dosimetry in promethium systems.

Other RadioisotopesShielding and In-Phantom Dose Prediction

The gamma interaction coefficients in QAD have been updated by using the values by E. Storm and H. I. Israel in LA-3753. QAD has been further modified to accommodate a larger number of boundaries, zones, and source points for calculating dose-rates from prototypic and actual circulatory support systems.

Neutron transport calculations have been made on LiH, LiF-KF, and LiCl-LiF eutectics with the ASISN code. Neutron removal cross-sections are being determined for comparison with those in the QAD library.

Neutron and gamma dose-rates in tissue from ^{252}Cf have been determined with the QAD code. Gamma abundances for ^{252}Cf were taken from Stoddard in DP-986. The neutron emission rate was measured with a Hanford standard long counter. The computations were for a ^{252}Cf needle having length of 2.0 cm and diameter of 0.06 cm, which is the size of the SRL Californium needle. The computed values for total rad dose were 20 to 30% higher than the experimentally determined dose by Fairchild in BNL-12452. The initial results are encouraging; the disparities that appear might be because the heavy ion recoils have been underestimated in the measurements, the fluence to dose conversion factors in QAD may be high, and/or the gamma abundances are in error.

In-Phantom Dosimetry Measurements

The calibration of the dosimeters designed for the plutonium measurements was nearly completed. Photon spectral measurements of LASL-1 source were initiated to determine the impurity level of ^{236}Pu .

Phantom dose-rate measurements were initiated with ^{252}Cf ; these, however, have not yet been compared with the QAD predicted values. In addition, long term neutron dose-rate measurements were started with the ^{252}Cf source implanted in the Remab phantom. Except for scaling factors, these measurements are expected to predict the neutron dose-rates from ^{238}Pu .

PROMETHIUM ISOTOPIC CHARACTERIZATION (Promethium-146 Burn-out)

J. C. Sheppard

All the ^{147}Pm targets irradiated in the Hanford K Reactor, to burn out the approximately 0.3 ppm of ^{146}Pm , have been radiochemically analyzed and have yielded a set of $^{146}\text{Pm}/^{147}\text{Pm}$ ratios used to calculate the ^{146}Pm cross-section.

The targets used in these experiments consisted of two parts: an inner, cylindrical target and an annular one which surrounded the inner one. Cross-sections of 3700 and 3300 b were calculated from the $^{146}\text{Pm}/^{147}\text{Pm}$ ratios for the inner and annular targets, respectively. These differences can be explained in terms of the hardening of the neutron flux by the annular promethium and resonance capture of epithermal neutrons by ^{146}Pm .

A 8400 b cross-section for ^{146}Pm was found by Pinajian.⁽¹⁾ The differences in cross-sections can be accounted for by flux distribution differences between the Oak Ridge Research Reactor and the Hanford K Reactor. It is thought that the former reactor is less well moderated than the latter and results in more resonance capture by ^{146}Pm and a larger observed cross-section.

A report on the Hanford promethium irradiations is in preparation.

The Savannah River promethium irradiations have been completed, and the targets now in transit should arrive at PNL in mid-May. Upon arrival, these targets will be radiochemically analyzed to determine their $^{146}\text{Pm}/^{147}\text{Pm}$ ratios.

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