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WANL-TNR-077

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WESTINGHOUSE
ASTRONUCLEAR LABORATORY
NEUTRON AND GAMMA DOSIMETRY
FOR IRRADIATION TESTING

NOVEMBER, 1962

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ABSTRACT

Current literature describing neutron and gamma dosimetry techniques for irradiation testing were reviewed. Tabulations of neutron activation detector characteristics and gamma dosimeters are given. The tabulations are intended to be usable for planning irradiation testing by listing the range, form, availability and limitations of the dosimeters. A recommended form, including units and terminology for reporting dosimetry measurements, is presented. A suggested dosimetric system is given. For additional information on any specific dosimeter, an extensive bibliography is included.

NEUTRON AND GAMMA DOSIMETRY FOR IRRADIATION TESTING

The properties of materials used for nuclear applications which are exposed to intense neutron and gamma radiation have been affected; for example:

1. The engineering properties of structural materials have been changed e.g. embrittlement.
2. The useful lifetime of semi-conductors has been altered by atomic displacements.
3. The glass envelopes of vacuum tubes have been damaged.
4. The characteristics of resistive materials have been altered.
5. The dielectric material in capacitors have been degraded.
6. Organic materials have been damaged.

To determine the adequacy of the design of components to be used for nuclear applications, irradiation tests are carried out to evaluate radiation effects. A thorough knowledge of radiation characteristics during an irradiation test is required to obtain meaningful results. The radiation characteristics to be determined are: type of radiation including intensity and energy spectrum of each type present. This is a report on the many dosimetry systems available for these measurements. A correlation of the best currently available data on neutron activation detector characteristics is included, and the information is presented in a form intended to be usable for planning irradiation testing dosimetry.

Irradiation tests are carried out by placing the test component in proximity to an intense source of neutrons and gamma rays such as an operating reactor. For the purpose of evaluating the radiation effects, each type of radiation must be identified as to intensity and energy spectrum. The degree of discrimination for the dosimetry system is dictated by practicality and availability; that is, how fine a discrimination is required for a specific case and what dosimetric material and evaluation equipment is available.

Irradiation testing is usually conducted in a mixed radiation field. The techniques and materials for determining the characteristics of neutron and gamma ray doses differ. Therefore, this report will consider the dosimetry for neutrons separate from gamma techniques.

NEUTRONS

Neutrons are not readily detected by direct means, rather they are detected from changes induced in materials by neutron absorption. A neutron absorbed by an atom will result in a new isotope which may emit a particle or ionizing radiation that may be measured. In some heavier atoms, the absorption of a neutron may cause the atom to split (fission) into two or more fission product isotopes; these product isotopes may produce a measureable particle or ionizing radiation. Neutron detection for irradiation dosimetry usually is accomplished by use of neutron sensitized ionization chambers or activation of target detectors.

NEUTRON IONIZATION CHAMBERS

The Bragg-Gray theory is the basis for the design of ion chambers used for routine dosimetry. The basic concepts for the design are discussed in the literature (1,2). Many basic ion chambers have been designed to measure gamma and X-radiation fields. A basic ion chamber may be modified to make it more sensitive to neutrons than gamma rays. The modification includes: filling the cavity with a gas with high neutron absorption cross-section, or by coating the walls and/or the central electrode with a high neutron absorption cross-section material or a fissionable material. The gas normally used is BF_3 ; the high absorption cross-section material is usually boron enriched with B^{10} ; the fissionable material may be uranium. Ionization resulting from the reaction or fission products produces a current which is interpreted as a neutron flux intensity. The energy of the flux may be interpreted from a knowledge of the nuclear characteristics of the absorbing material. Since any ionizing radiation may produce a current, these chambers must be compensated to nullify the gamma contribution when used in a mixed radiation field.

NEUTRON ACTIVATION DETECTORS

The neutron flux may be measured from the interaction of neutrons with target isotopes to produce (n, γ) , (n, p) , (n, α) , $(n, \text{fission})$, or $(n, 2n)$ reactions. This method involves the absolute counting of activation induced in detector materials. The flux in several energy increments is determined by using several materials with different threshold energies or different energy ranges for neutron capture. This technique provides excellent discrimination against gamma radiation below 6 Mev. The activation technique has disadvantages in that:

1. Information is available only after a significant time delay following irradiation;
2. Equipment required for evaluation is complex;
3. The integrated flux rather than an instantaneous flux is measured.

In spite of the disadvantages cited, the activation technique is the only method available at this time by which flux intensity and energy spectrum of a high neutron flux may be measured.

The flux is determined from the specific activity of the detector from the relationship:

$$\phi = \frac{A_0}{N\sigma (1-e^{-\lambda T}) (e^{-\lambda t})} \quad (1)$$

where: ϕ = flux ($\text{n cm}^{-2} \text{ sec}^{-1}$)
 A_0 = specific foil activity ($\text{dis. sec.}^{-1} \text{ gm.}^{-1}$)
 N = number of target isotopic atoms present (atoms gm.^{-1})
 σ = activation cross-section of the detector (barns)
 E = activation counter efficiency accounting for: window absorption, self absorption and back scattering, etc.
 λ = decay constant for isotope being formed (sec^{-1})
 T = time of irradiation (sec.)
 t = decay time (sec)

The specific detector activity (A_0) is determined as the zero time intercept of a semi-log plot of measured disintegration of the activated detector plotted as a function of time after irradiation.

The neutron energies for irradiation test considerations are rather arbitrarily divided into the following groups:

1. Thermal - 0.025 ev up to 0.1 ev
2. Resonance 0.1 ev up to 1000 ev
3. Fast - 0.1 mev up to 15 to 20 mev

The detector choice must consider the following basic criteria:

1. The energy of the neutrons to be measured;
2. The half life of the reaction product;
3. The physical and chemical properties of the detector;
4. The likelihood of interfering reactions;
5. The accuracy and completeness of knowledge of the detector nuclear characteristics.
6. The availability of the detector.

The expected energy of the neutrons to be measured will dictate the detector reaction expected. The degree of energy discrimination required will determine the number and types of different detector materials to be used.

The half life of the reaction product which will be counted is a factor in determining the time over which a detector analysis can be made. The reaction product half life becomes significant if a long time delay is expected between the end of irradiation and the availability of the detectors for evaluation.

The physical and chemical properties of the detector are significant with regard to irradiation conditions. The melting point, chemical activity, thermal stability or unusual cryogenic behavior is important for testing under extremes of temperature or when the detector will be in actual contact with the irradiation specimen.

The likelihood of extraneous reaction products whose activity might interfere with the analysis of the desired reaction is significant in the choice of a detector. The interferences are eliminated by:

1. Radiochemical separation prior to counting;
2. Choosing a detector pure in the desired target isotope;
3. Choosing suitable counting procedures or equipment.

The accuracy and completeness of knowledge of nuclear characteristics will influence the choice of detector. The limiting factors for the accuracy of flux measurements in many cases are: the activation cross-section at irradiation test conditions, the effective threshold energy and the effective threshold cross-section. Values for detector nuclear characteristics under extremes of temperature, e.g. cryogenic, may be significantly different from those currently available. Additional experimental work is required before meaningful measurements of flux under temperature extremes may be made.

Table I is the correlation of the characteristics of most of the promising detectors which might be used for neutron flux measurement by the activation technique. The values listed were obtained from the literature (3-16). The detectors are tabulated alphabetically by element name. Where values differed among the literature, the latest (literature date) value is listed. Uncertain values are prefixed by (?); characteristics for which values were not available are left blank. The preferred

TABLE I
NEUTRON ACTIVATION DETECTOR CHARACTERISTICS

Element	Atomic Number	Isotope	Isotopic Abundance (%)	Isotopic Activation Cross Section (barns)	Reaction	Product	Half-Life	Useful Energy Range	Main Resonance Energy (eV) or Threshold (MeV)	Principal Beta MeV (%)	Principal Gamma MeV (%)	Useful Temp. Max. (°C)	Detector Form	Potential Interferences	Nuclei Per Unit Volume (x10 ²⁴ /cm ³)	Potential Suppliers												
																See Table II												
Aluminum	13	Al ²⁷	100	0.116	(n, α)	Na ²⁴	14.97h	T	F	T-8.6	1.394	1.368	659	Foil	Hg ²⁰⁴ (n, p); secondary	0.0602	X	X	X	X	X	X	X	X	X	X	X	X
	13	Al ²⁷	100	0.21 ± 0.02	(n, p)	Al ²⁸	2.27m	R	F	R-5800	2.87(100)	1.78	659	Foil	Si ²⁸ (n, p); P ³¹ (n, α)	0.0602	X	X	X	X	X	X	X	X	X	X	X	X
	13	Al ²⁷	100	0.079	(n, p)	Mg ²⁷	9.45m	R	F	T-6.3	1.58(2)	0.834	659	Foil	Hg ²⁰⁴ (n, p); Si ³⁰ (n, α)	0.0602	X	X	X	X	X	X	X	X	X	X	X	X
Antimony	51	Sb ¹²¹	57.25	0.75	(n, 2n)	Sb ¹²⁰	5.8d ± 16.4m	F	F	T-9.25	(a) 1.70 (b) 1.36(51)	1.18	630	Oxide	Fe ¹²⁰ (n, p); secondary	0.0331	X	X	X	X	X	X	X	X	X	X	X	X
Arsenic	33	As ⁷⁵	100	0.545	(n, 2n)	As ⁷⁴	17.5d	F	F	T-10.3	(+) 0.92(89) (a) 1.61(98)	0.596(40) 0.94(74)	1465	Oxide	Se ⁷⁴ (n, p); secondary	0.0461	X	X	X	X	X	X	X	X	X	X	X	X
Cadmium	48	Cd ¹¹⁴	28.86	(a) 0.14 ± 0.03 (b) 1.13 ± 0.3	(n, p)	Cd ¹¹⁵	(a) 43d (b) 53h	R	F	R-120.2	0.968	no γ	320	Foil	In or Sn; matrix	0.0464	X	X	X	X	X	X	X	X	X	X	X	X
Carbon	6	C ¹²	98.89	(?) 0.005 ± 0.001	(n, 2n)	C ¹¹	20.4m	R	F	T-20.2	0.968	no γ	3705	Graphite	none for E < 25 Mev	0.0803	X	X	X	X	X	X	X	X	X	X	X	X
Chlorine	17	Cl ³⁵	75.4	0.19	(n, α)	F ³²	14.22d	F	F	T-1.8	1.707	no γ		Salt	P ³¹ (n, p); S ³² (n, p)	5.3 x 10 ⁵	X	X	X	X	X	X	X	X	X	X	X	X
	17	Cl ³⁷	24.6	0.56 ± 0.12	(n, p)	Cl ³⁸	37.29m	R	R	R-1.80 Mev	4.81(53)	2.15		Salt	S ³⁸ (n, p); S ³⁴ (n, α)	5.3 x 10 ⁵	X	X	X	X	X	X	X	X	X	X	X	X
Chromium	24	Cr ⁵⁰	4.31	16	(n, p)	Cr ⁵¹	27.8d	F	F	T-13.4	1.54(50)	0.063	1888	Oxide	Fe ⁵⁴ (n, α)	0.0822	X	X	X	X	X	X	X	X	X	X	X	X
	24	Cr ⁵⁰	4.31		(n, 2n)	Cr ⁴⁹	41.9m	F	F	T-13.4	1.54(50)	0.063	1888	Oxide	none for E < 20 Mev	0.0822	X	X	X	X	X	X	X	X	X	X	X	X
Cobalt	27	Co ⁶³	100	20 ± 3	(n, p)	Co ⁶⁴	5.24y	R	R	R-123	0.312	1.17, 1.33	1450	Foil	Ni ⁶⁴ (n, p); Cu ⁶³ (n, α)	0.0910	X	X	X	X	X	X	X	X	X	X	X	X
Copper	29	Cu ⁶³	69.1	4.51 ± 0.23	(n, p)	Cu ⁶⁴	12.80h	T	R	R-580	0.571	1.34(weak)	1000	Foil	Zn ⁶⁴ (n, p); secondary	0.0848	X	X	X	X	X	X	X	X	X	X	X	X
	29	Cu ⁶³	69.1	~ 0.9	(n, 2n)	Cu ⁶²	9.73m	F	F	T-11.7	2.91	0.69	1000	Foil	none for E < 20 Mev	0.0848	X	X	X	X	X	X	X	X	X	X	X	X
Erbium	66	Er ¹⁶⁴	28.18	2100 ± 300	(n, p)	Dy ¹⁶⁵	139.2m	T	F	R-14.6	1.25	0.0948	1400	Oxide	Ho ¹⁶⁵ (n, p); Er ¹⁶⁸ (n, α)	0.0317	X	X	X	X	X	X	X	X	X	X	X	X
Europium	63	Eu ¹⁵¹	47.77	1400 ± 300	(n, p)	Sm ¹⁵²	9.2h	R	R	R-0.461 ±	1.88	0.344	2355	Oxide	La ¹⁵² (n, p); secondary	0.0207	X	X	X	X	X	X	X	X	X	X	X	X
Fluorine	9	F ¹⁹	100	0.061	(n, 2n)	F ¹⁸	112m	F	F	T-10.4	0.649	no γ		LiF	none for E < 25 Mev	5.3 x 10 ⁵	X	X	X	X	X	X	X	X	X	X	X	X
Gold	79	Au ¹⁹⁷	100	96 ± 10	(n, p)	Au ¹⁹⁸	2.697d	T	R	R-4.906	0.960	0.4118	1000	Foil	Hg ¹⁹⁸ (n, p); secondary	0.0591	X	X	X	X	X	X	X	X	X	X	X	X
Iodine	53	I ¹²⁷	100	1.3	(n, 2n)	I ¹²⁶	13.3d	F	F	T-9.45	0.87 (29)	0.382	113	NH ₄ I	As ¹²⁶ (n, p); secondary	0.0234	X	X	X	X	X	X	X	X	X	X	X	X
Indium	49	In ¹¹⁵	95.77	155 ± 15	(n, p)	In ^{116m}	53.99m	T	R	R-1.457	100 (51)	1.274 (75)	155	Foil	Sn ¹¹⁶ (n, p)	0.0382	X	X	X	X	X	X	X	X	X	X	X	X
	49	In ¹¹⁵	95.77	*0.230	(n, n')	In ^{115m}	4.50h	T	F	T-0.85	0.83	0.335	155	Foil		0.0382	X	X	X	X	X	X	X	X	X	X	X	X
Iron	26	Fe ⁵⁶	91.68	0.44	(n, p)	Fe ⁵⁶	45.1d	F	F	T-2.9	0.271 (46) 0.462 (54)	1.098 (57)	1539	Foil	Co ⁵⁶ (n, p); Ni ⁶² (n, α)	0.0848	X	X	X	X	X	X	X	X	X	X	X	X
	26	Fe ⁵⁶	91.68		(n, p)	Fe ⁵⁶	2.576h	F	F	T-2.9	2.81 (50)	0.845	1539	Foil	Co ⁵⁹ (n, α); Ni ⁵⁵ (n, p)	0.0848	X	X	X	X	X	X	X	X	X	X	X	X
Lutetium	71	Lu ¹⁷⁶	2.60	4000 ± 800	(n, p)	Lu ¹⁷⁷	6.75d	T	R	R-0.142	0.497 (90)	0.1130	2000	Oxide	Tb ¹⁷⁷ (n, p); Bi ²⁰⁹ (n, p)	0.0335	X	X	X	X	X	X	X	X	X	X	X	X
Magnesium	12	Mg ²⁴	78.60	*0.19	(n, p)	Na ²⁴	14.97h	F	F	T-4.9	1.394	1.368	650	Foil	Na ²³ (n, p); Al ²⁷ (n, α)	0.0431	X	X	X	X	X	X	X	X	X	X	X	X
Manganese	25	Mn ⁵⁵	100	1.33 ± 0.2	(n, p)	Mn ⁵⁶	2.576h	T	R	R-337	2.81 (50)	0.85	951	Foil	Fe ⁵⁶ (n, p); Co ⁵⁹ (n, α)	0.0789	X	X	X	X	X	X	X	X	X	X	X	X
Molybdenum	42	Mo ⁹²	23.75	0.45 ± 0.10	(n, p)	Mo ⁹¹	66.0h	F	F	T-12.7	3.44	no γ	2627	Foil	Ru ¹⁰² (n, α); fission of U or Pu	0.0640	X	X	X	X	X	X	X	X	X	X	X	X
Neptunium	93	Np ²³⁷	100	0.13 0.019 ± 0.003	(n, 2n) (n, f)	Mo ⁹¹ P.P. ¹⁴⁰	15.5m 12.8d	F	F	T-12.7	3.44	no γ	2627	Foil	none	0.0640	X	X	X	X	X	X	X	X	X	X	X	X
						Te ¹³²	77.7h	F	F	T-0.75	1.02 (60)	0.0296			fissionable materials		X	X	X	X	X	X	X	X	X	X	X	X
Nickel	28	Ni ⁶⁴	1.16	1.6 ± 0.2	(n, p)	Ni ⁶⁵	2.564h	F	F	T-5.0	0.22	0.231	1461	Foil	Cu ⁶³ (n, p); Zn ⁶⁶ (n, α)	0.0913	X	X	X	X	X	X	X	X	X	X	X	X
	28	Ni ⁵⁸	68.0	1.23	(n, p)	Co ⁵⁸	72d	F	F	T-5.0	0.48	0.81, 1.64	1461	Foil		0.0913	X	X	X	X	X	X	X	X	X	X	X	X
Oxygen	8	O ¹⁶	99.99		(n, 2n)	O ¹⁵	124s	F	F	T-16.5	1.723	no γ	low	Cellophane		5.3 x 10 ⁵	X	X	X	X	X	X	X	X	X	X	X	X
Phosphorus	15	P ³¹	100	0.19 ± 0.01	(n, p)	P ³²	14.22d	F	F	R-15800	1.707	no γ	>1000	P with	S ³² (n, p); Cl ³⁵ (n, α)	0.0354	X	X	X	X	X	X	X	X	X	X	X	X
	15	P ³¹	100	0.142	(n, p)	Si ³¹	2.62h	F	F	T(?) ~ 0.7	1.471	1.26	60	P with	Si ³⁰ (n, p); S ³⁴ (n, α)	0.0354	X	X	X	X	X	X	X	X	X	X	X	X
Platinum	78	Pt ¹⁹⁸	7.2	3.9 ± 0.8	(n, p)	Pt ¹⁹⁹	31m	R	R	R-95	0.8	0.074	1770	Foil	none listed	0.0660	X	X	X	X	X	X	X	X	X	X	X	X
Plutonium	94	Pu ²³⁹		σ _f 1.8	(n, f)	EP. (See Neptunium)		R	R	R-10000	(See Neptunium)		350	Foil	fissionable materials	0.0498	X	X	X	X	X	X	X	X	X	X	X	X
										Shielded with B ¹⁰																		
Potassium	19	K ⁴¹	6.91	1.1 ± 0.1	(n, p)	K ⁴²	12.52h	R	R	R-5400	3.55 (82)	1.53	775	Pellet	Ca ⁴² (n, p); Sc ⁴⁵ (n, α)	0.0134	X	X	X	X	X	X	X	X	X	X	X	X
Praseodymium	59	Pr ¹⁴¹	100	2.06	(n, 2n)	Pr ¹⁴⁰	3.4m	F	F	T-9.4	2.23	no γ	935 ± 5	Oxide	none for E < 20 Mev	0.0290	X	X	X	X	X	X	X	X	X	X	X	X
Protactinium	91	Pa ²³¹			(n, f)	FF. (See Neptunium)		F	F	T-0.3	(See Neptunium)				fissionable materials	0.0402	X	X	X	X	X	X	X	X	X	X	X	X

TABLE I (cont'd)

NEUTRON ACTIVATION DETECTOR CHARACTERISTICS

Element	Atomic Number	Isotope	Isotopic Abundance (%)	Isotopic Activation Cross Section (barns)	Reaction	Product	Half-Life	Useful Energy Range	Main Resonance Energy (eV) or Threshold (MeV)	Principal Beta MeV (±)	Principal Gamma MeV (±)	Useful Temp. Max. (°C)	Detector Form	Potential Interferences	Nuclei Per Unit Volume ($\times 10^{24}/\text{cm}^3$)	Potential Suppliers See Table II												
																1	2	3	4	5	6	7	8	9	10	11	12	13
Samarium	62	Sm^{152}	26.63	140 ± 40	(n, γ)	Sm^{153}	47.1h	R	R-8.01	0.69 (70)	0.10318	2350	Oxide	$\text{Eu}^{153}(\text{n}, \gamma); \text{Gd}^{156}(\text{n}, \gamma)$	0.0309													
Silicon	14	Si^{28}	92.27	0.37	(n, p)	Al^{28}	2.27m	F	T-4.4	2.87 (100)	1.78	1430	Dioxide	$\text{Al}^{28}(\text{n}, \gamma); \text{Fe}^{51}(\text{n}, \gamma)$	0.0522													
Silver	47	Ag^{107}	51.35	0.54	(n, 2n)	Ag^{106}	24.0m	F	T-9.5	1.95 (89)	0.512	961	Foil	$\text{Cd}^{106}(\text{n}, \gamma); \text{secondary}$	0.0586													
Sodium	11	Na^{23}	100	0.53 ± 0.02	(n, γ)	Na^{24}	14.97h	T	R-2850	1.394	1.368	800	Pellet	$\text{Mg}^{24}(\text{n}, \gamma); \text{Al}^{27}(\text{n}, \gamma)$	0.0254													
Sulfur	16	S^{34}	4.215	0.138	(n, α)	S^{32}	2.62h	F	T-0.9	1.471	1.26	120-1580	Pellet	$\text{S}^{30}(\text{n}, \gamma); \text{Fe}^{51}(\text{n}, \gamma)$	0.0389													
Tantalum	73	Ta^{181}	99.918	0.065	(n, p)	Ta^{182}	14.22d	F	T-2.9	1.707	no γ	120-1580	Pellet	$\text{Ta}^{182}(\text{n}, \gamma); \text{Fe}^{51}(\text{n}, \gamma)$	0.0389													
Thorium	90	Th^{232}	100	19 ± 7	(n, γ)	Th^{232}	115.1d	F	T-0.514	0.514 (100)	0.03336	2996	Foil	$\text{Th}^{232}(\text{n}, \gamma); \text{Fe}^{51}(\text{n}, \gamma)$	0.0553													
Tin	50	Sn^{120}	32.97	0.001 ± 0.001	(n, γ)	Sn^{121}	400d	F	T(?) 1.1	(See Neptunium)			Foil	Fissionable materials	0.0293													
				0.014 ± 0.04	(n, γ)	Sn^{121}	27.5h	R	R-4.25	0.383	no γ	231	Foil	$\text{Sn}^{121}(\text{n}, \gamma); \text{Te}^{124}(\text{n}, \gamma)$	0.0330													
Titanium	22	Ti^{48}	73.45	0.0927	(n, p)	Sc^{48}	44h	F	T-1.3	0.65	0.986	1216	Foil	$\text{V}^{51}(\text{n}, \gamma)$	0.0566													
Tungsten	74	W^{186}	28.4	34 ± 7	(n, γ)	W^{187}	24.0h	T	R-18.8	0.63 (70)	0.07200	3370	Foil	$\text{Re}^{187}(\text{n}, \gamma); \text{Cs}^{130}(\text{n}, \gamma)$	0.0632													
Uranium	92	U^{236}	0.0057	0.0002	(n, γ)	U^{236}		F	T-0.1	(See Neptunium)		400	Foil	Fissionable materials	0.04783													
	92	U^{238}	$2.39 \times 10^7 \text{y}$		(n, γ)	U^{238}		F	T-0.7	(See Neptunium)		400	Foil	Fissionable materials	0.04783													
	92	U^{235}	99.3	0.54	(n, γ)	U^{235}		F	T-1.5	(See Neptunium)		400	Foil	Fissionable materials	0.04783													
Vanadium	23	V^{51}	99.76	1.5	(n, γ)	V^{52}	3.76m	R	R-3.0 eV	2.47	1.44	1710	Foil	$\text{Cr}^{52}(\text{n}, \gamma); \text{In}^{113}(\text{n}, \gamma)$	0.0704													
Zinc	30	Zn^{64}	48.89	0.44 ± 0.05	(n, γ)	Zn^{65}	245d			0.324	1.119	419	Foil	no primary reactions	0.0658													
	30	Zn^{66}	18.56	0.097 ± 0.010	(n, γ)	Zn^{66}	13.8h			IT	0.438	419	Foil		0.0658													
Zirconium	40	Zr^{90}	17.40	0.09 ± 0.03	(n, γ)	Zr^{90}	65d			0.396 (55)	0.722	1760	Foil	$\text{Mo}^{92}(\text{n}, \gamma)$	0.0423													

*GTR effective value

(?) questionable value

• preferred use

*GTR effective value
(?) questionable value
p preferred use

use in the useful flux range is indicated by (.). The listing of the principal gamma and beta emissions is intended to aid in the choice of evaluation method. The information as to availability of the detectors was obtained from the vendors listed in Table II. This list is not intended as a complete listing of all potential suppliers; rather it is a selection from August, 1962, issue of "Nucleonics Buyer's Guide."

Detailed procedures for determination of the flux using specific detectors are given in the literature (3,10). The evaluation of the activation of a detector may be accomplished electronically. The counting system to be used is determined from the accuracy required and the number of detectors to be processed. Table III lists several instruments and their applications. A discussion of counter calibration and a description of typical routine measurement techniques is given in reference (10).

Table II
Neutron Activation Detector
Potential Suppliers

<u>Designation</u> <u>Col. 17 Tab.1</u>	<u>Vendor</u>	<u>Address</u>	<u>Contact</u>
1	Aerojet-General, Nucleonics	San Ramon, California	
2	Alloys Unlimited, Inc.	21-01 43rd Avenue Long Island City 1, N. Y.	Stanley Bialek
3	Controls for Radiation, Inc.	130 Alewife Brook Pkwy. Cambridge 40, Mass.	Robert F. McMahon
4	Dresser Products, Inc.	Stockbridge Rd. Great Barrington, Mass.	William E. Ray
5	Edgerton, Germhausen & Grier, Inc.	Box 98 Goleta, California	A. A. Fleisher
6	General Astrometals Corp.	320 Yonkers Avenue Yonkers, N. Y.	A. M. Master, Jr.
7	Mac Kay, A. D., Inc.	198 Broadway New York 38, N. Y.	P. Nisley
8	Metals and Controls, Inc.	34 Forest Street Attleboro, Mass.	R. Morrow
9	Nuclear Science and Engineering Corp.	Box 10901 Pittsburgh 36, Pa.	R. C. Koch
10	Radiation Equipment and Accessories, Corp.	665 Merrick Road Lynbrook, N. Y.	F. A. Conklin
11	Reactor Experiments, Inc.	140 Harbor Blvd. Belmont, California	W. J. Heiman
12	United Mineral and Chemical Corp.	16 Hudson Street New York 13, N. Y.	S. Feierstein
13	U. S. Nuclear Corp.	Box 208 Burbank, California	J. L. Shepherd

Table III
Instruments for Activation Evaluation

<u>Instrument</u>	<u>Application</u>	<u>Remarks</u>
4π ion chamber high pressure	gamma, medium high intensity	decay analysis required; very accurate, reproducible results possible
4π scintillation counter gamma	gamma, low intensity	decay analysis required; very accurate, reproducible results possible; must be used in low background area
gamma ion chamber, portable	gamma, high intensity	accuracy limited, useful for relative measurements
gamma spectrometer multichannel analyzer	gamma, low intensity	operation complex; fast, accurate results possible
4π B ⁻ counter, proportional	beta	sample preparation complex, results very accurate
end window B ⁻ counter	beta	operation simpler than 4π ; less accurate

GAMMA

The ionizing effect produced by gamma rays provides a means by which they may be detected. Detectors in common usage for irradiation testing may be separated into two groups: those measuring the total integrated dose and those measuring the dose rate. Dosimeters used in irradiation testing are usually required to have a dose range greater than 10^6 ergs gm^{-1} (C) hr^{-1} and total integrated dose range from 10^5 to 10^{12} ergs gm^{-1} (C). In general, the dosimeters are not energy discriminating. Table IV lists gamma dosimeters currently in use and characteristics of each. Information as to availability of the dosimeters was obtained from the vendors listed in Table V. This is not intended as a complete listing of all potential suppliers; rather, it is a selection from the August, 1962, issue of the "Nucleonics Buyer's Guide." A brief discussion of each dosimeter type is given in the following paragraphs. Information included was obtained from the references (15, 16, 17).

TOTAL DOSE DEVICES

Dosimeters to measure the time integrated gamma dose are usually chemical, colorimetric or glass types. The dose is interpreted from a chemical or optical transmission effect induced by the radiation.

Ferrous-Ferric System: The ferrous sulfate (Fricke)⁽¹⁸⁾ dosimeter is generally considered as the standard for gamma measurement. This dosimeter is the most precise and foolproof of those available. The ferrous ions are oxidized to ferric by the radiation. The ferric ion concentration is then related to the dose. The ferric ion concentration is determined spectrophotometrically at 305 millimicrons. The usable range of this system is 10^5 to 4×10^6 ergs gm^{-1} (C) with the upper limit determined from the oxygen depletion from the system. This system may be used to a precision of ± 1 per cent. Additional information on this system and its use are given in the literature (19, 20, 21, 22, 24, 25).

Ceric-Cerous System: This system is based on the reduction of ceric to cerous ions by radiation. The dose is related to the difference in ceric ion concentration determined photometrically before and after irradiation. The dosimeter yield is relatively independent of radiation type and may be sensitive to fast neutron dose. This system has recently been studied to develop standard procedures for preparation and use so that its wide range (10^7 to 10^{10} ergs gm^{-1} (C)) may be used advantageously. This system may be used to a precision of ± 2 per cent. Additional information on this system is given in references (26, 27).

Nitrous Oxide System: This system is based on the decomposition of:

The dose is determined from a comparison of the number of moles of N_2 and O_2 produced per mole of N_2O and relating this to a calibration curve. This system is usable in the range of 10^9 to 2×10^{11} ergs/ gm^{-1} (C), with a precision to ± 5 per cent. The dosimeter shows good post-irradiation stability and is usable in a temperature range of -81° to 200°C . This system is discussed in detail in reference (28).

Table IV
Gamma Dosimeters

Dosimeter	Useful Range	Reaction	Remarks	Potential Suppliers (See Table V)
Integrated Dose	ergs gm ⁻¹ (C)	$\text{Fe}^{+2} \longrightarrow \text{Fe}^{+3}$	Commonly used as a standard, precision to + 1% Independent of rate up to 6×10^5 ergs gm ⁻¹ (C) hr ⁻¹ Limited by O ₂ depletion. References: 19, 20, 21, 22, 23, 24, 25	No supplier found, dosimeter must be custom constructed Information available in the references
Ferrous Sulfate (Fricke)	$10^5 - 4 \times 10^6$			
Ceric	$10^7 - 10^{10}$	$\text{Ce}^{+4} \longrightarrow \text{Ce}^{+3}$	Check with ferrous sulfate required System under development Precision to $\pm 2\%$ References: 26, 27	No supplier found, dosimeter must be custom constructed Information available in the references
Nitrous Oxide (N ₂ O)	$10^9 - 2 \times 10^{11}$	$\text{N}_2\text{O} \longrightarrow \text{N}_2, \text{O}_2, \text{NO}_2$	Good post-irradiation stability and reproducibility Useful for high doses Evaluation Complex, Precision to $\pm 5\%$ References: 28	No supplier found, dosimeter must be custom constructed Information available in the references
Dye Systems	$10^5 - 10^8$	Acid Formation with Indicator Dye Coloration	Several systems may be used Some are rate dependent (a libration is required) References: 29, 30, 31	Edgerton, Germeshausen & Grier, Inc. Box 98, Goleta, California Attention: A. A. Fleisher
Polymeric Systems	$10^5 - 10^{10}$	Coloration, polymerization or depolymerization	Several systems may be used. Sensitivity varies with system type, temperature and time of irradiation References: 32, 33, 34, 35, 36	No supplier found, dosimeter must be custom constructed Information available in the references
Glass	$10^6 - 5 \times 10^{10}$	Coloration	Several types available: Post-irradiation fading, stabilized by heat treating. Evaluation with microdosimeter reader is simple. References: 36, 37, 38, 39	Bausch & Lomb, Inc. Rochester 2, N. York Attention: R. F. Feitas Edgerton, Germeshausen & Grier, Inc. Box 98 Goleta, California
Luminors	$10^6 - 2 \times 10^7$	Degradation or Production of Luminescence	Several types available Good potential for routine dosimetry Evaluation with fluorimeter and photomultiplier References: 40, 41, 42	Harshaw Chemical Company 1945 E. 97th Street Cleveland 6, Ohio Attention: W. N. Managan
(Dose Rate)	ergs gm ⁻¹ (C) hr ⁻¹	Ionization current of a gas filled cavity is measured	Graphite lined chamber with CO ₂ is used as a standard. May be modified for neutron sensitivity References 1, 2	
Ionization Chambers	up to 10^{10}			
Calorimeter		Heat produced by absorbed radiation is measured	Used as secondary standard Not commonly used for routine dosimetry Reference: 17	No supplier found, dosimeter must be custom constructed
Semi-conductors	up to 10^9	Open circuit voltage is measured	Sensitive to neutrons Reference: 43	No supplier found, dosimeter must be custom constructed Information available in the references

Table V

Gamma Dosimeter

Potential Suppliers

<u>Designation</u>	<u>Vendor</u>	<u>Address</u>	<u>Contact</u>
1	Bausch & Lomb, Inc.	85062 Bausch Street Rochester 2, N. Y.	
2	Edgerton, Germeshauser & Grier, Inc.	160 Brookline Avenue Boston 15, Mass.	M. Alttillisch
3	Harshaw Chemical Co.	1945 E. 97th Street Cleveland 6, Ohio	S. Stewart
4	Hazelton-Nuclear Science Corporation	4062 Fabian Way Palo Alto, California	
5	Metcom, Inc.	76 Lafayette Street Salem, Mass.	P. Bagnell
6	Nuclear Materials & Equipment Corporation	196 Degraw Street Brooklyn 31, N. Y.	M. Felder

Dye Systems: Several types of dye dosimeters are available (15, 16) including a halogenated hydrocarbon type. This type of dosimeter is based on acid formation from radiation effects causing coloration of an indicator dye. The dose is determined from the amount of acid formed either by titration or by colorimetric measurement as determined from a calibration. These systems are usable over the range of $<10^5$ to 10^8 ergs gm^{-1} (C). These dosimeters are discussed in references (29, 30, 31).

Polymeric Systems: A polymeric dosimeter is based on radiation effects resulting in the formation of color, polymerization of low-molecular-weight compounds, or depolymerization of high-molecular weight compounds. The radiation sensitivity is variable within the limits of 10^5 to 10^{10} ergs gm^{-1} (C) depending on the particular composition chosen. The sensitivity of these systems is also dependent upon temperature and time of irradiation. This method is not applicable for routine measurements at this time due to the uncertainties of the system sensitivity. Additional information on this system is given in references: 32, 33, 34, 35, 36.

Glass Dosimeters: Glass dosimetry is based on radiation induced changes in optical properties. The most common of the radiation changes is coloration. The induced coloration tends to fade after irradiation in all glasses; however, it is possible to stabilize the color in many cases by post-irradiation heat treatment. Three types of glass have been studied to determine their applicability for dosimetry use:

1. Antimonate glass - usable range 10^7 to 10^{10} ergs gm^{-1} (C);
2. Corning glass (Code 8392) - usable range 8×10^6 to 5×10^{10} ergs gm^{-1} (C);
3. Cobalt glass (F-0621) - usable range 1×10^6 to 10^8 ergs gm^{-1} (C).

Glass dosimeters are described in the literature (36, 37, 38, 39).

Luminors: These dosimeters are based on the production or degradation of photoluminescence in the detector. Anthracene and naphthalene have been observed to suffer degradation of photoluminescence as a result of irradiation in the range of 3×10^7 to 2×10^{10} ergs gm^{-1} (C) for 1 mev gammas. The dose is determined for anthracene using a fluorimeter which excites with a 3650A ultraviolet radiation and measures the resulting 4420A fluorescence with a photomultiplier. This dosimeter shows a small post-irradiation recovery; however, the radiation effect may be stabilized by a 1 hour heat treatment at 100°C . Luminescence is produced by radiation in polystyrene scintillators with tetraphenyl butadiene either with or without paraterphenyl. The luminescence produced is proportional to exposure dose up to 6×10^7 ergs gm^{-1} (C). Additional information on these dosimeters is given in the literature (40, 41, 42).

DOSE RATE DEVICES

Dose rate is measured by a device which indicates continuously in the radiation field. Calorimeters, ionization chambers, or semiconductors may be used as dose rate devices. The temperature or current is measured to determine the dose rate from these dosimeters.

Calorimetry: A graphite sphere calorimeter is under development by the National Bureau of Standards for use as a primary standard for absorbed dose measurements. The calorimeter consists of a graphite sphere 1 cm. in diameter suspended in a spherical graphite shell by polystyrene pegs. The outer graphite shell has an interior diameter of 1.4 cm. and an outer diameter of 2.2 cm. The temperature rise of the inner sphere provides a measure of the absorbed dose rate. This calorimeter has been used for the calibration of high intensity sources and instruments to be used as secondary dosimeter standards. It is doubtful that the calorimeter will be widely used for routine dosimetry.

Gamma Ionization Chambers: Ion chambers are widely used for measuring gamma and neutron fluxes. The choice of materials may determine that the chamber will be more sensitive to one type of radiation. Dose rate measurements up to 10^{10} ergs $\text{gm}^{-1}(\text{C}) \text{hr}^{-1}$ are possible by choice of chamber parameters such as electrode spacing, gas pressure or operating voltage. To discriminate against neutrons and enhance the gamma sensitivity a carbon walled CO_2 filled chamber is used. This chamber has graphite electrodes and a flowing CO_2 system. The chamber design was initiated by ORNL and has recently been modified by Lockheed and Convair into a static gas type. The modified chamber will be used for standardization purposes.

Semiconductors: Semiconductors are affected by radiation and, therefore, may be used as dosimeters. Gamma dose rates up to 10^{12} ergs $\text{gm}^{-1}(\text{C}) \text{hr}^{-1}$ have been investigated using a p-n junction photovoltaic cell (43) produced by diffusing boron into n-type silicon. The internal impedance remained constant over the range of 10^4 to 10^9 ergs $\text{gm}^{-1}(\text{C}) \text{hr}^{-1}$ but approached saturation above the upper limit with a decrease in cell response. The cell will respond to dose rates up to 10^{12} ergs $\text{gm}^{-1}(\text{C}) \text{hr}^{-1}$; however, sensitivity is low for dose rates above 10^{10} ergs $\text{gm}^{-1}(\text{C}) \text{hr}^{-1}$.

The preparation and reporting of dosimetry results for irradiation testing is an important phase of a test program. Significant sections of irradiation test results reported in the past have been difficult to correlate due to lack of meaningful dosimetry data. The standardization of dosimetry reporting, particularly with regard to terminology and units, is important in comparing independent test results. The Radiation Effects Information Center at Battelle Memorial Institute published a memorandum (44) to standardize dosimetry reporting. This memorandum describes a suggested format for reporting radiation-effects data. The terminology suggested by the memorandum for describing dosimetry results is:

1. Thermal neutron flux will be described by the (nv_0) . When measured by activation detectors an absorption cross-section corresponding to a 2200 m sec^{-1} neutron velocity will be used.
2. Fast-neutron fields will be described by the number flux, fast $n \text{ cm}^{-2} \text{ sec}^{-1}$, with an additional statement describing the energy or energy distribution of the flux. When the flux is measured by activation detectors the cross section used should be given.

3. Gamma exposures will be reported in terms of the field, through its interaction with a reference material. The unit to be used is ergs per gram with carbon as the reference material. The dose unit is designated as $\text{ergs gm}^{-1} \text{ (C)}$.

The many routine calculations required to obtain an evaluation of the neutron flux from the activated detectors is applicable to the use of computers. If many detectors are used for each run, as in mapping experiments, the use of a computer for data reduction is mandatory if prompt results are required. The particular program used will depend, of course, on the computer available and the detector activation evaluation technique. In general, the data input for a computer will consist of:

1. Detector data - cross-section, decay constant and isotopic weights.
2. Activation data - power levels, change in power and shut-downs including time duration of each.
3. Run data - detector identification, weight and location during activation.
4. Counting data - number of counts for each detector, time of count after irradiation and counter correction factors.

The computer calculates the specific activity of each detector from the input data and then calculates the flux at each detector location by solving equation (1). Figure 1 is a sample data sheet that might be used for recording counting data that will be used for computer input.

Figure 1
Neutron Activation Detector
Counting Data

Counter No. _____
 Type _____

Date _____
 Activation Power Level _____

Detector				Time			Length of Count	Counts		Net counts per minute	Process efficiency	* Counter factor	Disintegrations per minutes	Disintegrations per second per gram	Curies per gram	Remarks (Changes in power level, etc.)
Identification	Weight or volume	Location	Run Start	Run End	Count Start	Count End		Gross	Net							
Counterfactor includes: 1. Efficiency 2. Self-absorption 3. Back scatter 4. Resorving time 5. Source size factor																

The techniques used will influence the accuracy of dosimetry results. The best accuracy that might be expected is:

1. Neutron flux by detector activation*
 - a. Thermal $\pm$ 10%
 - b. Resonance $\pm$ 10%
 - c. Fast $\pm$ 25%
2. Gamma dose
 - a. Integrated dose - to $\pm$ 1% (ferrous-sulfate method)
 - b. Dose rate 5% (carbon walled, CO₂ ion chamber)

The dosimetry system to be used should be carefully considered in the early planning stages of an irradiation testing program. The dosimetry will provide information as to the neutron flux and/or spectra and the gamma dose. Dosimetry for irradiation testing should attempt to fulfill the following criteria:

1. Dosimeter techniques and materials must be compatible with the irradiation environment.
2. The dosimeter system should be easily evaluated and provide accurate results.
3. The dosimetric results should be reported in units and a form to provide interpretation without ambiguity.

A neutron dosimetry system has been developed⁽¹⁶⁾ which establishes standard detectors and standard methods of activation analysis. At the present time, no similar standard program is available for gamma dosimetry. Therefore, the specific requirements for gamma dosimetry must be evaluated for each test.

A typical dosimetry system (neutron and gamma) for an irradiation test might include:

1. Neutron
 - A. Thermal cobalt or gold (n, γ)
 - B. Above 1 kev plutonium-239 (n,f)
 - C. Above 1.5 mev uranium-238 (n,f)
 - D. Above 2.9 mev sulfur-32 (n,p)
 - E. Above 6.3 mev aluminum-27 (n,p)
 - F. Above 8.6 mev aluminum-27 (n, α)
2. Gamma
 - A. Integrated dose - a chemical technique - ferrous sulfate (Fricke) or glass dosimeter.
 - B. Dose rate (optional) carbon walled, CO₂ ionization chamber.

A complete, accurate report of dosimetric measurements to describe without ambiguity the radiation environment should be prepared. Figure 2 is a sample radiation report sheet to be used to describe the environment. The units and terminology of the report should be carefully followed to conform with the suggested standard (44) format. The consistent usage of terms and units will simplify the interpretation of data from independent sources.

Dosimetry for irradiation testing is very important, for without knowledge of the radiation environment during a test, the test results would be meaningless. In the past a significant portion of reported radiation

*The accuracy limitation is determined from the accuracy available cross-section data.

Figure 2
Irradiation Testing
Dosimetric Measurements
Data Sheet

Exposure Time _____
Exposure Temp. _____ °C

Test no. _____
Date _____

Neutron Flux

Gamma Dose

Thermal detector	Above 1 Kev detector	Above 1.5 Mev detector	Above 2.9 Mev detector	Above 6.3 Mev detector	Above 8.6 Mev detector	Integrated Dosimeter type _____	Dosimeter type _____	Dose Rate Dosimeter	Remarks
_____	_____	_____	_____	_____	_____				
cross - section	cross- section	cross- section	cross- section	cross- section	cross- section			ergs gm ⁻¹ (C) hr ⁻¹	
(n _{v0})						ergs gm ⁻¹ (C)	ergs gm ⁻¹ (C)		

effects data has been difficult to correlate due to either a lack of associated dosimetry data, or information for proper interpretation of the reported data. A contributing factor to the confusion of reported data is the lack of standards for dosimetry. Recent developments as to standard neutron dosimetry⁽¹⁶⁾ and standard reporting format (44), if widely accepted, will be a significant step toward more usable radiation-effects data.

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