MAST

CONF. 786943--1

**PREPARATION AND CHARACTERIZATION OF RADIOACTIVE SAMPLES
FOR VARIOUS AREAS OF RESEARCH***

H. L. Adair

Solid State Division

Oak Ridge National Laboratory
Post Office Box X
Oak Ridge, Tennessee 37830

NOTICE

**PORTIONS OF THIS REPORT ARE ILLEGIBLE. IT
CONTAINS INFORMATION FROM THE BEST AVAILABLE
COPY TO PERMIT THE BROADEST POSSIBLE AVAIL-
ABILITY.**

By acceptance of this article, the
publisher or recipient acknowledges
the U.S. Government's right to
retain a nonexclusive, royalty-free
license in and to any copyright
covering the article.

DISTRICT OF COLUMBIA: 1978
UNLIMITED

*Research sponsored by the Division of Nuclear Sciences, U. S.
Department of Energy, under contract W-7405-eng-26 with the
Union Carbide Corporation.

PREPARATION AND CHARACTERIZATION OF RADIOACTIVE SAMPLES FOR VARIOUS AREAS OF RESEARCH*

H. L. Adair

INTRODUCTION

The demand for radioactive targets for many areas of research has increased steadily since the formation of the Isotope Research Materials Laboratory (IRML) in 1959. As the demand has increased, the need for better definition of the target materials has also increased. In particular, the need for better target definition has been generated by physicists who want to make more accurate cross-section measurements, metallurgists who want to make material property measurements, or by reactor engineers who want to use these materials as neutron monitors to map the neutron energy spectra, flux, and fluence at various locations in a reactor.

The hazards involved in processing the radioactive materials dictate the use of well-designed, contained facilities such as glove boxes or remotely operated hot cells. Such facilities are being used by IRML to prepare radioactive samples of ^3H , ^{232}Th , ^{233}U , ^{234}U , ^{235}U , ^{236}U , ^{238}U , ^{237}Np , ^{239}Pu , ^{240}Pu , ^{241}Pu , ^{242}Pu , ^{244}Pu , ^{241}Am , ^{243}Am , ^{244}Cm , and ^{252}Cf . The number of individual samples prepared has increased from <50 in 1959 to >2000 in 1978. The samples are available on an international basis subject to U. S. Department of Energy approval. The costs for the various samples are available upon request.

The techniques used to prepare and characterize some of the various radioactive samples will be described in this paper.

*Research sponsored by the Division of Nuclear Sciences, U. S. Department of Energy, under contract W-7405-eng-26 with the Union Carbide Corporation.

ACTINIDE DEPOSITS FOR CROSS-SECTION MEASUREMENTS

In order to answer many questions concerning fast flux reactor technology, more detailed neutron cross-section information is needed for actinide materials that will either be used as fuels or will be generated in such facilities. Accurate cross-section data is needed to answer such questions as how much fuel will be consumed as well as the quantities of highly radioactive materials that will be produced in the process.

Techniques used by the Isotope Research Materials Laboratory (IRML) to prepare actinide deposits used for cross-section measurements have been described at previous meetings of this society and thus will not be discussed in depth.^{1,2} Briefly, the inventoried chemical form of the actinide isotope (oxide) is pressed into a pellet containing approximately 1 g to 5 g and placed in an electron beam gun (Fig. 1). As shown in Fig. 2, the electron beam gun is mounted inside the vacuum chamber located in a stainless steel glove box. A substrate holder and masking assembly containing one or more targets can be placed at the desired distance from the evaporation source depending on the area of the target(s) and required uniformity. When only milligram quantities of material are available, a cylindrical tantalum crucible and radiofrequency induction heating is used rather than electron bombardment to vaporize the oxide material (Fig. 3). The amount of material deposited on the substrate is determined initially either by direct weighing (after evaporation is complete) or by use of a quartz crystal monitor (in situ). However, the final determination of isotope content is made by low geometry alpha counting. Typical weight ranges of actinide materials prepared by IRML are shown in Table 1.

ACTINIDE METAL PREPARATION

The preparation of high purity metals by the reduction-distillation technique is described in the literature and is the subject of a review paper at this meeting.^{3,4} However, since most of the work to be described

Table 1. Thickness Range of Actinide Deposits Prepared by IRML

Element	Compound Form	Substrate *	Weight Range (mg/cm ²)
²³² Th	ThO ₂	Most metals	1 x 10 ⁻³ - 10
²³⁵ U	UO ₂	Most metals	1 x 10 ⁻³ - 10
²³⁸ U	UO ₂	Most metals	1 x 10 ⁻³ - 10
²³³ U	UO ₂	Most metals	1 x 10 ⁻³ - 1
²³⁴ U	UO ₂	Most metals	1 x 10 ⁻³ - 1
²³⁶ U	UO ₂	Most metals	1 x 10 ⁻³ - 1
²³⁷ Np	NpO ₂	Most metals	1 x 10 ⁻³ - 10
²³⁹ Pu	PuO ₂	Most metals	1 x 10 ⁻³ - 10
²⁴⁰ Pu	PuO ₂	Most metals	1 x 10 ⁻³ - 1
²⁴¹ Pu	PuO ₂	Most metals	1 x 10 ⁻³ - 1
²⁴² Pu	PuO ₂	Most metals	1 x 10 ⁻³ - 1
²⁴⁴ Pu	PuO ₂	Most metals	1 x 10 ⁻³ - 1
²⁴¹ Am	AmO ₂	Most metals	1 x 10 ⁻⁴ - 1
²⁴³ Am	AmO ₂	Most metals	1 x 10 ⁻⁴ - 1
²⁴⁴ Cm	CmO ₂	Most metals	1 x 10 ⁻⁵ - 10
²⁵² Cf	Oxychloride	Most metals	1 x 10 ⁻⁶ - 1 x 10 ⁻³

*Thin deposits (<100 μg/cm²) can be prepared on thin carbon or thin nickel backings. Heavy deposits (≥1 mg/cm²) are normally prepared on titanium. Most substrates range in thickness from 0.0025 to 0.25 mm.

concerns stable materials, it is appropriate to briefly emphasize the types and purities of actinide metals prepared by IRML.⁴ Multigram quantities of americium, plutonium, and curium metals have been prepared by reducing the respective oxides with thorium metal. A typical spark source mass spectrographic analysis of ^{241}Am metal prepared by this technique is shown in Table 2.

Table 2. Spark Source Mass Spectrographic Analysis of High Purity ^{241}Am Metal Prepared by Reduction-Distillation

Element	Impurities (wt. ppm)	Element	Impurities (wt. ppm)
Al	1	Mn	0.3
B	1	Na	0.5
Bi	0.5	Ni	0.2
Ca	0.2	Pb	10
Cr	1	Si	10
Cu	3	Ta	≤5
Fe	5	Te	≤1
In	≤0.1	Zn	≤0.1
K	0.1	S	2
Mg	≤0.5	Tm	≤3

NEUTRON DOSIMETRY MATERIALS

The demand for high purity, well-characterized materials that can be used as neutron monitors in reactors has increased significantly over the past few years. This increased requirement can be attributed to the need to characterize neutron energy spectra, flux, and fluence in existing or proposed reactor cores and blanket regions and to determine accurate neutron energy and flux information in the exact volume region of an in-core reactor experiment.

IRML established a reactor neutron dosimeter program in 1971. To cover the neutron energy range of interest, many stable and radioactive materials are provided. Each material must be accurately characterized and must contain the minimum possible impurity level. In addition, the neutron monitor materials should meet several important criteria:^{5,6}

1. Materials should have peak activation characteristics at different neutron energies.
2. Dosimeter and encapsulating materials should have high melting points, normally $>1000^{\circ}\text{C}$.
3. Materials should produce product radioisotopes whose half-lives and radiation characteristics are accurately known and are amenable to direct counting through the capsule wall without loss of accuracy.

Each dosimeter material is encapsulated in high purity vanadium (99.9+%) which contains ≤ 10 ppm tantalum. The low tantalum content, as well as minimal content of trace impurities, is required to avoid neutron reactions resulting in extraneous nuclei which emit radiations which might interfere with the counting of specific gamma radiations needed to determine nuclear events of interest. Using this high purity vanadium, dosimeters of ^6Li , ^{10}B , ^{45}Sc , ^{232}Th , ^{235}U , ^{238}U , ^{237}Np , and ^{239}Pu are being prepared. Normally, the elemental form is used for the stable dosimeter materials while the radioactive isotopes are in oxide form. In addition, dilutions of the above materials in the form of alloys or co-precipitated oxides are provided for use in high fluences where the resultant activity level would be too high for direct counting.

All the dosimeter materials are encapsulated in either 0.89 mm outer diameter by 0.53 mm inner diameter, or in 1.27 mm outer diameter by 0.79 mm inner diameter vanadium capsules. Capsules vary in length from 3.05 mm to 8.64 mm. The amounts of dosimeter material loaded into each capsule can

vary from 1 mg to 10 mg, depending on the particular isotope. The specific isotope contained in the capsule is identified by an impressed dot code system.⁷ Some typical 1.27 mm diam capsules are shown in Fig. 4. Typical amounts of material loaded into the various vanadium capsules are given in Table 3.

Table 3. Quantities of Dosimeter Materials Loaded Into Various Vanadium Capsules

Material	Capsule Length (mm)	Capsule Diameter (mm)	Hole Depth (mm)	Hole Diameter (mm)	Weight Range Per Capsule (mg)
$^{235}\text{UO}_2$	3.05	1.27	2.29	0.79	1.5 ± 0.5
$^{235}\text{UO}_2$	4.83	0.89	3.81	0.53	1.5 ± 0.5
$^{238}\text{UO}_2$	7.87	1.27	7.11	0.79	10.0 ± 2
$^{238}\text{UO}_2$	7.87	0.89	6.86	0.53	7.0 ± 2
$^{237}\text{NpO}_2$	8.64	1.27	7.87	0.79	7.0 ± 2
$^{237}\text{NpO}_2$	7.11	0.89	6.10	0.53	5.0 ± 2
$^{239}\text{PuO}_2$	7.11	1.27	6.35	0.79	1.5 ± 0.5
$^{239}\text{PuO}_2$	6.35	0.89	5.33	0.53	1.5 ± 0.5
Sc_2O_3	6.35	1.27	5.59	0.79	2.0 ± 0.5
Sc_2O_3	8.64	0.89	7.62	0.53	2.0 ± 0.5

Initially, special fixtures were prepared for loading the vanadium capsules with oxide powders of the various dosimeter materials. Significant problems were encountered in this loading process including highly contaminated vanadium capsules and excessive errors in weight by difference since the weight of contained material was small as compared to the vanadium capsule weight. These problems were eliminated by the development and preparation of ceramic oxide "wire" containing the fissile and nonradioactive materials in compacted, handleable form suitable for cutting and direct

weighing.⁸ This development also reduced the personnel time required for loading the dosimeter capsules and has increased the accuracy of material content (since the ceramic material can be weighed directly).

Ceramic "wires" are prepared by mixing high purity oxide with a paraffin binder. The mixture is heated to approximately 60°C to melt the paraffin and the mixture is again thoroughly mixed; the resulting paste-like mixture is then extruded through a small diameter die (0.050 cm to 0.075 cm) into "green" wire. In this "green" state, the extruded material is very flexible; upon firing at temperatures in excess of 1000°C, the paraffin binder is vaporized or burned away leaving a solid, sintered oxide wire of uniform cross section.

For long-term reactor experiments where high fluences are involved, dilutions of the neutron-reactive nuclides are required because the amount of activation products or fission products would be excessive for accurate analysis by direct radiation counting. Ceramic dilutions have been prepared by making wires using magnesium oxide as the diluent. In some instances, metal alloy wires have been prepared using vanadium as the diluent.

Dilute ceramic oxide wires are prepared by mixing a small quantity (normally ≤ 1 wt. %) of the specific isotope (as oxide) with a matrix material of magnesium oxide. To assure maximum homogeneity, all oxides are dissolved in nitric acid; subsequent addition of urea (as solid) destroys the nitrate content and eventual coprecipitation occurs from molten urea at 180°C. Thus the nuclides exist in an atomic mixture of perfect homogeneity. The resultant mixture is calcined in air at 900°C for complete dehydration and removal of urea by thermal decomposition. After calcination the mixed oxides are added to molten wax containing dissolved polyethylene and the resultant mixture is extruded into green wire; subsequent sintering in magnesium oxide tubes at 1450°C for eight hours removes the binder and forms solid, uniform wires. The described procedure has been used to prepare 610 meters of 0.1 wt. % CoO-MgO, 610 meters of 0.1 wt. % Sc₂O₃-MgO, 98 meters of 0.7 wt. % NpO₂-MgO, and 110 meters of 1.0 wt. % ThO₂ in MgO.

Vanadium metal alloy wires containing 0.1-1.0 wt. % of several metallic nuclides of interest to be used as neutron dosimeters have been prepared by arc melting, casting, swaging, and/or drawing. Homogeneity of these alloys is assured by multiple melting and stirring of the carefully weighed component metals. Thus far 305 meters of 0.5 mm diam, 0.1 wt. % Co-V, and 0.1 wt. % Ta-V as well as 225 meters of 0.5 mm diam, 0.8 wt. % ^{238}U -V and 0.8 wt. % of ^{235}U -V have been prepared by this method.

CHARACTERIZATION OF VANADIUM-ENCAPSULATED DOSIMETERS

To insure reproducibility among dosimeter sets, quality assurance procedures were formulated for all phases of their manufacture. Preparative control and analytical characterization procedures were developed for qualifying vanadium capsule material, capsule fabrication, dosimeter materials, dosimeter wire fabrication, and the inspection of completed dosimeters.

Purity of both the vanadium capsule material and the contained dosimeter material is very important to avoid errors in determining reactor neutron spectral and fluence characteristics. Use of high purity materials minimizes interference from undesirable activation or fission products in the ultimate gamma analysis of the neutron irradiated dosimeters. Many analyses are required to characterize each dosimeter material. Initially, isotopic, spark source mass spectrographic (SSMS) and coulometric (where applicable) analyses were made on the oxides and elemental materials reserved for dosimeter use. Results of these analyses for radioactive dosimeter materials are shown in Tables 4 through 6. After IRML developed ceramic oxide "wire", similar analyses were performed on those materials. Examples of coulometric analyses on pure oxide radioactive materials are shown in Table 7. Activation analysis is used primarily in determining the elemental analysis of the dilute materials.

Table 4. Isotopic Analyses of High Purity Radioactive Dosimeter Materials

Isotope	Weight (%)
Batch ^{235}U -264C	
233	<0.0005
234	0.034
235	99.89
236	0.025
238	0.053
Batch ^{238}U -ES-Z	
233	<0.0001
234	<0.0001
235	0.0012
236	<0.0001
238	99.999
Batch ^{239}Pu -453-B-0	
238	<0.002
239	99.10
240	0.884
241	0.010
242	0.005
244	<0.005
Batch ^{237}Np -24-HP	
235	<0.0005
236	<0.0005
237	≥ 99.99
238	≤ 0.003
239	≤ 0.004

Table 5. SSMS Impurity Analyses of High Purity Dosimeter Materials (Wt. ppm)

	²³⁵ U 264C	²³⁸ U ES-2	²³⁹ Pu 453-7-0	²³⁷ Np Np-24-HP
Al	<1	5	15	2
As	<1	1	1	<1
B	<0.5	2	0.2	<0.5
Ba	<1	<1	<1	<1
Ca	2	15	8	2
Co	<1	<1	<1	<1
Cr	<1	1	<1	<1
Cu	5	2	6	2
Fe	8	25	10	4
K	<1	<1	10	1
Mg	<2	2	1	<1
Mn	<1	<1	<1	<1
Na	<1	<1	1	<1
P	1	10	<1	<1
Pb	1	<1	<1	
Si	<3	3	10	3
Sn		5	1	2
S	4	10	1000	4
Ti	10	10	<1	<1
²³⁸ U	100	M ^a	5	10
V	100	100	<1	<1
W	<1	<1		
Zn	<1	<1	2	<1
Zr	1	<1	<1	<1
²³⁵ U	M	4	1	<1
²³⁷ Np	<1	<1	1	M
²³⁹ Pu	<1	≤ 10		<1
²⁴⁴ Cm	<1	<1	<1	<1

^aM - Major.

Table 6. Coulometric Analyses of Dosimeter Materials. Analyses show the quantity of the major element per gram of material.

Element	Form	Batch Number	Elemental Analysis (mg/g)
^{235}U	U_3O_8	264C	846.8
^{238}U	U_3O_8	ES-Z	845.4
^{239}Pu	PuO_2	453-B-O	895.2
^{237}Np	NpO_2	Np-24-HP	874.5

Table 7. Coulometric Analyses of Oxide "Wire" Dosimeter Materials. Analyses show the quantity of the major element per gram of material.

Element	Form	Batch Number	Elemental Analysis (mg/g)
^{235}U	UO_2	264C	878.7
			888.6
			869.9
			876.2
			882.6
			882.3
^{238}U	UO_2	ES-Z	877.4
			878.9
			876.1
^{239}Pu	PuO_2	453-B-O	863.6
			868.5
			860.6
^{237}Np	NpO_2	Np-24-HP	874.0

The amount of isotope material contained in each dosimeter is primarily determined by weight. Using a substitution weighing scheme⁹, which compares the dosimeter material weight to a standard weight, accurate weight determinations of $\leq 0.5\%$ at the 95% CL are obtained for encapsulated dosimeter materials ≥ 1 mg. The quantity of each radioactive species contained in a single dosimeter capsule is verified by gamma counting using a NaI(Tl) detector. Although initially the dosimeter program was in support of the Fast Breeder Reactor program, dosimeter materials that have been developed can also be used to characterize Light Water Reactor (LWR) and Controlled Thermonuclear Reactor (CTR) environments.

TRITIUM TARGETS USED AS NEUTRON SOURCES

For many years the Isotope Research Materials Laboratory (IRML) has been preparing tritium targets by reacting various vapor-deposited metal layers with tritium in a high temperature environment. Detailed procedures for these preparations have been described previously.¹ Initially most of the tritium targets prepared by IRML were small in size (area), but as the need for more intense monoenergetic neutron sources increased, the requested targets also increased in size. Many of the more recent tritium target preparations have been for the rotating target neutron source (RTNS-I) at Lawrence Livermore Laboratory.¹⁰ The RTNS-I target design resulted from a development effort to increase the neutron yield and target lifetime in the deuteron beam beyond those of conventional systems.¹¹ One of the major changes incorporated in the use of the RTNS-I target was rotation of the target at high speed, which would reduce the dwell time of the deuteron beam at any one location on the target, thereby reducing the desorption of tritium caused by beam heating. A neutron yield of 6×10^{12} n/s can be produced from the 23 cm diam RTNS-I target using a 400 KeV, 20-mA D^+ beam. Present technology has been extrapolated in the design of a new deuteron accelerator (RTNS-II) which will be used to generate even higher yields for evaluation of component materials exposed to thermonuclear environments. Design goals for the RTNS-II targets are shown in Table 8 along with

present operational parameters of RTNS-I targets. Systems that will be used to fabricate the RTNS-II targets are shown in Figs. 5-7. The base layer of titanium will be deposited in a vac-ion system (Fig. 5) capable of maintaining a vacuum level of $\leq 10^{-7}$ torr during the evaporation. The system is pumped with a 200 l/sec ion pump in conjunction with a 200 l/sec titanium sublimator cryopump. A rod fed electron beam gun (Fig. 6) will be used to vaporize the titanium that is subsequently deposited on the 50 cm diam dish type copper alloy target that is located approximately 36 cm from the source. The target substrate will be heated to 450°C prior to and during the titanium deposition to provide an adherent titanium bond and will be rotated to provide the desired titanium uniformity.

Table 8. Operational and Designed Parameters for High Intensity Rotating Neutron Sources¹¹

Parameter	RTNS-I	RTNS-II
Beam energy (KeV)	400	400
HV supply current (mA)	60	300
Target current (mA)	22	150
Spot size (mm)	6	10
Target diameter (cm)	23	46
Target speed (rpm)	1100	5000
Source strength (n/s)	6×10^{12}	4×10^{13}
Maximum flux (n/cm ² · sec)	1.7×10^{12}	1.2×10^{13}
Target lifetime (h)	100	100

Each target will be removed from the deposition system in an argon atmosphere and placed inside the tritium sorption system. An exterior view of the tritium sorption chamber, tritium storage traps, and appropriate electronic components are shown in Fig. 7. The tritium sorption system is pumped with two 270 l/sec ion pumps and will maintain a vacuum level of $\leq 10^{-8}$ torr during the heating cycle before tritium sorption takes place.

During titanium vapor deposition film thickness will be determined with a resonating quartz crystal monitor and a resistance monitor. Change in resistance of thin films has been used satisfactorily to determine film thicknesses of titanium up to 1.0 mg/cm^2 .¹² In addition, it is hoped that the electrical resistance can be used during tritium sorption into the titanium layer to monitor this process as well. It has been shown that the conductance of the titanium-hydrogen layer passes through a minimum value and then steadily increases until saturation is achieved.¹² The uniformity of the tritium content over the active area will be measured using a small ion chamber detector.¹

MATERIAL COMPATIBILITY STUDIES

Ceramic "wire" techniques developed for the dosimeter program are directly applicable to the preparation of actinide oxide samples for compatibility studies. A typical example is the preparation of $^{244}\text{Cm}_2\text{O}_3$ bars which will be used to irradiate tensile strength specimens of tantalum, molybdenum, platinum, and iridium alloys. The Cm_2O_3 powder was mixed with a high temperature wax binder (melting point 205°C) and extruded through a 1.615 mm by 3.505 mm orifice. Experimentally it was determined that a maximum of 2 g increments of Cm_2O_3 could be used; larger amounts of oxide caused volatilization and decomposition of the binder, due to the decay heat from the ^{244}Cm (2.832 w/g), before extrusion could be accomplished.

Prior to each extrusion, a 2 g sample of oxide was heated to 800°C in argon to assure reduction to Cm_2O_3 . After cooling in argon, the material was ground, weighed, and mixed with 0.300 g of wax. After extrusion, the $^{244}\text{Cm}_2\text{O}_3$ bars were cut to 1.43 cm lengths for sintering. The bars were placed between thin aluminum oxide plates to prevent warping during sintering. Aging the "green" bars 24 hr before the sintering cycle was begun apparently allowed the wax to polymerize and prevented it from melting during the early stages of the sintering cycle, which produced a

better dimensionally defined product. The bars were sintered in a tube furnace at 1450°C in a glove box for 8 hr. During 2 hr of the 8-hr cycle, the glove box atmosphere was changed from argon to air to burn off any residual carbonaceous material. A total of 22 Cm_2O_3 bars were extruded and sintered according to the above procedure; one such bar is shown in Fig. 8.

SUMMARY

Over the past few years, many techniques have been applied by IRML to prepare radioisotope targets and related research materials from actinide and other radioisotopes. Many of these samples were prepared by ceramic technology--an R and D program extending over the past three years. Standard vapor deposition methods have been used to prepare hundreds of fission chamber and other planar samples used in determining cross sections. Refined methods of characterization of all samples have been applied to assure definition of contents to as low as $\pm 0.5\%$ in some cases. Tritium-containing target preparation in support of high yield neutron generators (10^{12} – 10^{13} n/sec) has been a continuing development effort both in manufacturing technology and in measurement of tritium content. Of all, however, the largest effort has been development and characterization of neutron dosimetry materials and samples useful for measurement of energy and fluence in reactor core and reflector regions. These dosimeter samples are available for monitoring LWR, CTR, and fast reactor systems.

REFERENCES

1. H. L. Adair, "Preparation and Characterization of $^{241}_{95}\text{Am}$ and ^3H Targets," presented at the Sixth Annual Conference, International Nuclear Target Development Society, Lawrence Berkeley Laboratory, Oct. 19-21 (1977), Berkeley, California.
2. H. L. Adair, J. R. Gibson, E. H. Kobisk, and J. M. Dailey, "Vapor Deposition of Large Area NpO_2 and UO_2 Deposits," presented at the Fifth Annual Conference, International Nuclear Target Development Society, Los Alamos Scientific Laboratory, Oct. 19-21 (1976), Los Alamos, New Mexico.
3. H. L. Adair and E. H. Kobisk, "Preparation and Characterization of Actinide Targets," proceedings of the 1974 Annual Conference of the Nuclear Target Development Society, Chalk River Nuclear Laboratories, Oct. 1-3 (1974), Chalk River, Ontario, Canada.
4. E. H. Kobisk and H. L. Adair, "Conversion of Isotope Compounds to Metals by Reduction-Distillation Methods," to be presented at the World Conference, International Nuclear Target Development Society, Universitat and Technische Universitat, Munchen, Sept. 11-14 (1978), Munchen, West Germany.
5. BNWL-SA-2887, "Neutron Dosimetry for Fast Reactor Applications," p. 3, Battelle Northwest Laboratories (1970).
6. J. L. Jackson et al., "EBR-II Dosimetry Test, Reactor Runs 50G and 50H," HEDL-TME 73-62, Hanford Engineering Development Laboratory.
7. CONF-711002, "Research Materials for Nuclear Measurements," p. 86, U. S. Atomic Energy Commission (1971).
8. U. S. Patent No. 3,971,944.
9. "Design and Test of Standards of Mass," Precision Measurement and Calibration, Optics, Meteorology, and Radiation Handbook, Vol. III, p. 77, U. S. Dept. of Commerce (1961).
10. R. Booth et al., Nuclear Instruments and Methods 145 (1977), pp. 25-39.
11. R. Booth, IEEE Trans. Nucl. Sci. NS-14 (1967), p. 943.
12. J. L. Provo, "Film Thickness Resistance Monitor for Dynamic Control of Vacuum Deposited Films," Ninth Annual New Mexico American Vacuum Society Symposium, April 9-11 (1973), Albuquerque, New Mexico.

FIGURE CAPTIONS

Figure No.

- 1 Electron Bombardment Source Used for Preparation of Adherent Actinide Deposits
- 2 Electron Bombardment Source Centrally Mounted in Stainless Steel Vacuum Chamber
- 3 Schematic Drawing of Vacuum Evaporation System Used for Evaporation of Small Quantities of Actinide Materials
- 4 Typical Vanadium Capsules and Materials Used for Accurate Determination of Neutron Energy Spectra, Flux, and Fluence
- 5 Exterior View of Vac-Ion Pumping System Used for Titanium Evaporations
- 6 Interior View of Titanium Evaporation System Showing the Electron Beam Gun, Quartz Iodide Heat Lamps, and Substrate Area
- 7 Exterior View of Tritium Sorption System Showing Sorption Chamber, Tritium Storage Traps, and Associated Electronic Components
- 8 Extruded $^{244}\text{Cm}_2\text{O}_3$ Bar Before Cutting and Mounting With Ta - 10 wt. % W Tensile Specimens

270° 10KW EVAPORATION SOURCE

Fig. 3.

Schematic drawing of vacuum evaporation system used for depositing actinide materials.

U235	Pu238	Pu241	Pu239	Np237	U238	Pu240
0.120	0.150	0.180	0.210	0.280	0.310	0.340

Pu Double Capsule

LABORATORY

11-11-11

11-11-11

11-11-11