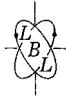


0 0 0 0 3 6 0 2 3 2 7

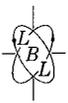


E. INSTRUMENT NOTES: GAMMA SPECTROMETRY

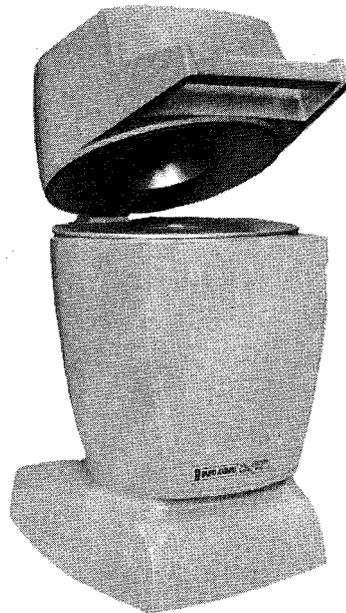
Included in the following pages are Instrument Notes for those instruments useful for gamma and x-ray spectrometry. The reader should be aware of the following facts, in order to find all Instrument Notes appropriate for gamma detection and measurement:

- 1) The filing system lists Instrument Notes alphabetically by manufacturer.
- 2) In the filing system, no distinction has been made between gamma, gamma/x-ray, and x-ray instruments. The choice of mnemonic (GAM, GAM-XRA, or XRA) has usually been made based upon the manufacturer's advertised specifications.
- 3) Instruments for gamma and x-ray monitoring are not included here. They have all been put together in a separate section, after the text on "X and Gamma Radiation Monitoring Instrumentation." Also, some similar instruments, also sensitive to betas or to alphas and betas, have been placed in the beta section (mnemonic RAD-BET, GAM) or under "Combination Instruments" (mnemonic RAD-ALP, BET, GAM).





Shielded Well Scintillation Counter  
Baird Atomic Model 988-10C



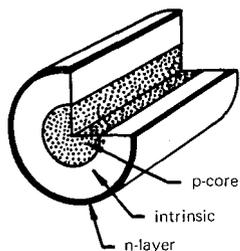
Class	Laboratory
Principle of Operation	Well-type NaI (Tl) scintillation detector system 1-3/4" diameter x 2" thick NaI (Tl) crystal. Well size 21/32" ID x 1-35/64" Deep
Sensitivity and Range	Efficiency: 35% for Iodine-131 Background: 250 cpm 20 cpm when used with spectrometer
Sampling	Batch
Performance	Accuracy: Operating temperature: meets minimum requirements
Requirements	Power: 110 Vac, 60 Hz (220 Vac, 50 Hz optimal) Size: 45.7cm H x 32.4cm diameter (18" height x 12-3/4" diameter) Weight: 131.5 kg (290 lbs.)
Features	Lead shield, tube mount, Model 810LX does not include detector
References	Manufacturer's specifications
Cost	\$1425.00
Address	Baird-Atomic, Inc. 125 Middlesex Turnpike Bedford, Mass. 01730 (617) 276-6204



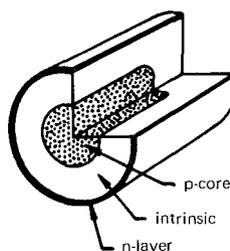


Ge(Li) Gamma Spectrometry Detectors

Canberra Industries



TRUE COAXIAL DETECTOR



CLOSED-END COAXIAL DETECTOR

Class

Laboratory

Principle of  
Operation

Ge(Li) semiconductor detectors used for gamma spectrometry.

Specifications

	Model 721X	Model 722
Shape	Circular True Coaxial	Closed End Coaxial
Active Area (cm <sup>2</sup> )	Up to 20	Up to 28
Volume (cm <sup>3</sup> )	10 to 100	10 to 150
Drifted Depth (mm)	Up to 20	Up to 25
Resolution, FWHM (keV) for 1.33 MeV	1.8 to 3	1.7 to 3
Efficiency*	Up to 20%	Up to 30%
Peak-to-Compton for 1.33 MeV	Up to 40:1	Up to 50:1
Price Range	\$9,282 - \$17,589	\$9,282 - \$17,589

\*Efficiency measured relative to 3" x 3" NaI(Tl), at 1.33 MeV and 25" distance.

Features

- (1) Cryostats and associated electronics also available.
- (2) Model 1427 electronic unit available for fast-timing applications.

References

Manufacturer's Specifications

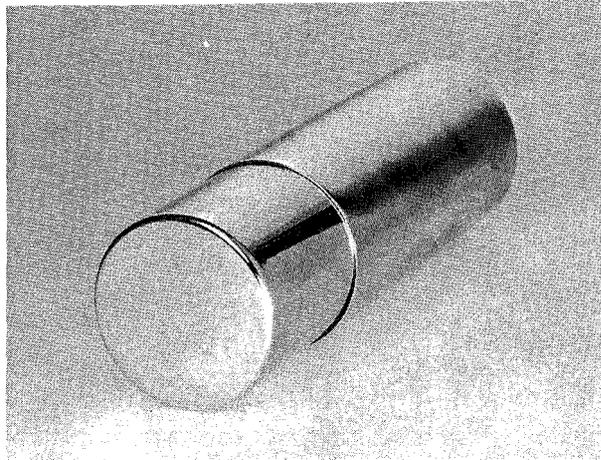
Address

Canberra Industries  
45 Gracey Avenue  
Meriden, CT 06450  
(203) 238-2351





NaI(Tl) Gamma Spectrometry Systems  
Canberra Industries Model 802



Class Laboratory  
Principle of Operation NaI(Tl) crystal, photomultiplier tube, and tube base voltage divider assembled into a unit for gamma spectrometry.

Specifications

Designation	Crystal Diameter		Crystal Thickness		Tube Size		Price for Assembly
	(cm)	(in.)	(cm)	(in.)	(cm)	(in.)	
802-1	3.8	1-1/2	2.5	1	5.1	2	\$247
802-3	5.1	2	5.1	2	7.6	3	\$344
A802-3W (well detector)	1.59	.625	3.65	1.438			\$381
802-4	7.6	3	7.6	3	7.6	3	\$420
B802-4W (well detector)	1.67	.656	5.24	2.063			\$1024

Features (1) Entrance window 0.019" aluminum  
(2) Aluminum housing  
(3) Auxiliary electronics also available

References Manufacturer's Specifications

Address Canberra Industries  
45 Gracey Avenue  
Meriden, CT 06450  
(203) 238-2351





X-ray Detectors  
Canberra Industries



Class

Laboratory

Principle of  
Operation

Si(Li) detectors used for x-ray spectroscopy

Specifications

Model	7313	7333	7383
Area(mm <sup>2</sup> )	10	30	80
Depth(mm)	3	3	3
Resolution <sup>1</sup> at 5.9 keV FWHM	160 eV	to 170 eV	185 eV
Price Range <sup>2</sup>	\$3,500 to \$5,700		

<sup>1</sup> Guaranteed worst case resolution @5.9 keV @1000 counts/sec. Typical performance exceeds specifications. <sup>2</sup> Includes Si(Li) detector, cryostat with choice of 1.0, 0.5, 0.3 mil Be window, Model 1708 pulsed optical feedback preamp.

Features

Liquid nitrogen cryostat, preamplifier, spectrum enhancer, system alarm, LN monitor, and bias supply also available

References

Manufacturer's specifications

Address

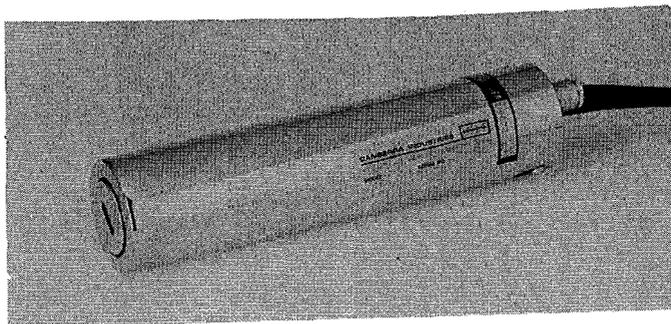
Canberra Industries  
45 Gracey Avenue  
Meriden, CT 06450  
(203) 238-2351





X-ray Scintillation Detectors

Canberra Industries



Principle of  
Operation

X-ray scintillation detector comprising a NaI(Tl) crystal photomultiplier tube and a voltage divider network

Specifications

	<u>Model 1701</u>	<u>Cost</u>
Window	0.005 in. Beryllium	\$624
Transmission	50% at 3.5 keV 88% at 6.0 keV	
Resolution	55% or better at 6 keV	
Collimator	Removable brass with 0.125" x 0.5 in. slit	
Crystal	NaI(Tl) 1" in dia., 1 mm thick	
Photomultiplier	Amperex XP 1010, selected	
	<u>Model 1702</u> with preamplifier	
Type	Charge sensitive, FET input	\$980
Power	24 V	
Output	0 to 20 V, 40 nsec risetime 60 usec fall time, 50 ohms impedance	

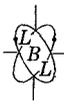
References

Manufacturer's Specifications

Address

Canberra Industries  
45 Gracey Ave.  
Meriden, CT 06450  
(203) 238-2351



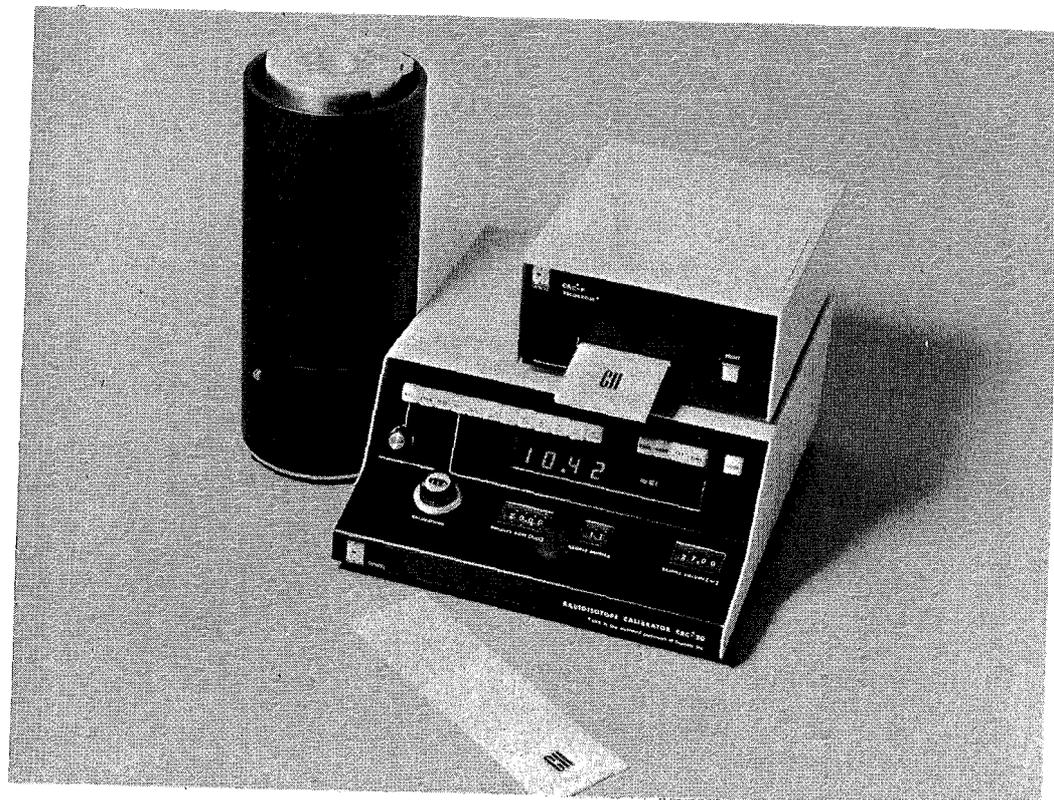


INSTRUMENTATION  
FOR ENVIRONMENTAL  
MONITORING

0 0 0 0 3 6 0 2 0 3 3

RAD-GAM  
Calibrator  
Capintec  
January 1978

### Radioisotope Calibrator



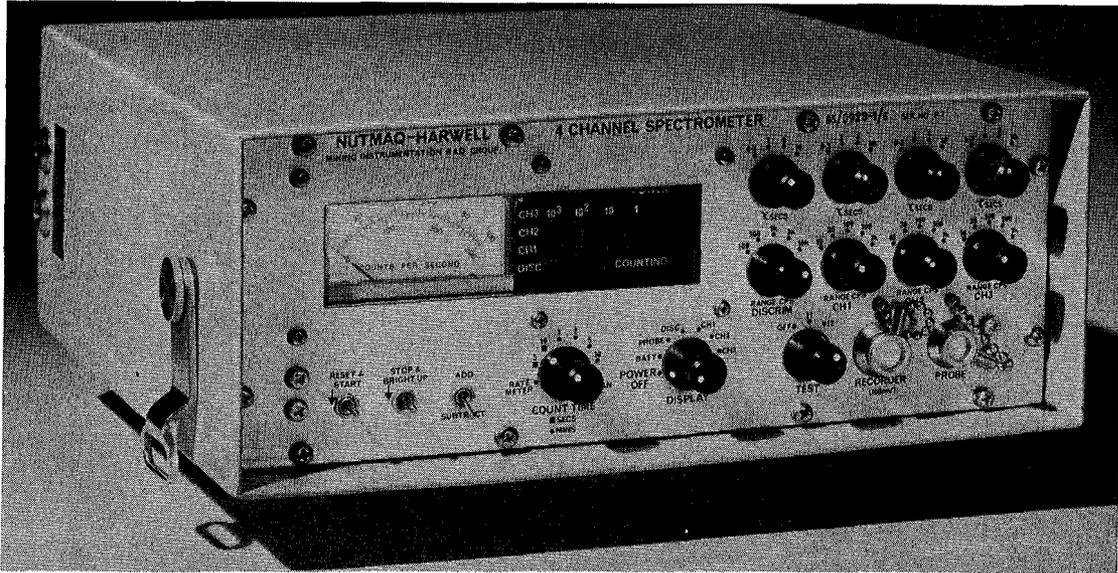
Class	Laboratory calibration for gamma-emitting radioisotopes		
Principle of Operation	Pressurized argon well-type ionization chamber and digital electrometer system.		
Sensitivity and Range	CRC-5/10	CRC-5M/10M	CRC-20
Function	Calibrator	Calibrator/radiation monitor	Calibrator
Energy Range	25 keV to 3 MeV		
Sensitivity Range	To 6 Ci maximum	To 6 Ci maximum (calibrator mode)	To 2 Ci maximum
Resolution	0.1 $\mu$ Ci		
Dose Range	NA	0 to 200 mR/hr 0 to 2 R/hr 0 to 20 R/hr (Radiation monitoring mode)	NA
Performance	<p>Each calibrates over 90 radionuclides. For CRC-20, computer memory can accommodate up to 19 samples and concentration recall displays decay adjusted concentrations for push-button isotopes with a ticket printed patient dose hard copy record.</p> <p>Accuracy: <math>\pm</math> 3% between calculated and observed values</p> <p>Operating temperature:</p>		
Requirements	CRC-5/10	CRC-5M/10M	CRC-20
Power:	117 VAC $\pm$ 10%, 50/60 Hz		
Size:	38.7 cm $\times$ 16.5 cm $\times$ 23.5 cm 15 1/4" W $\times$ 6 1/2" H $\times$ 9 1/4" D	38.7 cm $\times$ 16.5 cm $\times$ 23.5 cm 15 1/4" W $\times$ 6 1/2" H $\times$ 9 1/4" D	36.8 cm $\times$ 16.5 cm $\times$ 40.6 cm 14 1/2" W $\times$ 6 1/2" H $\times$ 16" D
Weight:	18.1 kg (40 lb)	20.4 kg (45 lb)	23.6 kg (52 lb)
Features	<ol style="list-style-type: none"> <li>1) Push button isotope selection</li> <li>2) Geometry independence</li> <li>3) <math>^{99}\text{Mo}</math> (interference w/ <math>^{99}\text{Tc}</math>) assay capability</li> <li>4) Accommodates small animals for total body assay</li> <li>5) CRC-5/10 field upgradable to computer/printer model (CRC-20)</li> </ol>		
Reference	<ol style="list-style-type: none"> <li>1) Suzuki, A., Suzuki, M.N., Weis, A.M., "Analysis of a Radioisotope Calibrator," Journal of Nuclear Medicine Technology, Volume 4, pg. 193-198, (December 1976).</li> <li>2) Manufacturer's Specifications</li> </ol>		
Cost	CRC-5/10 \$3250./\$3700	CRC-5M/10M \$3970./\$4225	CRC-20 \$6300
Address	Capintec, Inc. 136 Summit Avenue Montvale, New Jersey 07645 (201) 391-3930		



Four Channel Gamma-Ray Spectrometer

GEC-Elliott Process Model Type 0928

Model Name and Number

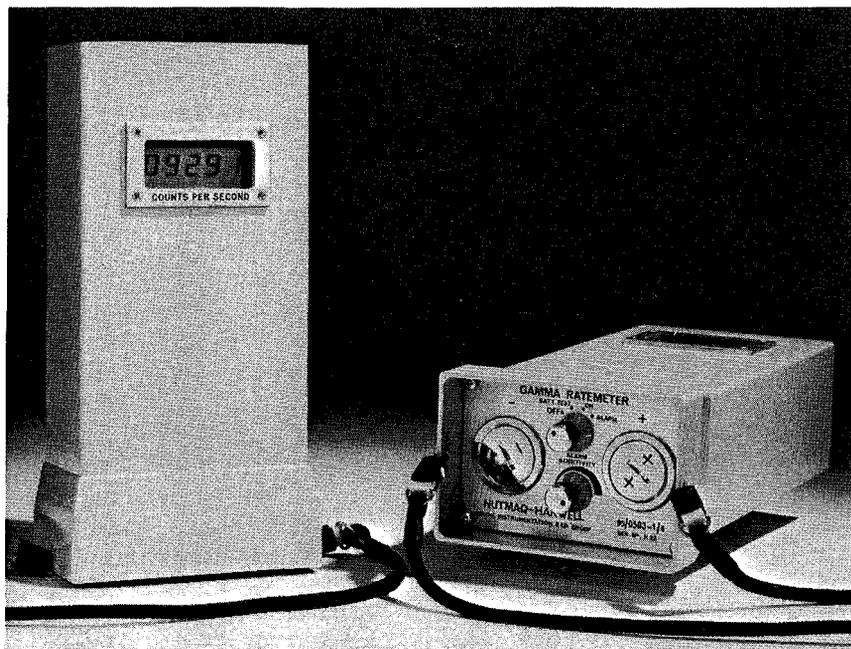


Class	Portable, designed for field use
Principle of Operation	Used with a variety of probes; it has a discriminator and three energy channels all of which may be preset within the range 50 keV to 3 MeV. Counts derived from $\beta$ -ray energies exceeding the discriminator threshold or lying within the channel boundaries, may be displayed one at a time on the front panel meter and, simultaneously, on an external 4-channel recorder; alternatively they may be displayed one at a time as an accumulated count on a digital display.
Sensitivity and Range	Range (for channels 1,2, and 3): 0-10, 0-30, 0-100, 0-300, 0-1000 cps Range (discriminator): 0-100, 0-300, 0-1000, 0-10,000 cps Sensitivity: depends on probe Energy limits: Discriminator 50 keV to 1 MeV Channel 1 Channel 1 1 MeV to 2 MeV Channel 2 Channel 2 1.25 MeV to 2.25 MeV Channel 3 Channel 3 2 MeV to 3 MeV
Features	1) Spectrum stabilized using 60 keV line from $^{241}\text{Am}$ reference source contained in probe 2) Front panel switch allow test pulser of a 1Hz or 136 Hz to be injected into digital counting circuits so that the correct operation of each counting chain may be monitored.
References	Manufacturer's specifications
Cost	Refer to Manufacturer
Address	GEC-Elliott Process Instruments Limited Nuclear Controls Division, Century Works Lewisham, SE13 7LN, England 01-692 1271 Telex 22469





Gamma Survey Meter  
GEC-Elliott Process Type 0593



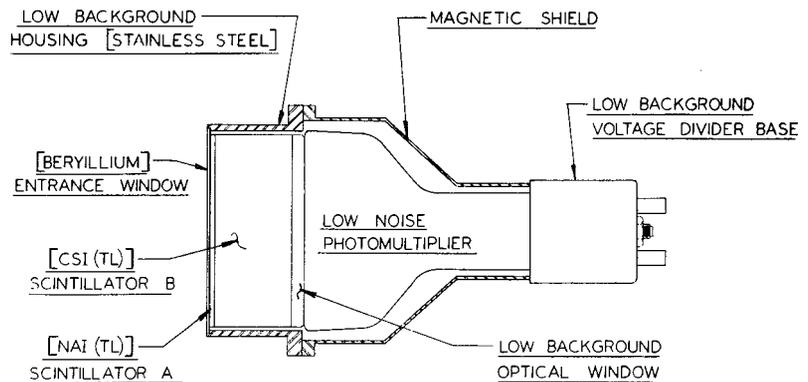
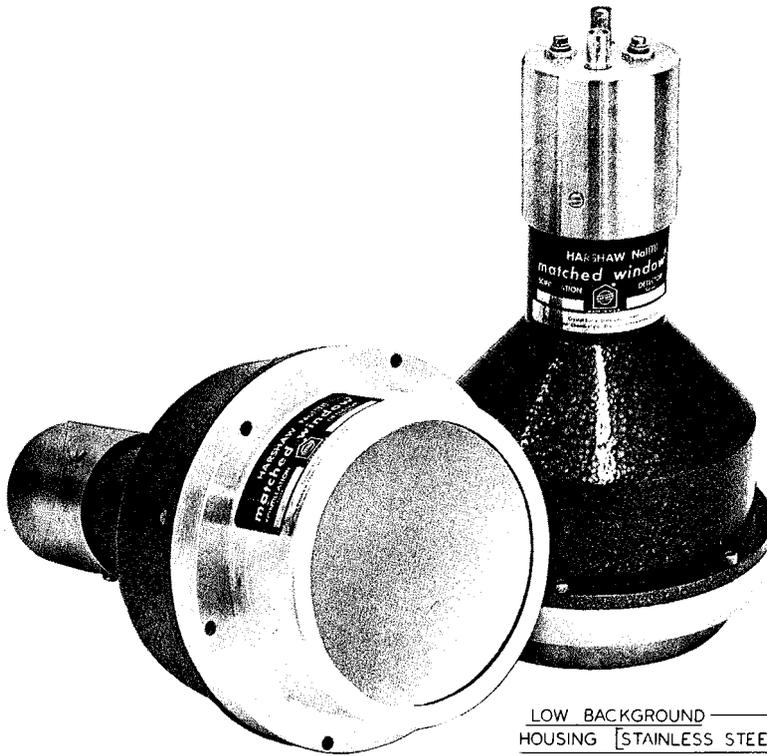
Class	Portable
Principle of Operation	$\gamma$ -ray detector, solid-state circuitry, scaler, liquid crystal display, and audible alarm
Sensitivity and Range	Range: 0-5 digit readout
Sampling	Continuous
Performance	Count cycling time: 2 seconds Operating temperature range: -5°C to 50°C
Requirements	Power: Six "C" cells Size: 7.4 cm $\times$ 12.4 cm $\times$ 24.0 cm (3 in. $\times$ 5 in. $\times$ 9 1/2 in.) Weight: 1.6 kg (3.5 lb.)
Features	1) Audible alarm has an adjustable threshold and pitch related to count rate 2) Hermetically sealed 3) Fully compensated to accommodate voltage changes between 9V and 6.6V 4) Battery test switch
References	Manufacturer's specifications
Cost	Refer to Manufacturer
Address	GEC - Elliott Process Instruments Limited Century Works Lewisham, London SE13 7LN, England Telephone: 01-692-1271





Phoswich Scintillation Detector

Harshaw Chemical Company



Class

Laboratory

Principle of Operation

The PHOSWICH detector is designed for high sensitivity radiation counting where low level, low energy radiation is normally obscured by background radiation. The detector consists of a thin primary scintillator and a thick secondary scintillator. Events that occur in the secondary crystal only, and those events that occur in both crystals simultaneously, are electronically rejected. Only the low level radiation that is totally absorbed by the primary scintillator is counted. Events are sorted out using the Harshaw NC-25 Pulse Shape Analyzer which recognizes events in the primary and secondary crystals by their different decay times.

By using a thin  $\text{CaF}_2$  (Eu) primary scintillator and a NaI(Tl) secondary scintillator, the Phoswich can detect alpha and beta particles in the presence of a gamma background.

Performance

Can suppress higher-energy backgrounds by factors of 3 to 10, compared to ordinary NaI(Tl) or CsI(Tl) detector.

Especially designed for low-energy x-rays from Americium-241, as used to determine plutonium.



Features                    Pulse shape analysis system Model TASC-10 also available.

References                Manufacturer's specifications

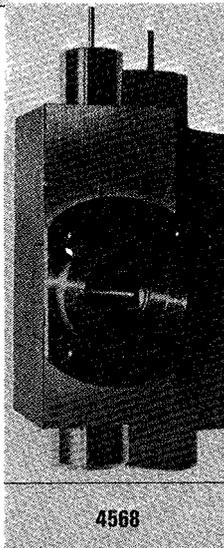
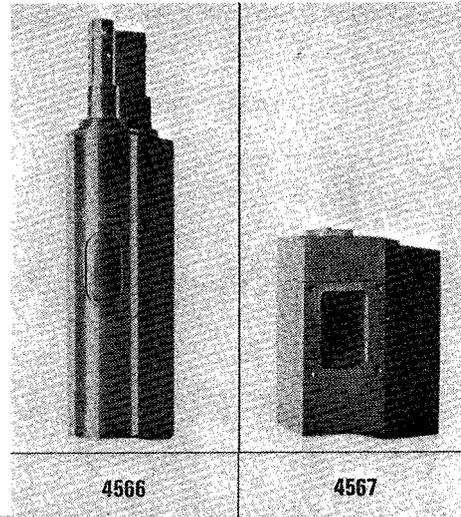
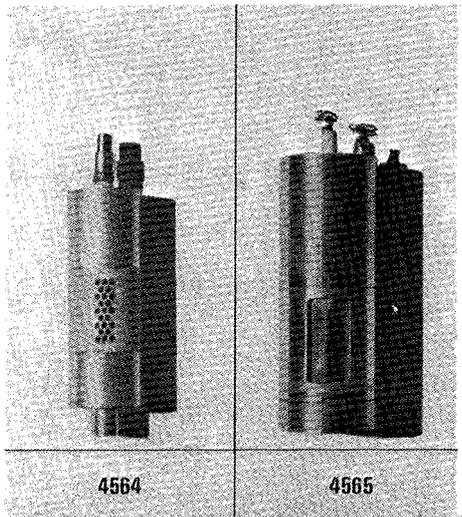
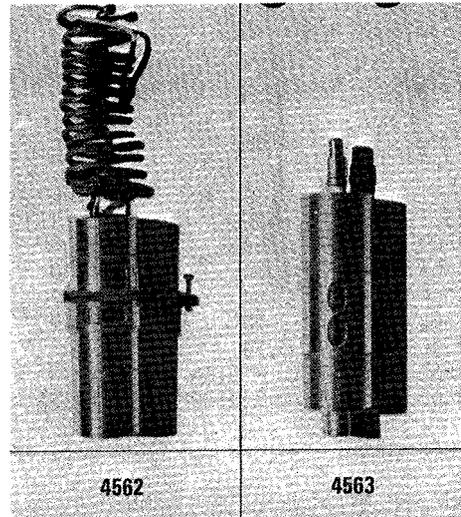
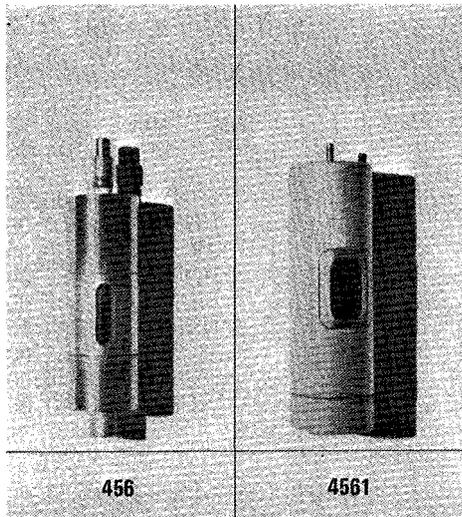
Cost                        Phoswich Detector-- ~ \$4500 - Depends on size

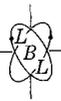
Address                    Harshaw Chemical Company  
                              6801 Cochran R-oad  
                              Solon, OH 44139  
                              (216) 248-7400



X-ray Proportional Detectors

LND Models





Class            Detector

X-ray Detectors

Model	Maximum Dia.		Overall Length		Operating Voltage	Price	Gas
	(cm)	(in.)	(cm)	(in.)			
456	2.54	1.00	10.16	4.00	1800	\$300.00	Xenon
4561	3.00	1.181	9.50	3.740	1850	\$300.00	Xenon
4562	2.54	1.00	7.62	3.00	1300	\$300.00	Xenon
4563	2.54	1.00	10.16	4.00	1400	\$450.00	Neon
4564	2.54	1.00	10.16	4.00	1400	\$500.00	Neon
4565	3.33	1.31	10.16	4.00	1200	\$300.00	Xenon
4566	2.54	1.0	13.97	5.5	1800	\$350.00	Xenon
4567	6.35 × 6.35	1.25 × 1.25	6.67	2.625	1850	\$300.00	Xenon
4568	4.20 × 4.20	1.653 × 1.653	14.47	5.697	1900	\$450.00	Xenon

Reference            Manufacturer's Specifications

Address            LND  
                     3230 Lawson Blvd.  
                     Oceanside, NY 11572  
                     (516) 678-6141



Ge(Li) Detectors  
Nuclear Enterprises

Class Laboratory  
Principle of Operation Ge(Li) semiconductor detector to be used with associated electronics, cryostat accessories  
Specifications

	Series Designation		
	GDP	GDC-T	GDC-C
Detector Type	Planar Ge(Li) for Low Energy X-Rays	Trapezoidal or Cylindrical 5-Sided Ge(Li)	Coaxial Ge(Li)
Area (mm)	4 to 15	10 to 40	4.5 to 15
Depletion Depth (mm)	5 to 15	5 to 15	8 to 15
Length (mm)	—	10 to 78	23 to 78
Sensitive Volume (cm <sup>3</sup> )	2 to 22.5	5 to 100	8 to 100
Energy Resolution at 1.33 MeV	2.5 to 5 keV	2.5 to 5 keV	2.5 to 5 keV
Peak/Compton at 1.33 MeV	4:1 to 12:1	5:1 to 18:1	6:1 to 17:1
Efficiency at 1.33 MeV Compared to 3" x 3" NaI(Tl)	—	1% to 11%	0.9% to 11%
Price Range	\$ to \$	\$ to \$	\$ to \$

Features Cryostats, associated electronics also available.  
3 grades of Ge(Li) available in each line.

References Manufacturer's Specifications

Address Nuclear Enterprises, Inc.  
931 Terminal Way  
San Carlos, CA 94070  
(415)592-8663. Telex: 348371

or

Nuclear Enterprises, Ltd.  
Bath Road, Beenham  
Reading  
ENGLAND





Si(Li) Detectors  
Nuclear Enterprises

Class Laboratory  
Principle of Operation Si(Li) semiconductor detector to be used with associated electronics  
Specifications

Series Designation	Area (mm)	Thickness (mm)	Resolution at 6.4 keV, 77°K (FWHM, eV)	Price, Detector Only
SDX 25-3A	25	3	280	\$
SDX 25-3	25	3	400	
SDX 50-3	50	3	450	
SDX 100-3	100	3	600	
SDX 200-5	200	5	750	
SDX 300-5	300	5	900	

Features Si(Li) detectors available for particle detection also.  
Associated electronics also available.  
Cryostats available separately.  
Resolution measured at LN (77°K).

References Manufacturer's Specifications

Address Nuclear Enterprises Inc.  
931 Terminal Way  
San Carlos, CA 94070  
(415) 592-8663 Telex: 348371

or

Nuclear Enterprises, Ltd.  
Bath Road, Beenham  
Reading  
ENGLAND





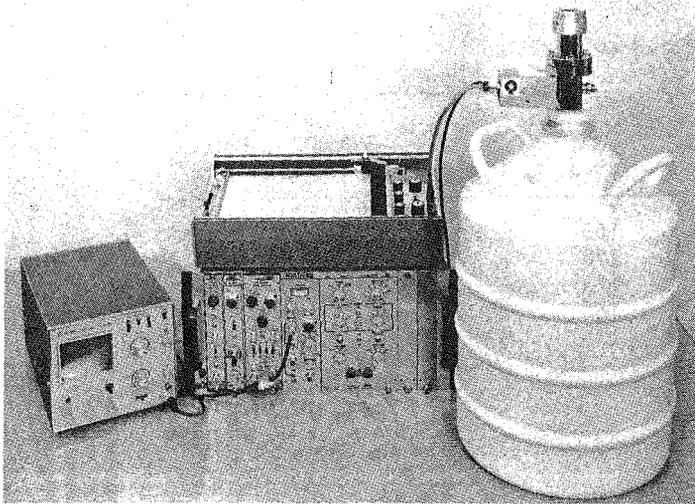
NaI(Tl) Crystals  
Nuclear Enterprises

Class	Laboratory
Principle of Operation	NaI(Tl) crystal coupled to photomultiplier tube
Specifications	<p>Crystal Sizes: Range from 1/2" x 1/2" to 6" x 4" Prices: Range from \$ to \$</p> <p>De-mountable Assemblies: Can accept crystal sizes from 1/2" x 1/2" to 6" x 4" Prices: Range from \$ to \$</p> <p>Integral assemblies: Can accept crystal sizes with diameter up to and including 3". Prices: Range from \$ to \$ Special configurations manufactured to order.</p>
Other Features Also Available	<p>Electronic accessories Low-background quartz windows &amp; stainless steel mounts, Beryllium windows Anti-Compton set-ups Large well set-ups Cesium-iodide (thallium) detectors also available</p>
References	Manufacturer's Specifications
Address	<p>Nuclear Enterprises Inc. 931 Terminal Way San Carlos, CA 94070 (415) 592-8663. Telex: 348371</p> <p>or</p> <p>Nuclear Enterprises, Ltd. Bath Road, Beenham Reading ENGLAND</p>

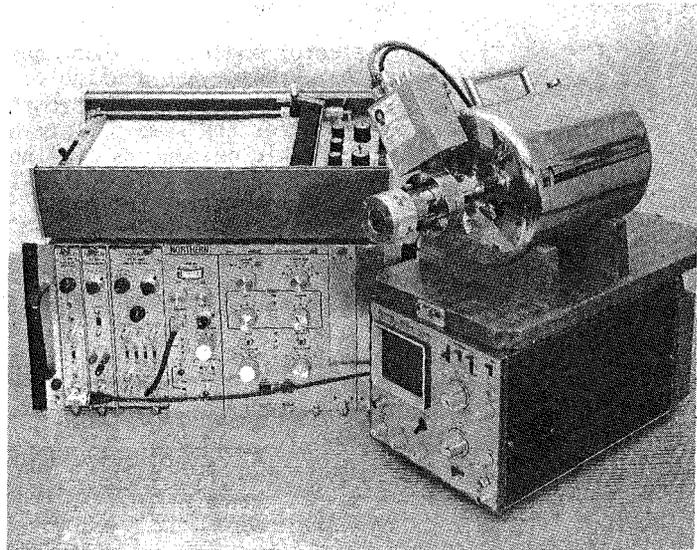




Si(Li) X-Ray Detector/Cryostat

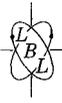


Nuclear Semiconductor 101



Nuclear Semiconductor 104

Class	Laboratory (101) and portable (104)
Principle of Operation	Cryogenically cooled Si (Li) detector and pulsed optical feedback preamplifier. Vertical dipstick (17ℓ) dewar



Sensitivity and Range                      Range: Oxygen X = 8 to americium Z = 95  
Sensitivity: 1 ppm = 100% full scale

RESOLUTION AT 5.9 keV  
(FWHM in eV)

10mm <sup>2</sup> area	30mm <sup>2</sup> area	80mm <sup>2</sup> area	200mm <sup>2</sup>	300mm <sup>2</sup>	Price
144	148	165			\$7280.
148	155	170			6880.
155	160	175			6080.
165	170	185			4070.
175	180	195	225	250	6490.
			260	290	4490.

Sampling                                      Continuous

Performance                                Accuracy: ±0.1%  
Temperature Range: 10-35°C  
Readout Time: Instantaneous

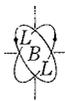
Requirements                                Power: 110 V ac, 2 A  
Size: 91 cm H × 48 cm W (36" H × 17" W)  
Weight: About 45 kg (100 lb)

Features                                      1) Detector mounted in shock absorber  
2) Other type dewars available  
3) Model 104-Portable spectrometer with 3 liter dewar, add \$1500 to above prices for the same performance  
4) Electronic components available, also.

References                                    Manufacturer's Specifications

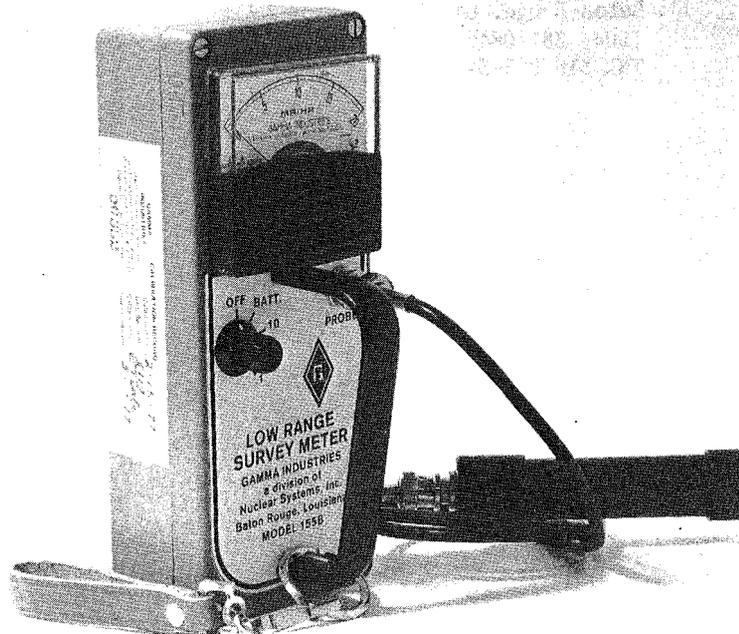
Cost    See table

Address                                        Nuclear Semiconductor  
1400 Stierlin Road  
P.O. Box 1389  
Mountain View, CA 94042  
(415) 969-9400



Survey Meter

Gamma Industries Models 150B, 155B, 250B



Class	Portable		
Principle of Operation	Geiger-Muller tube as detector, amplifier, electrometer, meter readout		
Sensitivity and Range	<u>150B, 155B</u>		<u>250B</u>
	Range:	0 - 2 mR/hr 0 - 20 mR/hr 0 - 200 mR/hr	0 - 10 mR/hr 0 - 100 mR/hr 0 - 1000 mR/hr
Sampling	Continuous		



Performance            150B - internal probe  
                          155B - external probe  
                          Accuracy:  $\pm 20\%$  FS  
                          Operating temperature range:  $-17.8^{\circ}\text{C}$  to  $75^{\circ}\text{C}$  ( $0^{\circ}\text{F}$  to  $167^{\circ}\text{F}$ )  
                          Time constants:  $\sim 5$  sec  
                          Energy range: 250 keV to 1.5 MeV

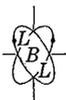
Requirements	<u>150B, 155B</u>	<u>250B</u>
Power:	Two "D" Cells	Two "D" Cells
Size:	8.25 cm $\times$ 20.32 cm $\times$ 10.16 cm (3 1/2" W $\times$ 8" L $\times$ 4" H)	8.26 cm $\times$ 20.32 cm $\times$ 10.16 cm (3 1/4" W $\times$ 8" L $\times$ 4" H)
Weight:	1.135 kg (2.5 lb)	1.25 kg (2.75 lb)

Features                1) Will not reverse in high intensity gamma radiation fields  
                          2) Low power consumption:  $> 400$  hours from ordinary batteries, up to  
                          1000 hrs on alkalines.

References             Manufacturer's Specifications

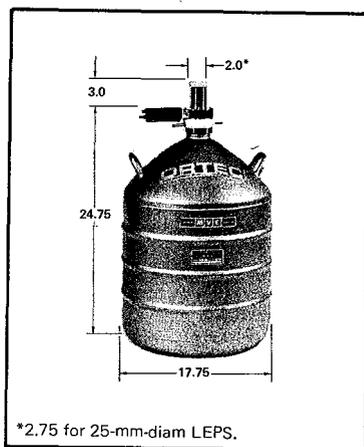
Cost                    Model 150B    \$275.00  
                          Model 155B    \$325.00  
                          Model 250B    \$250.00

Address                Nuclear Systems, Inc.  
                          924 Joplin St.  
                          P.O. Box 2543  
                          Baton Rouge, LA 70821  
                          (504) 387-0846  
                          TWX 510 993-3494

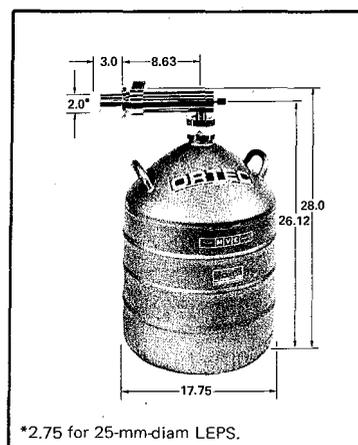


Ge(Li) Gamma Spectrometry Detectors, Systems

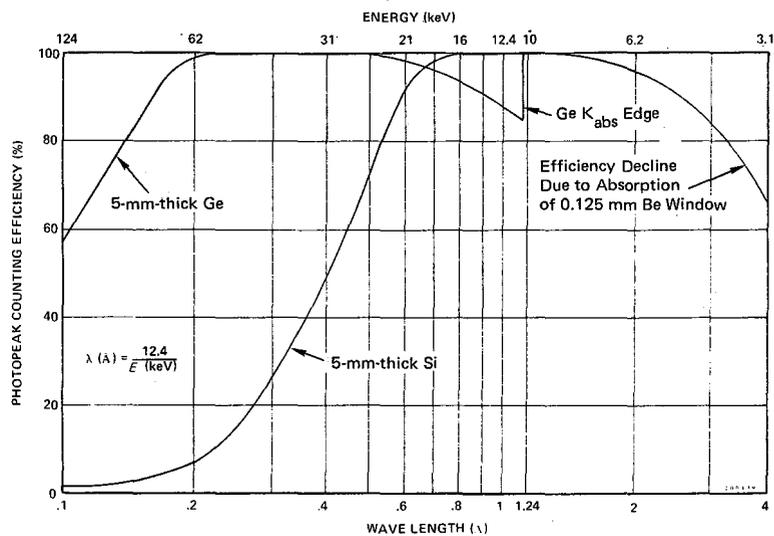
Ortec 8000 Series



Vertical Dip Stick



Horizontal Dip Stick



Comparison of Ge(Li) LEPS and Si(Li) Detectors for High-Energy X-Ray Analysis.

Class

Laboratory

Principle of Operation

Germanium (Lithium-Drifted) detectors used for gamma spectrometry, and full systems including dewar, preamplifier, cryostat.



Specifications

8000 Series Ge(Li) Coaxial Detectors

Detector Series	Vertical Cryostat Model No.	Horizontal Cryostat Model No.	Relative Photopeak Efficiency (%)	Resolution FWHM <sup>b</sup> (keV) for <sup>60</sup> Co 1.332 MeV Gammas	Peak-to-Compton Ratio	Price
VIP <sup>10</sup> Series Delivery: from stock	8001-1020V	8101-1020V	≥ 10	≤ 2.0	≥ 33:1	\$7,940
	8001-1022V	8101-1022V	≥ 10	≤ 2.2	≥ 30:1	6,960
WIN <sup>15</sup> Series Delivery: from stock	8001-1521W	8101-1521W	≥ 15	≤ 2.1	≥ 36:1	10,490
	8001-1523W	8101-1523W	> 15	≤ 2.3	≥ 33:1	9,770
TEC <sup>20</sup> Series Delivery: from stock to 60 days	8001-2021T	8101-2021T	≥ 20	≤ 2.1	≥ 41:1	13,090
	8001-2023T	8101-2023T	≥ 20	≤ 2.3	≥ 38:1	12,290
CUSTOM SYSTEMS Delivery: As per quotation	Single and multiple detector arrays available in a wide range of cryostat, end cap, and dewar configurations		To 75	To 1.7 Cooled FET Systems Available	200:1	Please ask for factory quotation

<sup>a</sup>The ORTEC Ge(Li) Detectors are true coaxial geometries to avoid the problems of "Dead Layer" and poor time resolution associated with closed-end type geometries.

<sup>b</sup>FWHM is ≤ 2 × FWHM (Full Width at Tenth Maximum is ≤ twice the Full Width at Half Maximum).

Specifications for Cryostats and Dewars

Model No.	Type	Cryostat Weight (kg)	LN <sub>2</sub> Volume (liters)	LN <sub>2</sub> Weight (kg)	Dewar Weight (kg)	Typical Holding Time (days)	Total Weight Full of LN <sub>2</sub> (kg)
80	Vertical Dipstick	2.45	31	24	15	28	41.5
81	Horizontal Dipstick	3.32	31	24	15	28	42.9



8000 Series Ge(Li) Low Energy Photon Spectrometers.<sup>a</sup>  
Delivery: stock to 30 days

Vertical Cryostat Model No.	Horizontal Cryostat <sup>b</sup> Model No.	Diameter (mm)	Drifted Depth (mm)	Active Area (mm <sup>2</sup> )	Energy Resolution FWHM <sup>c</sup> (eV)	
					At 5.9 keV ( <sup>55</sup> Fe)	At 122 keV ( <sup>57</sup> Co)
8013-06180	8113-06180	6	5	30	180	485
8013-06185	8113-06185	6	5	30	185	500
8013-06200	8113-06200	6	5	30	200	525
8013-10190	8113-10190	10	5	80	190	490
8013-10210	8113-10210	10	5	80	210	510
8013-10225	8113-10225	16	5	80	225	525
8013-16260	8113-16260	16	5	200	260	525
8013-16275	8113-16275	16	5	200	275	535
8013-16300	8113-16300	16	5	200	300	550
8013-25360 <sup>d</sup>	8113-25360 <sup>d</sup>	25	7	500	360	595
8013-25380 <sup>d</sup>	8113-25380 <sup>d</sup>	25	7	500	380	615
8013-25400 <sup>d</sup>	8113-25400 <sup>d</sup>	25	7	500	400	615
8013-32500 <sup>d</sup>	8113-32500 <sup>d</sup>	32 <sup>e</sup>	7	800	500	750

<sup>a</sup>All systems supplied with 5-mil-thick Be windows and 31-liter dewars except where noted.

<sup>b</sup>Horizontal common-vacuum configuration available special order.

<sup>c</sup>Using optimum time constant of 2 to 6  $\mu$ s FWHM  $\leq$  2 $\times$  FWHM. Total system resolution for 5.9 keV <sup>55</sup>Fe source at 1000 counts/s measured in accordance with IEEE Standard 325 (1971), using ORTEC standard electronics. Guaranteed for one year.

<sup>d</sup>10-mil-thick Be window.

<sup>e</sup>Special order.

- Features
- (1) Detector efficiency is compared to that of 3"  $\times$  3" NaI(Tl).
  - (2) Geometry is true cylindrical on all coaxial detectors.
  - (3) Electronic accessories (preamplifiers, etc) available.
  - (4) Spectrometer system has  $\leq$  1  $\mu$ m 'window'.
  - (5) End cap is 0.005" beryllium on all models except 0.010" on 25-mm- & 32mm diameter models.
  - (6) Cryosorption pumping used in all spectrometers.

References                      Manufacturer's Specifications

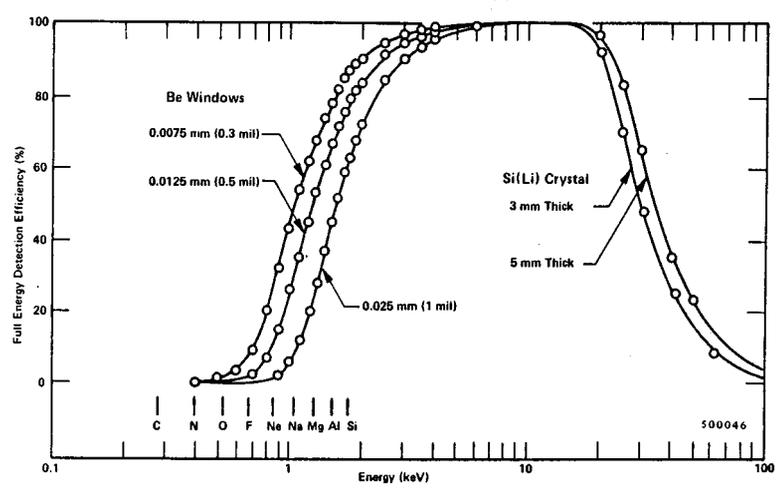
Address                              ORTEC, Inc.  
100 Midland Road  
Oak Ridge, TN 37830  
(615) 482-4411





Si(Li) Photon Detectors

Ortec Series 7000



Class Detectors

Principle of Operation Silicon (Lithium-Drifted) Detectors for Low-Energy Photon Spectrometry

Specifications

Cryostat Configuration	ORTEC Model No.	Dewar Size (Liters)	Liquid Nitrogen Use Time (Liters/Day)	Guaranteed Resolution* (eV) FWHM for ~ Area/Diam.				Range of Price
				12.5 mm <sup>2</sup> / 4 mm	30 mm <sup>2</sup> / 6 mm	80 mm <sup>2</sup> / 10 mm	200 mm <sup>2</sup> / 16 mm	
Vertical Dipstick	7013	30	1.1	205 160	215 175	230 190	325 285	\$5,570. 7,050.
Horizontal Dipstick	7113	30	1.1	205 160	215 175	230 190	325 285	5,720. 7,200.
Light Weight	7513	5	0.9	205 160	215 175	230 190	325 285	5,750. 7,230.

Detectors are available with a guaranteed resolution better than that shown in table.  
\*Total system resolution for <sup>55</sup>Fe 5.9 keV at 1000 counts/sec measured with an ORTEC 716A, a 459, and a dc-coupled 512-channel analyzer. The FWIM is ≤ 2 FWHM. Guaranteed for one year. Upper value is standard. Lower value is premium.

Beryllium Entrance Windows		Price
Standard	1 mil Be Window. Normally delivered with 4, 6, & 10 mm diameter detectors. 2 mil Be Window, also.	Included in System Price
Optional	0.5 mil Be Window. Thin, unsupported window for 4 and 6 mm diameter detectors.	\$220.00
	0.3 mil Be Window. Ultra-thin, unsupported window for 4 mm diameter detectors.	330.00



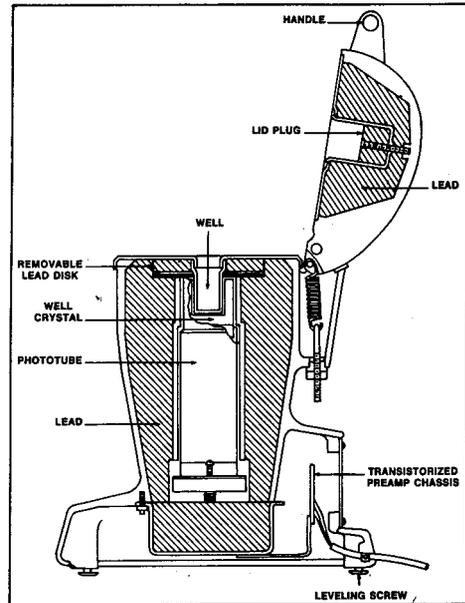
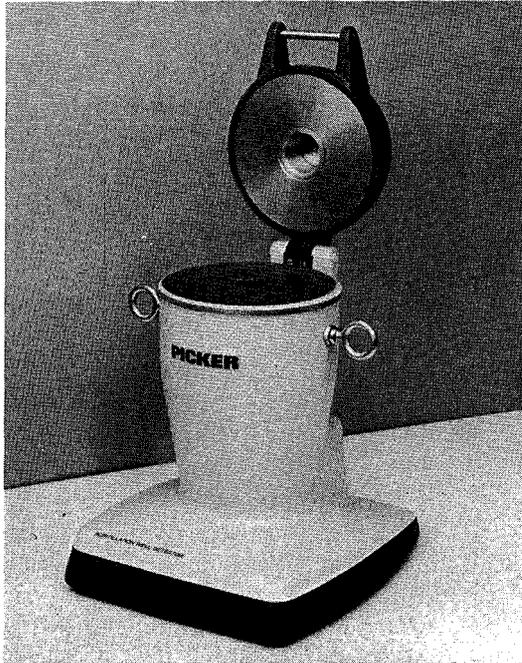
- Features
- (1) Price includes: (a) Si(Li) crystal
  - (b) Cryogenic Dynamic Charge Restoration Preamplifier
  - (c) Cryostat with entrance window
  - (d) Liquid nitrogen dewar
  - (2) Figure shows effect of different beryllium windows and Si(Li) thicknesses.

References           Manufacturer's Specifications

Address               ORTEC, Inc.  
                          100 Midland Road  
                          Oak Ridge, TN 37830  
                          (615) 482-4411



NaI(Tl) Well-Type Spectrometers  
 Picker Nuclear



Cross Section view of the 621-940 Scintillation Well Detector.

**Class** Laboratory Spectrometers

**Principle of Operation** NaI(Tl) well-type crystals with photomultiplier tubes and associated electronics, for gamma spectroscopy.

**Specifications**

Model Number	NaI(Tl) Crystal Size		Sample Volumes Accepted (cc)	Typical <sup>125</sup> I Background Window 20-40 keV	Price
	Diameter	Thickness			
621-940	2"	2"	15	70 CPM	
621-941	1.75"	2"	5	55 CPM	

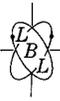
Dimensions: 30.5 cm x 33.0 cm x 41.2 cm  
 (12" W x 13" D x 16 1/2" H)  
 Net Weight: 81 kg (180 lbs)

**Features** Auxiliary accessories also available.  
 Shielding of at least 2" lead in all models.  
 Other models, automatic systems also available.

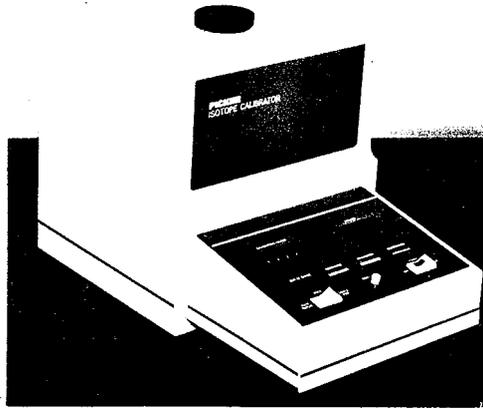
**References** Manufacturer's Specifications

**Address** Picker Corporation  
 333 North State Street  
 North Haven, CT 06473  
 (203) 484-2711





Isotope Calibrator  
Picker Model 632-507



Class	Stationary, Laboratory
Principle of Operation	Pressurized, well-type, argon filled, ion chamber, electrometer, dose computing circuit, digital display, calibration
Sensitivity and Range	Energy Range: Gammas from 25 KeV to 3 MeV Activity Range: 1 $\mu$ Ci to 999 mCi
Performance	Accuracy: $\leq \pm 5\%$ for any isotope Temperature: 12°C to 45°C Measurement Time: < 1 sec for samples above 200 $\mu$ Ci Repeatability: $\leq \pm 3\%$ for 99m Tc above 100 $\mu$ Ci Stability: $\leq \pm 1\%$ drift for $\pm 10\%$ line voltage variation
Requirements	Power: 115 Vac, $\pm 10\%$ , 60 Hz (optional 230 Vac, 50 Hz) 28 watts Size: 30.4cm H $\times$ 27.9cm W $\times$ 43.8cm D (12" $\times$ 11" $\times$ 17.3") Weight: 21.8 kg (48 lb)
Features	Calibration factor feature allows selection of the following radioisotopes and a variety of syringe and serum vial geometries. $^{18}\text{F}$ , $^{22}\text{Na}$ , $^{24}\text{Na}$ , $^{32}\text{P}$ , $^{51}\text{Cr}$ , $^{57}\text{Co}$ , $^{60}\text{Co}$ , $^{59}\text{Fe}$ , $^{67}\text{Ga}$ , $^{68}\text{Ga}$ , $^{75}\text{Se}$ , $^{85}\text{Sr}$ , $^{87\text{m}}\text{Sr}$ , $^{99\text{m}}\text{Mo}$ , $^{99\text{m}}\text{Tc}$ , $^{113\text{m}}\text{In}$ , $^{125}\text{I}$ , $^{131}\text{I}$ , $^{133}\text{Xe}$ , $^{137}\text{Cs}$ , $^{197}\text{Hg}$ , $^{203}\text{Hg}$ , $^{198}\text{Au}$ , $^{226}\text{Ra}$
References	Manufacturer's specifications
Cost	
Address	Picker Corporation 333 State Street North Haven, CN 06473 (203) 484-2711  Picker Corporation Marketing Services 6119 Highland Road Cleveland, OH Attention: Mr. Frank Dean





Spectrometer Components  
 Princeton Gamma Tech



Class Laboratory Spectrometers  
 Principle of Operation Ge(Li) and Intrinsic Germanium detectors, Amplifiers and other electronics, Cryostats and accessories  
 Specifications Ge(Li) Detectors

Geometry	Range of Efficiency Relative to 3 x 3 NaI(Tl) at 1.33 MeV	Energy Resolution (keV)		Peak/Compton Ratio at 1.33 MeV	Price
		1.33 MeV	7.6 MeV		
Modified Coaxial	5 to 26	1.7 to 2.5	4.5 to 6.0	20 to 50	\$5000 to \$28,000
True Coaxial	5 to 16	1.8 to 2.6	5.5 to 6.0	16 to 40	\$6000 to \$19,000

Standard Grade Coaxial Intrinsic Germanium Detectors,  
 Straight Dipstick Cryostat, Pre-amplifier, 17-Liter Dewar

Area (mm <sup>2</sup> )	Resolution at 5.9 keV	Resolution at 122 keV	Be Window Thickness	Price
25	155 eV	485 eV	.0013"	\$6,300
25	145	480	.0013"	7,300
50	175	490	.0013"	6,800
100	190	495	.0013"	7,900
200	240	540	.0023"	8,800
300	300	570	.006"	9,600
400	350	610	.006"	10,300
500	400	650	.006"	11,300
1000	600	800	.006"	13,300



Semi-Planar Ge(Li) Detectors, SP-Series

Model	Active Area (cm <sup>2</sup> )	Drift Depth (mm)	Energy Resolution (eV, FWHM, at 122 KeV)	Preamp Type	Price
LGSP10R	10	10	1000	RT	\$3,400
LGSP10C	10	10	750	CF	3,800
LGSP15R	15	10	1050	RT	4,400
LGSP15C	15	10	800	CF	4,800

Other sizes also available

Extra charges for other cryostats:

Model ED, horizontal-looking dipstick \$100.  
Model HT, horizontal-looking unitary (15-liter) \$250.  
Model DT, down-looking unitary (15-liter) \$250.

Features Many variations in full systems available.  
Special detector specifications available on request.

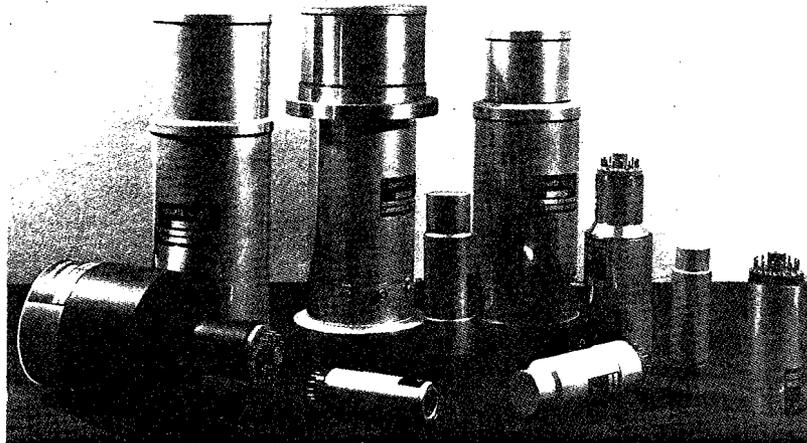
References Manufacturer's Specifications

Address Princeton Gamma-Tech  
Box 641  
Princeton, NJ 08540  
(609) 924-7310



Gamma Detectors

Quartz Products



**FOUR SEGMENT  
WELL-TYPE SCINTIFLEX**

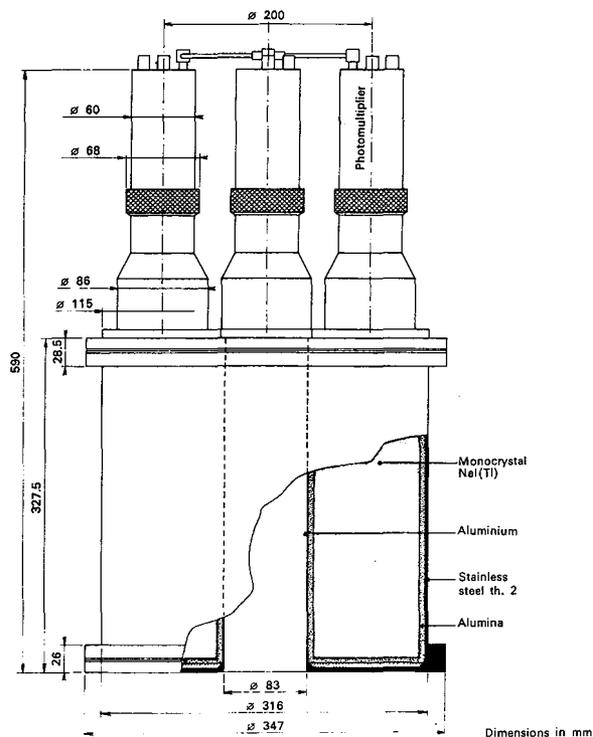
305 YPFE 305

for pair spectrometry operable as an  
anti-Compton annulus by series  
connexion of segments

Monocrystal NaI(Tl):  $\varnothing$  305 mm  
h 305 mm

4 Photomultipliers XP 1031 or RCA 8054

Well:  $\varnothing$  eff. 83 mm.





Class Laboratory

Principle of Operation NaI(Tl) scintillators and photomultipliers

Sensitivity and Range Energy Range: 10 keV to 3 MeV, gammas

Performance Accuracy: About 9%  
Temperature: -50°C to 50°C  
Calibration: <sup>137</sup>Cs resolution, <sup>60</sup>Co peak-to-valley ratio, <sup>55</sup>Fe for X-rays

Requirements Scintibloc Crystals (Solids & Well Types)

Diameter mm:	32	38	44	51	76	76	127
Height mm:	25	25	51	51	51	76	51
Energy							
Resolution:	7 - 9% for Cs-137 (661 keV)						
Cost:	\$137.- \$1194.						

Scintibloc X Crystals

Diameter mm:	25	32	44	76 to 250
Thickness mm:	1 to 5	1 to 5	1 to 5	1 to 5 mm
Energy				
Resolution:	45 - 60% for Fe-55 (220 keV)			
Cost:	\$265. - \$1194.			

Scintibloc and Scintiflex Crystals (Solids and well-types)

Diameter mm:	102	102	127	127	127	152	203	228	292
Height mm:	76	102	76	102	127	102	102	102	102
Energy									
Resolution:	7.5 - 9.5% for CS-137 (661 keV)								
Cost:	\$1159. - \$8500.								

Note: Scintibloc refers to Direct Coupling  
Scintiflex refers to Demountable Phototube

Features

1. Anti-Compton Annuli (Pair Spectrometer Annulus Option) from 8' x 8' to 12' x 12'
2. Phoswich detector (NaI[Tl]/CsI[Tl]) for Plutonium studies
3. Crystals up to 16" diameter

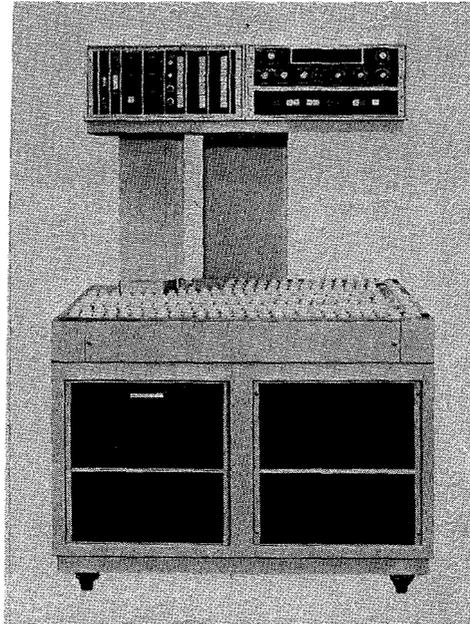
Other scintillating CsI (Tl), CsI (Na), Anthracene, Stilbene, ZnS (Ag) materials.

References Manufacturer's specifications

Address Quartz Products Corporation  
688 Somerset Street  
P.O. Box 628  
Plainfield, NJ 07061  
(201) 757-4545



Automatic Gamma Counter  
Seale Analytic Model 1185

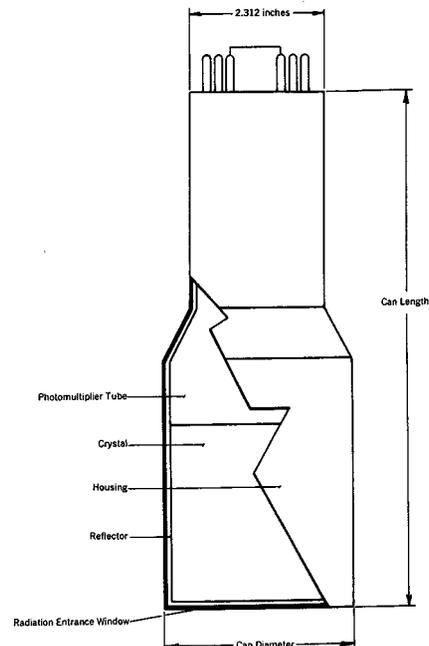
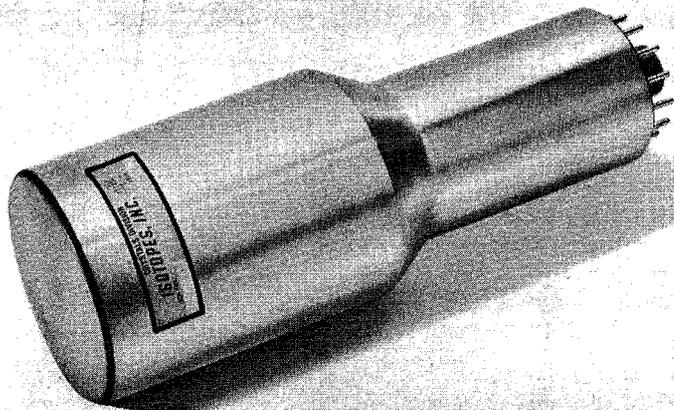


Class	Laboratory
Principle of Operation	A well-type scintillation detector and lead shield and a NaI(Tl) crystal with nearly $4\pi$ detection geometry
Sensitivity and Range	Sensitivity expressed in terms of $E^2/b$ ratio is at least 200 with a 2" crystal Dynamic Range - 100:1
Sampling	300 sample capacity, counted individually
Performance	Typical efficiency is 80% with 25 CPM background when a 2" crystal is used Temperature:
Requirements	Power: 115V $\pm$ 10V, 60 HZ or 230 V 50 HZ Size: 162.6 cm H $\times$ 111.8 cm W $\times$ 85.1 cm D (64" H $\times$ 44" W $\times$ 33.5" D) Weight: 907.2 kg (2000 pounds)
Features	Automatic background subtraction, two or three analysis channels with both preset and adjustable windows, automatic calibration, automatic correction for spillover in dual labeled experiments, ratemeter. RIA data reduction available
References	Manufacturer's Specifications
Cost	Not known
Address	Searle Analytic, Inc. 2000 Nuclear Drive Des Plaines, IL 60018 (312) 298-6600





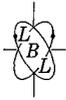
NaI(Tl) Gamma Spectrometry Systems  
Teledyne Isotopes



Typical Assembly

- |                        |  |
|------------------------|--|
| Class                  | Laboratory   |
| Principle of Operation | NaI(Tl) crystals, photomultipliers, mounts, and assembled systems for photon spectrometry.   |
| Specifications         | <p><u>NaI(Tl) Crystals</u></p> <ol style="list-style-type: none"> <li>(1) Wide variety of standard cylindrical crystals, from 1"(diam) × 1/2" (length) to 11-1/2" × 4".</li> <li>(2) Well shapes and thin crystals also available.</li> <li>(3) Table shows a few representative crystals:</li> <li>(4) <u>Response Time</u> can be pre-determined to satisfy customer needs through the use of specific Photomultiplier tubes.</li> </ol> |

Designation	Crystal Diameter		Crystal Thickness		Crystal Price
	(cm)	(in.)	(cm)	(in.)	
S-42	2.54	1"	1.27	1/2"	
S-44	2.54	1"	2.54	1"	
S-64	3.81	1-1/2"	2.54	1"	
S-88	5.1	2"	5.1	2"	
S-1212	7.6	3"	7.6	3"	
S-1616	10.2	4"	10.2	4"	
S-2020	12.7	5"	12.7	5"	
S-3216	20.3	8"	10.2	4"	
S-4616	29.2	11-1/2"	10.2	4"	
ST-41	2.52	1"		1 mm	
ST-46	2.54	1"		6 mm	
ST-81	5.1	2"		1 mm	
ST-86	5.1	2"		6 mm	



Specifications  
(Continued)

Assemblies

- (1) Many types of assemblies available.
- (2) All "Integral Series" assemblies have photomultiplier attached to NaI(Tl), all hermetically sealed.
- (3) Magnetic shield included.
- (4) Tube base available as extra option.
- (5) Aluminum housing (0.020") standard: copper or stainless steel also available.
- (6) Phototubes used are DuMont 3240 and 3248L, RCA 6342AV1, 2060, EMI 9778, 9578B and 4524.
- (7) Well-type and other geometries also available.

Designation	Crystal Diameter	Crystal Thickness	Price
S-64-I	1-1/2"	1"	
S-78-I	1-3/4"	2"	
S-84-I	2"	1"	
S-88-I	2"	2"	
S-1208-I	3"	2"	
S-1212-I	3"	3"	

References

Manufacturer's Specifications

Address

Teledyne Isotopes  
50 Van Buren Avenue  
Westwood, NJ 07675  
(201) 664-7070



NaI(Tl) Spectrometer Systems

Tennelec, Inc.

Class Laboratory Spectrometers  
Principle of Operation NaI(Tl) crystal, photomultiplier tube for gamma spectroscopy.  
Specifications

<u>Crystal Size</u>		<u>Performance</u>		
Diameter	Thickness	Model Type	Data	Photomultiplier Tube
				Price
<u>Thin Window NaI(Tl) X-Ray Detectors</u>				
( <sup>55</sup> Fe Peak/Valley Ratio, 5.9 keV)				
25mm × 1.5mm (1" × 0.2")		TD 25 SX 1/5	15:1	Philips XP1010 \$ 430
32mm × 1.5mm (1.25" × 0.2")		TD 32 SX 1/5	15:1	Philips XP1010 \$ 500
44mm × 1.5mm (1.75" × 0.2")		TD 44 SX 1/5	15:1	Philips 153 AVP 02 \$ 600
<u>Standard NaI(Tl) Assemblies</u>				
( <sup>137</sup> Cs Energy Resolution, 662 keV)				
25mm × 25mm (1" × 1")		TD 25 S 25	8.5%	RCA 6199 \$ 140
32mm × 25mm (1.25" × 1")		TD 32 S 25	8.5%	RCA 6199 \$ 150
38mm × 25mm (1.5" × 1")		TD 38 S 25	8.5%	RCA 6342A \$ 165
44mm × 51mm (1.75" × 2")		TD 44 S 51	8.5%	RCA 6342A \$ 270
51mm × 51mm (2" × 2")		TD 51 S 51	8.5%	RCA 6342A \$ 320
76mm × 51mm (3" × 2")		TD 76 S 51	8.5%	RCA 8054 \$ 700
76mm × 76mm (3" × 3")		TD 76 S 76	8.5%	RCA 8054 \$ 850
102mm × 102mm (4" × 4")		TD 102 SE 102	9.0%	RCA 8055 \$1,490
127mm × 127mm (5" × 5")		TD 127 SE 127	10.0%	RCA 8055 \$2,290
<u>Well Type Assemblies</u>				
( <sup>137</sup> Cs Energy Resolution, 662 keV)				
44mm × 51mm (1.75" × 2")		TD 44-SP-51	9.0%	RCA 6342A \$ 330
51mm × 51mm (2" × 2")		TD 51-SP-51	9.0%	RCA 6342A \$ 390
76mm × 76mm (3" × 3")		TD 76-SP-76	9.0%	RCA 8054 \$ 990
102mm × 102mm (4" × 4")		TD 102-SPE-102	10.0%	RCA 8055 \$1,645
127mm × 127mm (5" × 5")		TD 127-SPE-127	11.0%	RCA 8055 \$2,645

Features Electronic accessories also available. Individual NaI(Tl) crystals also available. All assemblies have magnetic shielding, hermetic seals.

References Manufacturer's Specifications

Address Tennelec, Inc.  
P.O. Box D  
Oak Ridge, TN 37830  
(615) 483-8405



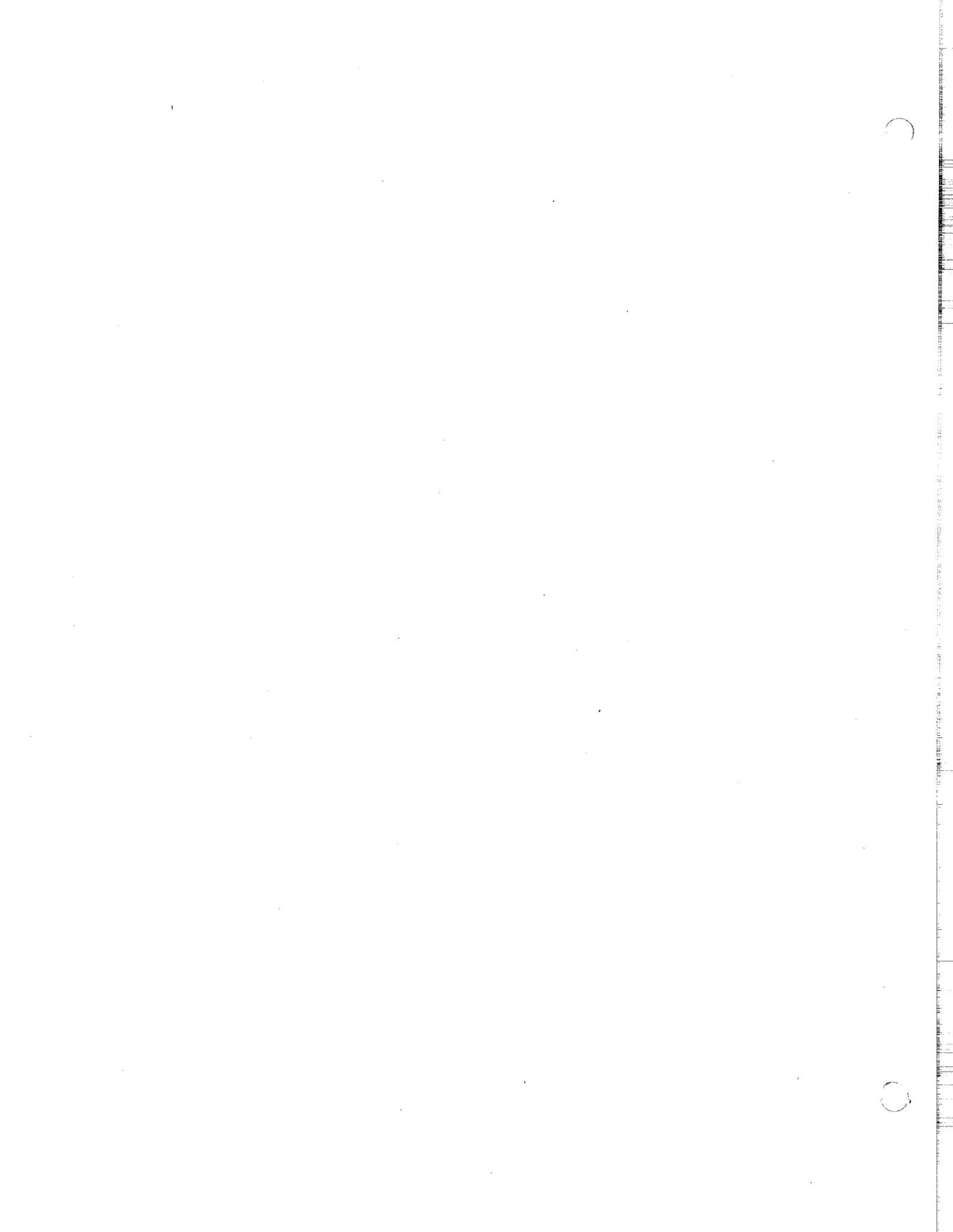
SECRET





NEUTRON MONITORING INSTRUMENTATION

1. Introduction
2. Radiation Protection Considerations
3. Measurement Considerations
  - A. Thermal Neutrons
  - B. Intermediate-Energy Neutrons
  - C. Fast Neutrons
  - D. Relativistic Neutrons
  - E. Dose-Equivalent Instruments
4. Sources for Calibration of Neutron Instruments
5. Summary and Recommendations
6. Acknowledgement
7. References
8. Instrument Notes





## 1. INTRODUCTION

There are a large number and variety of applications where a significant amount of radiation is due to the presence of neutrons. Among these are:

- a. charged particle accelerators, particularly proton and ion accelerators and high-energy accelerators;
- b. nuclear reactors;
- c. sealed neutron sources.

These sources vary widely in the intensities, energy spectra, and temporal durations of the neutron fluences involved. From the highest-energy proton accelerators, primary beam interactions can produce neutrons with energies up to the full energy of the primary protons themselves, while in reactor situations there can be a heavy weighting toward thermal neutrons. Given this wide energy range, it is remarkable that one can give any unified treatment to the general problems of measurement of neutron fluences. However, a unification is indeed possible, for a large class of applications, primarily because of the characteristics of neutron interactions in shielding materials.

Neutrons traversing through material can interact in one of five general ways:

- a. elastic scattering from nuclei;
- b. inelastic scattering through nuclear-level excitation;
- c. quasi-two-body interactions such as  $(n,\alpha)$  or  $(n,np)$  interactions;
- d. nuclear excitation or catastrophic break-up;
- e. the high-energy phenomena of many-particle production.

The relative importance of each of the five general types depends on both neutron energy and target material. Thus, elastic scattering from heavy nuclei is a very inefficient energy loss mechanism, as is easily seen from kinematical considerations. However, elastic scattering from hydrogen nuclei (protons) is one of the most useful

processes in degrading neutron energies in shielding (hence, the common use of concrete). Also, at energies much higher than those of typical nuclear levels (a few tens of MeV, say) a major energy loss mechanism is nuclear break-up or (above a few hundred MeV) multiparticle interactions, such as pion production. These facts have long been well known, and have influenced shielding design around major sources of neutrons and other particles. The unification of treatment mentioned above is primarily due to the effectiveness of energy-degradation processes in appropriately designed shielding, such that even at the highest energy accelerators the major components of neutron fluence are almost never at energies above a few tens of MeV.

Neutron interactions, then, generally tend to degrade energies continually, down to the region where the neutrons can be absorbed and eliminated by nuclear reactions. The ultimate energy degradation is neutron 'thermalization'; thermal neutrons are those with an energy spectrum in thermal equilibrium with the ambient material. Thermal neutrons can scatter without energy loss in, for example, air or concrete for relatively indefinite periods; while thermalization of a several-MeV neutron can take from a few to a few hundred microseconds, thermal-neutron lifetimes themselves can be as long as a few tens of milliseconds, until they are ultimately eliminated by nuclear absorption or reactions such as the  $(n,\alpha)$  mentioned above.

Historically, it has been common to divide the neutron energy spectrum into the following approximate categories:

Thermal neutrons	0 to 0.5 eV
Intermediate energy neutrons	0.5 eV to 200 keV
Fast neutrons	200 keV to 20 MeV
Relativistic neutrons	>20 MeV

This division is convenient both because of the different types of processes which are typical of these energy regions, and because instrumentation is often sensitive in only one of the categories. We shall discuss instrumentation partially according to these divisions, noting that in many applications one has a priori knowledge



that the neutron energy extends up to only a particular energy region.

For detailed information about techniques of neutron dosimetry, the reader is referred to several excellent papers and conference proceedings. (Ref. 1,2,3,4,5,6)

## 2. RADIATION PROTECTION CONSIDERATIONS

In the general case of occupational exposure to external radiation, the relevant Maximum Permissible Exposure guideline (discussed in more detail in the introductory part of this volume) is a basic limit to dose-equivalent of 5 rem/year, with no more than 3 rem to be accumulated in any 13-week quarter and no lifetime accumulation up to age N years in excess of 5 (N-18) rem.

The complication in measuring neutron rem (absorbed-dose equivalent) is the variation of quality factor (QF) with energy. Figure 1 shows the ICRP recommended QF for neutrons as a function of energy. Figure 2 gives the flux density of primary neutrons, in equilibrium with their secondaries, which is recommended by ICRP as being equivalent to 2.5 mrem/hour (Ref. 7). The inverse curve (Figure 3) is the energy response required of a neutron detector such that different energies are weighted properly for a direct determination of dose-equivalent (rem).

## 3. MEASUREMENT CONSIDERATIONS

There are three main tasks in neutron-monitoring:

1. Measurement of the intensity spectrum as a function of energy (neutrons  $\text{cm}^{-2} \text{MeV}^{-1} \text{sec}^{-1}$ ).
2. Measurement of the total absorbed-dose equivalent (rem or rem/sec).
3. Determination of the level of other possible particle species, and the response of neutron instrumentation to them.

There does not exist a single instrument or instrument type which can perform completely all of these functions (or indeed, any one of them separately). However, there does exist instrumentation more or less

adequate to the needs of the health physicist responsible for radiation protection around any of the sources mentioned above. The wide range of energies and fluences previously mentioned must be considered, along with several other concepts such as tissue-equivalence and particle equilibrium, which have been treated in the introductory section of this volume.

### A. Thermal Neutron Measurements (below 0.5 eV)

Truly thermal neutrons, which have an energy spectrum in thermal equilibrium with the ambient environment, are centered at an energy of about 0.025 eV. However, in actual practice, very few situations exist in which the neutron fluence is solely thermal: one possible case is that in which neutrons are almost totally thermalized for some application such as a thermal column. Many detection systems for higher-energy neutrons depend upon the process of (partial or complete) thermalization, with subsequent detection of the thermal neutrons using one of the techniques to be discussed in this section.

There is usually both a truly thermal (Maxwellian) component and a component with a continuous energy spectrum above thermal, typically with a ("epi-Maxwellian") distribution inversely proportional to energy. Also, most of the reactions used to determine thermal neutron flux densities have cross-sections which are almost exactly proportional to 1/velocity in the range from thermal up to the cadmium cut-off energy. (Since a cadmium absorber prevents neutrons below about 0.5 eV from reaching a detector, it allows a convenient separation at about that energy.) Because flux density (neutrons  $\text{cm}^{-2} \text{sec}^{-1}$ ) is proportional to velocity (if they are going twice as fast, they cross any area in half the time), the reaction rate per atom for a pure 1/v detector is independent of velocity and proportional only to the total neutron density. Table 1 (from Ref. 5) shows most of the common thermal neutron reactions; the coefficient  $g(20^\circ\text{C})$  in the right hand column indicates the departure from 1/v dependence for each reaction.

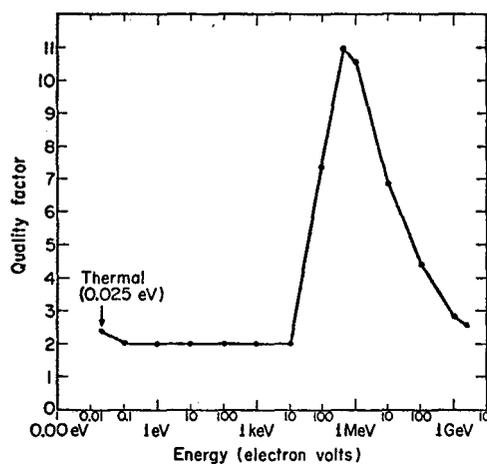
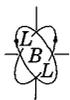


FIGURE 1: QUALITY FACTOR vs. ENERGY for monoenergetic neutrons, as recommended in ICRU Report No. 20.

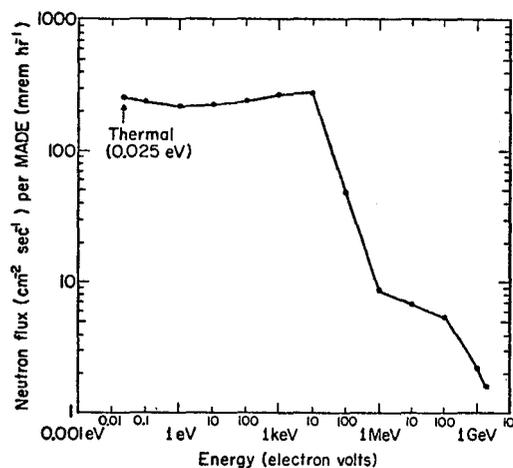


FIGURE 2: MONOENERGETIC NEUTRON FLUX vs. ENERGY which yields MADE of 1 mrem/hour.

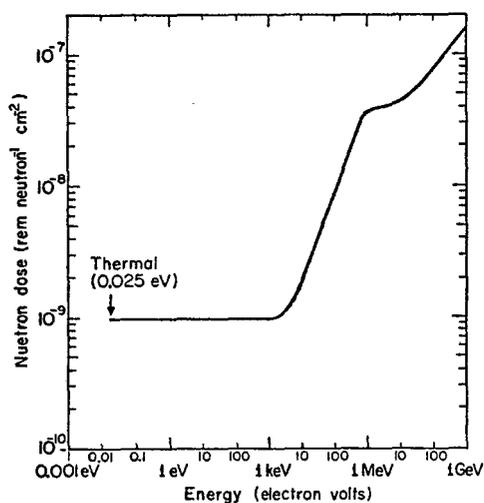


FIGURE 3: NEUTRON DOSE (rem) vs. ENERGY for 1 neutron/cm² (for monoenergetic neutrons incident normally on a 30 cm-thick tissue-equivalent phantom: ICRU Report No. 20).



Table 1

Useful Thermal Neutron Detectors

Nuclide or Element	Reaction	Half-Life of Nuclide Produced	$\sigma_0 \times 10^{24}$ for Production $\text{cm}^2$	g(20°C)
$^3\text{He}$	$(n, p)^3\text{H}$	12.3 y	5327	1.0
$^6\text{Li}$	$(n, t)^4\text{He}$	stable	945	1.0
$^{10}\text{B}$	$(n, \alpha)^7\text{Li}$	stable	3837	1.0
$^{23}\text{Na}$ (IS)	$(n, \gamma)^{24}\text{Na}$	15.0 h	0.534	1.0
$^{45}\text{Sc}$ (IS)	$(n, \gamma)^{46}\text{Sc}$	85d	22.3	
$^{51}\text{V}$	$(n, \gamma)^{52}\text{V}$	3.8 min	4.9	1.0
$^{55}\text{Mn}$ (IS)	$(n, \gamma)^{56}\text{Mn}$	2.58h	13.3	1.0
$^{59}\text{Co}$ (IS)	$(n, \gamma)^{60}\text{Co}$	5.24y	36.6	1.0
$^{63}\text{Cu}$	$(n, \gamma)^{64}\text{Cu}$	12.8h	4.5	1.0
$^{115}\text{In}$	$(n, \gamma)^{116}\text{In}^m$	54 min	157	1.019
$^{157}\text{Gd}$	$(n, \gamma)^{158}\text{Gd}$	stable	242000	0.854
$^{197}\text{Au}$ (IS)	$(n, \gamma)^{198}\text{Au}$	2.70d	98.8	1.005
$^{235}\text{U}$	fission	many	577	0.976

Notes: (IS) denotes natural monoisotopic element.

For a more detailed discussion of thermal neutron spectra, departures from Maxwellian distributions, and neutron temperature the reader is referred to ICRU Report No. 13 (Ref.5).

The most commonly used thermal neutron detector is the  $\text{BF}_3$  filled gas proportional counter, which is commercially available in a wide variety of sizes and sensitivities. The reaction employed is  $^{10}\text{B}(n, \alpha)^7\text{Li}$ , in which the alphas are detected. The Q (kinetic energy released) is 2.8 MeV, but most of the capture goes through an intermediate excited state of  $^7\text{Li}$ , which then decays to the ground state by emission of a 470 keV gamma. In most counters the  $\text{BF}_3$  gas is enriched with boron-10 to about 96%. (Natural abundance is 20%  $^{10}\text{B}$  and 80%  $^{11}\text{B}$ ). The cross-section of the above reaction is about 3800 barns with a  $1/v$  dependence in the thermal-energy range. Thus  $\text{BF}_3$  counters are relatively insensitive to particles with energies significantly above that of thermal neutrons (0.025 eV). The sensitivity depends on counter size, gas pressure, and enrichment. The response to gamma fields is very low. In fields of equal gamma and neutron fluxes, the gamma

response is typically two orders of magnitude smaller than the neutron response (Ref.1). An exception is when the  $\text{BF}_3$  detector is inside a moderator, and pulsed fields are being detected (such as near a low duty-cycle accelerator). In that case, photons can pile up and count.

The  $^3\text{He}$  gas proportional counter is another commonly used detector, similar to the  $\text{BF}_3$  counter. It relies upon the reaction  $^3\text{He}(n, p)^3\text{H}$ , with a Q value of 765 keV. Both the proton and the triton ( $^3\text{H}$ ) are available for ionization, but one of the problems is that near the walls efficiency can suffer, due to the rather low stopping power of the  $^3\text{He}$  gas. This deficiency can be overcome either by increasing the volume or gas pressure, or (more commonly) by adding an impurity (usually argon or krypton) to improve the stopping power. The disadvantage of the impurity is that gamma sensitivity is increased, while pulse rise-time is decreased. Because of the small Q value, gamma sensitivity is greater than in the  $\text{BF}_3$ -filled counter, but neutron sensitivity is also greater (Ref.1), until the recoil particles become too long (at neutron energies above about 10 MeV).



For neutron detection in high proton fields,  $^{10}\text{B}$ -lined proportional counters are used. The reaction is the same as for  $\text{BF}_3$  counters, except that the  $^{10}\text{B}$  is coated on the counter walls, in thicknesses that range from about 0.2 to 1.0 mg/cm<sup>2</sup>. Sensitivity is proportional to coating thickness, but self-absorption of the alphas prevents significant increases in sensitivity at greater thicknesses. These counters are extremely insensitive to photons, and are capable of neutron-sensitive operation in photon fields as high as  $10^4$  or even  $10^5$  R/hr. However, their sensitivity to neutrons is perhaps an order of magnitude smaller than that of the  $\text{BF}_3$  counter (Ref. 1). A typical fill gas is  $\text{CO}_2$  or Argon. The operating voltage is usually lower than that of the  $\text{BF}_3$  or  $^3\text{He}$  tubes, and typically larger output pulses with shorter rise times are produced.

Still another gas proportional neutron detector is the fission counter. A material such as  $^{233}\text{U}$ ,  $^{235}\text{U}$ , or  $^{239}\text{Pu}$  is coated on the walls. The kinetic energy of the fission fragments (approximately 200 MeV) produces large pulses and fast response time. The sensitivity of these counters is inherently lower by perhaps an order of magnitude than the  $^{10}\text{B}$  lined counters, and two orders of magnitude below that of the  $\text{BF}_3$  tubes (Ref. 1). However, operation at temperatures as high as 700°C and in extremely high gamma fluxes (up to  $10^6$  R/hr) makes them attractive for applications such as inside reactor shielding. It should be noted here that special licensing is often required to obtain the fissionable material within the counter.

Ionization chambers are also used for thermal neutron detection. Both  $^{10}\text{B}$ -lined and  $\text{BF}_3$ -filled ion chambers are used. Many of the same characteristics apply which have already been discussed in the case of the gas-proportional counters. The ion chambers are generally less sensitive than the gas proportional counters (Ref. 1). A fuller treatment of ionization chambers can be found below.

Scintillation counters (in which photomultiplier tubes detect the light produced by ionization in scintillator) are also used for detection of thermal neutrons (Ref. 1). Among the most sensitive are a lithium-iodide crystal with europium activation [ $^6\text{LiI}(\text{Eu})$ ];  $^6\text{Li}$ -loaded, cerium-activated silicate glass; and  $^6\text{Li}$ -loaded, zinc-sulfide crystals. In each case one counts the alpha and triton from the reaction  $^6\text{Li}(n,\alpha)^3\text{H}$ , with a Q of 4.8 MeV. If the detector is large in size there can be a significant perturbation of the field being measured. However, the gamma rejection is excellent, with gamma sensitivities typically one to two orders-of-magnitude below those for neutrons. The crystal itself (and the others as well) have negligible anisotropy of sensitivity, except for self-shielding and the flux perturbations already mentioned. A description of some of the properties of cerium-activated glass has been given by Ginther (Ref. 8); and a discussion of  $^6\text{Li}$ -loaded glasses is given by Bormann et al (Ref. 9).

Another technique is the activation of a NaI (Tl) crystal, in which one counts the decay after activation of  $^{24}\text{Na}$  (15 hour half-life,  $\beta$  with  $E_{\text{max}} = 2.12$  MeV). An advantage of this technique is the good detection efficiency for the decay betas. The sensitivity is very low, perhaps  $10^4$  below that of the others previously mentioned. Although common at one time, this technique is not often used today. Small crystals (say, 1 mm thick) must be used to minimize the problem of perturbation of the incident flux. For a good description of the NaI method, the reader is referred to Grimeland (Ref. 10).

Zinc sulfide glass with  $^{10}\text{B}$  loading is another material used as a scintillator; however, it suffers from two problems (Ref. 1) when compared to its counterpart,  $^6\text{Li}$ -loaded ZnS: first, poorer gamma discrimination, because of the lower Q value of the (n, $\alpha$ ) reaction; and second, typical sensitivities several orders of magnitude lower than for the lithium-loaded crystals. Thus, only in situations where neutron fluences are high and photon levels are known to be small can the  $^{10}\text{B}$ -loaded ZnS be useful.



Foil Activation is another common technique for detection of thermal neutrons. Table 1 above showed the properties of some of the useful reactions. Perhaps the most widely used foil material when high sensitivity is required is indium, due to its very large thermal neutron cross-section. Use of cobalt is common for the integration of neutron fluence over very long periods, because of the long (5.2 year) half-life of  $^{60}\text{Co}$ . Gold is used when high precision is required, since its capture cross-section is well-behaved for energies up to about 3 ev.

Thermoluminescence is still another detection technique for thermal neutrons (Ref. 11). The most commonly used thermoluminescence detector (TLD) for neutrons is  $^6\text{LiF}$ , but this detector also has substantial gamma sensitivity (Ref. 11). Therefore, it is common practice when gamma backgrounds are present to use two or three TLD's with varying fractional concentrations of  $^6\text{Li}$  and  $^7\text{Li}$ , in order to determine the gamma and neutron contributions separately. TLD's are integrating devices which must be read out in a separate operation after exposure. The general discussion of TLD's is found in the section entitled "Personal Dosimetry." The sensitivity of  $^6\text{LiF}$  is typically above 1000 neutrons/cm<sup>2</sup>, and the sensitivity to fast neutrons is small. Another common TLD device is  $\text{CaF}_2$ , but it has very small intrinsic neutron sensitivity and large gamma response; addition of  $^{10}\text{B}$  or  $^6\text{Li}$  is used to enhance the neutron response, but with the exception of pure neutron fields it is not recommended.

To summarize the discussion on thermal neutrons, it can be seen that there are many methods used for detection; however, complications can arise when higher-energy neutrons are present, since most thermal neutron detectors have at least some sensitivity to higher energies. When one suspects the presence of higher-energy fluences, a common technique to use is the "cadmium ratio" method. This involves shielding the detector with a layer of cadmium (an excellent absorber of neutrons below about 0.5 eV) and comparing detector response to the bare situation. To determine

the flux in the region near 1.5 eV, one can surround an indium foil with cadmium. The cadmium will absorb neutrons below 0.5 eV, and the indium foil (with its huge capture resonance of 26,000 barns near 1.5 eV) can be  $\beta$ -counted by any low background G-M counter to detect the beta from  $^{116}\text{In}$  decay after the reaction  $^{115}\text{In}(n,\gamma)^{116}\text{In}$ . Accuracies within better than a factor of 2 in flux measurements can be obtained. A further description of this procedure can be found in the article by Hankins (Ref. 12).

#### B. Intermediate-Energy Neutrons ( $\sim 0.5$ eV to 200 keV)

Intermediate energy neutrons are most often detected by moderating them down to the thermal region and then employing one of the thermal methods just described.

Direct detection of recoils from neutron scattering does not work well in this energy region, because it is difficult to observe the recoils against even a small gamma ray background. Instead, the most common technique is to detect the energetic products of an exothermic reaction such as  $(n,\alpha)$ . However, the cross sections for many of these reactions decrease rapidly at energies above a few eV, and thus neutrons in the keV range are usually detected by rather specialized means.

One such technique is the so-called "black detector" which absorbs all neutrons within a certain energy region. The most commonly used "black detectors" are a  $^{10}\text{B}$ -doped liquid scintillator or a detector consisting of a boron plate which uses the associated photon from the  $(n,\gamma)$  reaction for detection (Ref. 13). The problem is that these instruments absorb all neutrons below a certain maximum energy. To derive absorbed dose-equivalent from flux density involves a detailed knowledge of the energy spectrum (see Fig. 3).

The moderator approach has been used with reasonable success. Moderators can be appropriately designed to yield either rem-equivalent response or flux density response weighted to yield values of absorbed dose. An



instrument with the response of Fig. 4 has been reported by Leake (Ref. 14); however, the figure indicates a substantial fall-off in sensitivity at high and low energies, necessitating a correction for neutrons outside of the sensitive range. This device uses a  ${}^6\text{LiI}$  (Eu) crystal scintillator surrounded by polyethylene moderator. Another instrument (Ref. 15) uses a boron-loaded ZnS scintillator enclosed in a polyethylene sphere. Figure 5 (from Ref. 15) shows the energy response of this instrument for various moderator thicknesses. Note that the response can be made quite flat. Both of these instruments have excellent gamma discrimination (Ref. 1).

Instruments which cover the intermediate range in a rem-equivalent manner but are also sensitive over much broader energies will be discussed below in the section on dose-equivalent instruments.

### C. Fast Neutrons ( $\sim 200$ keV to 20 MeV)

In situations where fast neutrons require measurement, there is usually an accompanying flux of lower energy neutrons, and there may be gamma backgrounds as well. For this reason, a frequently used technique for isolating the fast-neutron component is foil activation, because one can choose processes with a variety of thresholds and sensitivities.

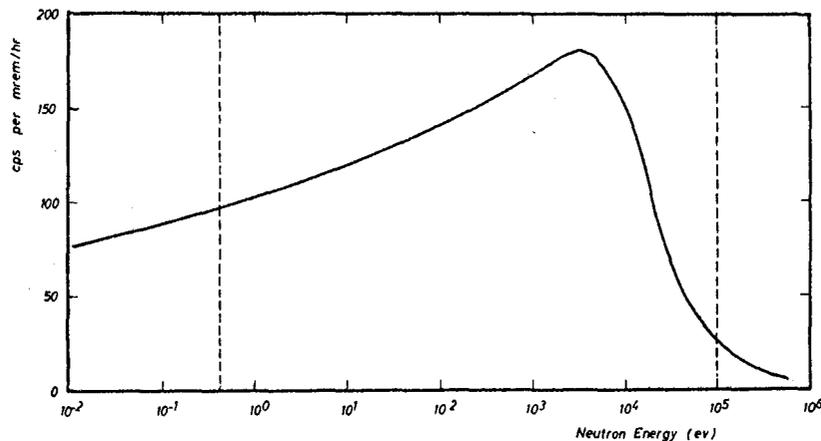


FIGURE 4: The response of a dosimeter designed to measure dose equivalent of neutrons in the intermediate energy range.

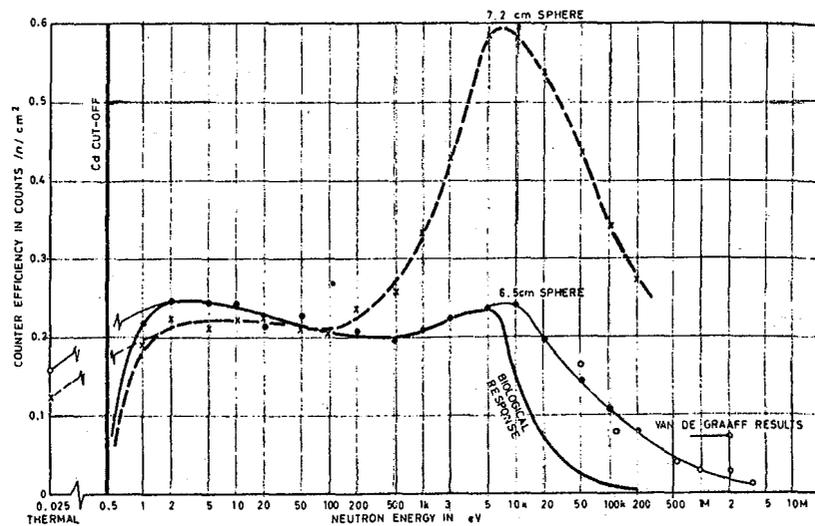
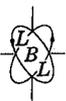


FIGURE 5: Variation of efficiency and dose dependence of the optimum arrangement with neutron energy.



Activation of foils is also used for relativistic neutrons. Figure 6 shows the cross section as a function of energy for some of the commonly used reactions (Ref. 16). Table 2 (from Ref. 17) gives some properties of the reactions, and the half-lives of the activation products. The last two columns in Table 2 show the sensitivity for 1 rad, and the background, for the detector used by Cross (Ref. 17). The sensitivity with In, Fe, and Zn is perhaps an order of magnitude better than for the fission detectors, because a decay gamma peak stands out above background to a greater extent than when counting a wide spectrum.

Gas proportional counters which rely upon the measurement of proton recoils have found widespread use for

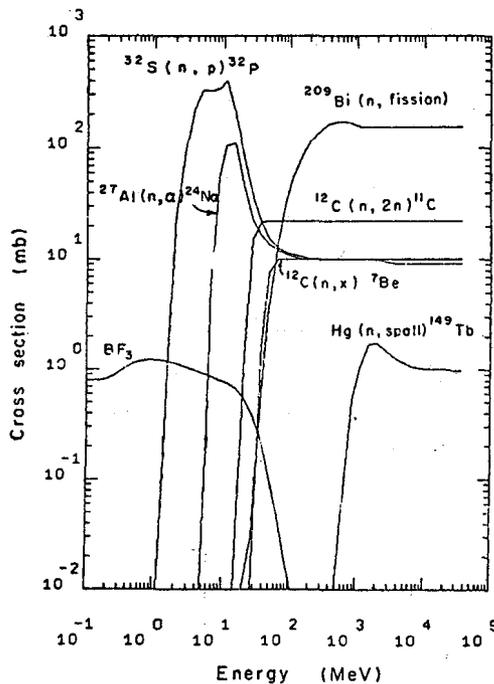


FIGURE 6: Response functions of high-energy neutron detectors.

fast neutron detection, because of their excellent gamma discrimination. Several such counters have been described in the literature. As an example, consider the counter built at Oak Ridge National Laboratory and described by Hurst and Wagner (Ref. 18). The counter, which reads out in rads, is completely lined with polyethylene ( $C_2H_4$ ) and filled with ethylene ( $C_2H_4$ ) to a pressure of 750 torr. Figure 7 (from Ref. 18) shows the response to 1.3 MeV  $Co^{60}$

Table 2  
Properties of Some Threshold Reactions

Reaction	$E_{th}$ (MeV)	$T_{1/2}$	$\bar{\sigma}$ (b)	$E_{\gamma}$ (MeV)	Sensitivity for 1 rad (cpm/g)	Background (cpm)
In <sup>115</sup> (n, n')In <sup>115m</sup>	1.2	4.5 h	0.3	0.385	121	14
Ni <sup>58</sup> (n, p)Co <sup>58</sup>	2.6	72 d	0.6	0.810	0.68	8
S <sup>32</sup> (n, p)P <sup>32</sup>	2.9	14.2 d	0.35	$\beta$	17	1
Zn <sup>64</sup> (n, p)Cu <sup>64</sup>	3.3	12.8 h	0.25	0.511	27	12
Fe <sup>56</sup> (n, p)Mn <sup>56</sup>	6.7	2.6 h	0.08	0.845	56	8
Mg <sup>24</sup> (n, p)Na <sup>24</sup>	7.2	16 h	0.15	2.76	13	1
Al <sup>27</sup> (n, $\alpha$ )Na <sup>24</sup>	7.6	15 h	0.1	2.76	9	1
In <sup>115</sup> (n, 2n)In <sup>114</sup>	10.6	50 d	1.4	0.192	0.46	17
Cu <sup>65</sup> (n, 2n)Cu <sup>64</sup>	11.4	12.8h	1.0	0.511	28	12
Ni <sup>58</sup> (n, np)Co <sup>57</sup>	12	267 d	0.7	0.122	0.58	20

$E_{th}$ : Effective threshold energy (in MeV) at which cross-section reaches 25% of  $\bar{\sigma}$ .

$T_{1/2}$ : Half life of product.

$\bar{\sigma}$ : Average cross-section (in barns) over the most sensitive energy region.

$E_{\gamma}$ : Energy of Decay  $\gamma$  (MeV)

Sensitivity: Counts/min per gram of detector for 1 rad of neutrons.

Bkgd.: Counter background (cpm).

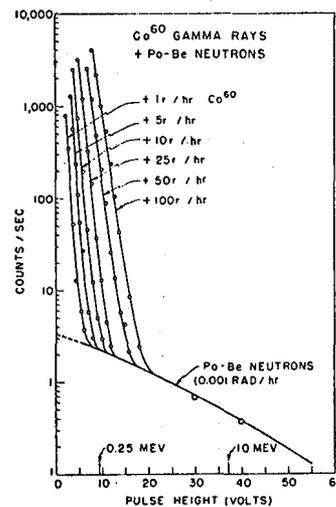


FIGURE 7: Integral pulse-height curves for mixtures of gamma-rays and fast neutrons.

gammas (at rates from 1 to 100 R/hour) and to the neutron spectrum from a Po-Be source, at rates of 0.001 rad/hour. If one is willing to set a bias level to eliminate signals from neutrons below about 0.5 MeV, the gamma discrimination is excellent.



Another counter, filled with methane-argon, has been described by Dennis and Loosemore (Ref. 19). Its energy response is shown in Fig. 8. Although this counter has sensitivity which is roughly dose-equivalent (rem) in the fast neutron region, it suffers from inaccuracies due to a loss of counts below the bias level. Theoretical techniques have been developed to estimate corrections for these losses (Ref. 20), but some independent measurements are probably required when the neutron spectrum is unknown.

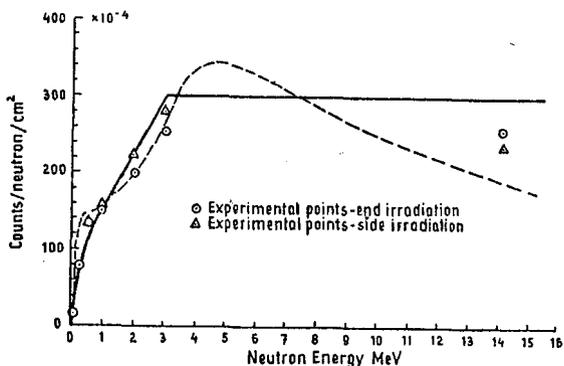


FIGURE 8: Average response of three type FN 2/3 counters as a function of neutron energy. The solid line is the response required according to the ICRP recommendations (ICRP Pub. 4) for a counting rate of 0.9c/sec/MPL. The dashed line is the theoretical response of the counter for end-on irradiation at a bias level equivalent to 0.09 MeV.

Other instruments of this type have been described by Anderson (Ref. 21) and Murthy (Ref. 22). Murthy's instrument can measure neutron fluxes up to 15,000 neutrons  $\text{cm}^{-2}\text{sec}^{-1}$ . These instruments can tolerate (Ref. 1) thermal/fast-neutron dose rate ratios of about 100/1 and gamma/fast-neutron dose rates of about 200/1.

Another fast-neutron detection technique is scintillation counting with plastic or liquid scintillators attached to a photomultiplier. Neutrons can be counted in a scintillator through their interaction and production of secondaries. The enhancement in high energy detection efficiency is accomplished by setting a rather high threshold. Figure 9 (from Ref. 23) shows the neutron detection efficiency of a typical liquid scintillator for various threshold values.

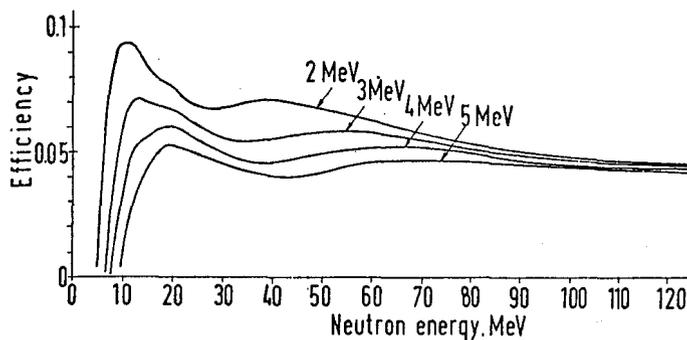


FIGURE 9: Neutron detection efficiency of a liquid scintillator, 4.7 cm x 4.7 cm, with thresholds of 2, 3, 4, and 5 MeV electron energy.

A problem with scintillation counting is that if any relativistic charged particles are present, they will count with nearly 100% efficiency. Therefore, to isolate the neutron component one must either determine that charged particles are absent or measure them separately.

Unfortunately, the photon response of some plastics, especially the organic plastics which contain significant fractions by weight of hydrogen, is also very large, making them unsuitable where gamma backgrounds are important.

Efforts have been made to overcome the problem of gamma sensitivity. The most useful method for distinguishing neutrons from gammas is pulse shape discrimination. As an example, Verbinski et al. (Ref. 24) have successfully differentiated the two species in the 1 to 10 MeV range using pulse shape discrimination on the response from NE-213 liquid scintillator manufactured by Nuclear Enterprises, Ltd. (Ref. 25). However, this technique suffers because it is delicate, and requires sophisticated electronic equipment.

Another fast neutron detector is the so-called 'Long Counter' originally developed in 1947 by Hanson and McKibben (Ref. 26), and improved in 1966 by DePangher and Nichols (Ref. 27). The Long Counter is a  $\text{BF}_3$  proportional counter surrounded by two concentric cylindrical hydrogenous moderators, with a thermal neutron shield in-between. It has high sensitivity, good gamma discrimination, and wide energy response (from about 25 keV to 14 MeV). The most significant drawback of the counter is its non-isotropy of response; it is insensitive except for neutrons incident along the axis.

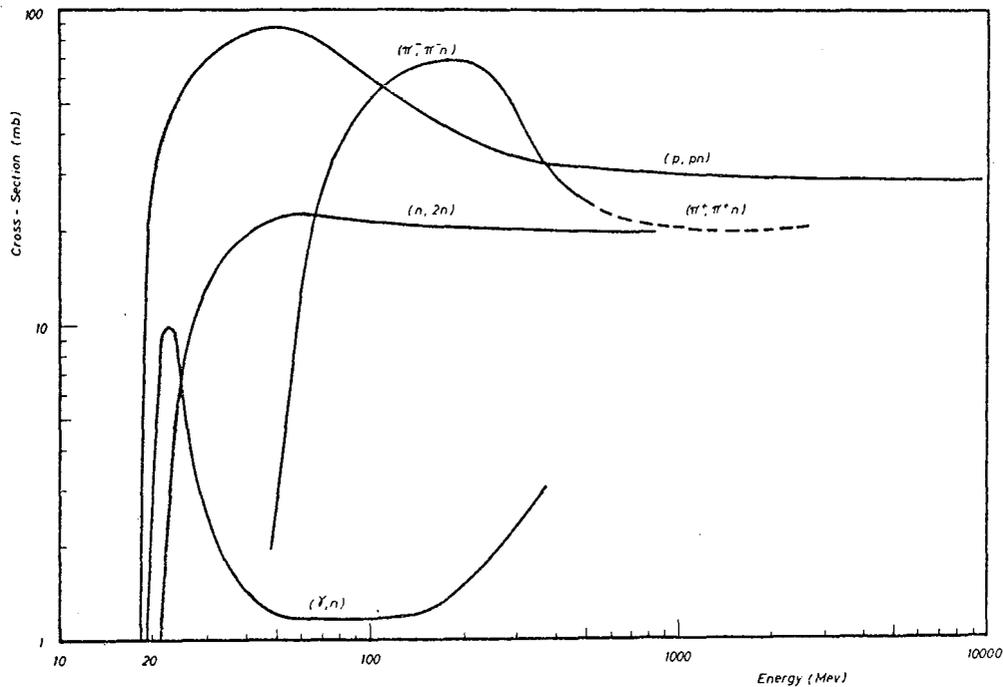
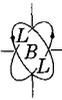


FIGURE 10: Cross-sections for the production of carbon-11 from carbon-12

As in the case of intermediate-energy neutrons, it is possible to moderate fast neutrons to thermal energies and apply any of the large number of thermal-detection schemes previously described. These instruments will be discussed in more detail in the section below on dose-equivalent rem-meters.

D. Relativistic Neutrons (Energies above 20 MeV)

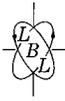
These neutrons are almost never found at levels which can pose a potential hazard to personnel except at high-energy accelerators. Carbon-11 activation (20.4 minute half-life) is one of the most widely used techniques for measuring the high-energy neutron components of a radiation field. A carbon-rich detector is exposed to the radiation field and the decay positrons (0.97 MeV  $E_{max}$ ) are then counted by standard techniques. The threshold for  $^{11}\text{C}$  excitation is about 20 MeV, and Fig. 10 (from Ref. 28) shows cross-sections for various particle types. Note that the  $(\gamma, n)$  cross-section is very small. Although the neutron cross-sectional behavior overemphasizes the region just above threshold, and high-energy charged particles can also produce  $^{11}\text{C}$ , this technique is often useful for a

rough check on the possible contribution from high-energy neutron fluences. A quantity termed the 'neutron flux density above 20 MeV' can be defined and estimated from the amount of Carbon-11 activity, assuming an average production cross-section of 22 mbarns (Ref. 28). The problem is to interpret this 'measured' quantity in terms of dose equivalent, which depends upon the assumed spectrum of neutrons. For 20 MeV neutrons, the ICRP recommended conversion factor (Ref. 7) states that 4 neutrons  $\text{cm}^{-2}\text{sec}^{-1}$  is equivalent to 1 mrem/hr (the number is 3.9 at 100 MeV). Various high-energy laboratories have recommended different conversion factors, which are summarized in the following table, reproduced from Ref. 28.:

Laboratory	Flux per mrem/hour
Brookhaven	4
Daresbury	4
CERN	10
Rutherford	3.6
Berkeley	$12.1/E(\text{MeV})^{1/4}$

This problem of exactly how to weight a measurement is inherent in all cases where activation is used.

Another high-energy detection technique depends upon fission. Ionization



chambers lined with bismuth are sometimes used for neutron measurements at very high energies (Ref. 1). Figure 11 (from Ref. 29) shows fission cross sections for U, Bi, Au, and Ta as a function of energy. Another technique is to coat thin evaporated layers of these materials on mica, which is subsequently scanned for fission events. To determine the energy spectrum, ratios of fission-probabilities are taken among the various materials (for example, U/Au, Bi/Au, U/Ta), and the different cross-sections of Fig. 11 are then used to determine the neutron energy spectrum.

Both foil activation and scintillation counting are commonly used for relativistic neutron detection; these have already been discussed under "fast neutrons".

#### E. Dose Equivalent Instruments

In this section, instruments will be discussed which are intended to provide a direct measurement of dose-equivalent (rem) over a wide range of neutron energies. Referring to Fig. 3, the neutron flux (particles  $\text{cm}^{-2}\text{sec}^{-1}$ ) is shown as a function of energy which provides proper weighting for dose-equivalent.

There are several different approaches to the direct measurement of rem. The most widely used method is the use of a suitably designed moderator surrounding a thermal-neutron detector. There are two geometries in common use: cylindrical and spherical. The cylindrical design was first described by Anderson and Braun (Ref. 30), who calibrated it using both thermal neutrons and 25 keV to 15 MeV neutrons. A similar instrument was evaluated by Block and Petrock (Ref. 31) who showed that in the intermediate-energy range the response was rem-proportional to within  $\pm 10\%$  and directionally isotropic to  $\pm 15\%$ . Another design, by Leake and Smith (Ref. 32), was an improvement over the original Andersson-Braun instrument. The Block design was in turn modified by Tracerlab (now Trapelo West, a division of Lab for Electronics). Trapelo West markets the commercially available "SNOOPY" (see the Instrument Note).

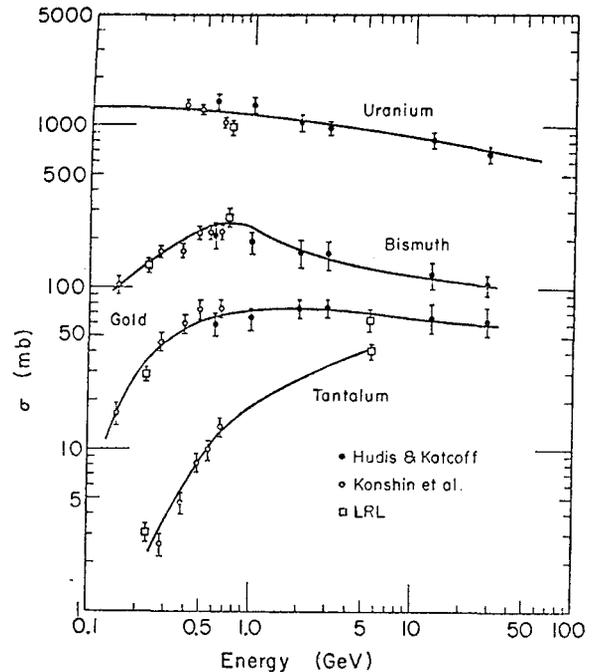


FIGURE 11: Fission cross sections vs. energy of incident nucleon (neutron or proton) for targets of U, Bi, Au, and Ta.

The generic instrument design is as follows: a  $\text{BF}_3$  proportional counter is surrounded by two cylindrical layers of polyethylene separated by a 200  $\text{mg}/\text{cm}^2$  layer of boron plastic. Similar disks cap the ends of the cylinder. The Andersson-Braun approach was to vary the two polyethylene thicknesses, and to drill a variable number of holes through the boron plastic layer, until the best empirical fit to the desired rem-response was achieved. Using a  $\text{BF}_3$  tube, 2.5 cm in diameter and filled to a pressure of 600 torr, the counter was found (Ref. 31) to be insensitive to  $^{137}\text{Cs}$  photons (660 keV) up to above 500 R/hour, with neutron sensitivities down to 2 mrem/hr full scale. Also, the energy response was within  $\pm 10\%$  of rem-equivalent all the way from thermal to 15 MeV, except for a slight region of oversensitivity near 10 keV. This is shown in Fig. 12, taken from Ref. 30.

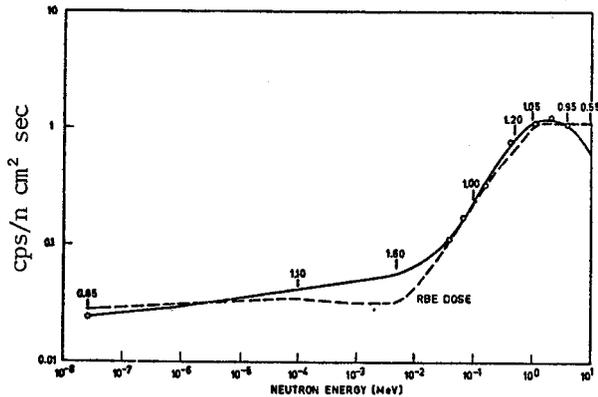


FIGURE 12: Neutron detection efficiency as a function of neutron energy. The figures give the counting rate relative to that from a counter with an ideal rem dose response of 19 cps/2.5 mrem h.

The cylindrical instrument is now a widely used survey instrument for mixed-field neutron dosimetry measurements (Ref.33). It is rugged, portable (about 11 kg), and sensitive. It has one important disadvantage: -the electronics associated with the BF<sub>3</sub> tube cannot follow extremely brief, high-intensity radiation bursts such as those around poor-duty-factor linear accelerators. A similar instrument is manufactured in Britain by Twentieth Century Electronics, Ltd, of Surrey. Another Andersson-Braun type detector has been reported by Leake (Ref.34). This instrument, specifically designed for use around pulsed-radiation sources, replaces the BF<sub>3</sub> counter with a BF<sub>3</sub>-filled ion chamber run in current mode. The output is measured using a d.c. amplifier with a 30-second integrating time constant. However, the greatest sensitivity is about an order of magnitude less sensitive than that of the BF<sub>3</sub>-counter instrument.

The use of a number of different detectors in spherical geometry was first described by Hankins (Ref.35). The basic instrument is commonly called the "Bonner sphere" (Ref.36): a <sup>6</sup>LiI(Eu) crystal, 4mm in diameter and 4mm thick surrounded by polyethylene moderating spheres. "About 80% of incident thermal neutrons are absorbed in 1mm of <sup>6</sup>LiI, so that thermal neutron detection is essentially a surface effect, whereas the efficiency for detection of gamma radiation and fast neutrons is roughly proportional to the volume of the crystal" (Ref.36). In Hankins' description

(Ref. 35) the detector is surrounded by a succession of polyethylene spheres, 2,3,5, 8, and 12 inches in diameter. Figure 13 from Ref. 35 shows the relative count rate vs. energy for the various sphere sizes, while Fig. 14 shows the estimated response of a single 10-inch sphere. While the gamma discrimination is slightly less impressive than for the cylindrical instrument, it is still very good: 5 R/hour of photons produces negligible response. That is quite sufficient for most purposes, since one seldom needs to measure 1 mrem/hr of neutrons in the presence of photon exposure rates as large as 5 R/hr.

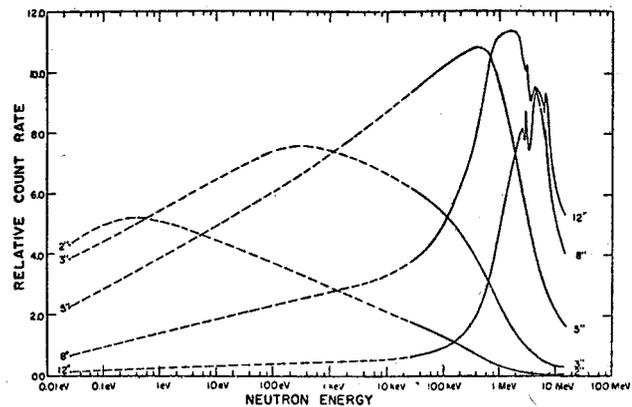


FIGURE 13: Relative counting rate plotted against energy for moderating spheres.

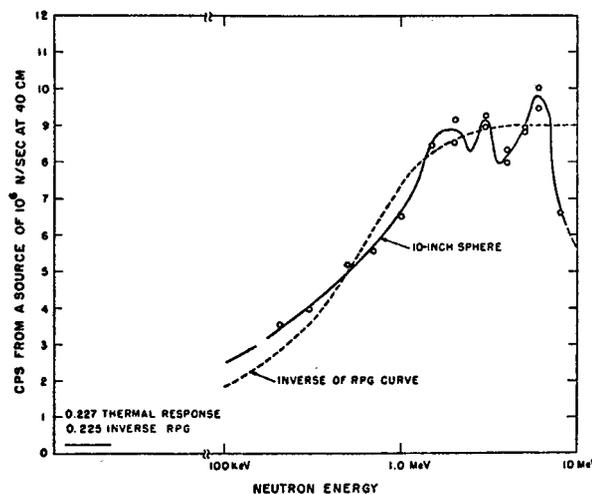


FIGURE 14: Estimated response of a 10-inch sphere plotted against energy. The points shown are experimental points.



The directional sensitivity of the spheres is, of course, essentially isotropic except on the side where the electronic circuitry is located. From Fig. 14 it can be seen that if one can tolerate errors of about factors of two or three in the fast energy range, a single 10-inch sphere will suffice; the multi-sphere technique must be used when greater precision is required. These spheres are now commercially available, from both Ludlum and Nuclear Chicago (see the Instrument Notes section). Because of the use of LiI with resolution times of about 1 microsecond, these devices suffer from an inability to measure properly in poor-duty-factor situations, a failing they share with the cylindrical (BF<sub>3</sub> counter) instrument. Eberline manufactures a 9" sphere with a BF<sub>3</sub> counter as the detector.

Leake (Ref. 34) has reported a spherical-geometry detector with a moderator similar to the Andersson-Braun design, except that the boron plastic is replaced by cadmium between the polyethylene layers. The detector is LiI(Eu), similar to that of Hankins (Ref. 35) and Bramblett et al. (Ref. 36). Commercial manufacture has been undertaken by Isotope Developments, Ltd., of Reading, U.K. (Ref. 34).

Nachtigall and Rohloff (Ref. 37) have succeeded in extending the multi-sphere technique up to an energy of 50 MeV, using four spheres of diameter 2, 5, 11, and 18 inches.

Dose-equivalent instruments which employ several other principles of operation have been reported in the literature. One of these is the use of gas proportional counters sensitive to proton recoils. These instruments have been discussed above under "fast neutrons." They follow the desired response reasonably well up to about 15 MeV, but are insensitive to neutrons below about 100 keV. Three different efforts to develop so-called "Universal Dose Equivalent Instruments" (not specific to neutrons but rather intended for all types of radiation) have been discussed in the section Particle Accelerators. These are the approaches of LET spectrometry, columnar recombination in an ion chamber, and use of a liquid-filled ion chamber.

#### 4. SOURCES FOR CALIBRATION OF NEUTRON INSTRUMENTS

Some of the general problems with calibration of instruments have been discussed in the introductory part of this volume. Here we shall focus attention on a few of the aspects specific to the use of neutron sources for calibration purposes.

The central problem in any calibration measurement is to understand the source of neutrons: its energy spectrum, directionality, presence or absence of equilibrium secondaries, and possible secondary sources from scatter of the primary neutron flux.

One common calibration technique in the fast-neutron region is the use of radioactive neutron sources from ( $\alpha, n$ ) reactions. Considerable research has been done on a determination of energy spectra from these sources, and a review paper by Hanson (Ref. 40) contains an excellent summary of information up to 1960. More recent work is summarized in ICRU Report No. 13 (Ref. 5). The most common sources are Po-Be, Ra-Be, Pu-Be, and Am-Be. Figure 15, from ICRU 13, shows the energy spectrum of neutrons from <sup>241</sup>Am-Be as measured by several groups. The purpose of the figure is to show how much difference exists among various investigators for Am-Be; this situation is also true for each of the other sources mentioned. The spectral uncertainties imply that absolute accuracies better than about  $\pm 10\%$  are probably not achievable with this method; but this is probably sufficient for most radiation protection work. One remaining problem is that the spectral studies have not generally extended below about 1 MeV, despite the fact that 20 to 30% of the neutrons in a typical source are in that low energy region (Ref. 5). The Am-Be source has one important advantage over the others, in that there is insignificant contamination from decay of daughters subsequent to Americium decay.

Fission is another common source of fast neutrons. Most data on energy spectra seem to fit the Maxwellian distribution function (Ref. 5), where  $E$  is the fission neutron energy and  $E_n$  is an experimental parameter:

$$dN(E)/dE \propto E^{1/2} \exp(-E/E_n)$$

Experimental numbers for  $E_n$  are in the range 1.2 to 1.6 MeV for several fission nuclei, such as <sup>233</sup>U, <sup>235</sup>U, <sup>239</sup>Pu, <sup>240</sup>Pu, <sup>241</sup>Pu, and <sup>252</sup>Cf. A compilation of studies on fission spectra was done by Barnard et al. (Ref. 41), and summarized in ICRU No. 13 (Ref. 5).

Use of accelerators for production of nearly mono-energetic neutrons has been common; a wide variety of reactions is available leading to a variety of neutron energies. Figure 16 and Table 3, from ICRU No. 13 (Ref. 5), show the properties of some of the reactions used.

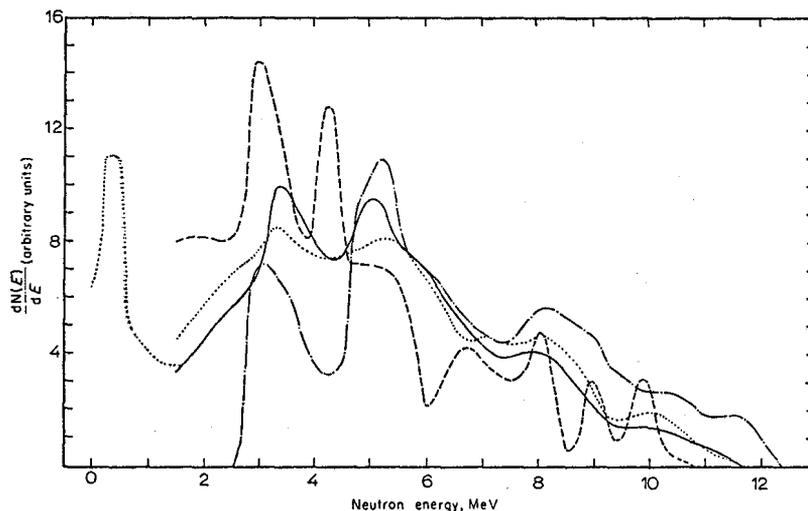


FIGURE 15: Energy distribution of  $^{241}\text{Am-Be}$  neutrons according to different authors. The curves have been normalized to give the same total number of neutrons between energies 2.5 and 10.5 MeV: — Geiger and Hargrove (1964), proton-recoil counter telescope and time of flight; --- Greiss (1968), nuclear emulsions; ···· Thompson and Taylor (1965), stilbene crystal with pulse-shape discrimination; - · - · - · Salgir and Walker (1967), proton-recoil spectrometer (silicon detector). The dotted low energy peak has been drawn according to the suggestion of Geiger and Hargrove.

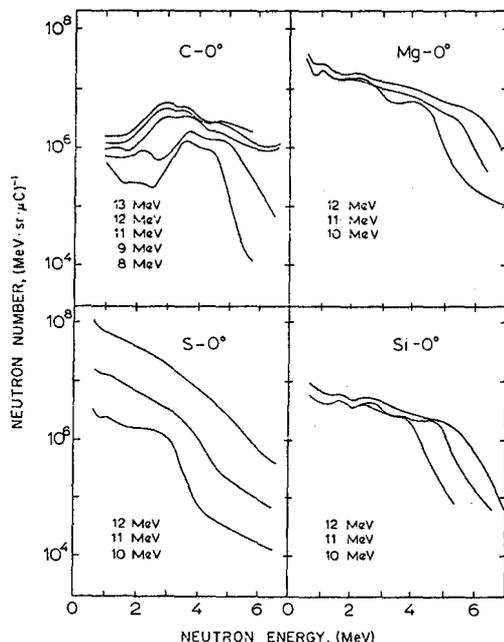


Fig. 16 . Neutron energy spectra resulting from the bombardment of carbon, magnesium, sulphur and silicon with protons of several energies. (Borchers et al., 1964. Publisher: North Holland Publishing Company, Amsterdam.)

TABLE 3 — Monoenergetic neutron-producing reactions (all energies in MeV)

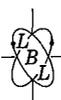
Reaction	Q-Value	Thresh- old Energy	Minimum Neutron Energy at 0°	Maximum Neutron Energy at Bombarding Energy		Maxi- mum Bom- barding Energy for Mono- ener- getic Neu- trons
				2 MeV	3 MeV	
$^2\text{D}(d, n)^3\text{He}$	3.265	—	1.8 <sup>a</sup>	5.24	6.26	—
$^3\text{T}(p, n)^3\text{He}$	-0.764	1.019	0.0639	1.20	2.21	—
$^3\text{T}(d, n)^4\text{He}$	17.6	—	12.4 <sup>a</sup>	18.25	19.57	3.7
$^7\text{Li}(p, n)^7\text{Be}$	-1.647	1.882	0.0294	0.228	1.3	2.38
$^{13}\text{C}(d, n)^{14}\text{N}$	-0.281	0.328	0.00195	1.6	2.6	3.0
$^{45}\text{Sc}(p, n)^{45}\text{Ti}$	-2.79	2.908	0.00140	—	0.112	3.67
$^{51}\text{V}(p, n)^{51}\text{Cr}$	-1.562	1.565	0.00059	0.215 <sup>b</sup>	—	2.36
$^{65}\text{Cu}(p, n)^{65}\text{Zn}$	-2.136	2.169	0.00051	0.214 <sup>c</sup>	—	2.29

<sup>a</sup> Measured at 2.0 MeV, 150 degrees.

<sup>b</sup> At bombarding energy = 1.762 MeV.

<sup>c</sup> At bombarding energy = 2.368 MeV.

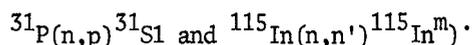
Sources of thermal neutrons are usually made by the moderation of any of the fast neutron sources just mentioned (Ref. 1). Alternatively, a thermal column from a reactor can be used. The fluence rate of any of these thermal sources can be calibrated by the foils-activation technique; gold foils are commonly used when high accuracies are required.



The directionality of the neutron source is sometimes difficult to determine, especially for thermal neutrons. The important problem here is the possible presence of scattered particles in addition to the primary flux. For accelerator-generated neutron beams the beam direction is known well, but beam halo is difficult to understand or reduce. For isotopic sources of small size, one can use a small detector to study departures from inverse-square fall-off of the flux. To minimize scattering problems, the physical layout of the calibration set-up should be carefully planned: for example, apparatus placed near a wall can be subject to large scattering corrections. A discussion of this, and of methods for measuring the possible scattering correction, can be found in ICRU Report No. 20 (Ref. 1).

In all calibrations, cross-comparison with standard instruments is extremely important. This is necessary to complement the understanding of the properties of the neutron source, because of possible unknown changes in the source.

As of this time, there does not exist a standard mechanism of calibrations for neutron instruments through any of the national standards bodies. Further, there are large regions in the neutron energy spectrum for which no well-understood neutron sources exist (Ref. 5): this is true in the region around 1 eV, and also near 2 keV. The available data are poor for many of the important cross-sections in the fast neutron region (e.g.,



ICRU Report No. 13 contains a summary of basic research (primarily cross-section measurements) needed to make possible accurate neutron instrument calibrations.

## 5. SUMMARY AND RECOMMENDATIONS

Neutron monitoring comprises two distinct tasks: (a) making detailed area surveys to determine the properties of the radiation fields; and (b) conducting routine surveillance monitoring. The latter should be supplemented with detailed surveys, undertaken occasionally and whenever there have been important changes in operating conditions.

The purpose of this section has been to enumerate the various classes of instruments, the conditions under which each class is appropriate and the changes in operation which require a detailed re-evaluation. The reader is directed to the original references for relevant details about specific techniques.

It is important to remember that whenever a new installation begins operation, it is essential to perform a detailed survey. Thus, separate instruments dominantly sensitive to thermal, intermediate, fast and possibly to relativistic neutrons are needed, as well as an instrument sensitive to photons. The relative weighting of neutrons in the various energy ranges can be assessed using a combination of techniques discussed in the text. There should by all means be an overlap in the means of measuring the various neutron energies to provide some redundancy of information. Also, these spectral studies are an important aid in studies for reducing levels for radiation protection purposes.

For routine surveillance an instrument whose response is proportional to dose-equivalent (rem) is recommended. Among those commercially available the cylindrical Anderson-Braun type is widely used, specifically that manufactured by Trapelo West. Also available are the Bonner-sphere instruments manufactured by Ludlum, Nuclear Chicago, and Eberline.

An instrument sensitive to thermal neutrons is frequently required:  $\text{BF}_3$  and  $^3\text{He}$  gas proportional counters, boron-lined chambers,  $\text{LiI}(\text{Eu})$  scintillation detectors, as well as several other types are discussed. Activation of foils is another good technique for monitoring thermal neutrons because it is both simple and sensitive. A variety of foils applicable to a wide range of neutron energies is available commercially. All of these techniques are represented among the individual Instrument Notes.

The Indium/Cadmium technique, together with any of the thermal neutron detectors, can be used to provide a rapid assessment of fluxes of thermal- and intermediate-energy neutrons.

In the text no attempt has been made to comment on the commercially available instruments individually: the advantages and disadvantages are discussed in the classes of detectors (e.g.,  $\text{BF}_3$  tubes). However, sufficient information is not available at this time to make valid comparisons concerning the regions of applicability of many instruments. Also, we have attempted to give a brief discussion of some instruments and techniques which are not now commercially available, but are worthy of commercial development.

Finally a comment about electronics design should be made. It is generally true that monitoring instruments have not yet incorporated a number of the most recent advances in electronic technology. When exploited, these advances will lead to many improvements: Increased stability and reliability, reduced maintenance, smaller size,



weight and cost, and better means of data read-out. It is urgently recommended that these improvements be incorporated commercially wherever possible.

#### 6. ACKNOWLEDGMENT

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RAD-NEU  
Neutron Monitoring  
Instruments  
September 1979

F. INSTRUMENT NOTES: NEUTRONS

In this section are found Instrument Notes for those commercial instruments which are capable of measuring neutrons. The filing system lists Instrument Notes alphabetically by manufacturer. These notes include instruments that are appropriate for neutron spectroscopy as well as dose-equivalent measurements, neutron flux measurements, etc.





INSTRUMENTATION  
FOR ENVIRONMENTAL  
MONITORING

RAD-NEU  
Stationary Monitor  
Eberline  
January 1978

Neutron Dose-Equivalent Monitor  
Eberline Radiation Monitor RM-16  
Neutron Rem Detector, NRD-1



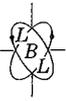
Class	Stationary
Principle of Operation	Polyethylene (cadmium-loaded) sphere of 9" diameter, $\text{BF}_3$ tube inside, used with rate meter.
Sensitivity and Range	NRD-1: (1) Energy response close to rem-equivalent from thermal to about 10 MeV. (2) Range 1 to 10,000 mrem/hour. RM-16: (1) Input sensitivity is charge sensitive, adjustable from approximately $2 \times 10^{-13}$ to $10^{-11}$ Coulomb. (2) Range of four logarithmic decades. (3) Window width is adjustable from 0 to at least double the input sensitivity.
Sampling	Continuous
Performance	NRD-1: (1) Gamma rejection up to 500 R/hour dependent on high voltage (2) About 50 counts/minute corresponds to 1 mrem/hour. (3) Plateau approximately 300 volts with a slope of 3% per 100 volts. (4) Directional response within $\pm 10\%$ (5) Operation voltage in typically 1600 to 2000 volts. RM-16: (1) Response time varies continuously with count rates. (2) Linearity within $\pm 10\%$ of reading typically, $\pm 25\%$ maximum. (3) High voltage regulated and adjustable from less than +200 volts to +2500 volts. (4) Temperature range is $-1^\circ\text{C}$ to $60^\circ\text{C}$ ( $+30^\circ\text{F}$ to $\pm 140^\circ\text{F}$ )



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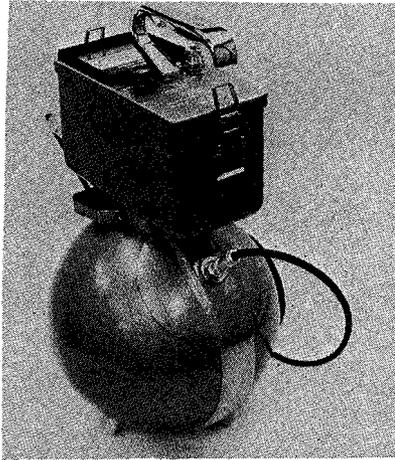
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	<u>NRD-1</u>	<u>RM-16</u>
Requirements	Power: 105 = 125V, 60 Hz Size: 22.9 cm diameter x 25.1 cm overall height (9" x 9 7/8") Weight: 6.25 kg (13 3/4 lb)	105 - 125V, 50-60 Hz and rechargeable batteries 13.6 cm H x 26.4 cm D x 21.6 cm W (5 3/8" x 10 3/8" x 8 1/2") 3.69 kg (8 lbs. 2 oz.)
Features	(1) Detector can be wall-mounted (2) Cable up to 100 feet long from detector to electronics package (3) Alarm indication, high, red light and more penetrating tone; low, blue lamp	
References	Manufacturer's specifications	
Cost	NRD-1 Sphere w/cable \$695.00 RM-16 Rate Meter \$890.00 w/meter face 10472-B42	
Address	Eberline Instrument Corporation P.O. Box 2108 Santa Fe, NM 87501 (505) 982-1881	



Fast-Slow Neutron Counter

Eberline Model PNC-4



Class Portable, hand-held

Principle of Operation Uses  $\text{BF}_3$  slow neutron tube and electronic counter

Sensitivity and Range Neutrons detected: slow and fast  
Range: 0-500,000 cpm in four, linear, continuously progressive Lin-Log ranges: 0-500, 500-5,000 0-50,000, and 50,000 - 500,000 (equivalent to approximately 0-10,000 n/cm<sup>2</sup>sec for 1 MeV neutrons)

Performance Linearity:  $\pm 10\%$   
Temperature:  $-40^\circ\text{F}$  to  $+140^\circ\text{F}$   
Moderator: Paraffin block enclosed in a 0.03 inch Cadmium shield in a 0.06 inch aluminum can  
Gamma rejection: Will not detect  $^{60}\text{Co}$  gamma radiation below 10R/hr (at normal setting on detector plateau)  
High Voltage: Variable ( $+1200$  volts to  $+2000$  volts) and regulated

Requirements Power: Five "D" cells, 200 hours  
Size: 22.9 cm  $\times$  10.2 cm  $\times$  29.9 cm (9" L  $\times$  4" W  $\times$  11-3/4" H)  
Weight: 5.44 kg (12 lbs)

Features The detector may be slipped in or out of the moderator for monitoring fast or slow neutrons, respectively. By correlation of these readings with a known neutron flux-rate and a known energy, the user may evaluate fast and slow neutron hazards.

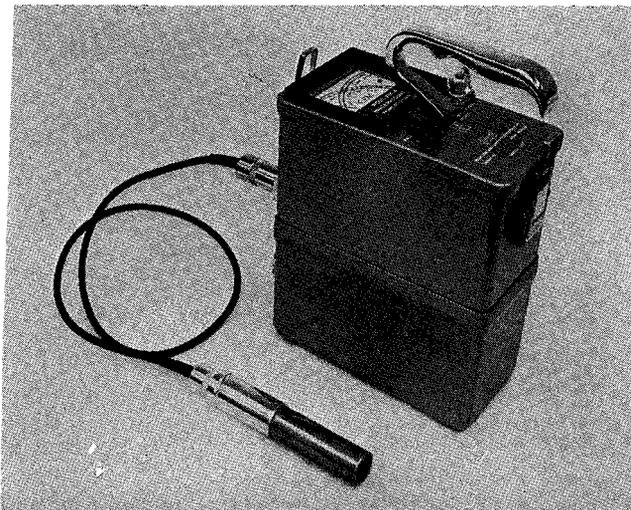
Cost \$925.00

Address Eberline Instrument  
P.O. Box 2108  
Santa Fe, New Mexico 87501  
(505) 982-1881





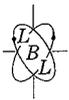
Neutron Rem Counter  
Eberline Model PNR-4



Class	Portable, hand-held
Principle of Operation	Uses a $\text{BF}_3$ tube and electronic counter
Sensitivity and Range	Energy sensitivity: Thermal to approx. 10 MeV Range: 0-5000 mRem/hr in four, linear continuously progressive, Lin-Log ranges: 0-5, 5-50, 0-500, 500-5,000 mRem/hr.
Performance	Linearity: $\pm 8$ of full scale of decade being read Temperature: $-40^\circ\text{C}$ to $60^\circ\text{C}$ ( $-40^\circ\text{F}$ to $+140^\circ\text{F}$ ) Gamma rejection: up to 500 R/hr, dependent on high voltage setting Response time nominal: 12 seconds, 1st decade 6 seconds, 2nd decade 1.5 sec, 3rd decade 0.3 sec, 4th decade Directional response: within $\pm 10\%$ High voltage: regulated and adjustable, 1300 volts to 2000 volts
Requirements	Power: Five "D" cells, 200 hours Size: 43.2 cm $\times$ 24.1 cm $\times$ 22.9 cm (17" H $\times$ 9-1/2" L $\times$ 9" W) instrument 10.2 cm W (4") Weight: 8.9 kg (19-1/2 lbs.)
Features	Detachable detector, lin-log presentation; aural monitoring capability
Reference	Manufacturers' specifications
Cost	\$1125.00
Address	Eberline Instrument Corporation P.O. Box 2108 Santa Fe, New Mexico 87501 (505) 982-1881







Cylindrical BF<sub>3</sub> Counting Tubes (continued)

Model	Outside Dia. (cm)	Outside Dia. (inches)	Overall Length (cm)	Overall Length (inches)	Operating voltage	General	Price
2024	2.62	1.031	60.64	23.875	1200	General Purpose	\$175.00
2025	2.62	1.031	30.48	12	1200	Reactor Source Range	175.00
2026	2.62	1.031	76.52	30.125	1200	Reactor Source Range	175.00
202ER	2.62	1.031	37.78	14.875	1600	Diffraction	550.00
203S	5.16	2.031	40.32	15.875	2200	General Purpose	190.00
203A	5.16	2.031	40.32	15.875	2200	General Purpose	190.00
2031	5.16	2.031	27.94	11	3000	AN/PDR-58	550.00
20310	5.16	2.031	64.45	25.374	2700	General Purpose	225.00
2032A	5.16	2.031	30.56	12.375	1300	General Purpose	225.00
2032S	5.16	2.031	30.56	12.041	1300	General Purpose	225.00
212E	5.16	2.031	33.63	12.031	2500	Diffraction	750.00
212ER	5.16	2.031	39.69	28.25	2500	Diffraction	750.00
20327	2.0				2200-2900	Low energy neutron detection	200.00

<sup>10</sup>B Lined Proportional Counters

Model	Max. Dia. (cm)	Max. Dia. (inches)	Overall Length (cm)	Overall Length (inches)	Operating voltage	Sensitivity (cps/nv)	Price
230	1.35	0.531	As Specified		700	0.6	\$350.00
232	2.54	1.0	20.00	7.875	800	1.0	300.00
23210	2.54	1.0	38.74	15.25	950	4.0	400.00
23211	2.54	1.0	76.2	30.0	800	10	475.00
23011	1.27	0.50	29.61	11.656	600-900	0.6	300.00
23110	1.27	0.50	69.06	27.188	600-950	7.5	300.00
23113	1.27	0.50	160.82	63.313	600-950	7.5	500.00

Proton Recoil Fast Neutron Detectors

Model	Max. Dia. (cm)	Max. Dia. (inches)	Overall Length (cm)	Overall Length (inches)	Operating voltage	Price
270	4.10	1.615	11.75	4.625	1500-3200	\$400.00
2801	5.08	2.00	17.15	6.75	1400	350.00
2802	5.08	2.00	20.80	8.19	1400	400.00
2803	5.08	2.00	31.75	12.50	1400	430.00
2804	5.08	2.00	53.72	21.15	1400	475.00
281	3.81	1.50	25.40	10.0	2500	300.00
2821	2.54	1.0	17.15	6.75	1100	300.00
2822	2.54	1.0	20.80	8.19	1100	330.00
2823	2.54	1.0	31.75	12.50	1100	400.00
2824	2.54	1.0	53.75	21.15	1100	440.00
28114	5.1	2.0	70.00	27.75	2500	250.00
28115	5.1	2.0	85.34	33.6	2500	285.00

Fission Counter

Model	Max. Dia. (cm)	Max. Dia. (inches)	Overall Length (cm)	Overall Length (inches)	Operating voltage	Remark	Range	Price
3075	0.64	0.25	2.54	1.0	200-800	Stainless	10 <sup>3</sup> to 10 <sup>8</sup> n	\$640.00
3076	1.27	0.5	13.97	5.5	200-800	Aluminum	10 <sup>3</sup> to 10 <sup>8</sup> n	700.-725.00
3078	5.08	2.0	30.48	12	200-800	Aluminum	10 <sup>3</sup> to 10 <sup>8</sup> n	850.00



Model	<u>One-dimension Position Sensitive Proportional Counter</u>				Operating voltage	Anode	Price
	Max. Dia. (cm)	Max. Dia. (inches)	Overall Length (cm)	Overall Length (inches)			
25225	2.54	1.0	112.40	744.25	1000	quartz	\$1800.00
25228	2.54	1.0	112.40	44.25	1000	molecul---	1800.00
25229	2.54	1.0	43.18	17.00	1000	quartz	1200.00
25230	2.54	1.0	43.18	17.00	1000	molecul---	1200.00
26211	7.62	3.0	25.40	10.00	1000	molecul---	5000.00

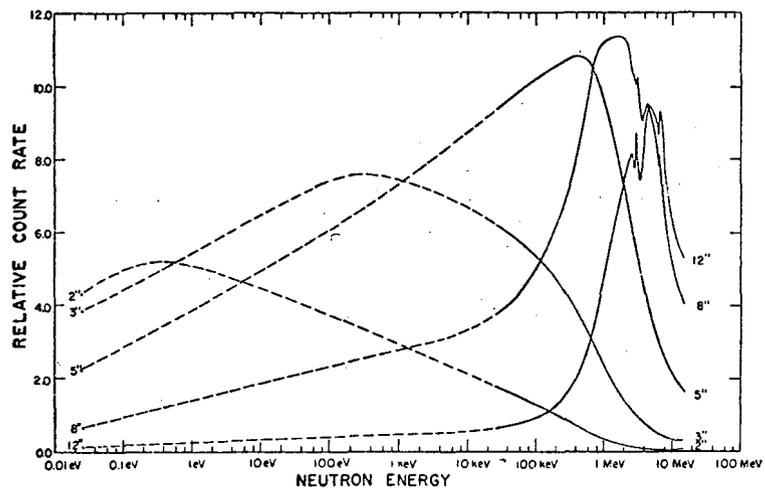
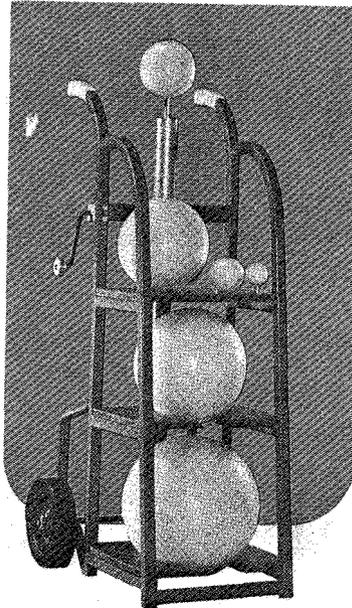
Reference                      Manufacturers' specifications

Address                         LND  
                                     3230 Lawson Blvd.  
                                     Oceanside NY 11572  
                                     (516) 578-6141  
                                     Telex 14-4563





Neutron Spectrometer  
Ludlum Model 42-5



*Relative counting rate plotted against energy  
for moderating spheres.*

Class

Mobile

Principle  
of Operation

Six different diameter spheres moderate various energy neutrons LiI(Eu) crystal, light pipe and RCA 6199 photomultiplier tube to be used with any counting system capable of accepting 2 millivolt minimum signals and providing 900 V dc to the detector

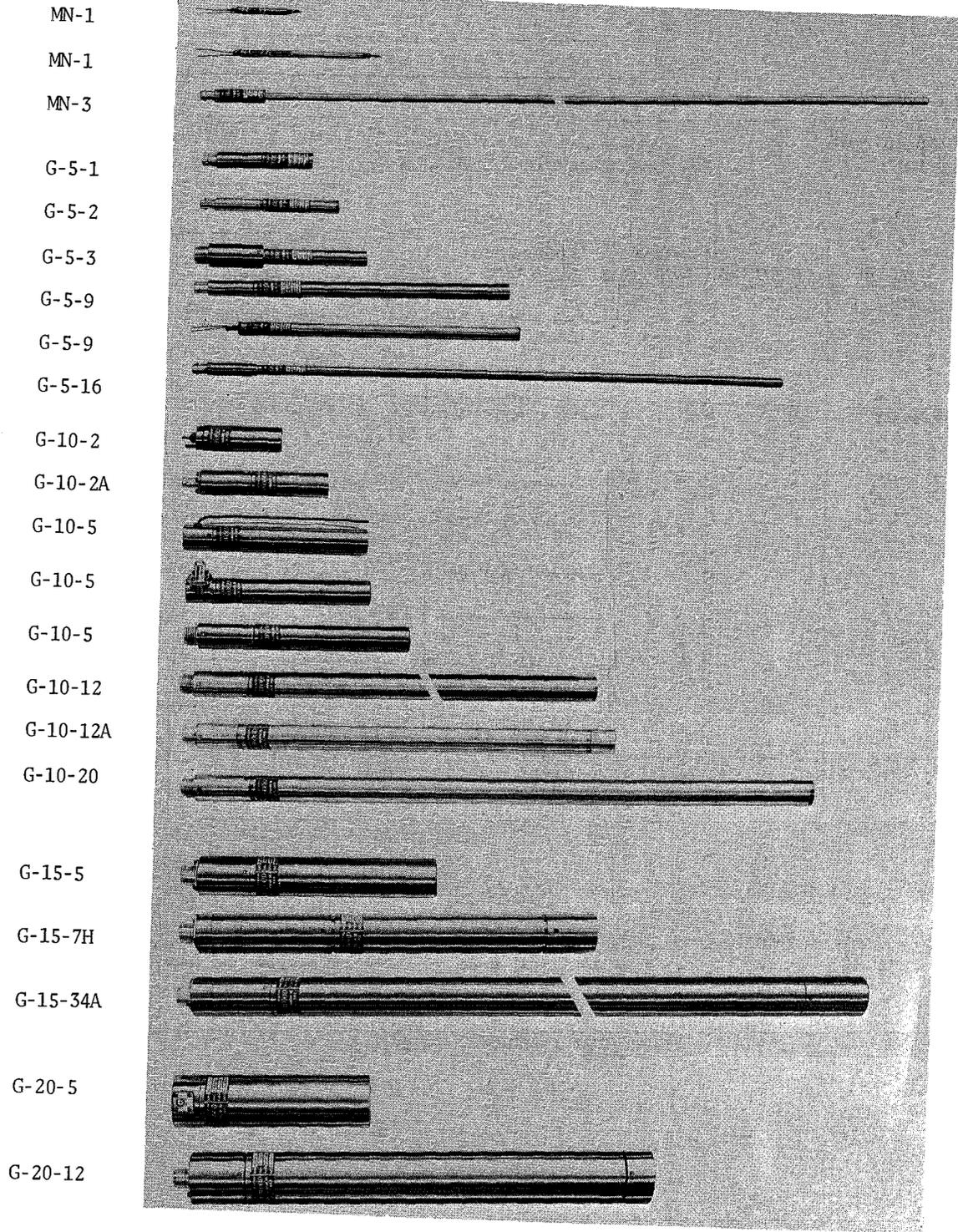


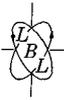
Sensitivity and Range	Sphere Diameter (in)	Energy Range
	2	
	3	
	5	See Figure
	8	
	10	
	12	
Sampling	Continuous	
Performance	Accuracy: Efficiencies, response time, etc. (see references)	
Requirements	Power: 900 V dc to be supplied by counting system Size: 112 cm H x 45 cm W x 52 cm D (45" x 18" W x 21" D) Weight: 55 kg (120 lbs)	
Features	X-ray inspection of all spheres insures void-free moderators. A crank operated elevator assembly adjusts detector center from 44 to 73 inch working height.	
References	a) Manufacturers' specifications b) D. E. Hankins, "The Multisphere Neutron-Monitoring Technique," Los Alamos Report LA-3700, UC-41 Health and Safety, TID-4500, Los Alamos, New Mexico 1968.	
Cost	\$1,695.00	
Address	Ludlum Measurements Inc. 501 Oak Street Sweetwater, Texas 79556 (915) 235-5494	



Neutron Counters

N. Wood Models





Neutron Counters

N. Wood Models

Class Portable, Laboratory

Principle of Operation  $BF_3$  Neutron proportional counters

MODEL	DIAMETER	ACTIVE LENGTH	OPERATING VOLTAGE	GENERAL	PRICE	
MN-1	1/4"	1/4" to 5"	1100 to 2000 volts, for 20 to 60 cm. Hg. pressure of $BF_3$ .	Pigtail lead. 931/U connector. (MN-3 has 5/16" casing up to 56" long.)	Microneutron counters for reactor instrumentation.	\$100.00
MN-3	1/4"	as specified.				\$200.00
G-5-3	5/8"	3"	1400 to 2100 volts, for 20 to 60 cm. Hg.	931/U connector.	Small general purpose neutron detectors.	\$100.00
G-5-9	or	9"		Caps same O.D. as counter.		\$107.00
G-5-12	1/2" as	12"				\$113.00
G-5-16	specified.	16"				\$120.00
G-10-2	1"	2"	1300 v. for 20 cm. Hg.	Pigtail lead.	Widely used in portable instruments.	\$107.00
G-10-2A	1"	2"	1800 v. for 60 cm. Hg.	931/U connector.		\$120.00
G-10-5	1"	4 1/2"	1500 to 2200 v. for 20 to 60 cm. Hg. For lower oper. volt. specify small dia. anode.	HN 82-805 or	Standard multiple purpose neutron detectors.	\$133.00
G-10-12	1"	12 1/2"		931/U connector.		\$147.00
G-10-20	1"	20"				\$160.00
G-15-5	1 1/2"	5"	1700 to 2250 v. for 20 to 45 cm. Hg. Higher pressure may be specified as well as lower voltage.	HN 82-805 or	Standard research counters. High sensitivity low background.	\$133.00
G-15-12	1 1/2"	12"		931/U connector.		\$167.00
G-15-20	1 1/2"	20"				\$187.00
G-15-34A	1 1/2"	34"		[G-15-34A was developed for cosmic radiation studies (Simpson counter).]		\$233.00
G-20-5	2"	5"	1900 to 2300 v. for 20 to 45 cm. Hg. Higher pressures and lower voltages may be specified.	General purpose high sensitivity detectors. Useful in moisture meters and environmental monitoring. Connectors same as above models.		\$153.00
G-20-12	2"	12"				\$187.00
G-20-20	2"	20"				\$213.00
G-20-38	2"	38"				\$247.00

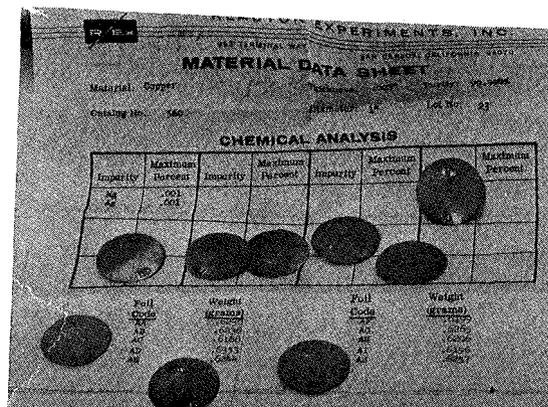
Prices listed are for brass or copper cathode, prices for aluminum cathode - add \$10.00; prices for stainless steel on request.

Prices listed for Model G (Enriched  $BF_3$  - 97% Boron 10) are the same for Model F (Depleted  $BF_3$  - 11% Boron 10).

Address N. Wood Counter Laboratory, Inc.  
1525 E. 53rd Street  
Chicago, Illinois 60615  
(312) 324-1114



Reactor Experiments, Inc.



Class Portable  
Principle of Operation Neutron dosimeter foils 12.7 mm (0.5") diameter

Material	Cat. No.	Thickness		Nominal Purity (percent)	Price per Box of 10
		(mm)	(inches)		
Aluminum	505	0.76	0.030	99.963	\$60.00
Aluminum	505A	0.13	0.005	99.991	55.00
Aluminum	505B	0.05	0.002	99.999	65.00
Ammonium-Sulfate	538A	3.2	0.125	99.998	65.00
Cadmium Cover Set	531A	1.0	0.040	99.9	\$70.00
Cadmium Cover Set	531	0.5	0.020	99.9	55.00
Cobalt	506A	0.05	0.002	99.896	\$85.00
Copper	509A	0.25	0.010	99.998	65.00
Copper	509	0.13	0.005	99.998	70.00
Copper	509B	0.013	0.005	99.997	\$ 70.00
Dysprosium	581	0.025	0.001	99.995	125.00
Gold	521B	0.025	0.001	99.995	\$100.00
Hafnium	553	0.05	0.002	99.957*	140.00
Indium	501	0.03	0.005	99.99	65.00
Iron	515	0.13	0.005	98.643	\$ 95.00
4.45% Lutetium-Aluminum	508	0.1	0.004	99.963	180.00
Magnesium	514	0.13	0.005	99.781	60.00

\* Includes 2.94% Zirconium

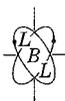


Reactor Experiments, Inc.

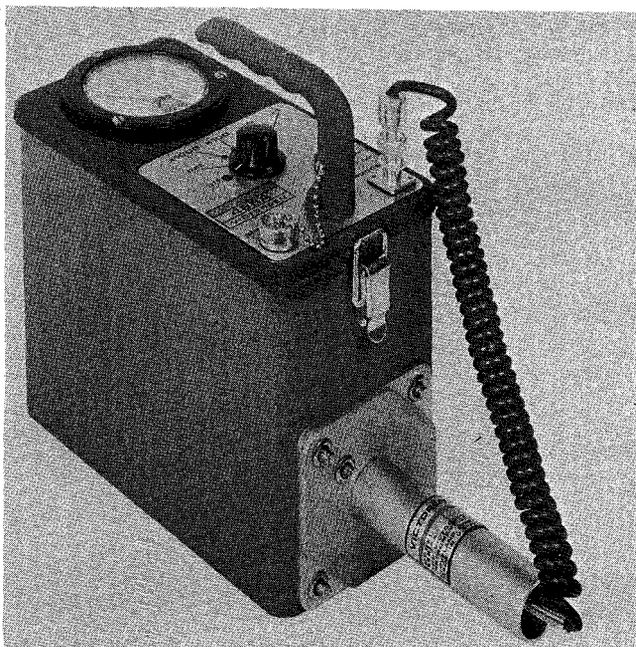
Material	Cat. No.	Thickness		Diameter		Nominal Purity (percent)	Price per Box of 10
		(mm)	(inches)	(mm)	(inches)		
80.2% Manganese-Copper	516	0.05	0.002			99.722	\$65.00
Molybdenum	527	0.075	0.003			99.985	60.00
Neptunium-237	578	50 $\mu\text{g cm}^2$		19	0.75		275.00/foil
Nickel	513A	0.25	0.010			99.981	60.00
Nickel	513B	0.075	0.003			99.981	\$65.00
Niobium	576	0.13	0.005			99.837	85.00
Plutonium-239	575	30 $\mu\text{g cm}^2$		19	0.75	99.22	275.00/foil
Rhodium	540	0.025	0.001			99.97	145.00
Scandium	580	0.13	0.005			99.64	225.00
Silver	511	0.13	0.005			99.96	65.00
Sodium Chloride	543B	2.0	0.075			99.99	\$65.00
Sulfur (1" diam.)	522A	4.75	0.187			99.99	60.00
Sulfur	522B		0.075			99.99	55.00
Tantalum	534	0.13	0.005			99.98	65.00
Thorium	512B	0.1	0.004			99.885	65.00
Tin	551	0.1	0.004			99.57	\$60.00
Titanium	535	0.25	0.010			99.876	105.00
Tungsten	526	0.15	0.006			99.96	115.00
Uranium (natural)	503	0.175	0.007			99.97	110.00
Uranium (depleted)	504L	0.025	0.001			99.958	145.00 for 5 foils
Uranium (enriched)	577	30 $\mu\text{g cm}^2$		19	0.75	99.06	275.00/foil
Vanadium	517	0.075	0.003			99.968	75.00
Zinc	552	0.13	0.005			99.99	75.00
Zirconium	541	0.13	0.005			99.82	75.00

Features Each foil is thoroughly cleaned and accurately weighted to 0.0001 g with a weight code embossed directly on the foil and 28 page "Foil Instruction Manual". Flux wires, fission foils, and cadmium tubing is also available.

Address Reactor Experiments, Inc.  
963 Terminal Way  
San Carlos, CA 94070  
(415) 592-3355



Neutron Survey Meter  
Victoreen Model 488A



Class	Portable, handheld
Principle of Operation	Uses a 4" boron-lined neutron proportional counter, moderator, amplifier, ratemeter, meter readout
Sensitivity and Range	Range: Thermal to 14 MeV Thermal neutrons, four ranges: 0-800 cpm 0-8,000 cpm 0-80,000 cpm 0-800,000 cpm These ranges correspond to 0-12 thermal neutrons/cm <sup>2</sup> /sec, x1, x20, x100, x1000. Gamma Sensitivity: Completely insensitive to gamma radiation fields to 500 R/hr.
Sampling	Continuous
Performance	Accuracy: ±10% fullscale deflection Time Response: Switch selected, 1.5 seconds, 5 seconds, 15 seconds Moderator: high density polyethylene encased in 1/32 in. cadmium chilled. 2.54cm (1 in.) wall thickness overall moderator diameter 7.9cm (3 1/8"). Drift: less than 5% over 24 hours Temperature operating range: -30°C to +50°C -20°F to +120°F Temperature dependence: less than 5% over above temperature range Humidity range: 0-90%
Requirements	Power: Two "D" cells Size: 30.5cm L × 11.4cm W × 25.4cm H (12" × 4 1/2" × 10") Weight: 3.85 kg (8.5 lbs)
Features	The detector assembly may be utilized in three modes -- either bare tube, tube with polyethylene moderator, or tube with moderator and cadmium shield. With these combinations the 488A detects thermal neutrons, moderated fast and thermal neutrons, and only fast neutrons respectively.

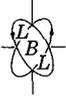


INSTRUMENTATION  
FOR ENVIRONMENTAL  
MONITORING

RAD-NEU  
Survey  
Victoreen  
Page 2

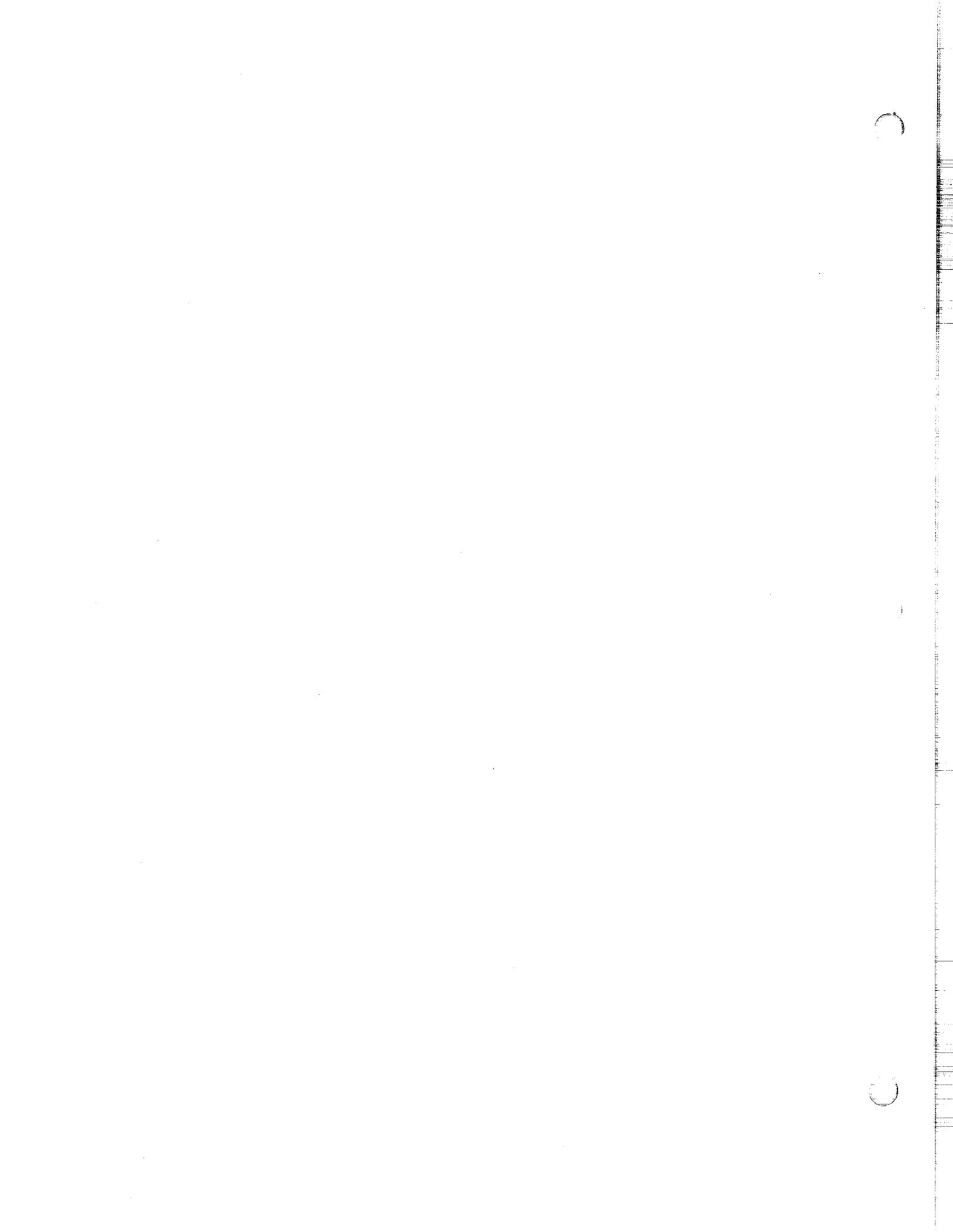
Cost \$1,090.00

Address Victoreen Instrument Division  
10101 Woodland Avenue  
Cleveland, Ohio 44104  
(216) 795-8200



PERSONNEL DOSIMETRY

1. Introduction
2. Design Considerations
3. Dosimeter Types
4. Gamma/Beta Dosimetry
  - a. Film
  - b. TLD
  - c. Pocket Ionization Chambers
  - d. Other Dosimeters
    - (i) RPL
    - (ii) TSEE
    - (iii) Track Etching
5. Neutron Dosimetry
  - a. Fast Neutron Detectors
  - b. Thermal Neutron Detectors
  - c. Albedo Dosimetry
6. Special Dosimetry Problems
7. Summary
8. Acknowledgment
9. References





## 1. INTRODUCTION

In this section we shall discuss instrumentation used to measure integrated external radiation doses to personnel. The archetype of such instrumentation is the "film badge", but of course this discussion will be of broader scope; we shall deal with the general measurement problem of integrating dose instruments designed to be pinned to clothing, carried in pockets, worn on fingers, or otherwise portable without the user's constant attention or even constant consciousness.

There are several reasons why such instruments are used. Perhaps the most commonly expressed reason is related to legal liability: in routine use at low integrated dose levels, the dosimeter protects the employer against liability for negligence, and is in many cases required by law for just such a purpose. A second important motivation, and certainly the prime operational motivation, is the assessment of dose in the event of an accident, as this can be a crucial aid in post-accident treatment. Third, when doses may approach maximum permissible levels, dosimeters are often used to limit individual exposures. Finally, the psychological benefit of dosimetry is often cited; some individuals ignorant as to the true meaning of radiation dose somehow feel "safer" wearing a dosimeter, as if "big brother" were looking after them.

## 2. DESIGN CONSIDERATIONS

Each of these motivations places a different set of constraints upon the dosimetry system. Thus the whole question of accuracy is related both to the legal-liability situation and to the requirements imposed by decision-making in post-accident treatment. Also, the desirability of long-term dosimeter storage for legal purposes is sometimes cited. Further, the portability required of the dosimeter imposes size-and-weight restrictions, while cost considerations demand that large numbers of readings be possible at low unit-cost. All of these factors must be considered in any discussion of personnel dosimetry.

Obviously, "the more accurate the better". However, certain basic limitations prevent the attainment of extremely precise measurements. The most important is the problem of sampling. A dosimeter can only measure the radiation which strikes it, but the problem arises when one relates the dosimeter reading to an "equivalent whole body external dose." Several factors contribute here; for example, low-energy beta activity in the air can register in a sensitive dosimeter and yet be effectively shielded by a couple of layers of clothing. Also, the angle of incidence of the radiation is important: exposures from one side of the body may be misrepresented by some dosimeter

geometries, especially if the dosimeter is at the front while radiation is incident at the back of the body. Another problem occurs when dose is delivered to a part of the body (e.g., hands or feet) distant from the dosimeter.

Dosimetry thus presents a nearly impossible problem if accuracies in the few-percent range were to be required. Fortunately, no such need exists, from the viewpoint of either post-accident treatment or legal responsibility.

To illustrate these points, we shall quote from ICRP Publication 12 (Ref. 1):

"A dosemeter carried on the surface of the worker's body is best regarded as a sampling device. It provides a sample of the dose received at the surface of the body during the movement of the worker through his environment. It does not provide a direct measurement of the radiation dose to organs or tissues, except for the dose to the skin in the immediate vicinity of the dosemeter. Neither does it necessarily provide an adequate assessment of the situation in the workplace because it measures the dose only at one point on the body of each worker.

"It is only in exceptional cases, usually associated with substantial accidental exposures, that attempts need be made to estimate the actual organ doses, rather than doses at the surface of the body. The greatest uncertainties in the assessment of organ doses from routine monitoring are caused by the fact that a small number of doseimeters, often only one, has to be taken as representative of the exposure of the whole surface of the body. Unless the doseimeter results are representative, there is little point in conducting detailed calculations of the depth dose in a composite, finite medium -- the human body.

"The need to use many doseimeters and the complexity of the necessary interpretation to assess the actual organ doses prohibit the making of these assessments as part of routine individual monitoring. A simplified system of interpretation has to be adopted...and this will usually have the effect of assessing upper limits to organ doses rather than actual values...

"It has become conventional to note and record all radiation doses above the threshold of detection of the personal doseimeter. This detailed attention to low doses may, in some cases, focus unrealistic attention onto exposures which, if they cause any risk at all, cause risks which are trivial in comparison with other risks of life. Other exposures, often of the same order of magnitude, are not recorded,



either because they are below the threshold of detection or because the individuals are not monitored. The Commission [ICRP] has now recommended that individual monitoring and associated records are not needed if the doses are most unlikely to exceed 3/10 of the annual Maximum Permissible Doses. It is possible to make use of this recommendation to make individual monitoring simpler and more logical...

"In principle, the choice of an investigation level would correspond exactly to 3/10 of the appropriate annual Maximum Permissible Dose...

"In practice, however, such a precise equivalence cannot be achieved because most individual monitoring for external radiation is carried out over periods substantially shorter than a year. It is therefore recommended that, for the whole body, the investigation level for penetrating radiation should be set somewhat lower--50 mrem for a dosimeter issued for 2 weeks, 100 mrem for 1 month, or 300 mrem for 3 months. In the rare cases when a substantial part of the dose to an individual results from neutrons or from internal contamination, the investigation level should be applied to the total recorded dose in a control period rather than to each of the separate contributions...

"The uncertainties acceptable in routine individual monitoring should be somewhat less than the investigation level and can best be expressed in relation to the annual dose. The uncertainty in assessing the upper limits to the annual dose equivalent to the whole body or to the organs of the body...should not exceed 50%. Where these doses are less than 2 rems an uncertainty of 1 rem is acceptable. This uncertainty includes errors due to variations in the dosimeter sensitivity with incident energy and direction of incidence, as well as intrinsic errors in the dosimeter and its calibration."

For the purposes of our discussion here, suffice it to say first that the obvious accuracy limitations on external personnel dosimetry must be recognized by the user; second, that when high accuracies are required other methods must be called upon to complement the dosimeters of concern here; and finally, that most people realize that the need for great intrinsic accuracy is not the most significant factor in the choice of dosimeters.

What is important is an assurance that no substantial component of the absorbed dose is missed entirely by the dosimetry system

being used. Thus one needs to base a dosimetry system upon detailed information about the types of radiation likely to be encountered.

We shall concentrate here upon a discussion of the types of dosimeters; their relative sensitivities to various particle species and energy ranges; and their advantages and disadvantages.

### 3. DOSIMETER TYPES

For routine daily monitoring, there are two basic types of dosimeters in common use today: film and thermoluminescent dosimeters (TLD's). Film is the more commonly used, and historically has been the mainstay of personnel dosimetry systems. TLD's are now achieving increasingly widespread use. The most important subject of debate today in this field seems to concern the relative merits of film and TLD.

Several other types of dosimeters are also in use, usually for special applications. Among these are pocket ionization chambers, activation detectors, radiophotoluminescent (RPL) dosimeters, and thermally-stimulated-exoelectron-emission (TSEE) devices.

We shall attempt here to discuss the basic properties of each of these various types, as well as their various regions of applicability. We shall also indicate those new techniques which show possible promise for dosimetry, but still require extensive investigations and development.

There are two basic types of external radiation fields which are commonly monitored: beta/gamma and neutron fields. We shall discuss these separately, since the latter involves techniques and considerations completely different from the former.

### 4. GAMMA/BETA DOSIMETRY

#### A. Film

The basic property of film dosimeters is the ionization-induced darkening of emulsion. The intrinsic limitations of film are due to four problems: first, some (unavoidable) background darkening occurs in the absence of any ionization; second, film must be wrapped to avoid light leaks, and even for thin wrappings the dosimeter is typically insensitive to alpha particles, low energy betas, and low energy x-rays; third, the high atomic number of typical emulsions makes the response non-tissue-equivalent for low-energy photons; and finally, emulsion darkening ultimately saturates at high doses.



## 1. INTRODUCTION

In this section we shall discuss instrumentation used to measure integrated external radiation doses to personnel. The archetype of such instrumentation is the "film badge", but of course this discussion will be of broader scope; we shall deal with the general measurement problem of integrating dose instruments designed to be pinned to clothing, carried in pockets, worn on fingers, or otherwise portable without the user's constant attention or even constant consciousness.

There are several reasons why such instruments are used. Perhaps the most commonly expressed reason is related to legal liability: in routine use at low integrated dose levels, the dosimeter protects the employer against liability for negligence, and is in many cases required by law for just such a purpose. A second important motivation, and certainly the prime operational motivation, is the assessment of dose in the event of an accident, as this can be a crucial aid in post-accident treatment. Third, when doses may approach maximum permissible levels, dosimeters are often used to limit individual exposures. Finally, the psychological benefit of dosimetry is often cited; some individuals ignorant as to the true meaning of radiation dose somehow feel "safer" wearing a dosimeter, as if "big brother" were looking after them.

## 2. DESIGN CONSIDERATIONS

Each of these motivations places a different set of constraints upon the dosimetry system. Thus the whole question of accuracy is related both to the legal-liability situation and to the requirements imposed by decision-making in post-accident treatment. Also, the desirability of long-term dosimeter storage for legal purposes is sometimes cited. Further, the portability required of the dosimeter imposes size-and-weight restrictions, while cost considerations demand that large numbers of readings be possible at low unit-cost. All of these factors must be considered in any discussion of personnel dosimetry.

Obviously, "the more accurate the better". However, certain basic limitations prevent the attainment of extremely precise measurements. The most important is the problem of sampling. A dosimeter can only measure the radiation which strikes it, but the problem arises when one relates the dosimeter reading to an "equivalent whole body external dose." Several factors contribute here; for example, low-energy beta activity in the air can register in a sensitive dosimeter and yet be effectively shielded by a couple of layers of clothing. Also, the angle of incidence of the radiation is important: exposures from one side of the body may be misrepresented by some dosimeter

geometries, especially if the dosimeter is at the front while radiation is incident at the back of the body. Another problem occurs when dose is delivered to a part of the body (e.g., hands or feet) distant from the dosimeter.

Dosimetry thus presents a nearly impossible problem if accuracies in the few-percent range were to be required. Fortunately, no such need exists, from the viewpoint of either post-accident treatment or legal responsibility.

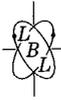
To illustrate these points, we shall quote from ICRP Publication 12 (Ref. 1):

"A dosimeter carried on the surface of the worker's body is best regarded as a sampling device. It provides a sample of the dose received at the surface of the body during the movement of the worker through his environment. It does not provide a direct measurement of the radiation dose to organs or tissues, except for the dose to the skin in the immediate vicinity of the dosimeter. Neither does it necessarily provide an adequate assessment of the situation in the workplace because it measures the dose only at one point on the body of each worker.

"It is only in exceptional cases, usually associated with substantial accidental exposures, that attempts need be made to estimate the actual organ doses, rather than doses at the surface of the body. The greatest uncertainties in the assessment of organ doses from routine monitoring are caused by the fact that a small number of dosimeters, often only one, has to be taken as representative of the exposure of the whole surface of the body. Unless the dosimeter results are representative, there is little point in conducting detailed calculations of the depth dose in a composite, finite medium -- the human body.

"The need to use many dosimeters and the complexity of the necessary interpretation to assess the actual organ doses prohibit the making of these assessments as part of routine individual monitoring. A simplified system of interpretation has to be adopted...and this will usually have the effect of assessing upper limits to organ doses rather than actual values...

"It has become conventional to note and record all radiation doses above the threshold of detection of the personal dosimeter. This detailed attention to low doses may, in some cases, focus unrealistic attention onto exposures which, if they cause any risk at all, cause risks which are trivial in comparison with other risks of life. Other exposures, often of the same order of magnitude, are not recorded,



either because they are below the threshold of detection or because the individuals are not monitored. The Commission [ICRP] has now recommended that individual monitoring and associated records are not needed if the doses are most unlikely to exceed 3/10 of the annual Maximum Permissible Doses. It is possible to make use of this recommendation to make individual monitoring simpler and more logical...

"In principle, the choice of an investigation level would correspond exactly to 3/10 of the appropriate annual Maximum Permissible Dose...

"In practice, however, such a precise equivalence cannot be achieved because most individual monitoring for external radiation is carried out over periods substantially shorter than a year. It is therefore recommended that, for the whole body, the investigation level for penetrating radiation should be set somewhat lower--50 mrem for a dosimeter issued for 2 weeks, 100 mrem for 1 month, or 300 mrem for 3 months. In the rare cases when a substantial part of the dose to an individual results from neutrons or from internal contamination, the investigation level should be applied to the total recorded dose in a control period rather than to each of the separate contributions...

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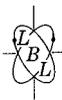
We shall attempt here to discuss the basic properties of each of these various types, as well as their various regions of applicability. We shall also indicate those new techniques which show possible promise for dosimetry, but still require extensive investigations and development.

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The advantages of film include the ability to re-scan emulsions after an exposure to determine details of dose depositions; use of filter shadowing and exposure-uniformity information to measure directionality of exposure and source distance; and the very wide variety of possible film types and grades (although in actual fact the variety of films now commercially available is quite limited).

Besides the physical or engineering considerations, two other properties of film are often cited: low unit cost, and long-term storage (for re-measurement and legal record-keeping).

Here we shall present data about some typical films to indicate the approximate ranges of various parameters. We shall not try to discuss and compare all of the various commercially available films and film services; because of their large number, that would be an almost insurmountable task. Suffice it to say that film badge services have come under the study of the International Atomic Energy Agency (IAEA) and the U.S. Public Health Service, and documents containing overviews of the important considerations have been in the open literature for several years (Ref. 2, 3, 4). Also, a 1962 book by Becker (Ref. 5) contains an excellent summary of the basic properties and considerations in film dosimetry.

The three basic operational characteristics to consider are accuracy, dynamic range, and energy response.

The intrinsic accuracy of film is quite good. In one study (Ref. 6) under carefully controlled conditions, and changing films every week, x-ray exposures to Kodak Type 3 film of 10 to 200 mR/week (0.5 to 10.0 R/year) have been determined to  $\pm 20\%$  for a single film and  $\pm 10\%$  (standard deviation) on a quarterly basis. For a single 10 mR exposure, standard deviations of  $\pm 25\%$  are possible. Of course, such "carefully controlled conditions" are not typical of actual operational use. One very important quality-control practice is to set aside and develop a few blanks and controlled-exposure films with each new batch of emulsions, to determine if any changes in sensitivity have occurred. The main point to be made here is that intrinsic accuracy is not a significant contributor to error in the use of film dosimeters.

The dynamic range of the more sensitive films is typically from a few mrad to a few thousand mrad, with films available which respond up to the megarad region; the typical dosimeter system might use films of both the most and least sensitive types in order to cover as large a dynamic range as possible. Film's linearity of response (emulsion density vs. dose) is excellent, until saturation sets in at the high-dose end. This is illustrated

in Figure 1 (from Ref. 7).

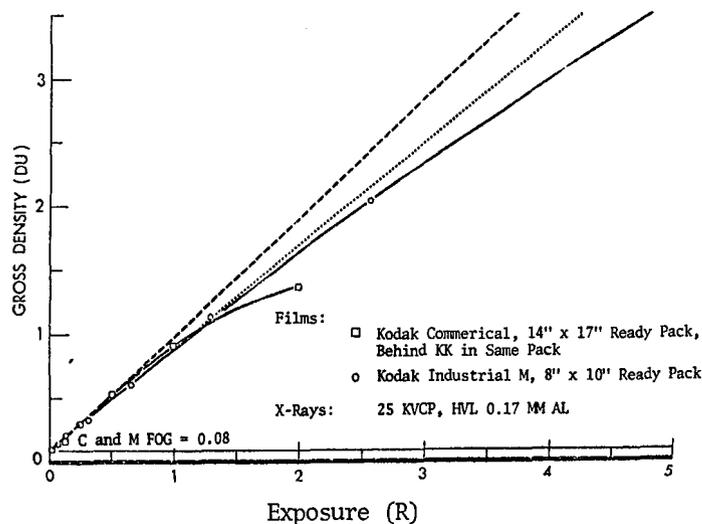


FIGURE 1. Optical density against roentgen exposure plotted on linear grid for Kodak Commercial film and for Kodak Type M films in Ready Packs (from Ref. 7).

The humidity and temperature effects are significant, however. Studies of DuPont 555 and Kodak Type 2 films have shown rapid losses of up to 40% in sensitivity at high humidity (Ref. 8). The effect is probably on latent image formation, possibly from "the action of the various radiolytic products of water upon the silver halide crystal, affecting either the migration of conduction band electrons, the fraction of silver ions free to migrate to the sensitivity centers, or both" (Ref. 8). Figure 2, from Becker (Ref. 14) shows the effect of 75 to 95% humidity, 23 to 32°C on optical density for a typical film.

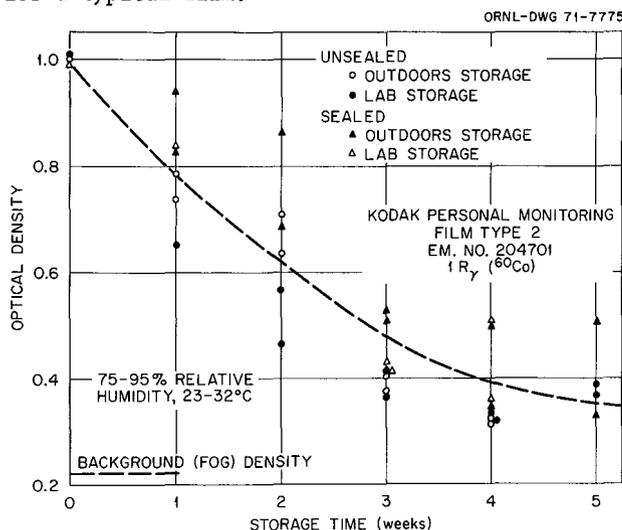
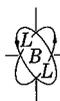


FIGURE 2. Effect of 75 to 95% humidity, 23 to 32°C on optical density of Kodak Type 2 Film, as a function of storage time (from Ref. 14).



The non-uniformity of photon sensitivity of films as a function of quantum energy is considered by some to be their greatest drawback. At low energies, the intrinsic sensitivity of emulsions is much larger than at medium energies, in some cases by big factors. Much effort has gone into the design of various types of filters to compensate for this energy dependence.

A typical example of the use of filters for decreasing the energy dependence is the work of Storm and Shlaer at Los Alamos (Ref. 9). Figure 3 shows the way in which the unfiltered response is modified by using filters with one element (Bi), two elements (Bi, Er) and six elements (Bi, Au, Ta, Er, Cd, and Nd). The idea is that the incident x-rays are attenuated by the K-edge absorption of each successive material, decreasing the effective dosimeter sensitivity. Figure 4 shows the measured response of four commercial films using this technique. Use of a lead frame back was necessary to eliminate a scattering problem in the plastic encasing the filters.

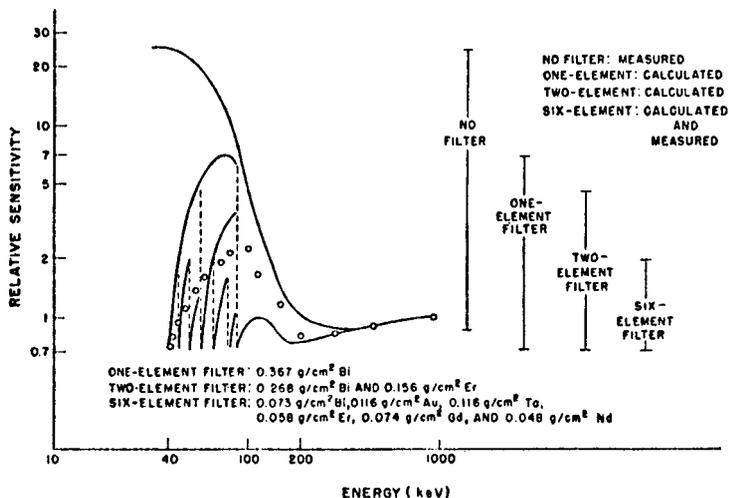
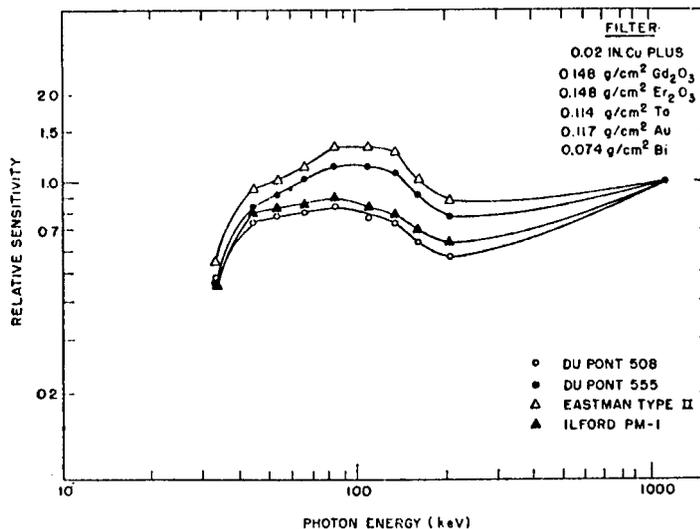


FIGURE 3. The photon response of DuPont 502 film unfiltered and filtered by one-, two- and six-element filters (from Ref. 9).

FIGURE 4. Response of DuPont 508 and 555, Eastman Type II and Ilford PM-1 films in the lead-frame plastic badge (from Ref. 9).



The conclusion of Storm and Shlaer is that it is possible to eliminate much variation in sensitivity at the low energies in film dosimeters, at least down to energies of a few tens of keV. It should be noted that multi-filter holders are relatively expensive, which has discouraged their wide use. At the other (high) end of the energy scale, response becomes poor at energies high enough that particle-equilibrium is not established upstream of the dosimeter (of course, this is true for all dosimeters under consideration in this section).

### B. TLD

Thermoluminescent dosimetry (TLD) is becoming more and more widely used for routine gamma/beta dosimetry. The basic physical principle is now reasonably well understood. Ionizing radiation incident upon a crystal elevates an electron from the valance band to the conduction band, leaving a hole in the valance band. The electron and the hole move through the crystal until they either recombine



or are trapped in metastable states, usually associated with impurity sites. When the crystal is heated, sufficient energy may be given to the electron (or the hole) so that it wanders around, eventually recombining with a counterpart hole (or electron). The recombination energy is released as visible light (a thermoluminescent photon).

A good description of basic TLD dosimetry can be found in the excellent book by Cameron, Suntharalingam, and Kenney (Ref. 10). Also, Cameron has compiled a bibliography on the subject (Ref. 11).

In perusing the literature on TLD, one is impressed by a tremendous amount of enthusiasm for TLD among its proponents. This impression can be had from even a brief study of the proceedings of recent conferences on the subject of TLD (Ref. 12, 13, 14). We would like to emphasize here that both film and TLD seem to have some advantages in particular situations, and can be used to complement one another for mixed-field dosimetry.

The advantages of TLD over film are several: (a) reproducible accuracies under laboratory conditions have been achieved at the  $\pm 5\%$  level; (b) the dynamic range is large, with doses of from a few mrem to many thousand rem measurable in the same crystal; (c) directional uniformity surpasses that of film: it is easier to design a dosimeter to reduce the directional dependence to  $\pm 20\%$  down to low x-ray energies; (d) fading is minimal: TLD's do not fade significantly with extreme climatic conditions; (e) energy response is more uniform at low x-ray energies; (f) TLD dosimeters are rugged and reusable.

The disadvantages include the destructive readout (once read out, the information is lost); the inability to determine directionality of dose deposition; and slightly higher costs than film. The commonest way to avoid the problem of data lost by accident is use of a back-up chip; this is now found in many systems.

The most common thermoluminescent crystal used today is lithium fluoride. There is extensive documentation of the properties of LiF; we shall try to show some of the typical data. For example, Figure 5 (Ref. 15) shows that the useful range of LiF extends up beyond  $10^5$  rem; a typical sensitive film is shown for comparison. The linearity of response is good, extending down to below 10 mrem on the low-dose end. The directional sensitivity is shown in Figure 6 (Ref. 15); again, the uniformity is excellent.

The energy response of a typical TLD is shown in Figure 7 (from Ref. 16). The enhanced sensitivity in the 20 keV region is much more easily handled by filtering techniques than is the (more extreme) problem with film.

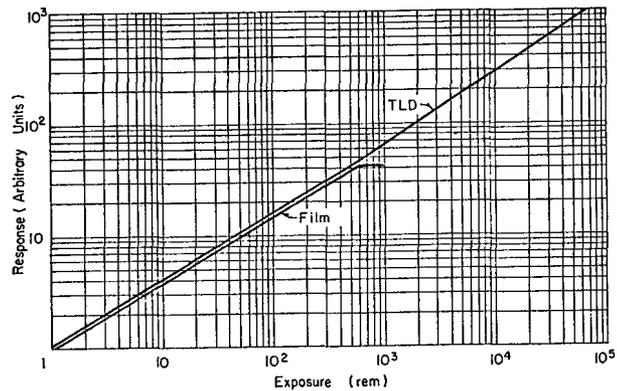


FIGURE 5. Useful range of TLD and film (from Ref. 15).

Still another advantage of TLD is its sensitivity to betas. LiF can be fabricated in very thin layers, and requires little wrapping, so that beta dosimetry is quite feasible.

Perhaps the two main reasons cited at laboratories where TLD has not replaced film are its difficulties when neutrons are present, and its destructive readout. We shall discuss the neutron measurement problem separately below.

The destructive readout is intrinsic to the method. One heats up the TLD chip, measures the integrated signal, and is then ready to re-use the chip. Systems are now commonly used to semi-automate the process; however, all information is lost, and in the event of some failure there is no second opportunity to fall back upon, except the back-up chip already mentioned.

Another frequently cited "disadvantage" of TLD (compared to film) is the legal, record-keeping requirement. The argument is given that possession of the actual exposed film dosimeter itself is somehow intrinsically preferable to the mere record from a TLD reading. However, it is now generally recognized that well-maintained records are sufficient evidence in any legal situation (Ref. 17).

The "fading" problem is an important consideration if dosimeters are not changed often. There are many situations in which large groups

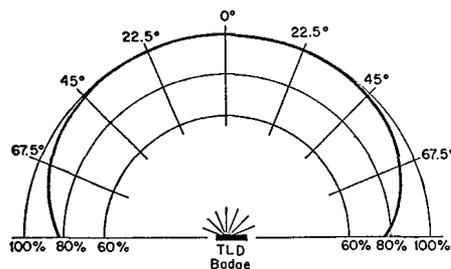


FIGURE 6. TLD dosimeter response vs. angle of incidence (from Ref. 15).

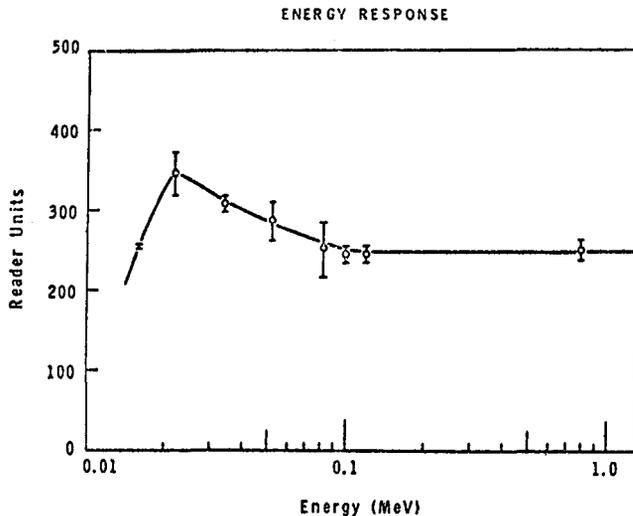
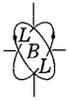


FIGURE 7. Energy Response of TLD-700 thermoluminescent material (from Ref. 16).

of employees are monitored with no expectation that any detectable exposure will occur. For example, at Lawrence Livermore Laboratory, semi-annual or annual processing cycles were desired for many employees whose exposure risk is minimal, but it was estimated that "the maximum period that film should be used is three months. In practice, a one-month exposure period was never exceeded because of the significant aging and fading characteristics of the film dosimeter" (Ref. 18). The fading of LiF, by contrast, is considered to be of only minor concern. However, in order to double-check for anomalous or incorrect readings, a back-up TLD chip (of calcium fluoride,  $\text{CaF}_2$ ) is used at Livermore; it is read out only in the event of such an anomaly (Ref. 18).

In addition to LiF, several other TLD materials have been studied and applied in personnel dosimetry. Among those in use today are  $\text{CaF}_2:\text{Mn}$ ,  $\text{CaSO}_4:\text{Tm}$ ,  $\text{CaSO}_4:\text{Dy}$ ,  $\text{Al}_2\text{O}_3$ , and fluorite. Some of these are used in combination with or as back-ups for LiF. Some of them also have properties which are improvements over one or more of the less desirable characteristics of LiF.

For example, although LiF is subject to tribothermoluminescence (mechanically induced luminescence) and other possible spurious luminescence,  $\text{CaSO}_4:\text{Dy}$  is significantly less susceptible: a factor of 30 less tribothermoluminescent response was measured in one study (Ref. 19). Also, this study showed excellent low-dose sensitivity for  $\text{CaSO}_4:\text{Dy}$  (down to  $\sim 0.2$  mR), which can be of occasional usefulness in dosimetry.

Another drawback of LiF is that its thermoluminescence is strongly dependent upon the

thermal history of the material (Ref. 20, 21), so that the annealing technique must be quite carefully controlled. This is much less true of  $\text{CaF}_2:\text{Mn}$  and  $\text{CaSO}_4$  dosimeters.

### C. Pocket Ionization Chambers

A frequently-used back-up or auxiliary dosimeter is a pocket ion chamber, usually pen-sized. It is typically used to assess (*in situ*) the dose being received by a worker in a high-radiation-level environment. These instruments are understood to be for x and gamma radiation only (although many will respond as well to beta radiation), and to be capable of either direct read-out or immediate indirect read-out with a separate instrument.

Specifications for such dosimeters have recently been established by the American National Standards Institute, Subcommittee N13.5 (Ref. 22). The electrodes of the ion chamber form a capacitor which is charged to a pre-determined voltage; ionization occurring in the chamber surrounding the electrodes causes a decrease in the charging voltage which can be read by an electrometer; or alternatively causes motion of a movable electrode, which motion can be read directly.

A typical instrument today can measure exposure (Roentgen) to about  $\pm 15\%$  of full scale for 200 mR full scale settings, and undergoes discharge (leakage) at rates of at most a few tenths of a milliroentgen per hour, for 200 mR full scale (Ref. 22). Since it is meant for use on at most a daily (8-hour or perhaps 24-hour) basis, it is a very useful instrument for daily or even hourly dose assessment. Pocket ion chambers can now be obtained in a variety of rugged, sealed, environment-insensitive design; their disadvantage is an almost total insensitivity to betas, alphas, and neutrons.

### D. Other Dosimeters

Several dosimetry techniques are now in the research-and-development stage, and are not yet fully operational. Among the concepts being studied for gamma dosimetry are radiophotoluminescence (RPL) and thermally stimulated exoelectron emission (TSEE). Neither RPL nor TSEE is now used in any commercially available systems.

#### (i) RPL

In radiophotoluminescence, radiation produces defects in a solid (usually a glass); ultraviolet light can subsequently be used to produce fluorescence in the glass. The amount of fluorescence is related to the radiation absorbed. UV stimulation is not a destructive readout, however, which is a clear difference from



TLD. On the other hand, because dose is continually integrated an RPL dosimeter eventually saturates and must be discarded.

Figure 8 (Ref. 23) shows the absorption and luminescence response curves for one typical RPL glass.

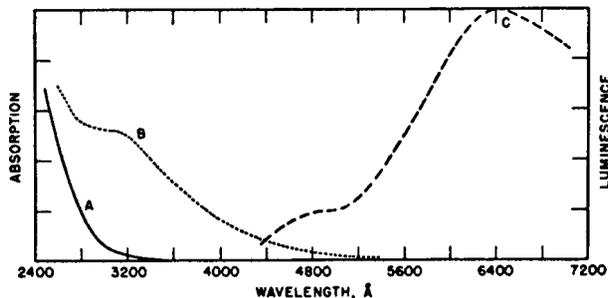


FIGURE 8. Typical absorption and luminescence curves for an RPL glass: (A) absorption characteristics prior to irradiation, (B) absorption after irradiation, (C) luminescence produced by illumination in the absorption band (from Ref. 23).

Most RPL glasses suffer from fading over long periods of time. Another problem is the "pre-dose" RPL, which is the amount of response even before irradiation. Today, pre-doses equivalent to a few hundred mrad are about the lowest available (Ref. 24). Among the glasses being investigated are aluminum and lithium phosphates and lithium borate, all doped with silver phosphate. Figure 9 shows the calculated energy dependence of a lithium borate RPL glass containing various amounts of  $\text{AgPO}_3$ . It can be seen that this material is quite energy-independent down to a few tens of keV. Among other drawbacks (Ref. 26) of glass dosimeters are the poor sensitivity to  $\beta$ 's of encapsulated glasses; and the relative insensitivity to low energy x-rays ( $\leq 50$  keV).

Recent summaries by Cameron (Ref. 23), Becker (Ref. 24, 27) and Yokota (Ref. 26) can be consulted for more detailed information about RPL. We will summarize here by stating that RPL has several attractive features, which still require development work to produce a workable beta/gamma dosimetry system.

#### (ii) TSEE

Thermally-stimulated exoelectron emission (TSEE) is a process quite similar physically to thermoluminescence; indeed, many substances exhibit both TSEE and thermoluminescence. The basic concept is

that radiation impinging on the TSEE material elevates an electron from the valance band to the conduction band, leaving a hole in the valance band. The electron and hole move about until trapped in impurity sites. When the crystal is heated, the electron may pick up enough energy to overcome the work-function and be emitted: hence, thermally-stimulated exoelectron emission. The number of emitted electrons is measured in some counting or integrating device (e.g., G-M tube, ionization chamber).

Reviews of recent research in TSEE dosimetry have appeared in the two most recent conferences on TLD (Ref. 13, 14). Becker (Ref. 29, 30) has also recently reviewed the subject. Among the features which commend TSEE are extreme sensitivity (down to even the  $10^{-6}$  R region); significantly reduced problems with poor crystal purity; low background and "pre-dose" characteristics; the small thickness of the sensitive layer (perhaps less than 10 nm); and the possibility of excellent energy response down to low x-ray energies, by the use of several materials. The disadvantages, at present, include poor "fading", so that readout must take place within at most a few days of exposure; and the more complicated read-out system required. At present, TSEE still requires considerable work on both the basic physics and the engineering details, before it can supplement the more widely-used systems.

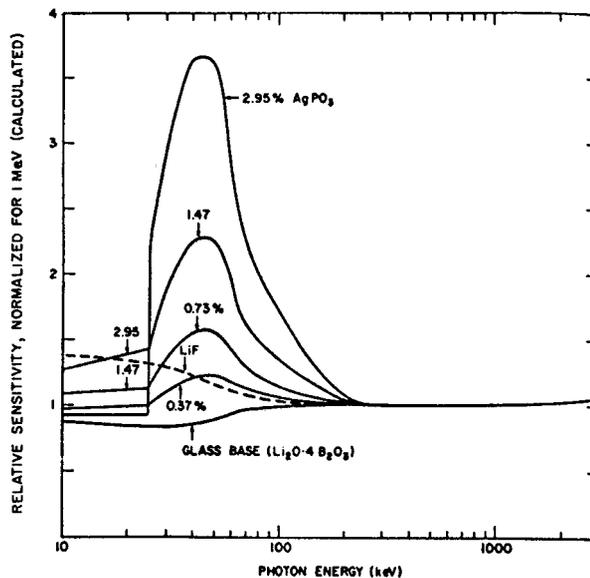


FIGURE 9. Calculated energy dependence of a lithium borate RPL glass containing different amounts of silver phosphate (from Ref. 23).



(iii) Track Etching

This technique relies upon the fact that certain materials suffer local radiation damage from very high-LET radiation, the sites of which are preferentially attacked by certain etchants. This technique, which is principally used for neutron dosimetry, will be discussed below under "Thermal Neutrons".

5. NEUTRON DOSIMETRY

The problem of neutron personnel dosimetry is discussed separately from the gamma/beta problem for three reasons: First, a large fraction of radiation exposure involves no neutron fluxes at all; second, the present instrumentation for neutron dosimetry is in many important cases not adequate to the task, unlike that for gamma/beta dosimetry; and third, the dosimetry of high-LET radiations (of which neutron dosimetry is but one part) has been of increasing concern in recent years.

For details on the various energy divisions, properties, and relative importances of neutron fluences, the reader is referred to "Neutron Monitoring" elsewhere in this volume. In what follows here, we shall assume a familiarity with that section.

Baarli (Ref. 31) has discussed the problem of personnel dosimetry in mixed radiation fields around high-energy accelerators; the conclusion is drawn that present techniques are adequate only if the approximate composition and energy spectra of the component radiations are understood reasonably well.

At the present time, separate dosimeters are in common use for thermal and for fast neutron detection. Intermediate-energy personnel dosimetry is not often attempted, and high-energy neutron personnel dosimetry is performed using techniques (mainly, foils activation) already described in detail in "Neutron Monitoring."

There are basic physical reasons for the difficulty in monitoring intermediate-energy neutrons using a portable dosimeter. Thermal neutrons can be captured in materials with high thermal-neutron cross-sections, and fast neutrons can be detected by the charged secondaries produced in (n,p) interactions in hydrogenous (or other low-Z) material. However, neutrons in the intermediate-energy region (between ~0.5 eV and ~200 keV) are not easily detected except by the use of a moderator and a thermal-neutron detector.

As discussed in "Neutron Monitoring," relatively few situations exist in which a

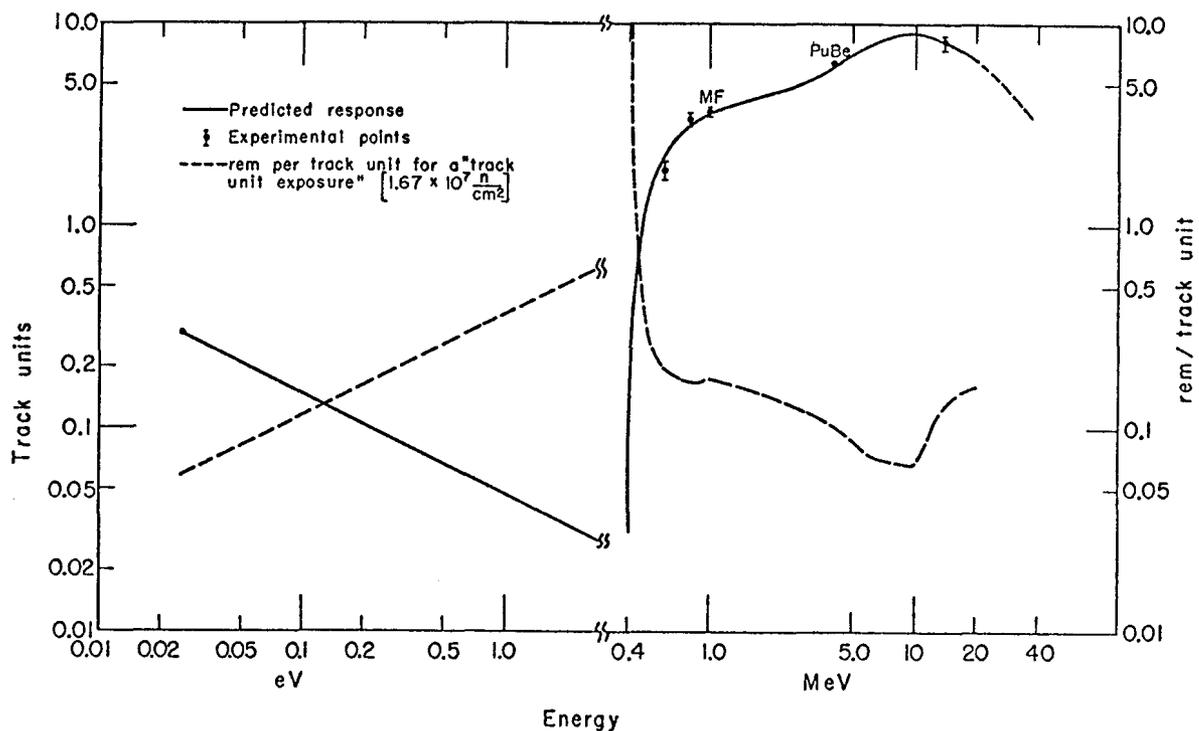


FIGURE 10. Response of NTA packets to neutrons of various energies (from Ref. 32).



substantial fraction of the dose is from thermal neutrons; the main exception is near a thermal column at a reactor. However, thermal-neutron detectors are important because many indirect methods of assessing higher-energy neutron fluences actually rely on thermal-neutron detectors.

We shall begin by discussing fast-neutron and then thermal-neutron dosimeters, and finally body backscattering and albedo dosimeters.

### A. Fast Neutrons

The basic technique most commonly used for fast neutrons (~200 keV to ~20 MeV) is the nuclear track emulsion. A typical emulsion of this type is Eastman Kodak Type A (often called "NTA") emulsion, which has been described in detail by Lehman (Ref. 32) and Becker (Ref. 5). NTA is a special fine-grained film, typically 25 to 35 microns thick. Neutrons are detected by the trail of ionization from charged particles released by one of three mechanisms: (a) elastic (n,p) interactions with hydrogen, which register in NTA above about 0.4 MeV; (b) exoergic (n,p) reactions with nuclei, which register in NTA below about 10 eV and then only on nitrogen nuclei; and (c) inelastic nuclear interactions resulting in a "star", significant about 20 MeV.

The energy response of NTA is shown in Figure 10, from Lehman (Ref. 32). The sensitivity is seen to be significant only for fast and thermal neutrons. Standard practice is to report the response of neutron film in tracks per unit area, as seen through a microscope. There are difficulties in relating the measured quantities to neutron flux (and hence to dose-equivalent), mainly because some of the response comes from interactions in the emulsion wrapping, and some from interactions in the emulsion itself; varying emulsion thicknesses (batch to batch) complicate calibration procedures. One problem with this class of detectors is that it is susceptible

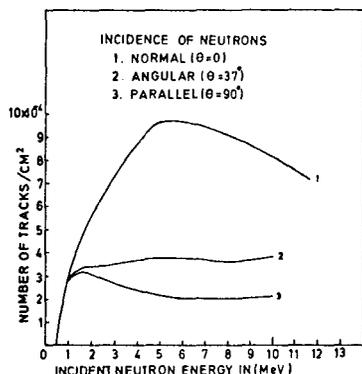


FIGURE 11. Estimated NTA film response (from Ref. 34).

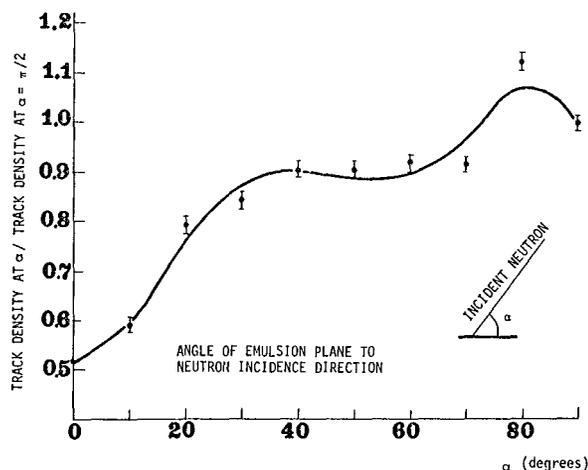


FIGURE 12. NTA film badge response as a function of neutron incidence angle (from Ref. 35).

to hydration at high humidities, which increases the hydrogen content and hence the fast-neutron sensitivity. This and other problems (temperature effects, storage problems) have been discussed by Lehman (Ref. 32) and others (Ref. 33, 34, 35). Figures 11 (Ref. 34) and 12 (Ref. 35) show the fast-neutron response of NTA film as a function of incident angle.

Figure 13 (Ref. 35) shows the effect of 70% relative humidity on latent image stability of NTA film. Desiccation followed by the heat-sealing of Kodak NTA Type B film in moisture-proof bags has been successfully used for several years by a British group. Their unpublished results (Ref. 45) show that the 2- to 4-week latent-image half-life of NTA film can be extended to about 8 weeks using this technique. The bags are made of a laminate of paper/0.012 mm Al foil/polyethylene. Marshall and Stevenson (Ref. 36) have considered the edge-effects problem in scanning of emulsion, while Oshino (Ref. 37) has studied the energy dependence of NTA worn on a phantom.

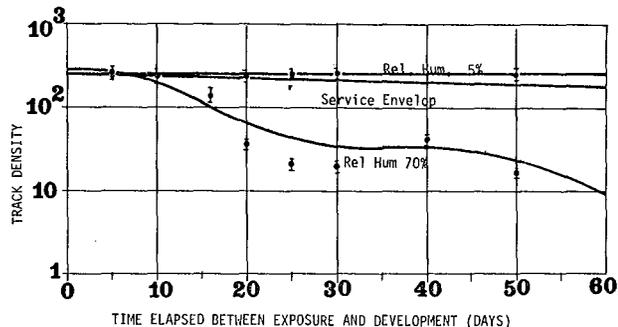


FIGURE 13. Effect of 70% relative humidity on latent image stability of NTA film (from Ref. 35).



Although fast neutron dosimetry has been carried out for almost two decades using nuclear track emulsions, it is clear that some features are in need of improvement. However, in situations where neutron spectra are known to be dominated by fast neutrons, and where high humidity is not a problem, this technique can be used with reasonably good results.

Two other methods of fast neutron personnel dosimetry involve foils activation or fission fragment foils, in either case resulting in darkening of film by charged particles (from decay or fission fragments). These will be touched on briefly in the next section, on thermal neutron detectors.

### B. Thermal Neutron Detectors

Thermal neutron personnel dosimetry is commonly performed today with one of three means: thermoluminescent dosimeters, fission foil dosimeters, or film surrounded by a foil which is activated and then decays.

The commonly used TLD method relies upon two isotopic compositions of lithium fluoride,  ${}^6\text{LiF}$  and  ${}^7\text{LiF}$ . Lithium-6 has a thermal-neutron cross-section of about 960 barns for the process  ${}^6\text{Li}(n,\alpha){}^3\text{H}$ ; the cross-section of lithium-7 is smaller by a factor of about 25,000. The  ${}^7\text{LiF}$  TLD responds to photons with the characteristics already described, while the  ${}^6\text{LiF}$  responds to both photons and thermal neutrons. Harvey et al. report that the response of a  ${}^6\text{LiF}$  TLD to 1 rem of thermal neutrons is approximately 80 times the response to 1 R of gamma radiation (Ref. 28), and that the minimum detectable thermal neutron dose is  $\leq 0.5$  mrem. However, "it is necessary to calibrate each batch of  ${}^6\text{LiF}$  separately, as significant differences in sensitivity may occur between batches, and the ratio of neutron to gamma sensitivity is not constant" (Ref. 28). Fast neutron sensitivity of  $\text{LiF}$  is insignificant. "The response in equivalent gamma rads per fast neutron tissue rad is roughly proportional to neutron energy, ranging from 1% at 1 MeV to 10% at 10 MeV" (Ref. 28).

Many of the basic advantages of and problems with TLD's are identical to those of TLD's in general, which have already been discussed.

The fission foil technique exploits the property of certain crystalline materials, plastics, and glasses, whereby radiation damages the materials so that certain etchants will preferentially attack the damaged areas. In particular, some materials only produce "tracks" (etched pits or holes) when the linear energy transfer (LET) exceeds some critical rate. Neutrons themselves, and gammas, do not provide the minimum LET, while fission fragments do; hence, the fission fragment technique, in which a fissionable foil is placed next to a plastic

detector, and the integrated thermal neutron fluence subsequently determined by etching and microscopic scanning or electronic counting. Typical track sizes of  $0.1 \mu$  are observed. Becker (Ref. 38) has prepared a review of track etching.

These dosimeters are relatively unaffected by humidity, high temperature, and mechanical shock. Using  ${}^{235}\text{U}$ , detection limits of as low as  $10^{-6}$  rem have been reported for thermal neutrons (Ref. 30); and using other nuclides ( ${}^{238}\text{U}$ ,  ${}^{232}\text{Th}$ ,  ${}^{237}\text{Np}$ ) with thresholds in the MeV region, detection limits in the 0.5 to 1 mrem range for fast neutrons are also reported. A discussion by Unruh gives some of the important considerations in applying this technique (Ref. 39). Another discussion by Widell (Ref. 40) indicates that mica is an excellent substance for image formation, but that residual trace uranium in some natural micas produces a high latent background, which must be "erased" before use by baking for several hours at  $500^\circ\text{C}$ .

Still another thermal-neutron technique is activation. One approach uses film surrounded by cadmium, with emulsion darkening from the gammas due to the reaction  ${}^{113}\text{Cd}(n,\gamma){}^{114}\text{Cd}$  (Ref. 34). Unruh, et al. (Ref. 39) describe a system in which a series of three foils is activated, with subsequent exposure of emulsion by foil decay. The three foils are tin-rhodium (sensitive to gammas, thermal, and "body moderated" fast neutrons), cadmium-rhodium (gammas, "body moderated" fast neutrons), and tin-iron (mainly gammas). The various radiations are determined independently using three film-darkening measurements, and by solving a set of simultaneous algebraic equations.

### C. Albedo Neutron Dosimeters

A fundamental consideration in neutron personnel dosimetry is that of backscatter or albedo. We quote from Hankins (Ref. 41):

"When the human body is exposed to neutrons, some of the incident neutrons are backscattered to create a flux of neutrons of various energies leaving the body. These neutrons are called albedo neutrons, and a dosimeter placed on the body to measure this flux of backscattered neutrons is called an albedo-neutron dosimeter. Such dosimeters are usually designed to detect thermal neutrons, and when cadmium is used to eliminate incident thermal neutrons, the dosimeter response is largely from the thermal neutrons that return from the body."

The power of albedo dosimetry is that it has sensitivity in the energy range where very few other techniques work: for intermediate-



energy neutrons (~0.5 eV to ~200 keV). There are many situations in which a significant fraction of the dose equivalent is from intermediate-energy neutrons, and in which personnel dosimetry is desired.

Unfortunately, many of the albedo-neutron dosimeters in present use have a significant energy dependence, which decreases their usefulness in situations where the neutron energy spectrum is unknown. Where the spectrum is known, correction factors can be applied to compensate for energy dependence and arrive at a reasonably accurate ( $\pm 30\%$ ) dose-equivalent measurement.

A recent study by Hankins (Ref. 41) examined the way in which various design parameters affect dosimeter response. Hankins used small TLD chips ( $^6\text{LiF}$ ) encased in various thicknesses of 2" diameter polyethylene (CH), all surrounded by cadmium. He studied the effects of placement of the TLD in front of, within, and behind the CH, and the effect of removing the cadmium on the back of the badge. His most important conclusion was that almost none of his parameters significantly affected the rather large energy-dependent sensitivity. On the positive side, he found that adding CH increased the dosimeter sensitivity; that with total encasement of cadmium the sensitivity is not significantly affected by distance away from the body, at least up to 3 cm; that sensitivity is greatly affected by shielding thickness between source and dosimeter; that if cadmium is not present on the back (body side), most of the response is from albedo rather than incident neutrons, except for thermal neutron exposures on a totally cadmium-enclosed dosimeter; and that the placement of both  $^6\text{LiF}$  and  $^7\text{LiF}$  chips in the same or equivalent positions is critical.

The energy dependence as found by Hankins shows a monotonic decrease of a factor of about 10 between 200 keV and 3 MeV neutron incident energies, with comparably high sensitivities for the intermediate and 200 keV cases.

Similar results have been reported by Hoy (Ref. 42) at Savannah River, where one of the main monitoring problems is around a plutonium production plant, with a typical  $^{238}\text{PuF}_4$  neutron spectrum mainly between 0.3 and 25 MeV. The dosimeter (a 2"-diameter two-section concentric hemisphere, separated by cadmium, using  $^6\text{LiF}$  and  $^7\text{LiF}$  TLD, and worn on a belt) gave adequate dose-equivalent measurements.

Another albedo dosimeter has been developed in the United Kingdom (Ref. 28). TLD's ( $^6\text{LiF}$  and  $^7\text{LiF}$ ) are housed in a plastic cover loaded with boron carbide. The design criterion was as follows: for an unshielded 1/velocity (1/v) detector at the surface of the body, the response to the albedo from incident thermal

neutrons exceeds that to the albedo from incident 10 keV neutrons by a factor of about 10. The dosimeter was designed to modify this response to decrease the thermal and epithermal sensitivity: hence the boron. The resulting sensitivity is shown in Figure 14 (Ref. 28). Note that the sensitivity is reasonably flat from thermal energies to 10 keV, which is a wide range indeed.

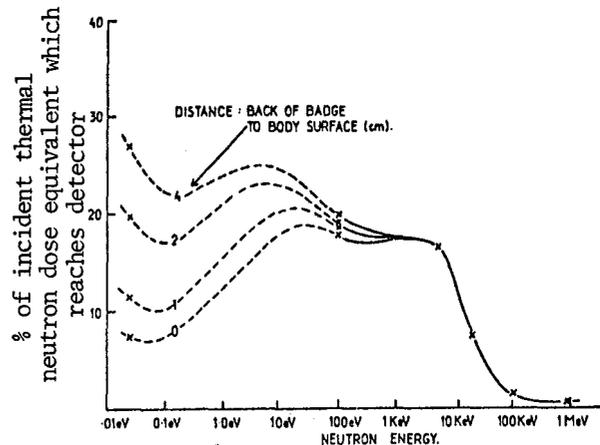


FIGURE 14. Response of albedo dosimeter worn on the body as a function of incident neutron energy (from Ref. 28).

## 6. SPECIAL DOSIMETRY PROBLEMS

Although the most common monitoring problem in personnel dosimetry is whole-body exposures, there are often requirements for dosimetry in other situations. Perhaps the most familiar of these is measurements of doses to the extremities (particularly to the hands).

For this purpose, several specially designed classes of integrating dosimeters have been developed. Thus, dosimeters to be worn on the fingers are now in common use, and dosimeters for mounting in shoes have also been designed.

Here we will not treat these in detail; instead the reader is referred to the literature. The main operational difficulties with such instruments are related to the validity of the measurement: the accuracy of the particular dosimeter used is by no means the most important consideration. For example, dose may be accrued through contact with radioactive material, a common problem in radiochemistry laboratories. Here even finger dosimeters can give incorrect results in many cases, since low-energy betas can give surface skin doses undetectable by some dosimeters.

The main point to be made here is that users must investigate very carefully the particular radiation field being monitored, to assure that the measured dose is meaningfully related to the dose received.



A discussion of finger dosimetry can be found in Ref. 43 and 44.

## 7. SUMMARY

The aim of this section has been to discuss instrumentation for the measurement of integrated external radiation doses to personnel. We have attempted to outline the types of measurements needed; the accuracies required; the intrinsic limitations imposed by the sampling problem; and the capabilities and drawbacks of presently existing instrumentation.

We will summarize by reiterating several points made in the main text:

a) The accuracy required of personnel dosimetry systems is not excessively great: the ICRP (Ref. 1) has recommended maximum uncertainties of less than 50% (or less than 1 rem, whichever is greater) in assessment of annual dose equivalent.

b) One key requirement of any dosimetry system is to make sure that no significant component of the radiation field goes completely unmeasured: in cases of large accidental exposure, such an omission would be regrettable indeed.

c) For routine monitoring of external beta/gamma radiations, adequate instrumentation exists today. While improvements are

definitely possible (and in some cases imminent, or at least probable from the current research program in dosimetry), the basic task of beta/gamma dosimetry is now being performed reasonably well.

e) There is still need for a dosimeter useful in those situations dominated by x-rays below about 15 keV.

f) Thermal-neutron dosimetry can now be performed adequately with any of several dosimetry systems.

g) Fast-neutron dosimetry has largely relied for more than a decade upon the nuclear-track-emulsion technique, which suffers from several important drawbacks. Research on a number of possible alternate systems ought to be encouraged.

h) Intermediate-energy neutrons (~0.5 eV to ~200 keV) are still a difficult measurement problem. Albedo dosimeters are not now sufficiently well understood to allow their use in neutron fields with unknown energy spectra, and no other systems now seem capable of performing this task adequately.

i) More work is required to develop a simpler way of assessing the energy spectra of external radiation fields, to complement the measurement of dose-equivalent. For cases of accidental exposure this is particularly important, and has not (in our opinion) been given sufficient attention.

## 8. ACKNOWLEDGMENT

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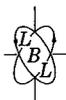


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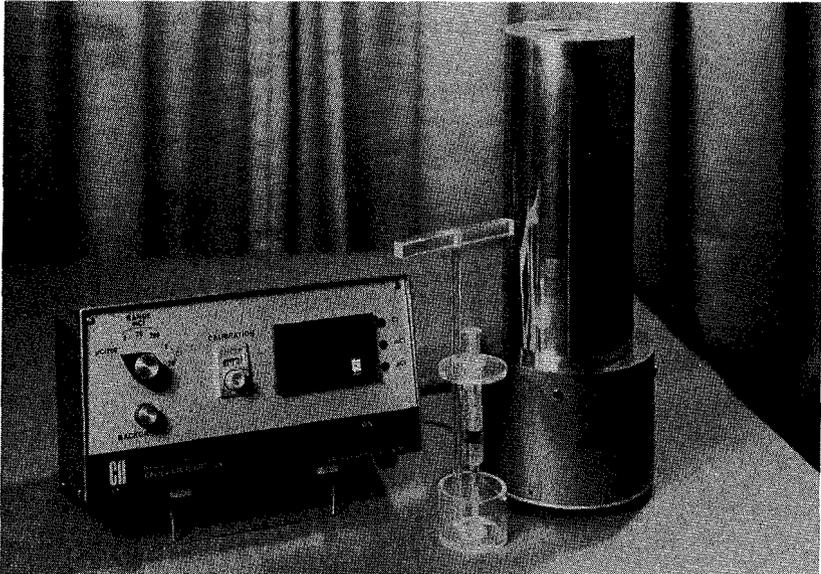




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FOR ENVIRONMENTAL  
MONITORING

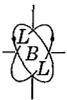
RAD-DOS  
Gamma, X-Ray  
Capintec  
Jan. 1973

Exposure Rate Meter  
Capintec Model 192

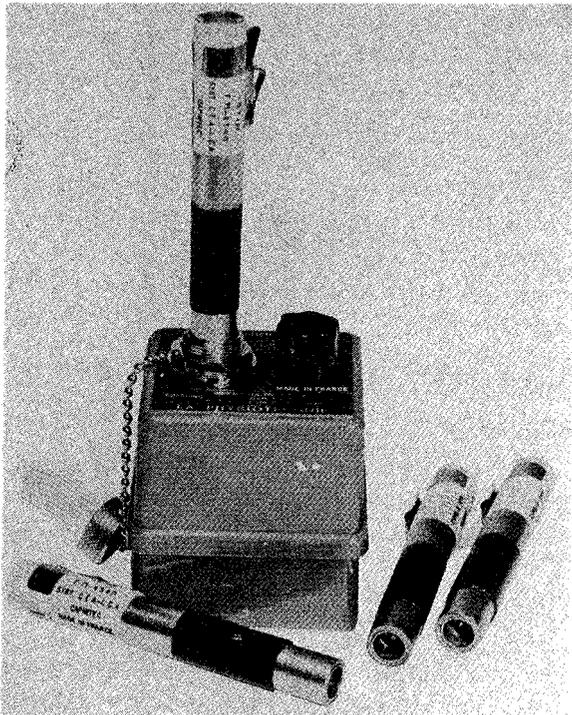


Class	Dosimeter System
Principle of Operation	Thimble-type ionization chambers and electrometer electronic system. Direct reading digital output in mR, R, mR/minute, R/minute.
Sensitivity and Range	20 mR (mR/min) full scale, to 2000 R (R/min) full scale
Performance	Temperature stability $\pm 0.1\%$ per $^{\circ}\text{C}$
Requirements	Power: 117 V ac, 50/60 Hz, 10 W
Features	<ol style="list-style-type: none"> <li>1. Calibrate X-ray diagnostic machines.</li> <li>2. Calibrate X-ray/cobalt-60 therapy machines.</li> <li>3. X-ray head leakage checks.</li> <li>4. Scattered radiation surveys.</li> <li>5. Intracavitary measurements.</li> </ol>
References	Manufacturer's specifications
Cost	Model 192 System: \$1750. Special Purpose Probes: \$500. each
Address	Capintec Inc. 63 East Sandford Boulevard Mount Vernon, NY 10550 (914) 664-6600

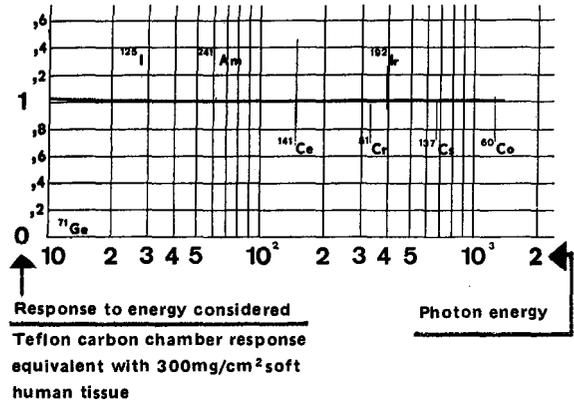




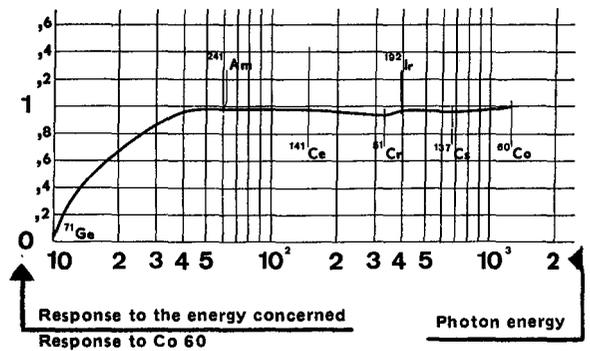
Personnel Dosimeter  
Capintec SEQ-5, 6, 7 and Charger



SEQ-6 SPECTRAL RESPONSE



SEQ-5 SPECTRAL RESPONSE



Class	Portable Dosimeter		
Principle of Operation	Tissue-equivalent ionization chambers, with charging system. Direct-reading output in rad.		
Sensitivity and Range	<u>Model Number</u>	<u>Sensitivities (Full Scale, in Rad)</u>	<u>Effective Wall Thickness (mg/cm<sup>2</sup>)</u>
	SEQ-5	0.2, 0.5, 2.0	300
	SEQ-6	0.2, 0.5, 2.0	330
	SEQ-7	0.5, 0.6	7 or 300
Sampling	Continuous		
Performance	Temperature: Range: -20°C to +60°C		
Requirements	Charger Power: 1.5 V battery Size: Dosimeter Sizes: Pen-sized		
Features	<ol style="list-style-type: none"> <li>1. Charger has a "press-to-charge" system to increase battery life.</li> <li>2. Dosimeters have tissue-equivalent walls.</li> <li>3. Dosimeters are direct-reading.</li> <li>4. Low energy response: is flat down to 10 keV.</li> </ol>		
Cost	Charger: \$55. Dosimeters: \$75. to \$175.		
Address	Capintec Inc. 63 East Sandford Boulevard Mt. Vernon, NY 10550 (914) 664-6600		



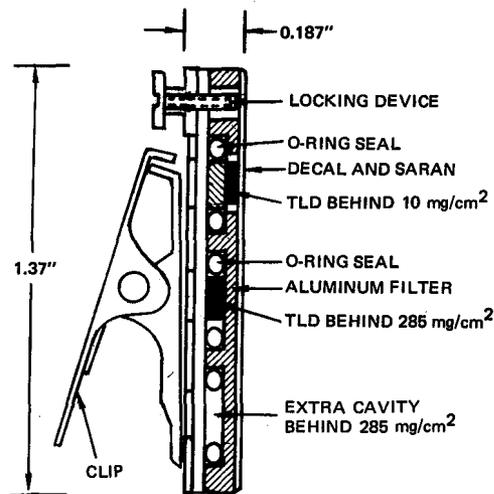


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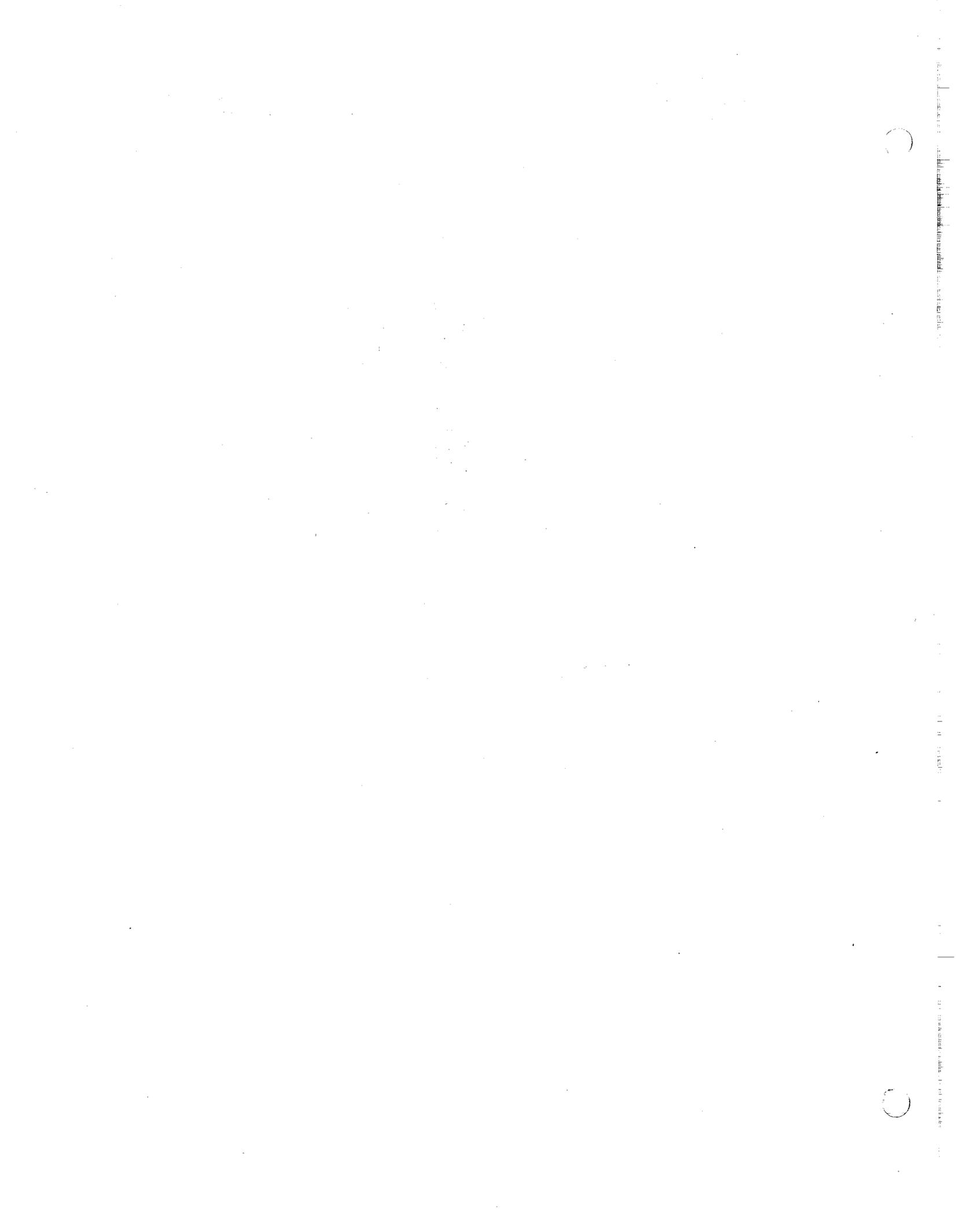
RAD-DOS  
Dosimetry Service  
Eberline  
July 1973

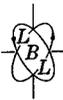
TLD Dosimetry Service

Eberline



Class	Thermoluminescent Dosimetry Service (TLD)														
Principle of Operation	TLD dosimeters with various parameters are supplied, returned for processing, read out, and results reported. Read out is by heating and observing thermoluminescence.														
Sensitivity and Range	<table border="0"> <tr> <td>Exposure, mrem:</td> <td>5</td> <td>10</td> <td>50</td> <td>300</td> <td>3,000</td> <td>30,000</td> </tr> <tr> <td>Standard Deviation:</td> <td>20%</td> <td>10%</td> <td>4%</td> <td>4%</td> <td>4%</td> <td>4%</td> </tr> </table>	Exposure, mrem:	5	10	50	300	3,000	30,000	Standard Deviation:	20%	10%	4%	4%	4%	4%
Exposure, mrem:	5	10	50	300	3,000	30,000									
Standard Deviation:	20%	10%	4%	4%	4%	4%									
Features	<ol style="list-style-type: none"> <li>(1) Badge has two TLD chips, behind 10 and 285 mg/cm<sup>2</sup>, to enable calculation of beta or non-penetrating x-ray dose.</li> <li>(2) Dosimeter contains natural lithium.</li> <li>(3) Wrist, ankle, ring badges also available.</li> <li>(4) Environmental badges (5 LiF chips) available in weatherproof bag.</li> <li>(5) Quarterly exchange is typical. Other cycles available.</li> </ol>														
References	Manufacturer's specifications														
Cost	\$1.50 per exchange														
Address	Eberline Instrument Corporation P.O. Box 2108 Santa Fe, NM 87501 (505) 982-1881														

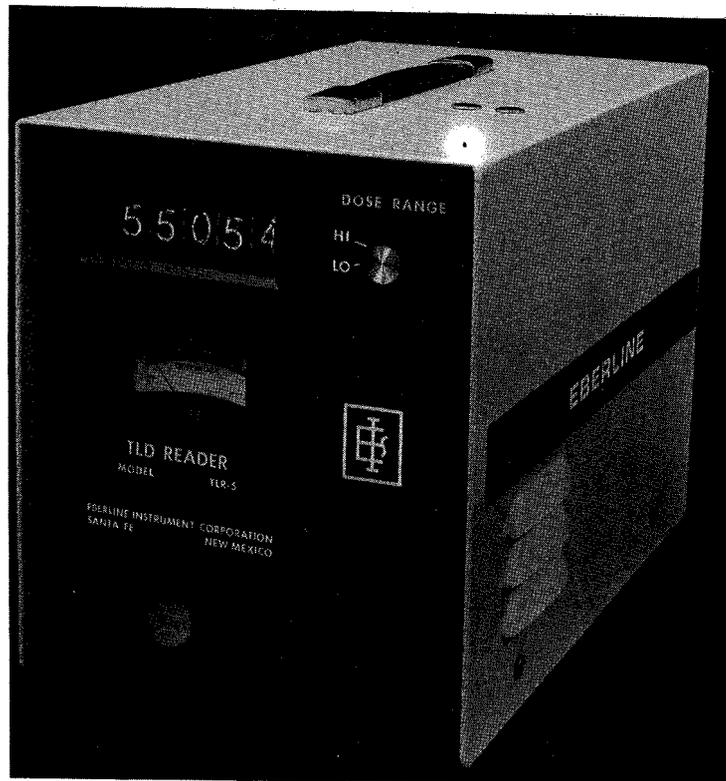




INSTRUMENTATION  
FOR ENVIRONMENTAL  
MONITORING

RAD-DOS  
TLD Reader  
Eberline 2  
July 1973

TLD Reader  
Eberline TLR-5



Class	Laboratory
Principle of Operation	Thermoluminescent Dosimeter (TLD) Reader
Sensitivity and Range	1 mR to $10^5$ R in two ranges: 1 mR to 100 R 1 R to $10^5$ R
Sampling	Each TLD requires 6 to 60 second cycle
Requirements	Power: 117 Vac, 50/60 Hz, 120 W Size: 12" x 10" x 18" (30 x 25 x 45 cm) Weight: 36 lb (16 kg)
Features	(1) 5-digit display. (2) Glow curve signal available on rear banana jacks, 0 to 10 mV. (3) Background less than one count plus 10 mR, referred to 35 mg LiF. (4) Gas flowmeter available as option. (5) All solid-state circuitry.
References	Manufacturer's Specifications
Cost	\$2300
Address	Eberline Instrument Corporation P.O. Box 2108 Santa Fe, NM 87501 (505) 982-1881





INSTRUMENTATION  
FOR ENVIRONMENTAL  
MONITORING

RAD-DOS  
Gamma  
E.G. & G.  
Mar. 1972

TLD Dosimeter Reader  
E.G. & G. 8506B

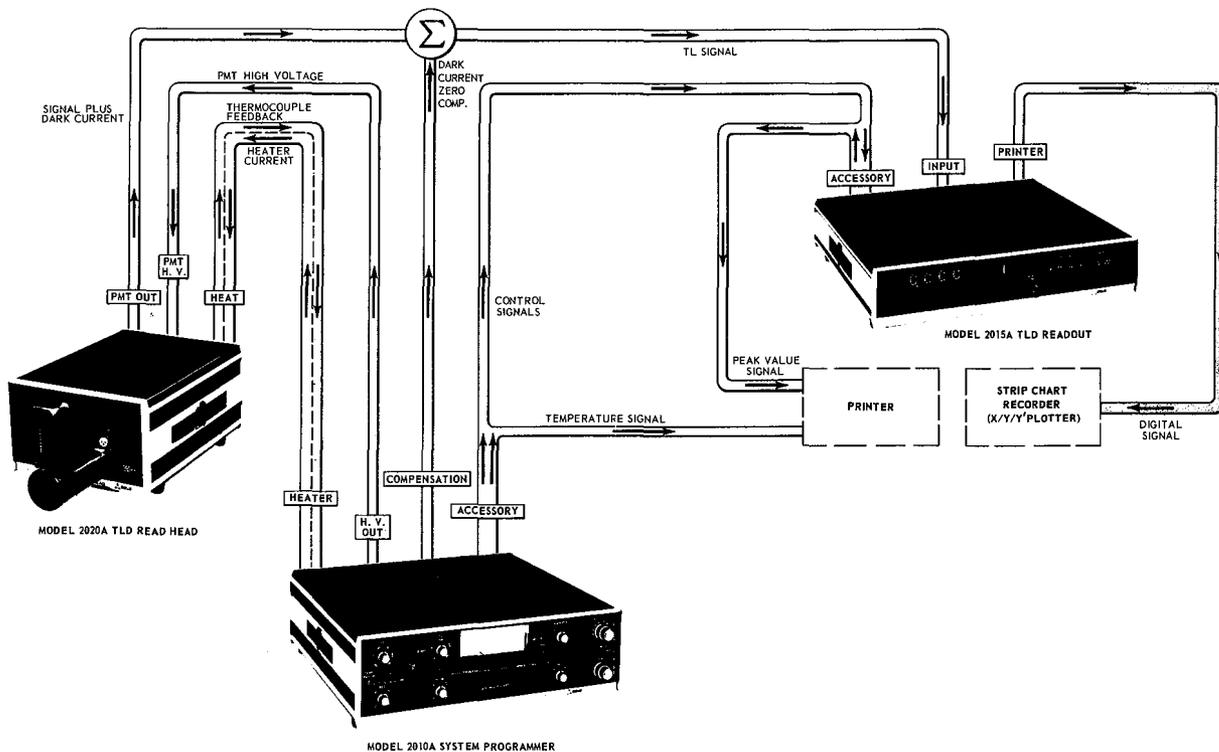


Class	Laboratory, TLD
Principle of Operation	Accepts a TLD (thermoluminescent detector) in a light-tight chamber, heats it up with a regulated current, samples the glow curve peak, digital electronic display
Sensitivity and Range	Energy Sensitivity: See tables for TLD's Range: 0.1 R to 18 KR
Sampling	Peak of glow curve
Performance	Accuracy: $\pm 2\%$ or 0.1 R whichever is greater Read out time: < 20 sec Temperature range: $-10^{\circ}\text{C}$ to $40^{\circ}\text{C}$ Humidity Range: 0-70%
Requirements	Power: 115 V ac, 50-60 Hz, 1A Size: 13 cm, 28 cm, 40 cm (5", 11", 17") Weight: 11.2 Kg (25 lbs)
Features	Digital presentation
Cost	\$2950
Address	E.G. & G. Inc. Electronic Product Division 35 Congress Street Salem, Mass. 01970 (617) 745-3200





Dosimeter System  
E.G. & G. Model 2005A System



Class	Lab
Principle of Operation	The system comprises a model 2020A Read head, a model 2010A Programmer, and a model 2015A Readout. The Read head is capable of housing, heating, and collecting light from a TLD, the programmer allows the operator to control the heat/time functions, and the readout is a digital, peak current reading pico-ammeter.
Sensitivity and Range	Energy Sensitivity: See Tables for TLD's Range: 0.5 mR to $10^8$ mR
Sampling	Peak of the glow curve
Performance	Accuracy: +0.05% to +12% for the model 2015A Temperature: $25^\circ\text{C} \pm 5^\circ/\mu\text{C}$ Readout time: Relative humidity: 0-50%

Requirements	Model	2020A	2010A	2015A
Power:	From 2010		117V ac, 60Hz, 15° W max	117V ac, 60Hz
Size:		8"W, 11.5"L 6"D	17"W, 17"D, 5.3"H	17"W, 17"D, 3.5"H
Weight:				20 lbs.



Features Optional printer, stripchart recorder, or X-Y plotter

Cost

Address EG & G Inc.  
Electro-Optics Division  
35 Congress Street  
Salem, Mass., 01970  
(617) 745-3200

### LIF DOSIMETERS

Model	TL Phosphor	Config-uration	Dose Range			Dose Rate Dependence R/sec	Fast Neutron Response P <sub>0</sub> B <sub>0</sub> R/n/cm <sup>2</sup>	Thermal Neutron Response R/n/cm <sup>2</sup> Co <sup>60</sup> Rads	Linearity	Repro-ducibility for Dosimeter	Re-sponse Curve	Figure Refer-ence	Size
			Model 2005A	Model TL-3B	Model 8506B								
TL-021	Lif	Mini	10mR-10 <sup>9</sup> R Nonlinear >10 <sup>9</sup> R	10mR-10 <sup>9</sup> R Nonlinear >10 <sup>9</sup> R	200mR-10 <sup>9</sup> R Nonlinear >10 <sup>9</sup> R	Negligible to 10 <sup>9</sup>	~ 2 x 10 <sup>-10</sup>	4.8 x 10 <sup>-8</sup>	5% over the total dose range to 10 <sup>9</sup> R	± 3% >1R ± 20% at 10mR	B	1	a=12mm b=1.4mm
TL-021-A	Lif	½ Mini	15mR-10 <sup>9</sup> R Nonlinear >10 <sup>9</sup> R	10mR-10 <sup>9</sup> R Nonlinear >10 <sup>9</sup> R	2R-10 <sup>9</sup> R Nonlinear >10 <sup>9</sup> R	Negligible to 10 <sup>9</sup>	~ 2 x 10 <sup>-10</sup>	4.8 x 10 <sup>-8</sup>	5% over the total dose range to 10 <sup>9</sup> R	± 3% >1R ± 20% at 10mR	B	2	a=6mm b=1.4mm
TL-028-A	Lif	Hot Press Chip	10mR-10 <sup>9</sup> R Nonlinear >10 <sup>9</sup> R	10mR-10 <sup>9</sup> R Nonlinear >10 <sup>9</sup> R	200mR-10 <sup>9</sup> R Nonlinear >10 <sup>9</sup> R	Negligible to 10 <sup>9</sup>	~ 2 x 10 <sup>-10</sup>	4.8 x 10 <sup>-8</sup>	5% over the total dose range to 10 <sup>9</sup> R	± 3% >1R ± 20% at 10mR	B	3	a=b=¼ inch c=0.030 inch
TL-029	Lif	Micro	250mR-10 <sup>9</sup> R Nonlinear >10 <sup>9</sup> R	1R-10 <sup>9</sup> R Nonlinear >10 <sup>9</sup> R	N/A	Negligible to 10 <sup>9</sup>	~ 2 x 10 <sup>-10</sup>	4.8 x 10 <sup>-8</sup>	7% over the total dose range to 10 <sup>9</sup> R	± 6% or 0.2R	B	5	a=6mm b=0.9mm
TL-203	Lif	Square Rod	5mR-10 <sup>9</sup> R Nonlinear >10 <sup>9</sup> R	10mR-10 <sup>9</sup> R Nonlinear >10 <sup>9</sup> R	200mR-10 <sup>9</sup> R Nonlinear >10 <sup>9</sup> R	<10%-10 <sup>11</sup>	~ 2 x 10 <sup>-10</sup>	4.8 x 10 <sup>-8</sup>	5% over the total dose range to 10 <sup>9</sup> R	± 3% >1R ± 20% at 5mR	B	7	a=b=1mm c=6.0mm
TL-022	*Lif	Mini	10mR-10 <sup>9</sup> R Nonlinear >10 <sup>9</sup> R	10mR-10 <sup>9</sup> R Nonlinear >10 <sup>9</sup> R	500mR-10 <sup>9</sup> R Nonlinear >10 <sup>9</sup> R	Negligible to 10 <sup>9</sup>	~ 2 x 10 <sup>-10</sup>	7.0 x 10 <sup>-8</sup>	5% over the total dose range to 10 <sup>9</sup> R	± 3% >1R ± 20% at 10mR	B	1	a=12mm b=1.4mm
TL-022A	*Lif	½ Mini	200mR-10 <sup>9</sup> R Nonlinear >10 <sup>9</sup> R	20mR-10 <sup>9</sup> R Nonlinear >10 <sup>9</sup> R	2R-10 <sup>9</sup> R Nonlinear >10 <sup>9</sup> R	Negligible to 10 <sup>9</sup>	~ 2 x 10 <sup>-10</sup>	7.0 x 10 <sup>-8</sup>	5% over the total dose range to 10 <sup>9</sup> R	± 3% >1R ± 20% at 10mR	B	2	a=6mm b=1.4mm
TL-026	*Lif	Micro	300mR-10 <sup>9</sup> R Nonlinear >10 <sup>9</sup> R	1R-10 <sup>9</sup> R Nonlinear >10 <sup>9</sup> R	N/A	Negligible to 10 <sup>9</sup>	~ 2 x 10 <sup>-10</sup>	7.0 x 10 <sup>-8</sup>	7% over the total dose range to 10 <sup>9</sup> R	± 6% >10R ± 20% at 500mR	B	5	a=6mm b=0.9mm
TL-028-B	*Lif	Hot Press Chip	10mR-10 <sup>9</sup> R Nonlinear >10 <sup>9</sup> R	10mR-10 <sup>9</sup> R Nonlinear >10 <sup>9</sup> R	N/A	<10%-10 <sup>11</sup>	~ 2 x 10 <sup>-10</sup>	7.0 x 10 <sup>-8</sup>	5% over the total dose range to 10 <sup>9</sup> R	± 3% >1R ± 20% at 10mR	B	3	a=b=¼ inch c=0.030 inch
TL-201	*Lif	Hot Press Chip	10mR-10 <sup>9</sup> R Nonlinear >10 <sup>9</sup> R	10mR-10 <sup>9</sup> R Nonlinear >10 <sup>9</sup> R	N/A	<10%-10 <sup>11</sup>	~ 2 x 10 <sup>-10</sup>	7.0 x 10 <sup>-8</sup>	5% over the total dose range to 10 <sup>9</sup> R	± 3% >1R ± 20% at 10mR	B	3	a=b=1mm c=6mm
TL-023	*Lif	Mini	5mR-10 <sup>9</sup> R Nonlinear >10 <sup>9</sup> R	10mR-10 <sup>9</sup> R Nonlinear >10 <sup>9</sup> R	200mR-10 <sup>9</sup> R Nonlinear >10 <sup>9</sup> R	Negligible to 10 <sup>9</sup>	~ 2 x 10 <sup>-10</sup>	9.0 x 10 <sup>-11</sup>	5% over the total dose range to 10 <sup>9</sup> R	± 3% >1R ± 20% at 10mR	B	1	a=12mm b=1.4mm
TL-023-A	*Lif	½ Mini	10mR-10 <sup>9</sup> R Nonlinear >10 <sup>9</sup> R	200mR-10 <sup>9</sup> R Nonlinear >10 <sup>9</sup> R	2R-10 <sup>9</sup> R Nonlinear >10 <sup>9</sup> R	Negligible to 10 <sup>9</sup>	~ 2 x 10 <sup>-10</sup>	9.0 x 10 <sup>-11</sup>	5% over the total dose range to 10 <sup>9</sup> R	± 3% >1R ± 20% at 10mR	B	2	a=6mm b=1.4mm
TL-027	*Lif	Micro	300mR-10 <sup>9</sup> R Nonlinear >10 <sup>9</sup> R	1R-10 <sup>9</sup> R Nonlinear >10 <sup>9</sup> R	N/A	Negligible to 10 <sup>9</sup>	~ 2 x 10 <sup>-10</sup>	9.0 x 10 <sup>-11</sup>	7% over the total dose range to 10 <sup>9</sup> R	± 5% >1R ± 20% at 10mR	B	5	a=6mm b=0.9mm
TL-028-C	*Lif	Hot Press Chip	10mR-10 <sup>9</sup> R Nonlinear >10 <sup>9</sup> R	10mR-10 <sup>9</sup> R Nonlinear >10 <sup>9</sup> R	N/A	<10%-10 <sup>11</sup>	~ 2 x 10 <sup>-10</sup>	9.0 x 10 <sup>-11</sup>	5% over the total dose range to 10 <sup>9</sup> R	± 3% >1R ± 20% at 10mR	B	3	a=b=¼ inch c=0.030 inch
TL-028-C-20	*Lif	Hot Press Chip	10mR-10 <sup>9</sup> R Nonlinear >10 <sup>9</sup> R	10mR-10 <sup>9</sup> R Nonlinear >10 <sup>9</sup> R	N/A	<10%-10 <sup>11</sup>	~ 2 x 10 <sup>-10</sup>	9.0 x 10 <sup>-11</sup>	5% over the total dose range to 10 <sup>9</sup> R	± 3% >1R ± 20% at 10mR	B	3	a=b=¼ inch c=0.020 inch
TL-202	*Lif	Square Rod	5mR-10 <sup>9</sup> R Nonlinear >10 <sup>9</sup> R	10mR-10 <sup>9</sup> R Nonlinear >10 <sup>9</sup> R	200mR-10 <sup>9</sup> R Nonlinear >10 <sup>9</sup> R	<10%-10 <sup>11</sup>	~ 2 x 10 <sup>-10</sup>	9.0 x 10 <sup>-11</sup>	5% over the total dose range to 10 <sup>9</sup> R	± 3% >1R ± 20% at 10mR	B	7	a=b=1mm c=6mm



### CaF<sub>2</sub>:Mn DOSIMETERS

Model	TL Phosphor	Config-uration	Dose Range			Dose Rate Dependence R/sec	Fast Neutron Response P-B. R/n/cm <sup>2</sup>	Thermal Neutron Response R/n/cm <sup>2</sup> Co <sup>60</sup> Rads	Linearity	Repro-ducibility for Dosimeter	Re-sponse Curve	Figure Refer-ence	Size
			Model 2005A	Model TL-3B	Model 8506B								
TL-031	CaF <sub>2</sub> :Mn	Mini	3mR-10 <sup>4</sup> R	10mR-10 <sup>4</sup> R	200mR-10 <sup>4</sup> R	Negligible to 10 <sup>4</sup>	2.3 x 10 <sup>-10</sup>	1.2 x 10 <sup>-10</sup>	5% over dose range	±3% >1R ±20% at 10mR	A	1	a = 12mm b = 1.4mm
TL-031-A	CaF <sub>2</sub> :Mn	½ Mini	5mR-10 <sup>4</sup> R	20mR-10 <sup>4</sup> R	1R-10 <sup>4</sup> R	Negligible to 10 <sup>4</sup>	2.3 x 10 <sup>-10</sup>	1.2 x 10 <sup>-10</sup>	5% over dose range	±3% >1R ±20% at 10mR	A	2	a = 6mm b = 1.4mm
TL-033	CaF <sub>2</sub> :Mn	Hot Press Chip	2mR-3 x 10 <sup>4</sup> R	5mR-3 x 10 <sup>4</sup> R	N/A	<10%-10 <sup>11</sup>	2.3 x 10 <sup>-10</sup>	1.2 x 10 <sup>-10</sup>	3%-10 <sup>4</sup> R 10%-3 x 10 <sup>4</sup> R	±2% >1R ±20% at 20mR	A	3	a = b = ¼ inch c = 0.035 inch
TL-035	CaF <sub>2</sub> :Mn	Tube	0.5mR-10 <sup>4</sup> R	0.5mR-10 <sup>4</sup> R	N/A	<10%-10 <sup>11</sup>	2.3 x 10 <sup>-10</sup>	1.2 x 10 <sup>-10</sup>	2% over dose range	±2% or 0.4mR	A	4	—
TL-039	CaF <sub>2</sub> :Mn	Micro	100mR-10 <sup>4</sup> R	1R-10 <sup>4</sup> R	N/A	Negligible to 10 <sup>4</sup>	2.3 x 10 <sup>-10</sup>	1.2 x 10 <sup>-10</sup>	7% over dose range	±3% or 0.1R	A	5	a = 6mm b = 0.9mm
TL-072	CaF <sub>2</sub> :Mn	Powder	0.2mR-3 x 10 <sup>4</sup> R	N/A	N/A	<10%-10 <sup>11</sup>	2.3 x 10 <sup>-10</sup>	1.2 x 10 <sup>-10</sup>	7%-10 <sup>4</sup> R	±15% >100 mR ±40% at 10mR	A	6	200-400 mesh
TL-301	CaF <sub>2</sub> :Mn	Square Rod	1mR-3 x 10 <sup>4</sup> R	5mR-3 x 10 <sup>4</sup> R	200mR-10 <sup>4</sup> R	<10%-10 <sup>11</sup>	2.3 x 10 <sup>-10</sup>	1.2 x 10 <sup>-10</sup>	3%-10 <sup>4</sup> R 10%-3 x 10 <sup>4</sup> R	±2% >1R ±20% at 20mR	A	7	a = b = 1mm c = 6mm

### CaF<sub>2</sub> DOSIMETER TYPES

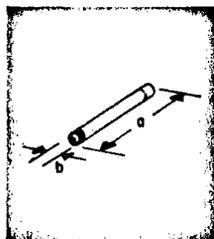


FIG. 1 MINI

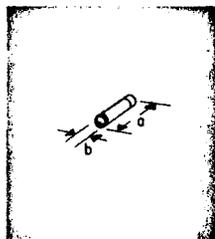


FIG. 2 1/2 MINI

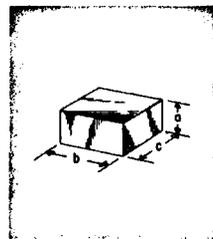


FIG. 3 HOT PRESS CHIP

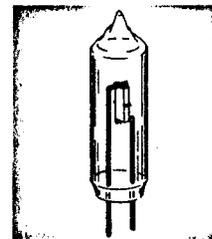


FIG. 4 TUBE

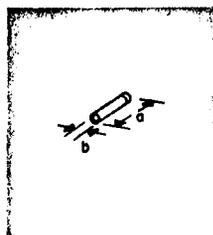


FIG. 5 MICRO



FIG. 6 POWDER

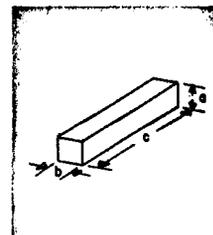
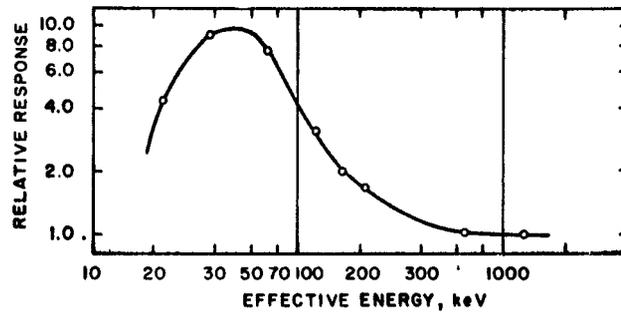


FIG. 7 SQUARE ROD

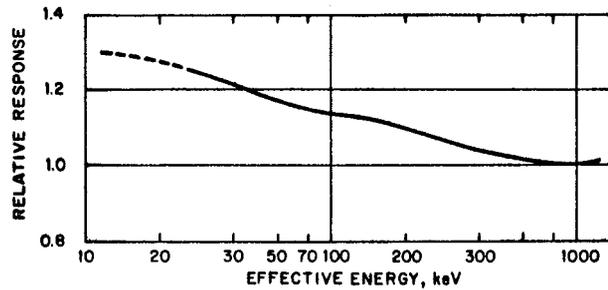
CaF<sub>2</sub>: Mn - GOLD



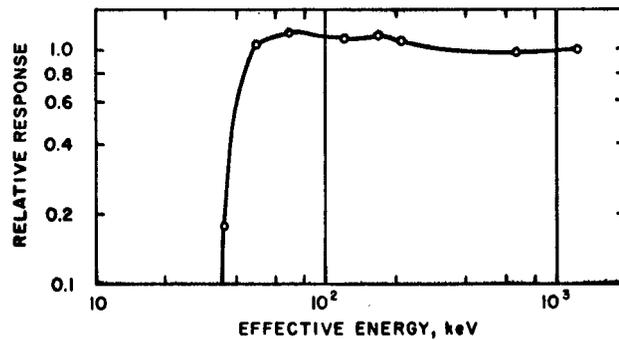
### EFFECTIVE ENERGY RESPONSE CURVES



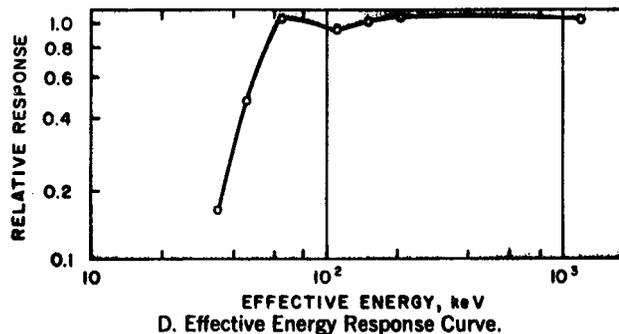
A. Effective Energy Response Curve.



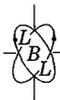
B. Effective Energy Response Curve.



C. Effective Energy Response Curve.



D. Effective Energy Response Curve.



### LiF DOSIMETER TYPES



FIG. 1 MINI



FIG. 2 1/2 MINI

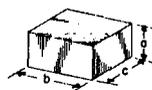


FIG. 3 HOT PRESS CHIP

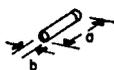


FIG. 5 MICRO

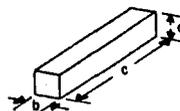


FIG. 7 SQUARE ROD

LiF-GREEN  
LiF-BLUE  
LiF-ROSE

### SHIELDS



FIG. 8 MINI



FIG. 9 TUBE



FIG. 10 HOT PRESS CHIP



FIG. 11 MINI  
(HOLDS 3 DOSIMETERS)



FIG. 12 MICRO  
AND 1/2 MINI

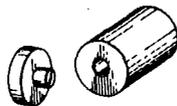


FIG. 13 EQUILIBRIUM  
SHIELD FOR MINIS

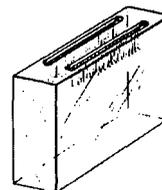


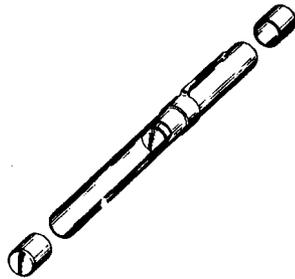
FIG. 14 ELECTRON  
EQUILIBRIUM SHIELD  
(HOLDS ALL MINIS AND MICROS)



### DOSIMETER/SHIELD COMBINATIONS

Model	TL Phosphor	Config-uration	Dose Range			Dose Rate Dependence R/sec	Fast Neutron Response P <sub>n</sub> , R/n/cm <sup>2</sup>	Thermal Neutron Response R/n/cm <sup>2</sup> Co <sup>60</sup> Rads	Linearity	Repro-ducibility for Dosimeter	Re-sponse Curve	Figure Refer-ence	Shield/ Dosimeter
			Model 2005A	Model TL-3B	Model 8506B								
TL-011	Caf <sub>2</sub> :Mn, Shield	Mini	3mR-10 <sup>4</sup> R	10mR-10 <sup>4</sup> R	200mR-10 <sup>4</sup> R	Negligible to 10 <sup>4</sup>	—	—	5% over dose range	±3% >1R ±20% at 10mR	C	8	TL-051, TL-031
TL-011-A	Caf <sub>2</sub> :Mn, Shield	½ Mini	5mR-10 <sup>4</sup> R	20mR-10 <sup>4</sup> R	1R-10 <sup>4</sup> R	Negligible to 10 <sup>4</sup>	—	—	5% over dose range	±3% >1R ±20% at 10mR	C	12	TL-059, TL-031-A
TL-013	Caf <sub>2</sub> :Mn, Shield	Hot Press Chip	2mR-3 x 10 <sup>4</sup> R	5mR-3 x 10 <sup>4</sup> R	N/A	<10%-10 <sup>11</sup>	—	—	3%-10 <sup>4</sup> R 10%-3 x 10 <sup>4</sup> R	±2% >1R ±20% at 20mR	C	10	TL-053, TL-033
TL-014	Caf <sub>2</sub> :Mn, Shield	Square Rod	1mR-3 x 10 <sup>4</sup> R	5mR-3 x 10 <sup>4</sup> R	200mR-10 <sup>4</sup> R	<10%-10 <sup>11</sup>	—	—	3%-10 <sup>4</sup> R 10%-3 x 10 <sup>4</sup> R	±2% >1R ±20% at 20mR	C	12	TL-059, TL-031
TL-015	Caf <sub>2</sub> :Mn, Shield	Tube	0.5mR-10 <sup>4</sup> R	0.5mR-10 <sup>4</sup> R	N/A	Negligible to 10 <sup>11</sup> R/sec	—	—	2% over dose range	±2% or 0.4mR	D	9	TL-052, TL-035
TL-019	Caf <sub>2</sub> :Mn, Shield	Micro	100mR-10 <sup>4</sup> R	1R-10 <sup>4</sup> R	N/A	Negligible to 10 <sup>4</sup>	—	—	7% over the total dose range	±3% or 0.1R	C	12	TL-059, TL-039 (3 each)
TL-020	Caf <sub>2</sub> :Mn, Shield	Mini	3mR-10 <sup>4</sup> R	10mR-10 <sup>4</sup> R	200mR-10 <sup>4</sup> R	Negligible to 10 <sup>4</sup>	—	—	5% over the total dose range	±3% >1R ±20% at 10mR	C	11	TL-055, TL-031 (3 each)

### ACCESSORIES



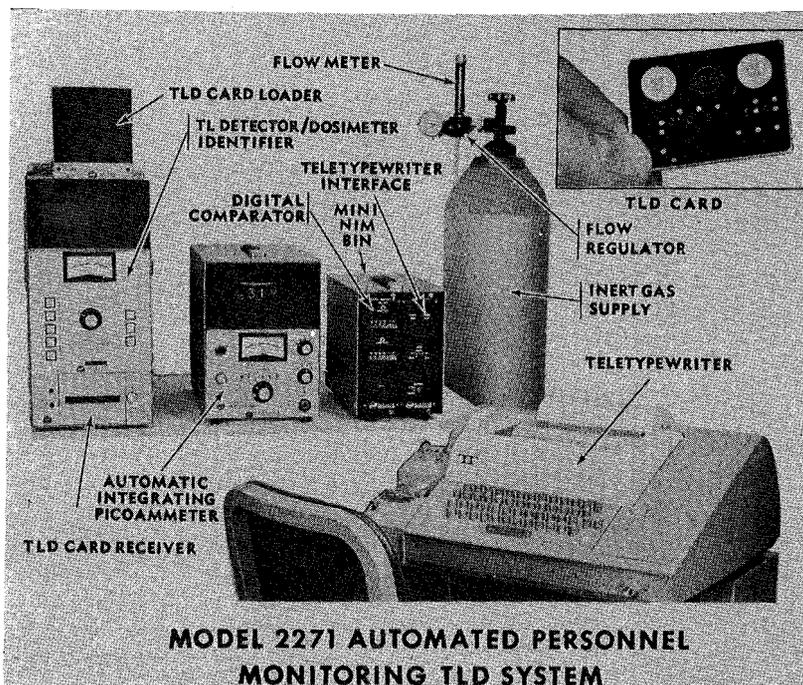
MODEL TL-001  
TWO-COMPARTMENT DOSIMETER  
HOLDER (HOLDS ALL DOSIMETER  
TYPES, EXCEPT TL-035)



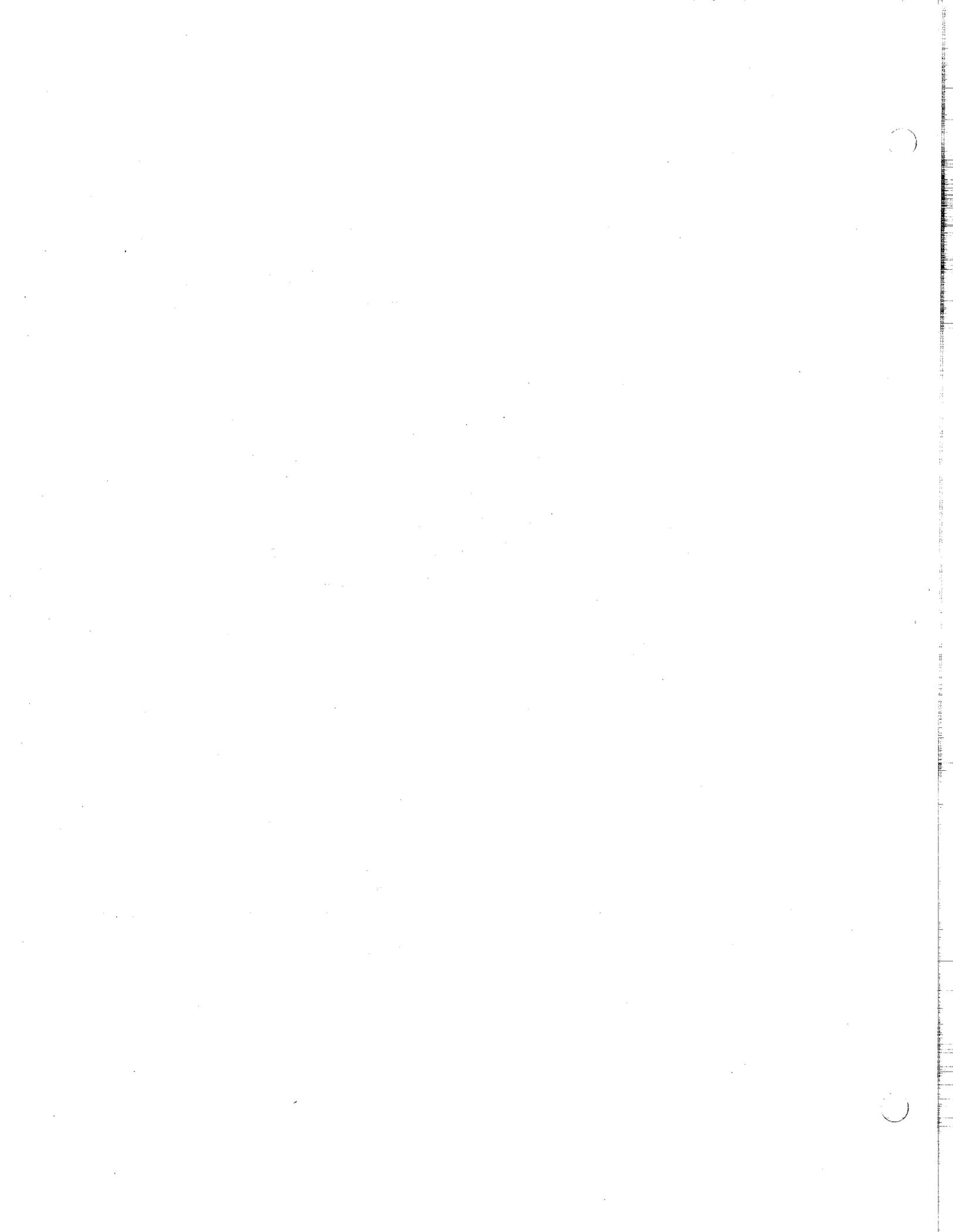
MODEL TL-82-B  
REFERENCE LIGHT



TLD Dosimetry System  
Harshaw Model 2271



Class	Laboratory
Principle of Operation	Thermoluminescent Dosimetry (TLD) system. TLD chips are read out by heating which generates luminescence, detected by a phototube. System is automated. TLD chips are lithium fluoride (LiF).
Sensitivity	TLD dose from 10 mR to 10,000 R
Sampling	TLD card containing 2 chips can be read out in 40 seconds
Requirements	Power: Size: Weight:
Features	(1) System can accommodate 250 TLD cards (2 chips each) automatically. (2) BCD identification of each card also automated. (3) Manual mode also available. (4) TLD reader, integrating picoammeter, flow regulator, flow meter, teletypewriter all included.
Reference	(1) Manufacturer's specifications
Cost	\$
Address	Harshaw Chemical Company 6801 Cochran Road Solon, OH 44139 (216) 248-7400





INSTRUMENTATION  
FOR ENVIRONMENTAL  
MONITORING

RAD-DOS  
Dosimetry Service  
ICN  
Aug. 1972

Film TLD Dosimetry Service

ICN Inc.

Class Film, TLD Dosimetry Service

Principle of Operation Film, TLD, or both distributed to user, developed after exposure, results of analysis reported to user.

Sensitivity and Range Film Sensitivity: 5 mrem (soft X-rays), 10-20 mrem (gamma), 30 mrem (beta)  
TLD Sensitivity: 10 mrad to 100,000 rad  
Film Energy Range: 20 keV to several MeV for gammas  
TLD Energy Range: ±15%, 20 keV to 1.3 MeV for gammas

Performance Film Accuracy: 15-25%  
TLD Accuracy: ±20% at 10 mrad, ±5% above 1 rad

Features

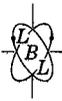
1. Dosimeters cycle weekly, biweekly, or monthly
2. Reports mailed to user within 48 hours of receipt of dosimeter
3. Film dosimeter has a backup film for use in case of accidental damage
4. Finger-ring and wrist badge dosimeters also available

References Manufacturer's specifications

Cost	Units	β, γ, X Film	Ring Badge	TLD	
				TLD	and Film
	1	\$1.75	\$1.75	\$2.50	\$3.75
	2	1.10	1.50	2.00	2.75
	3	.85	1.20	1.60	2.00
	4-5	.75	1.10	1.40	1.80

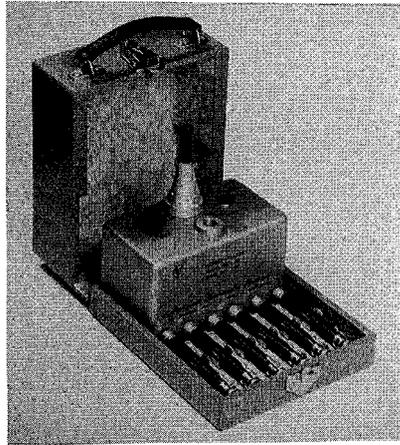
Address ICN Tracerlab  
26201 Miles Road  
Cleveland, OH 44128  
(216) 662-0218





Dosimeter Charger-Reader

Johnson Model L-60



Class	Portable															
Principle of Operation	Ion chamber charger-reader															
Sensitivity and Range	Pocket Chambers															
	<table><thead><tr><th><u>Model</u></th><th><u>Range</u></th><th><u>Type of Radiation</u></th></tr></thead><tbody><tr><td>L-65</td><td>200 mr</td><td>X, gamma</td></tr><tr><td>L-69</td><td>200 mrem</td><td>X, gamma, thermal neutrons</td></tr><tr><td>L-70</td><td>200 mrem</td><td>thermal neutrons only</td></tr><tr><td>L-81</td><td>2 r</td><td>X, gamma</td></tr></tbody></table>	<u>Model</u>	<u>Range</u>	<u>Type of Radiation</u>	L-65	200 mr	X, gamma	L-69	200 mrem	X, gamma, thermal neutrons	L-70	200 mrem	thermal neutrons only	L-81	2 r	X, gamma
<u>Model</u>	<u>Range</u>	<u>Type of Radiation</u>														
L-65	200 mr	X, gamma														
L-69	200 mrem	X, gamma, thermal neutrons														
L-70	200 mrem	thermal neutrons only														
L-81	2 r	X, gamma														
Sampling	Continuous															
Performance	Stability															
Requirements	Power: One "D" cell and six 22.5 V No. 412 Minimax batteries Dosimeter size: 1.3 cm D, 4 cm or 9 cm L (0.5" D, 1.5" or 3.5" L) Weight:															
Features	Chambers are hermetically sealed. Sleeves are included to provide electron equilibrium at higher energies.															
Cost	\$150															
Address	William B. Johnson and Associates Research Park Boonton Avenue Montville, New Jersey 07045 (201) 334-9221															



0 0 0 0 3 6 0 2 3 9 7

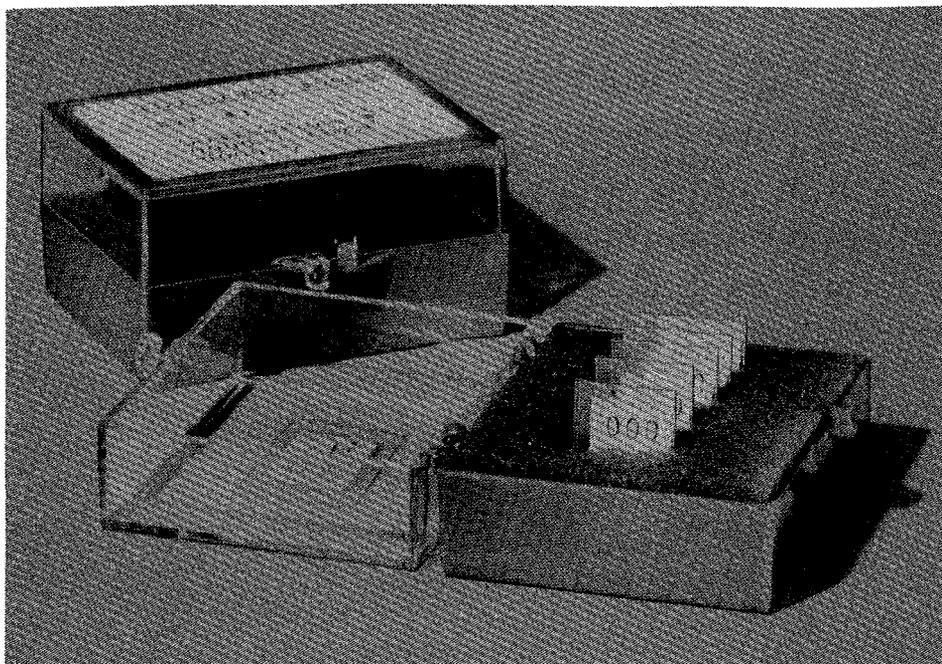


INSTRUMENTATION  
FOR ENVIRONMENTAL  
MONITORING

RAD-DOS  
X, GAM  
Landauer  
Oct. 1972

Personnel Dosimetry Service

Landauer





Class Personnel Dosimetry Service

Principle of Operation LiF (TLD) dosimeters, readout by Landauer after mail-exchange

Sensitivity and Range Sensitivity: Exposure range 10 mR to 100,000 R  
Energy Range: Energy independent above 100 keV, 40% duration down to 18 keV

Performance Accuracy: Standard error <25% (10 mR), <10% (>100 mR)

Features

1. Weekly, biweekly, monthly schedules.
2. Dosimeters can be furnished sealed in sterile tubing.
3. TLD ring dosimeter service also available.

References Manufacturer's specifications

Cost	<u>Number of Five-Dosimeter Kits</u>	<u>Price Per Kit</u>
	1-5	\$25.00
	6-10	21.50
	11-25	17.50
	26-50	15.00
	51-100	12.50
	over 100	on quote

Address R.S. Landauer Jr. & Co.  
Glenwood Science Park  
Glenwood, IL 60425  
(312) 755-7000



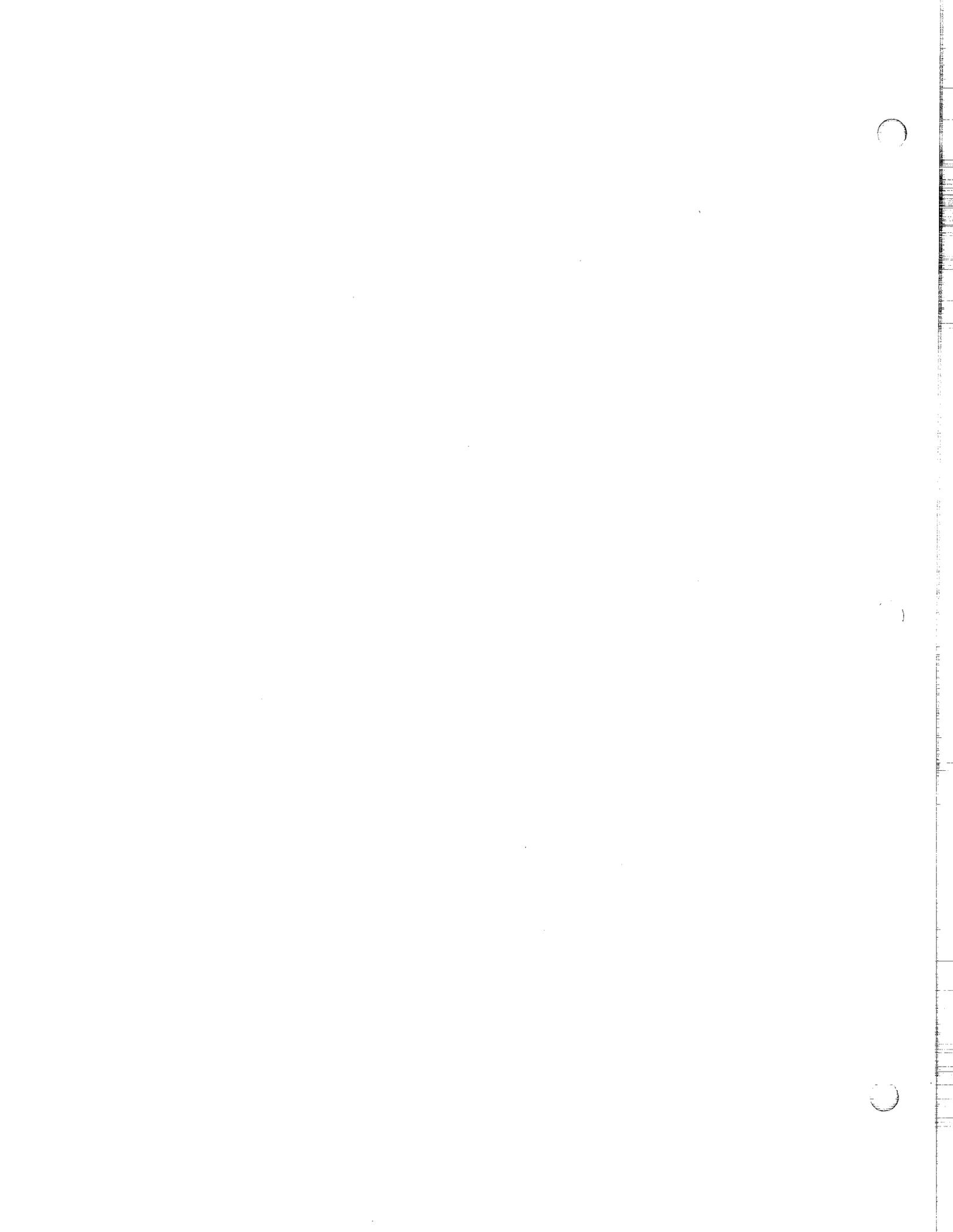
INSTRUMENTATION  
FOR ENVIRONMENTAL  
MONITORING

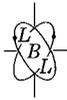
RAD-DOS  
Neutrons  
Nuclear Associates Inc.  
Oct. 1972

Thermal Neutron Dosimeter  
Nuclear Associates Model 06-609



Class	Portable, Fountain Pen Size
Principle of Operation	Boron-coated ion chamber measured first-collision thermal neutron dose in a gamma field, reads in mrem.
Sensitivity and Range	Energy Sensitivity: Thermal Calibration: 670 thermal neutrons $\text{cm}^{-2}\text{sec}^{-1}$ = 120 mrem for 40 hr Exposure Range: 0-120 mrem
Sampling	Continuous
Performance	Accuracy: $\pm 10\%$ of true dosage
Requirements	Power: Use charger Size: Four inches x 9/16 in. diameter Weight: Two ounces
References	Manufacturer's specifications
Cost	\$130.
Address	Nuclear Associates 35 Urban Avenue Westbury, NY 11590 (516) 333-9344

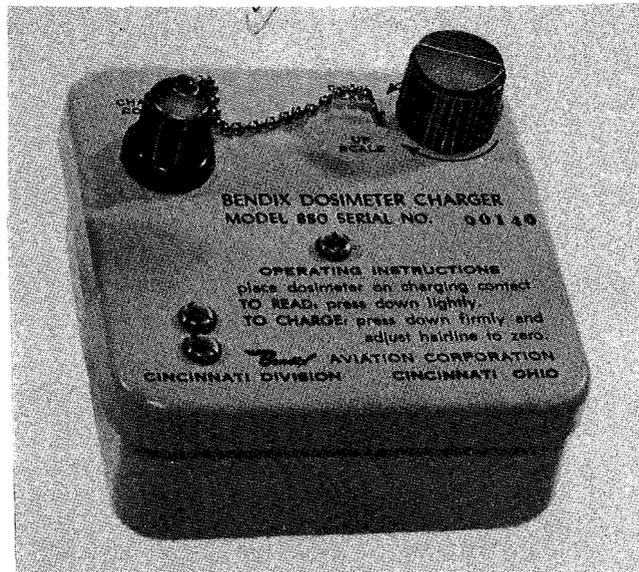
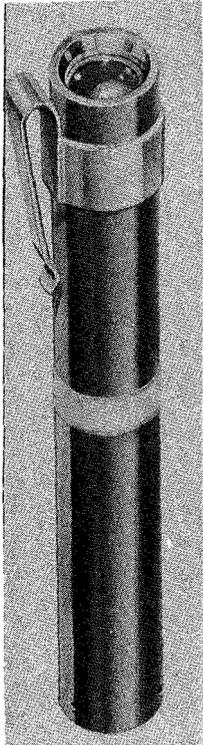




INSTRUMENTATION  
FOR ENVIRONMENTAL  
MONITORING

RAD-DOS  
Gamma, X-Ray  
Nuclear Associates Inc.  
Oct. 1972

Direct Reading Dosimeters  
Nuclear Associates Inc.



Class Portable, Pocket, Fountain Pen Size

Principle of Operation A quartz fiber electrometer, compound microscope calibrated readout.



Sensitivity  
and Range

Energy Sensitivity:

Model	Range	Cost 1-9 ea.
06-862	200 mr	\$45.00
883	500 mr	55.00
866	1 r	60.00
608	10 r	67.50
619	100 r	75.00
638	200 r	75.00
686	600 r	75.00

Sampling

Continuous

Performance

Accuracy:  $\pm 10\%$  of dose  
Leakage: 2% per 24 hours

Requirements

Power: Use charger  
Size: 4" L x 9/16" D  
Weight: About 2 ounces

Features

Optional Charger, Model 06-906, \$55.00  
Optional Calibrator, Model 06-200, \$37.50

References

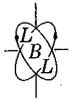
Manufacturer's specifications

Cost

See chart above

Address

Nuclear Associates Inc.  
35 Urban Avenue  
Westbury, NY 11590  
(516) 333-9344



INSTRUMENTATION  
FOR ENVIRONMENTAL  
MONITORING

RAD-DOS  
X-Ray  
Nuclear Associates Inc.  
Oct. 1972

X-Ray Dosimeter

Nuclear Associates Model 06-002, 050, 099

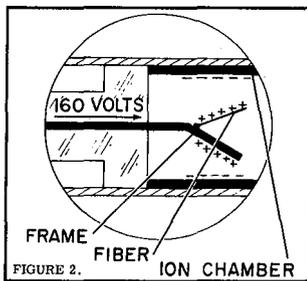
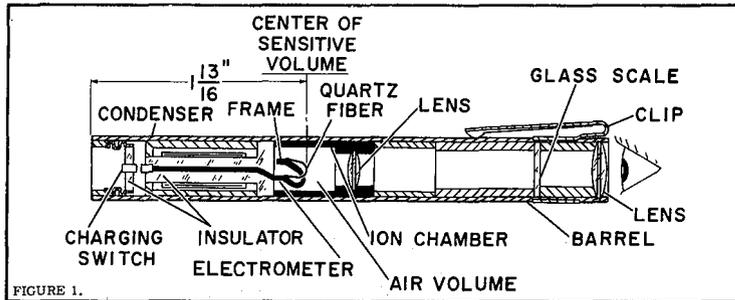


FIGURE 2.

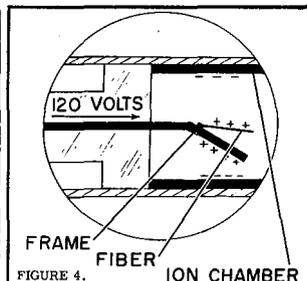


FIGURE 4.

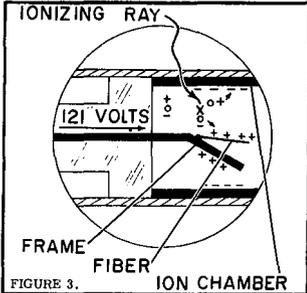


FIGURE 3.

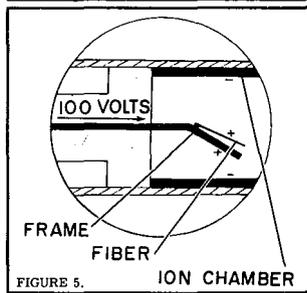
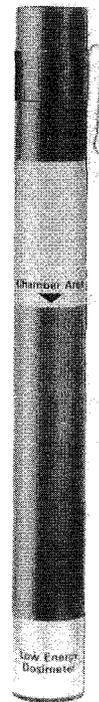
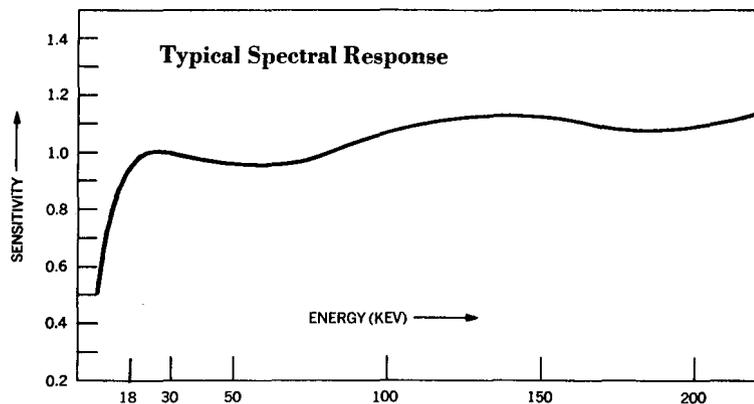


FIGURE 5.

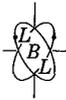


Low Energy  
Direct Reading  
Dosimeter





Class	Portable, Pocket								
Principle of Operation	Plastic walls allow low energy X-rays to penetrate the ion chamber.								
Sensitivity and Range	<table><thead><tr><th><u>Model</u></th><th><u>Range</u></th></tr></thead><tbody><tr><td>06-002</td><td>0-200 mR</td></tr><tr><td>050</td><td>0-5 R</td></tr><tr><td>099</td><td>0-100 R</td></tr></tbody></table>	<u>Model</u>	<u>Range</u>	06-002	0-200 mR	050	0-5 R	099	0-100 R
<u>Model</u>	<u>Range</u>								
06-002	0-200 mR								
050	0-5 R								
099	0-100 R								
Sampling	Continuous								
Performance	Accuracy: $\pm 15\%$ of true dose for 30 keV X-rays $\pm 10\%$ of 30 keV reading for 15 to 100 keV $+20\%/-10\%$ of 30 keV reading for 30 to 225 keV Leakage: 2% of full scale/24 hr								
Requirements	Power: Use charger Size: Four in. length x 9/16 in. diameter Weight: About 2 oz								
References	Manufacturer's specifications								
Cost	\$75.								
Address	Nuclear Associates Inc. 35 Urban Avenue Westbury, NY 11590 (516) 333-9344								

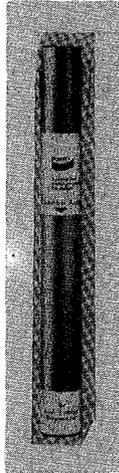


INSTRUMENTATION  
FOR ENVIRONMENTAL  
MONITORING

RAD-DOS  
X-ray  
Nucleus, Inc.  
Mar. 1972

X-Ray Dosimeter

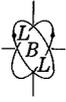
Nucleus Model 06-002, 050, 099  
(Bendix)



06-050  
Low-Energy Dosimeter

Class	Portable, pocket								
Principle of Operation	Plastic walls allows low energy X-rays to penetrate the ion chamber								
Sensitivity and Range	<table border="1"> <thead> <tr> <th>Model</th> <th>Range</th> </tr> </thead> <tbody> <tr> <td>06-002</td> <td>0-200 mR</td> </tr> <tr> <td>050</td> <td>0-5 R</td> </tr> <tr> <td>099</td> <td>0-100 R</td> </tr> </tbody> </table>	Model	Range	06-002	0-200 mR	050	0-5 R	099	0-100 R
Model	Range								
06-002	0-200 mR								
050	0-5 R								
099	0-100 R								
Sampling	Continuous								
Performance	Accuracy: $\pm 15\%$ of true dose for 30 keV x-rays $\pm 10\%$ of 30 keV reading for 15 to 100 keV $+20\%$ $-10\%$ of 30 keV reading for 30 to 225 keV Leakage 2% of full scale/24 hours								
Requirements	Power: use charger Size: 4" in length Weight: about 2 ounces								
Cost	\$75								
Address	Nucleus, Inc. P.O. Box R Oak Ridge, Tenn. 37830 (615) 483-0008								





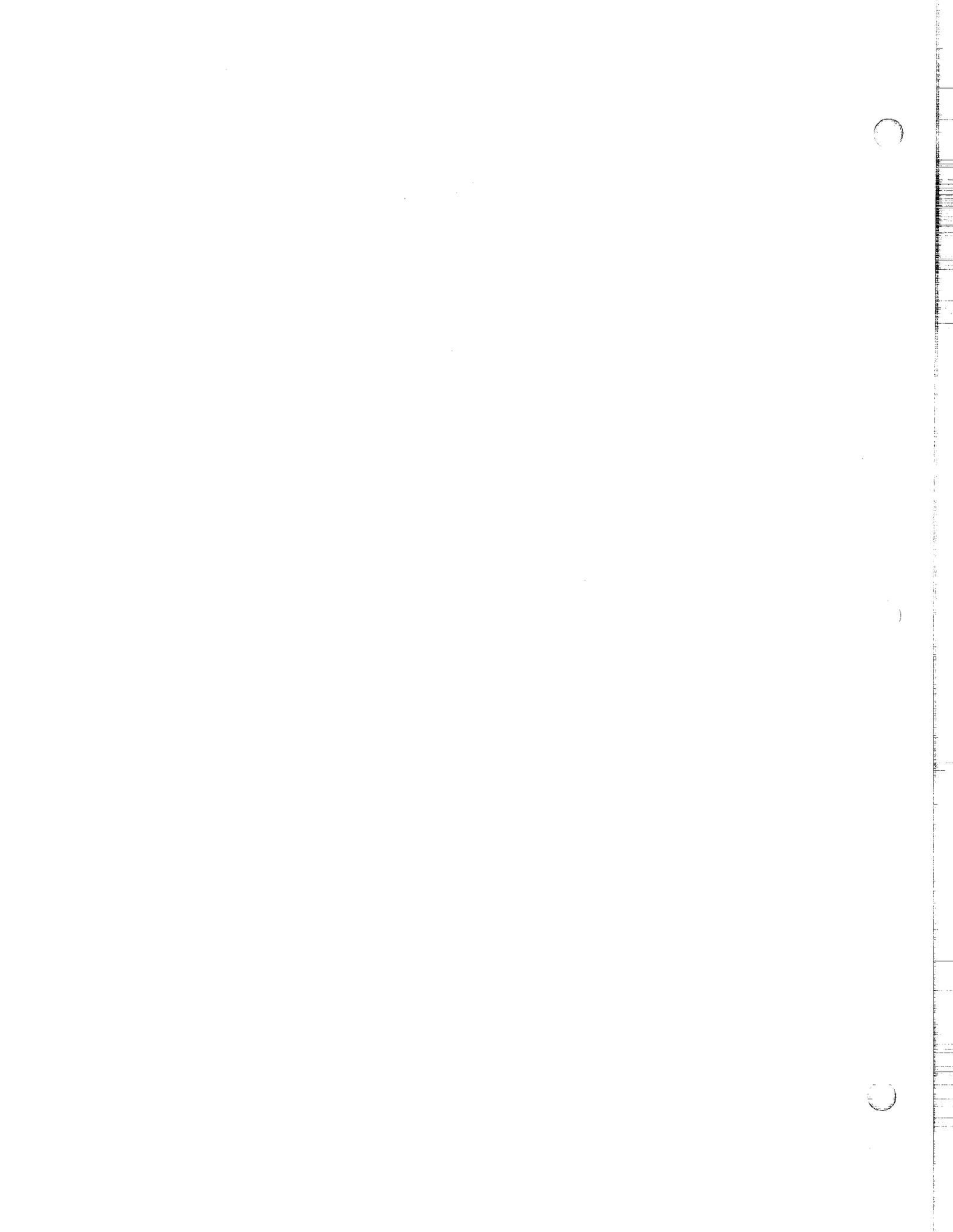
INSTRUMENTATION  
FOR ENVIRONMENTAL  
MONITORING

RAD-DOS  
Neutrons  
Nucleus Inc.  
Mar. 1972

Thermal Neutron Dosimeter  
Nucleus Model 06-609



Class	Portable, fountain pen size
Principle of Operation	Boron coated ion chamber measured first-collision thermal neutron dose in a gamma field, reads in rem.
Sensitivity and Range	Energy Sensitivity: thermal Calibration: 670 thermal neutrons/cm <sup>2</sup> /sec=120 mrem, for 40 hours Exposure range: 0-120 mrem
Sampling	Continuous
Performance	Accuracy: Unavailable
Requirements	Power: Use charger Size: 4 inches Weight: 2 ounces
Features	
Cost	\$130
Address	Nucleus Inc. P.O. Box R Oak Ridge, Tenn. 37830 (615) 483-0008

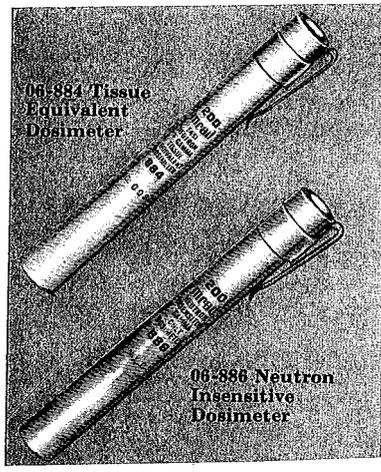




INSTRUMENTATION  
FOR ENVIRONMENTAL  
MONITORING

RAD-DOS  
Gamma, Neutron  
Nucleus, Inc.  
Mar. 1972

Direct Reading Dosimeter  
Nucleus Model 06-884, 886



Class	Portable, pocket
Principle of Operation	Tissue equivalent dosimeter, measures both fast neutron and gamma dose in rads, based on collisions between fast neutrons and atoms of soft tissue with the empirical formula $C_5H_4O_{18}N$ .
Sensitivity and Range	Neutron energies: 20 keV to 15 MeV Exposure: 20 mrad to 600 rads Gamma energies: 80 keV to 3 MeV
Sampling	Continuous
Performance	Accuracy: 20% of actual dose for neutrons 10% of actual dose for Cobalt-60 and Cesium-137 gammas 15% of actual dose for gammas from 80 keV to 3 MeV Leakage: Unavailable
Requirements	Power: Use charger Size: 4 inches, length Weight: 2 ounces
Features	To measure the first-collision dose of fast neutrons, subtract the neutron insensitive dosimeter reading (model 06-886, \$115) from the tissue equivalent dosimeter reading
Cost	\$130 each
Address	Nucleus, Inc. P.O. Box R Oak Ridge, Tenn. 37830 (615) 483-0008



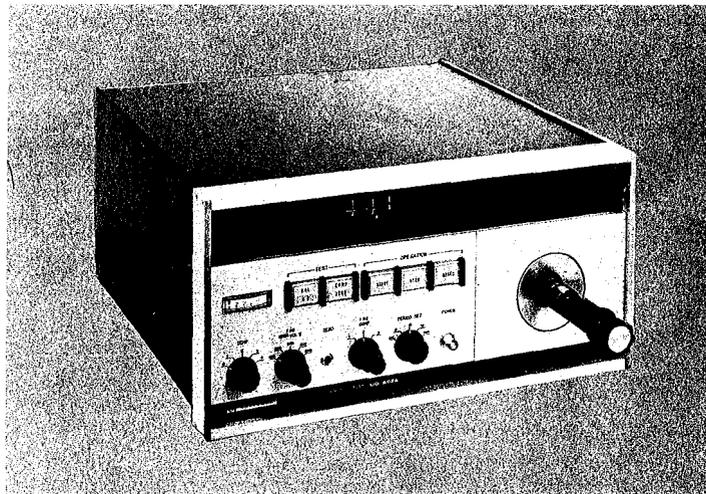


INSTRUMENTATION  
FOR ENVIRONMENTAL  
MONITORING

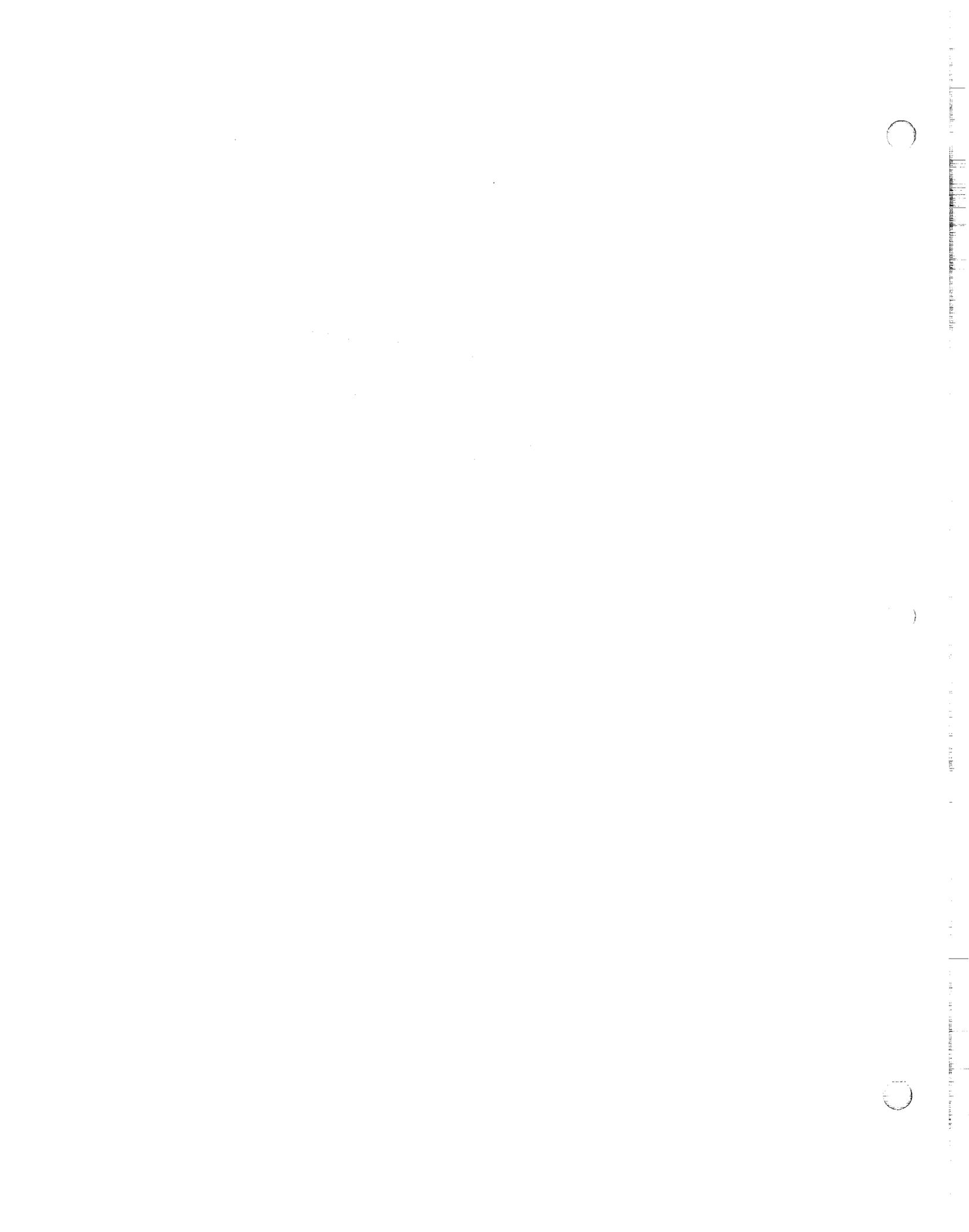
RAD-DOS  
X,GAM  
Panasonic  
Aug. 1972

TLD Personnel Dosimetry Readout System

Panasonic Model UD-502A

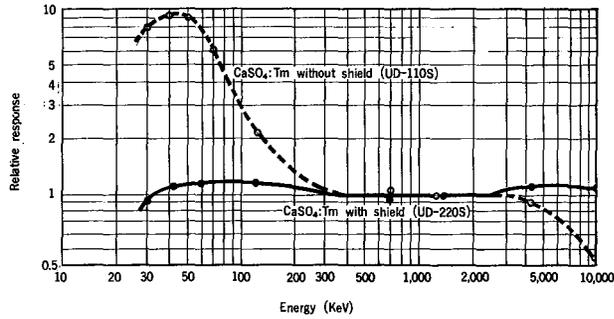


Class	Laboratory TLD Dosimeter Readout System
Principle of Operation	Precise heating by convection: hot flowing air passes over TLD dosimeter. TLD light output detected by photomultiplier tube, displayed on digital readout of <u>area</u> under TLD glow curve.
Performance	Accuracy: 0.2% of full scale Readout Time: 6, 10, or 12 seconds Display: Four digit nixie-tube readout, automatic ranging (0.001 mR to 199.9 R, 4 ranges) Calibration: Radioisotope light source
Requirements	Power: AC 100/115 V, 200 W Size: 43 x 43 x 23 cm (16.9" x 16.9" x 8.9") Weight: 18 kg (40 lb)
Features	BCD output connection for auxiliary printer; adaptable to many different TLD geometrical shapes
References	Manufacturer's specifications
Cost	
Address	Matsushita Electric Corporation of America Pan Am Building 200 Park Avenue New York, NY 10017

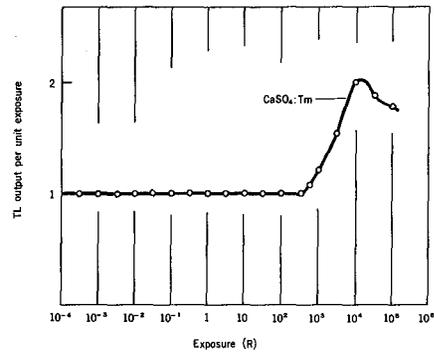




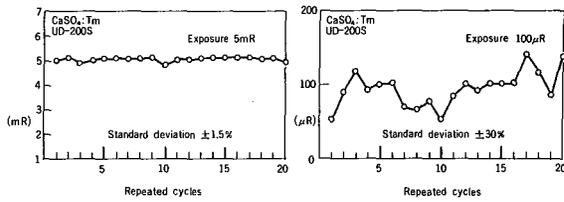
Personnel Dosimeters  
Panasonic TLD Dosimeters



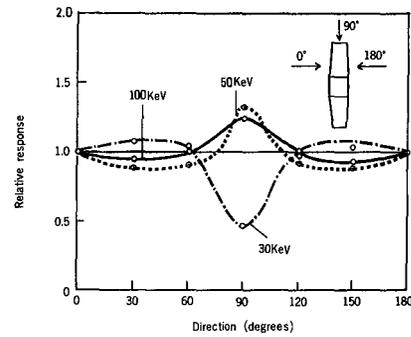
ENERGY RESPONSE OF  $\text{CaSO}_4:\text{Tm}$  DOSIMETERS ; (UD-110S, UD-200S)



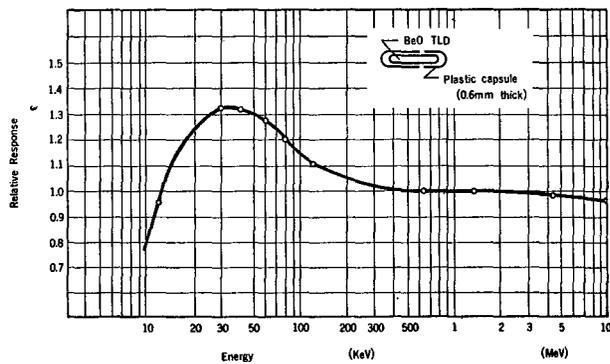
LINEARITY OF  $\text{CaSO}_4:\text{Tm}$



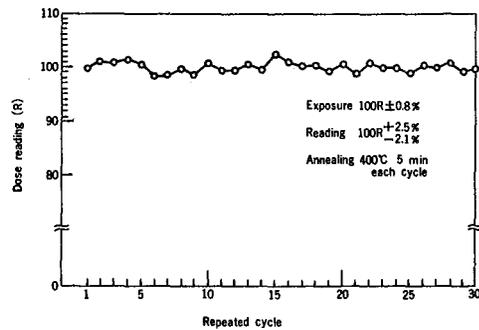
REPEATED APPLICATION OF  $\text{CaSO}_4:\text{Tm}$  DOSIMETER (UD-200S)



DIRECTIONAL RESPONSE OF  $\text{CaSO}_4:\text{Tm}$  DOSIMETER (UD-200S)



ENERGY RESPONSE OF BeO (UD-170A, UD-170B)



REPRODUCIBILITY OF BeO TLD

———— CLINICAL DOSE ————



Class Personnel Dosimeters

Principle of Operation Thermoluminescent dosimeters (TLD) of  $\text{CaSO}_4:\text{Tm}$  and  $\text{BeO}$

Sensitivity and Range

Model Number	Phosphor	Dimensions		Range	Repro-ducibility	Energy with Shield	Fading in 6 Months
		Diam	Length				
UD-200S	$\text{CaSO}_4:\text{Tm}$	11 mm	60 mm	0.1 mR to 20 R	0.1 mR $\pm 30\%$ 5 mR $\pm 1\%$	$\pm 40\%$ ( $>30$ keV)	8%
UD-110S	$\text{CaSO}_4:\text{Tm}$	2	12	0.1 mR to 20 R	0.1 mR $\pm 30\%$ 5 mR $\pm 1\%$	$\pm 20\%$ ( $>200$ keV)	8%
UD-100M	$\text{CaSO}_4:\text{Tm}$	10	0.06	10 mR to 2000 R	10 mR $\pm 30\%$ 100 mR $\pm 2\%$	$\pm 20\%$ ( $>200$ keV)	8%
UD-170A	BeO	2.2	12	2 mR to 200 R	2 mR $\pm 30\%$ 100 mR $\pm 1\%$	$\pm 25\%$ ( $>15$ keV)	6%
UD-170B	BeO	1.2	8	10 mR to 1000 R	10 mR $\pm 30\%$ 1 R $\pm 1\%$	$\pm 25\%$ ( $>15$ keV)	6%

Features

References Manufacturer's specifications

Cost

Readout Instruments

Model UD-505A \$4800.

Dosimeter

Model UD-200S	21.70
UD-110S	6.64
UD-100M-8	1.55
UD-170A	6.64
UD-170B	6.64
UD-610F	2.58
UD-611F	4.25
UD-612	3.26

Dosimeter Annealing Furnace

Model 602 822.

[All prices F.O.B. Westwood, NJ]

Address

Teledyne Isotopes  
50 Van Buren Avenue  
Westwood, NJ 07675  
(201) 664-7070 Telex 134-474

Matsushita Electric Corporation of America  
Pan Am Building  
200 Park Avenue  
New York, NY 10017

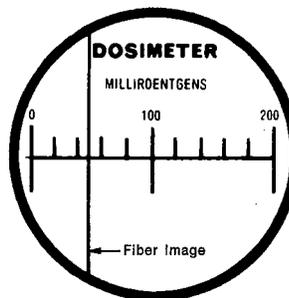
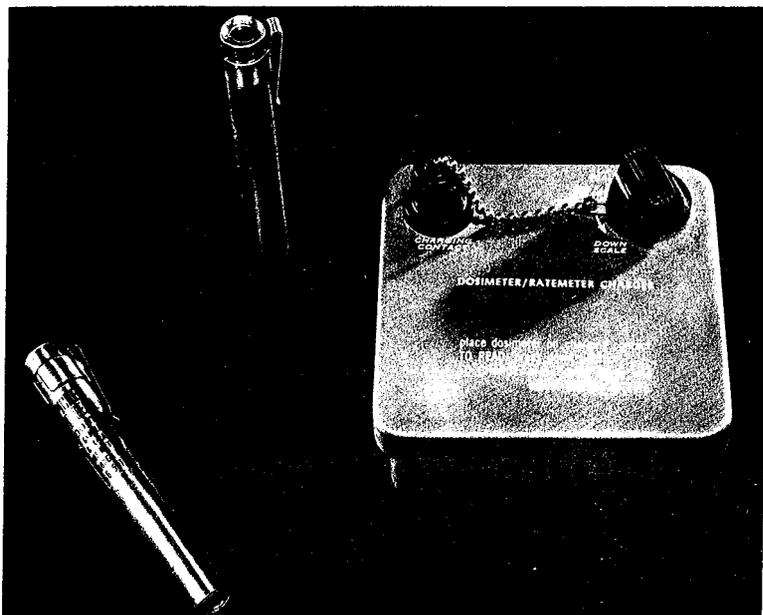


INSTRUMENTATION  
FOR ENVIRONMENTAL  
MONITORING

RAD-DOS  
Gamma, Neutron, X-ray  
Picker Nuclear  
Oct. 1972

Pocket Dosimeter & Charger

Picker 655-010, 019



Typical appearance of graduated scale in 655-010 Dosimeter (enlarged).

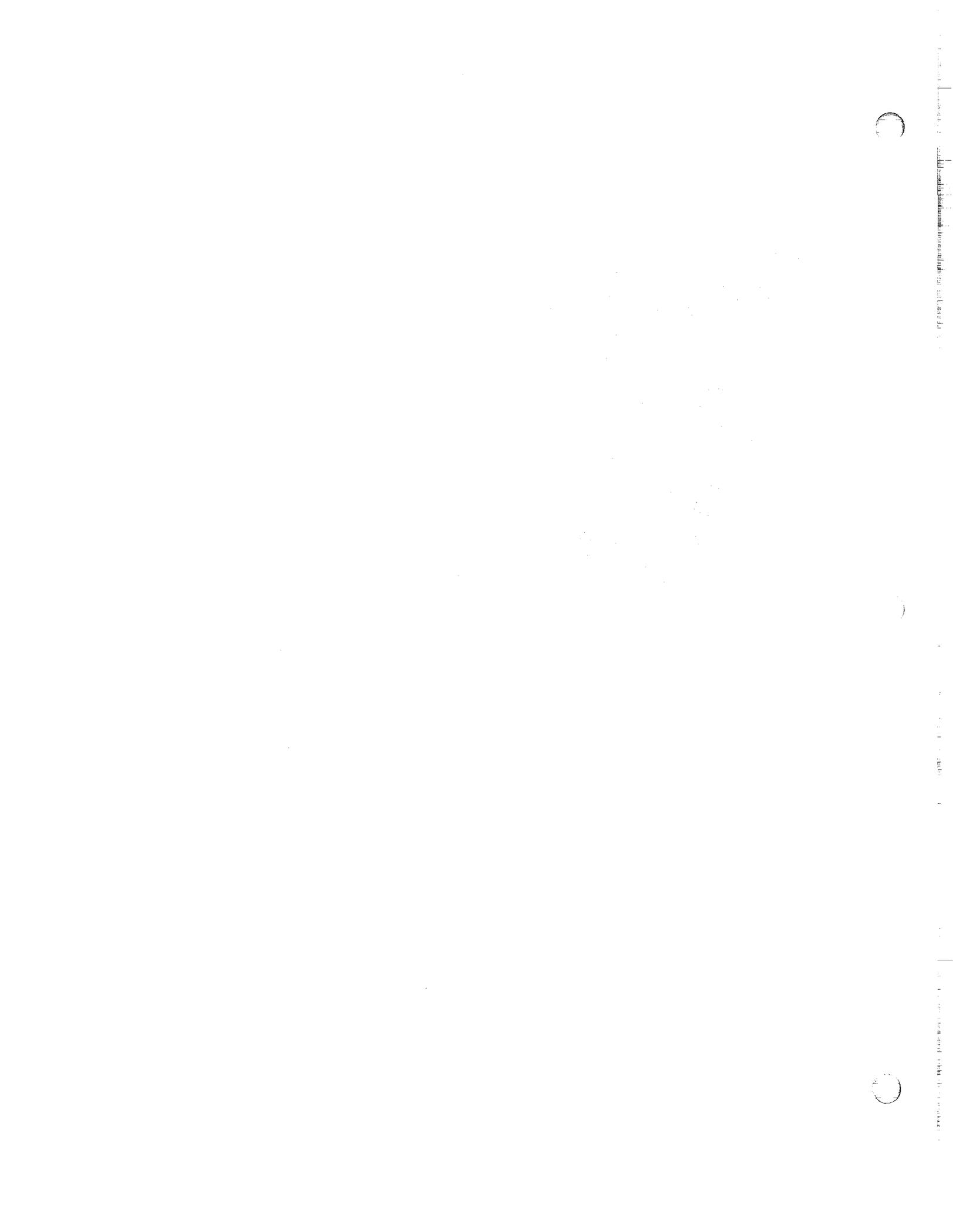
Class Pen size  
Principle of Operation Ionization chamber with built in microscope

Sensitivity and Range

MODEL	PARTICLE	ENERGY	RANGE
655-010	Gamma, X	80 keV to 300 MeV	0-200 mR
655-012	Neutrons	0.025 eV	0-120 m Rem

Sampling Continuous  
Performance Accuracy: 10% of true dose  
Leakage: 2% per day  
Requirements Power: One 1.5 V battery for charger  
Size: 11 cm L, 1 cm D (4.4"L, 0.5"D)  
Weight: 150 grams (1.25 oz)  
Cost 655-010 \$42  
655-012 \$85  
655-019 \$50-charger

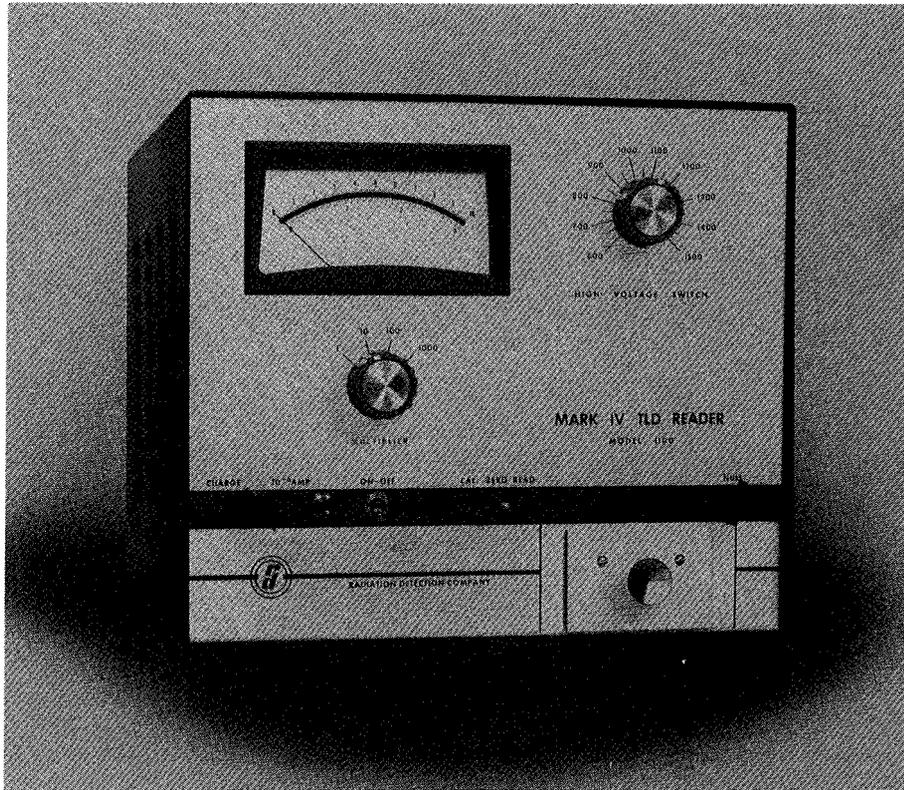
Address Picker Corporation  
333 State Street  
North Haven, CN 06473  
  
Picker Corporation  
Marketing Services  
6119 Highland Road  
Cleveland, OH  
Attention: Mr. Frank Dean





### TLD Reader

#### Radiation Detection Mark IV Model 1100



Class	Laboratory
Principle of Operation	TLD reader for thermoluminescent dosimeters. Accepts a wide variety of powdered or solid TLD's, mainly designed for LiF. Constant voltage heater supply, chopper stabilized operational amplifier.
Sensitivity and Range	Sensitivity: 10 mR to 50,000 R for LiF
Performance	Accuracy: 10 mR to 100 mR better than $\pm 10$ mR 100 mR to 1 R better than $\pm 10\%$ Above 1 R better than $\pm 5\%$ Readout Time: 10 to 15 seconds
Requirements	Power: 115 VAC, 60 cycles Size: 12.4" height, 14.1" width, 18.6" depth (31 x 36 x 47 cm) Weight: 42 lb (19 kg)
Features	<ol style="list-style-type: none"> <li>1. Removable planchet for powder samples</li> <li>2. Internal reference light</li> <li>3. Provision for recording glow curves</li> <li>4. Inert gas flow to suppress nonradiation-induced luminescence</li> </ol>
References	Manufacturer's specifications
Cost	\$1595.
Address	Radiation Detection Company 162 Wolfe Road Sunnyvale, CA 94088 (415) 967-2837

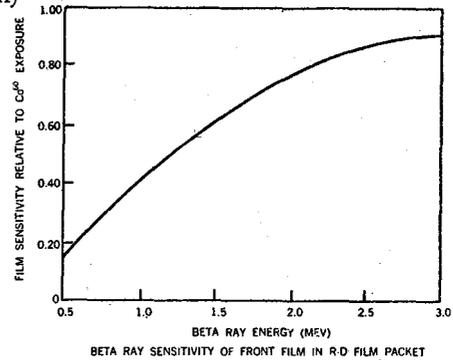
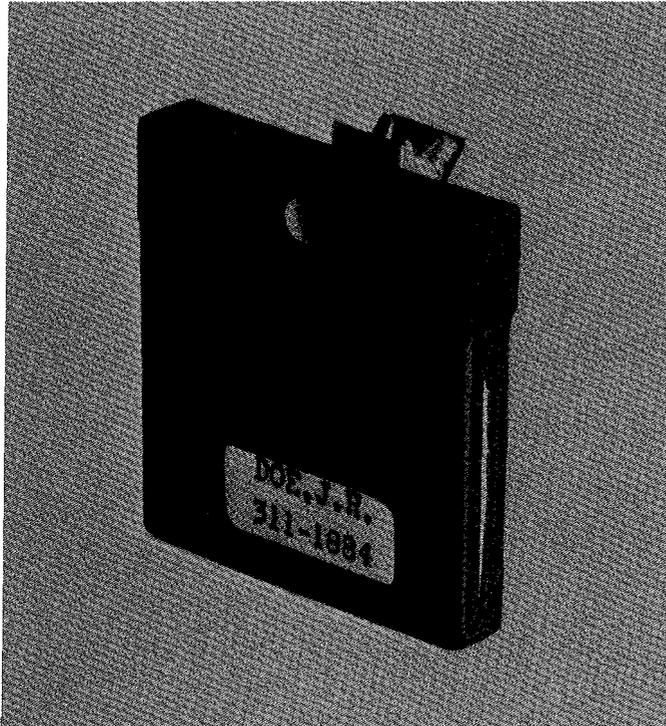




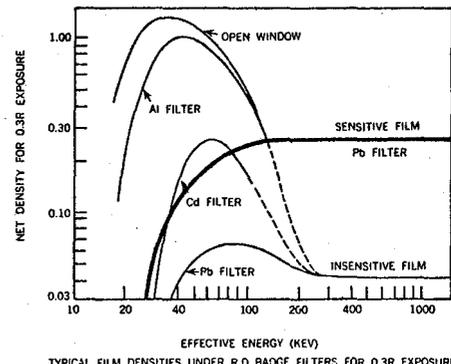
INSTRUMENTATION  
FOR ENVIRONMENTAL  
MONITORING

RAD-DOS  
Dosimetry Service  
Radiation Detection  
Oct. 1972

Film Badge Dosimetry Service  
Radiation Detection Company



BETA RAY SENSITIVITY OF FRONT FILM  
IN R-D FILM PACKET



FILM DENSITIES UNDER R-D BADGE  
FILTERS FOR 0.3R EXPOSURE

Class	Film Badge Dosimetry Service		
Principle of Operation	Film is distributed to user, developed after exposure, exposure reading reported. Film is sensitive to gamma, X-ray, and beta radiation, with four filters.		
Sensitivity and Range	<u>Radiation</u>	<u>Energy Range</u>	<u>Exposure or Dose Equivalent</u>
	X, Gamma	25 keV to 70 keV 70 keV to 3 MeV	0.005 to 50 R 0.01 to 1000 R
	Beta	Above 1 MeV	0.01 to 1000 rem
Filters	Open Window Region 0.125" Plastic 0.032" Aluminum 0.015" Cadmium plus Aluminum 0.030" Lead plus Aluminum		
Features	<ol style="list-style-type: none"> <li>1. Films cycle weekly, biweekly, monthly.</li> <li>2. Reports mailed to user within three days of receipt of film.</li> <li>3. Ring dosimeters also available.</li> <li>4. Neutron dosimeters using several activation foils also available.</li> </ol>		
References	Manufacturer's specifications		
Cost	Upon request		
Address	Radiation Detection Company 162 Wolfe Road Sunnyvale, CA 94086 (408) 735-8700		



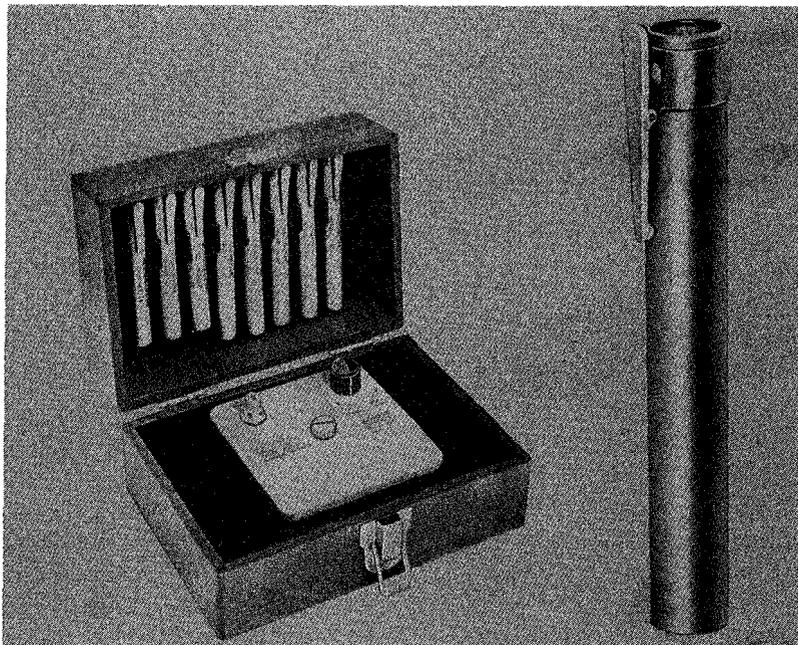


INSTRUMENTATION  
FOR ENVIRONMENTAL  
MONITORING

RAD-DOS  
Beta, Gamma, X-Ray  
REM  
Aug. 1972

Dosimeter/Charger System

REM Incorporated



Class	Portable Dosimeter
Principle of Operation	Quartz-fiber ion chamber, charger-reader system, output in rad
Sensitivity and Range	Sensitivity: Eight models with full-scale sensitivities of 0.2, 5, 10, 20, 50, 100, 200, 600 rad; one neutron-sensitive model Energy Range: 15% maximum energy dependence, 80 keV to 2 MeV (gammas)
Sampling	Continuous
Performance	Accuracy: $\pm 10\%$ for gammas, $\pm 20\%$ for neutrons
Requirements	Power: 1.5 volt battery, lifetime a few hundred dosimeter charges Kit Size: 6" x 8" x 3" (15 cm x 20 cm x 7.5 cm) Kit Weight: 2 lb (1 kg) Dosimeter Size: 4-1/2" length, 1/2" diameter Dosimeter Weight: 1/2 oz (14 g)
Features	Leakage 2%/24 hours; geotropism $\pm 2\%$ ; hermetically sealed chamber
References	Manufacturer's specifications
Cost	\$65.
Address	REM Incorporated 2000 Colorado Avenue Santa Monica, CA 90404 (213) 828-6078

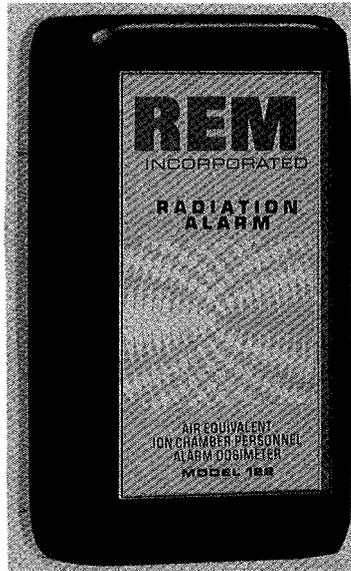




INSTRUMENTATION  
FOR ENVIRONMENTAL  
MONITORING

RAD-DOS  
X-Ray, Gamma  
REM, Inc.  
Nov. 1972

Pocket Dosimeter  
REM Squeeler Model 122



Class	Personal, Portable
Principle of Operation	Air equivalent ionization chamber measures the integrated dose, audible alarm at preselected adjustable dose.
Sensitivity and Range	Energy Response: $\pm 15\%$ 25 KeV to 50 KeV $\pm 10\%$ 50 KeV to 2 MeV Range: Model 122A 20-300 mR alarm level settings Model 122B 200 mR to 3 R alarm level settings
Sampling	Continuous
Performance	Accuracy: $\pm 5\%$ Leakage: Less than 3 mR per 8 hours
Requirements	Power: Two 22.5 V batteries, 100 hours Size: 6.5 cm, 15 cm, 2.5 cm (2.5", 6", 1") Weight:
References	Manufacturer's specifications
Cost	
Address	REM Incorporated 3107 Pico Boulevard Santa Monica, CA 90405 (213) 828-6469





Dosimeters

Searle Analytic Models NC-401, 402, 405

Class Portable, personnel dosimeter  
Principle of Operation Model 401, 402, indirect reading quartz fiber volt meter  
Model 405, indirect reading ion chamber

Model: 401 402 405

		401	402	405
Sensitivity	Radiation Measured:	Thermal Neutrons	Gamma x-ray	Gamma x-ray
	Full-scale Range:	120 mrem	200 mr	200 mr
Performance	Accuracy:	±10%	±10%	
	Average leakage per 24 hours:	Less than 1% of full scale	Less than 1% of full scale	
Requirements	Power:	Battery, 1.5 v	Battery, 1.5 v	Charger required
	Size:	1.5 cm dia. 10.5 cm long	1.5 cm dia. 10.5 cm long	1.3 cm dia. 9 cm long
	Weight:	45 g (1.25 oz)	45 g (1.25 oz)	30 g (1 oz)

References Manufacturer's Specifications

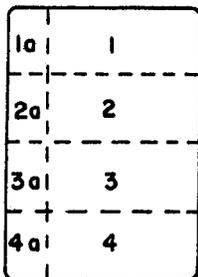
Cost Dosimeter Model No. 401 \$  
402 \$\$  
405 \$  
Charger Model No. 403 \$  
404 \$

Address Searle Analytic, Inc.  
(formerly Nuclear-Chicago)  
2000 Nuclear Drive  
Des Plaines, Illinois 60018  
(312) 827-4456

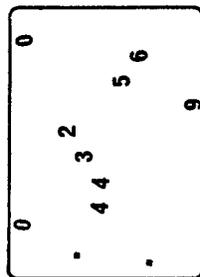




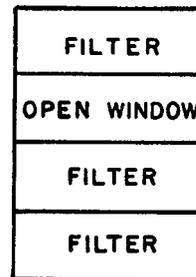
TLD Dosimetry System and Badge Service  
Teledyne Radi-Guard and Model 8300



Multi-area dosimeter.  
4 main readout areas  
(1, 2, 3, 4) and 4 backup  
areas (1a, 2a, 3a, 4a).

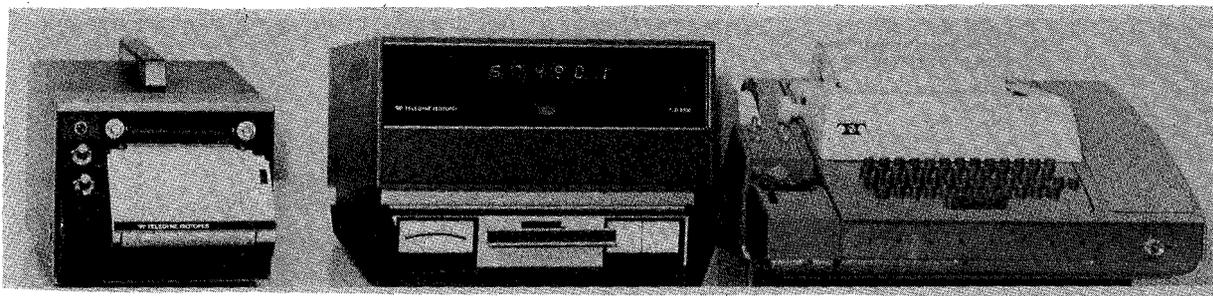


Rear view of dosimeter with  
identification numbers. 2  
dots insure dosimeter is  
properly inserted in reader.



Standard personnel  
badge filter array.

*RADI-GUARD DOSIMETER*



*Model 8300 Reader*

Class Thermoluminescent Dosimetry System (TLD)

Principle of Operation TLD dosimeters with various parameters are read out by heating and observing thermoluminescence. Badge service is provided. Standard Badge (see figure) has following filters:

1. 2 mm plastic + 1 mm Al
2. open
3. 1 mm plastic + 1 mm Cu + 1 mm Al
4. 1 mm plastic + 1 mm Pb + 1 mm Al

Sensitivity and Range (a) Personnel  
With Cobalt-60 exposure, maximum error (90%) =  $\pm 8$  mR or  $\pm 10\%$ , whichever is greater. Dosimeter is 15% loaded LiF.

(b) Environmental  
Standard dosimeter is  $\text{CaSO}_4:\text{Dy}$ , 30% loaded.

Features (a) Model 8300 Reader

- (1) Integrator linear to  $\pm 1.5\%$  over entire range
- (2) 6-decade display
- (3) Glow curve output (0 to 8 volts) provided, with automatic range changing
- (4) Inert gas flow is used
- (5) Built-in annealing feature with on-off switch
- (6) Fast calibration (2 seconds)
- (7) 4 programmed temperature cycles available, cycle times 12 to 18 sec
- (8) Switch can enable reading of LiF,  $\text{CaSO}_4:\text{Dy}$ , and finger dosimeters
- (9) Model 9100 automatic system is compatible



Features (Cont'd)

(b) TLD Dosimeters

- (1) Rods (1 mm diameter, 6 to 150 mm long) available in  $^6\text{LiF}$ ,  $^7\text{LiF}$ ,  $\text{CaSO}_4:\text{Mn}$ ,  $\text{Li}_2\text{B}_4\text{O}_7:\text{Mn}$
- (2) Tape (25 mm wide, 0.4 mm thick) available in  $^7\text{LiF}$
- (3) Discs of various sizes available in  $^7\text{LiF}$ ,  $^6\text{LiF}$ ,  $\text{Li}_2\text{B}_4\text{O}_7:\text{Mn}$ ,  $\text{CaF}_2:\text{Mn}$ ,  $\text{CaSO}_4:\text{Mn}$

(c) Dosimetry Service

- (1) Issue period: 1, 2, 4 weeks, 1 month or 3 months
- (2) Radi-Guard dosimeters (see figure) are used
- (3) Calibration is done with cobalt-60
- (4) Zero control badges supplied free with each shipment

References

- 1) Manufacturer's specifications

Cost

Model 8300 Reader	\$
Dosimetry Service: processing per badge	\$ 0.65
report charge per reporting period	\$ 5.00
replacement for lost badge	\$ 2.50
initial charge per badge	\$ 0.75

Dosimeters -- prices vary with particular models

Address

Teledyne Isotopes  
50 Van Buren Avenue  
Westwood, NJ 07675  
(201) 664-7070

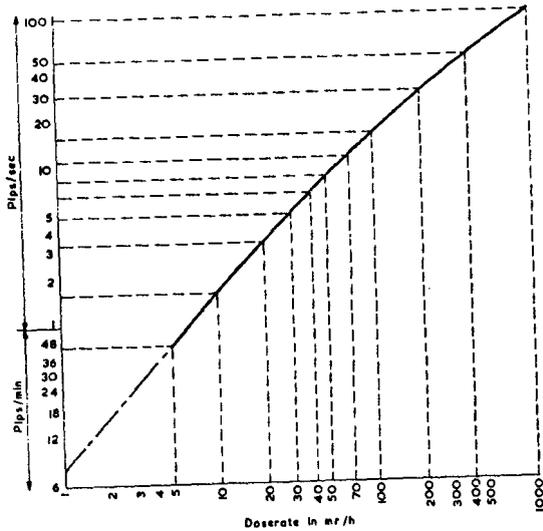
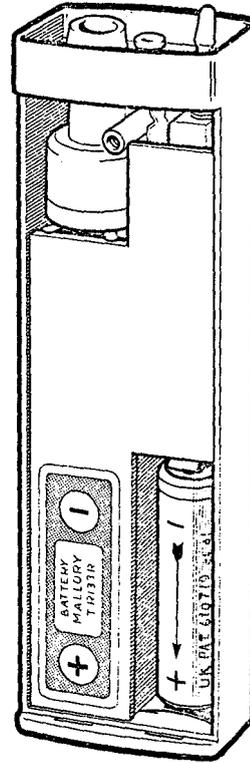
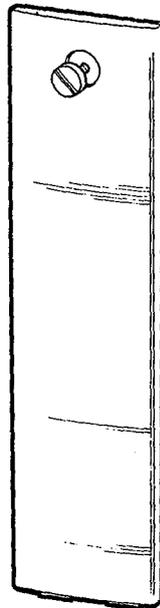
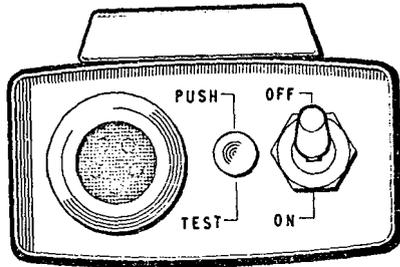
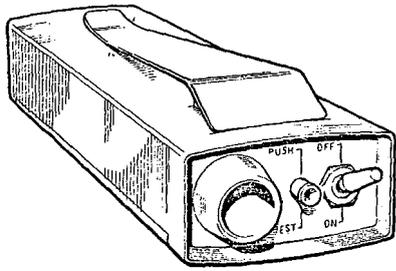


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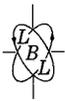
0 0 0 0 3 6 0 2 4 1 3

RAD-DOS  
Pocket  
UKAEA  
Aug. 1972

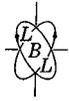
Gamma Dose-Rate Alarm  
UKAEA Model 0023



Pip rate against doserate



Class	Portable, Pocket	
Principle of Operation	Miniature GM counter, energy compensating filter, audio oscillator, and speaker.	
Sensitivity and Range	Energy Range: 60 keV to 2 MeV Sensitivity: 0 to 250 mR/hr, proportional; 250 to 2000 mR, nonlinear; 2000 R/hr maximum	
Performance	Accuracy: Temperature: -10°C to +55°C	
Requirements	Power: One 4 V mercury cell, Mallory TR133R; 6 months Size: 15 cm x 4.5 cm x 2.6 cm (5.8" x 1.8" x 1.1") Weight: 140 g (0.3 lb)	
Features	Built in Sr-90 test source operated by push button, equivalent to 2-4 mR/hr. Model 95/0061/6 has suppressed response below 120 mR/hr, no test source, threshold device.	
References	Manufacturer's specifications	
Cost	\$240.	
Address	Nuclear Enterprises Ltd. Bath Road, Beenham Reading, Berks, ENGLAND Tel.: Woolhampton 2121	Nuclear Enterprises Ltd. 935 Terminal Way San Carlos, CA 94070 (415) 593-1455



INSTRUMENTATION  
FOR ENVIRONMENTAL  
MONITORING

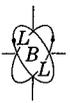
RAD-DOS  
Gamma  
Victoreen 3  
Mar. 1972

Personal Dosimeter  
Victoreen Model 541-A



Class	Portable
Principle of Operation	Pocket ion chamber
Sensitivity and Range	Energy Sensitivity: $\pm 10\%$ from 55 keV to 2 MeV, gamma $\pm 15\%$ from 45 keV to 2 MeV, gamma Range: 0 - 0.2 Roentgen
Performance	Leakage rate: 2% per 24 hrs
Requirements	Power: None Size: 4" long, 1/2" dia.
Features	Uses Model 2000A charger, \$55
Cost	\$45
Remarks	A direct reading personal dosimeter. The roentgen scale is built into the dosimeter and can illuminated by an external light source.
Address	Victoreen Instrument Division 10101 Woodland Avenue Cleveland, Ohio 44104 (216) 795-8200





0 0 3 0 3 6 0 2 4 1 5

Personal Dosimeter Reader/Charger  
Victoreen Model 687C Minometer II



**Class** Laboratory

**Principle of Operation** Radiation penetrating a pocket chamber capacitor causes current to leak off in proportion to the amount of radiation. The resulting voltage drop is calibrated in milliroentgens and is read on the Minometer II.

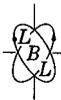
**Sensitivity and Range** Full scale ranges: 0-40 mR 0-200 mR  
Minimum detectable energy:

**Performance** Instrument accuracy:  $\pm 10\%$   
Leakage rate:

**Dosimeters**

Model	362	362A
Radiation Detected	Gamma & X-ray	Gamma & X-ray
Gamma Energy Dependence	$\pm 15\%$ from 30 keV to 1.2 MeV	$\pm 15\%$ from 30 keV to 1.2 MeV
Cs <sup>137</sup> Calibration	$\pm 15\%$	$\pm 15\%$
Range	0-0.2 R	0-0.5 R
Overall Dimensions	5-1/2" long, 1/2" dia.	5-1/2" long, 1/2" dia.
Price	\$12	\$12

**Requirements** Power: 100-130 V ac, 50 or 60 Hz, 15 VA  
Size: 10.75" long x 7.5" wide x 5.25" high  
Weight: 9 lbs



STRAY RADIATION CHAMBERS  
(Supplement to Victoreen Minometer II)

MODEL	208	208A	239	239A
Radiation Detected	X-Ray & Gamma	X-Ray & Gamma & Beta	X-Ray & Gamma	X-Ray, Gamma & Beta
Accuracy	+10%	+10%	+10%	+10%
Range	0.001R and 0.002R Full Scale	0.001R and 0.002R Full Scale	0.01R and 0.002R Full Scale	0.01R and 0.002R Full Scale
Energy Dependence	+5%, 40 KeV to 1.2 MeV +15% Down to 30 KeV	+10% From 5 to 50 KeV	+10% Down 35 KeV +25% Down 20 KeV	+10% from 5-50 KeV
Overall Dimensions	6 1/2" Long 4 1/2" Dia.	6 1/2" Long 4 1/2" Dia.	4 1/2" Long 2" Dia	4 1/2" Long 2" Dia
Cost	\$110	\$175	\$95	\$145

Features Reader and charger for pocket chambers

Cost \$525

Address Victoreen Instrument Division  
10101 Woodland Avenue  
Cleveland, Ohio 44104  
(216) 795-8200



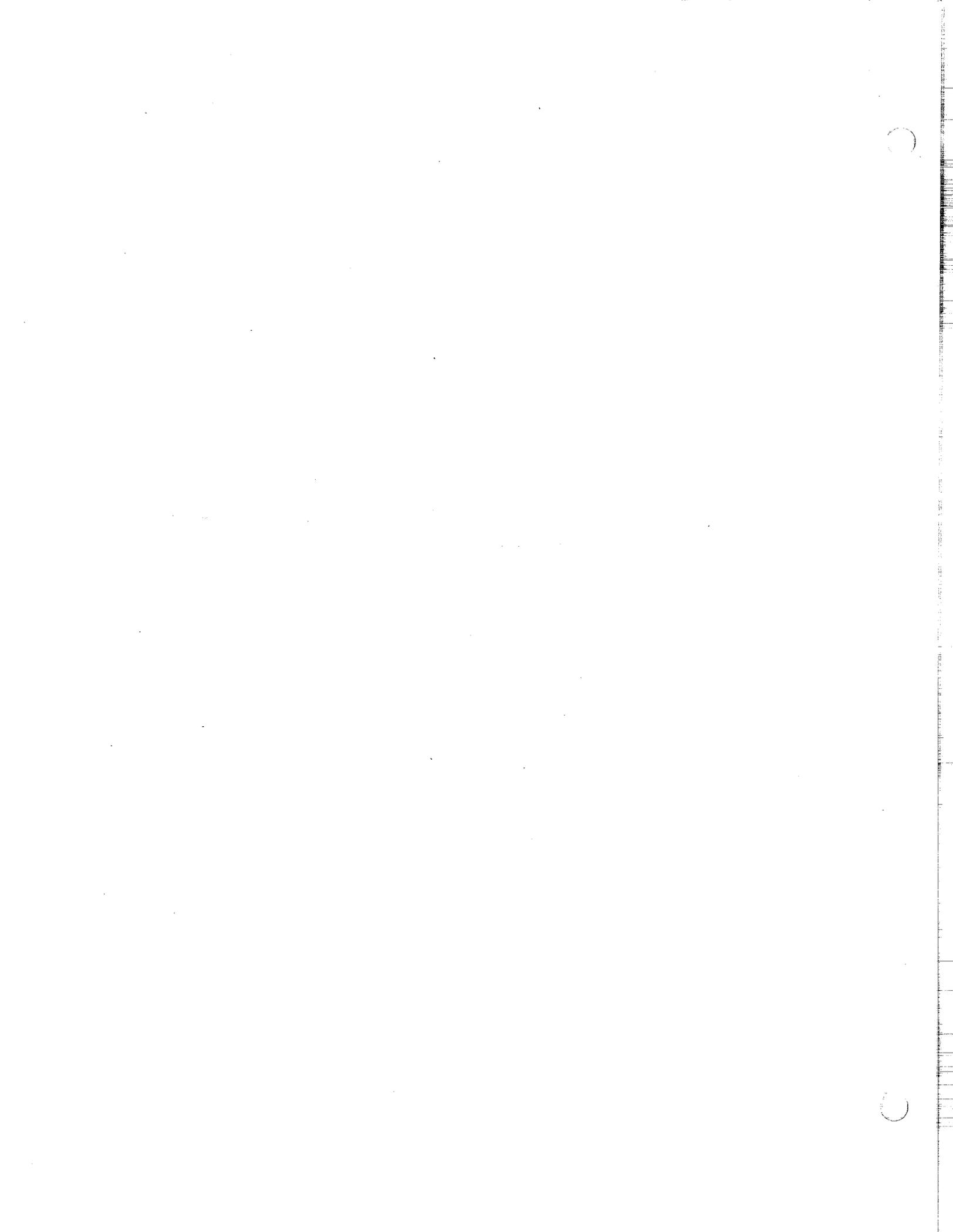
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INSTRUMENTATION  
FOR ENVIRONMENTAL  
MONITORING

RAD-NUC  
Radionuclides  
Contents  
Oct. 1973

G. RADIONUCLIDES

1. Tritium
2. Krypton - 85
3. Radon - 222 and Its Daughters
4. Strontium - 90 and 89
5. Iodine - 131 and 129
6. Radium
7. Uranium
8. Plutonium
9. Instrument notes

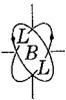




## T R I T I U M

1. Introduction
2. Tritium in Water
  - a. Liquid Scintillation Counting
  - b. Tritium in Flowing Water
  - c. Urinalysis
  - d. Solid Scintillation Counting
3. Tritium in Air
  - a. Ionization Chambers
  - b. Proportional Counters
  - c. Silica Gel/Liquid Scintillation
  - d. Bubbler Exchange
  - e. Combustion for HT Determinations
4. Enrichment Techniques
5. Summary and Conclusions
6. Acknowledgment
7. References





## 1. INTRODUCTION

Tritium (the hydrogen isotope of mass 3) is one of the most important radioactive nuclides released to the environment during the normal operation of light water nuclear reactors and their fuel reprocessing plants. Tritium is produced principally in ternary fission in the reactor core and to a much lesser extent by activation of the primary cooling water. The fission yield of tritium (tritium atoms per fission) is about 0.01% (Ref. 1), and in fuel ready for reprocessing ( $\leq 150$  days of cool-down) it is present at the level of about 700 curies/metric ton of fuel (Ref. 1). Tritium is also produced by neutron irradiation of the deuterium,  $^3\text{He}$ ,  $^{10}\text{B}$ , and  $^6\text{Li}$  in the reactor coolant (Ref. 2); Table I (from Ref. 3) shows the estimated tritium production (Ci/yr) at typical nuclear power stations of the boiling water (BWR) and pressurized water (PWR) types. Essentially all of the tritium produced is ultimately released to the environment, either at the reactor or most probably during fuel reprocessing. Tritium appears predominantly as tritiated water (HTO), and to a lesser extent as hydrogen gas (HT). With the use of Zircaloy-clad fuel rods in light-water reactors, tritium diffusion from the spent fuel should be very small ( $\leq 1\%$ ), so that most of the release occurs at the fuel reprocessing plant (Ref. 4). With the increasing number of nuclear power plants, the world-wide inventory of tritium is expected to rise dramatically. One projection is shown in Figure 1, from Ref. 1. Krypton-85 is also shown for comparison.

Tritium is also produced at heavy-water reactors and at tritium production facilities, at which its relative importance is generally greater than that of all other radionuclide hazards combined (Ref. 4).

Tritium's radioactive decay has a half-life of 12.36 years. The decay is by  $\beta^-$  emission. Figure 2 (from Ref. 5) shows the beta energy spectrum. The very weak beta has a maximum energy of 18.6 keV and a most probable energy of 5.7 keV. (The maximum range of an 18 keV beta is about 6 microns in water or 0.5 cm in air.) There is no other decay mode. One gram of pure tritium would have an activity of about 9600 curies.

Tritium appears naturally in the environment at an estimated world-wide level of from 20 to 60 megacuries (Ref. 4), corresponding to a mass of 2 to 6 kg. All but a fraction of one percent is in the form of HTO in the air and water. The fractional atomic concentration relative to ordinary hydrogen is about  $4 \times 10^{-15}$  (HT/H<sub>2</sub>) in the atmosphere; about  $8 \times 10^{-17}$  (HTO/H<sub>2</sub>O) as vapor in air; and about  $1 \times 10^{-18}$  in surface water (Ref. 4). Most of the naturally-occurring tritium is produced by interactions in the upper atmosphere of the primary cosmic ray protons (Ref. 6).

Production of tritium by atmospheric nuclear tests has dwarfed the natural levels: the tritium yield of a thermonuclear reaction (fusion) has been estimated at about 1 kg per megaton equivalent (Ref. 4, 7). Essentially 100% of the tritium produced in this way is converted to HTO water vapor (Ref. 8). The concentration in stratospheric water of tritium rose from about 0.03 pCi/cm<sup>3</sup> (in the period prior to nuclear testing) to about 20,000 pCi/cm<sup>3</sup> in 1960, an increase of almost six orders of magnitude (Ref. 4). Although it is chemically identical to normal hydrogen for most purposes, stratospheric tritium may not mix quickly and thoroughly with much of the naturally-occurring water.

TABLE I.

*Estimated tritium ( $^3\text{H}$ ) production at nuclear power stations, Ci/yr (Ref. 3).  
(Power level approximately 1000 MWe and 3300 MWt)*

Source	Reaction	BWR	PWR
coolant water	$^2\text{H}(n,\gamma)^3\text{H}$	8	3
dissolved boron	$^{10}\text{B}(n,2\alpha)^3\text{H}$	0	600
uranium fuel rods	fission	13,000	13,000
boron control rods	$^{10}\text{B}(n,2\alpha)^3\text{H}$ and	5,000	0
	$^{10}\text{B}(n,\alpha)^7\text{Li}$ ,		
	$^7\text{Li}(n,n\alpha)^3\text{H}$		

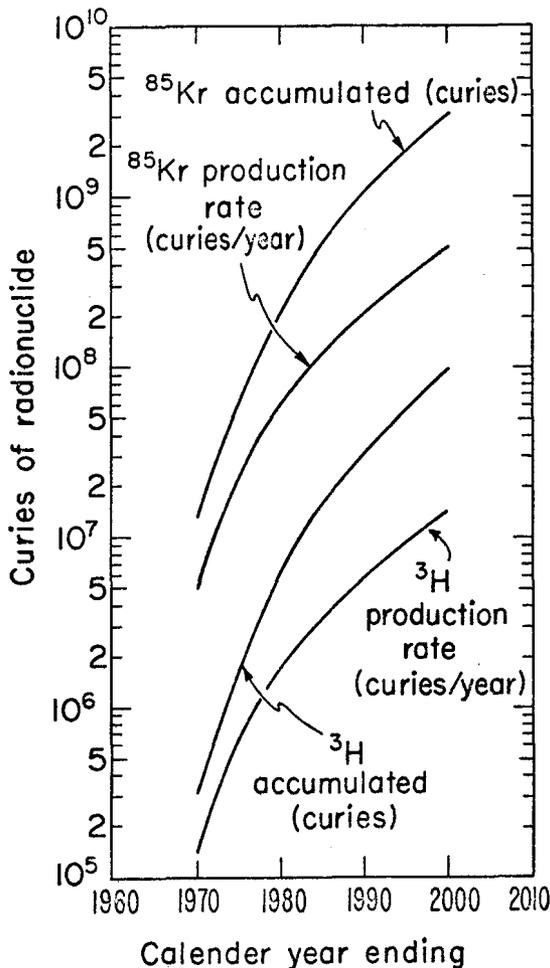
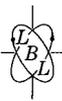


FIGURE 1. Estimated production of tritium ( $^3\text{H}$ ) and Krypton-85 from the nuclear power industry of the free world (from Ref. 1).

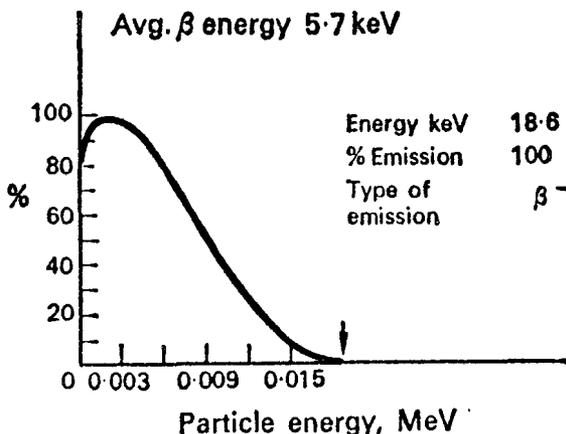


FIGURE 2. Beta energy spectrum for tritium (from Ref. 5).

For 168-hour occupational exposures, the International Commission on Radiological Protection (Ref. 9) has established maximum permissible concentrations in air and water  $(\text{MPC})_a$  and  $(\text{MPC})_w$ , as well as maximum permissible body burdens  $(\text{MPBB})$ . For individuals in the general public, the applicable MPC values are a factor of 10 smaller than for occupational exposure; and for exposure to a suitably large sample of the general public, another factor of 3 smaller still. The U.S. Atomic Energy Commission (Ref. 10) has established MPC levels identical to those of the ICRP.

The occupational MPBB values are 1 mCi of HTO in body tissue, or 2 mCi in the total body. The MPC values are as follows:

	168-hour occupational limits	168-hour limits, individuals in general public
as HTO in water	30,000 pCi/cm <sup>3</sup>	3000 pCi/cm <sup>3</sup>
as HT gas in air	400 "	40 "
as HTO vapor in air	2 "	0.2 "

In terms of curies, these permissible levels seem at first glance to be very high in comparison to MPC levels for other radionuclides. There are three reasons for this: first, there is a large amount of water in biological systems, in which ingested tritium can be diluted; second, there is no known biological or physical reconcentration mechanism; and third, the tritium betas have very low energy.

Tritium seems to have a biological half-life of about 8 to 10 days in humans (Ref. 11, 12, 13), which is similar to that of the gross water content of the body.

Measurements of tritium are difficult mainly because of the low energy of the tritium betas, which can be counted by only a few of the many detectors sensitive to ionization energy loss. The discussion below is intended to point out the advantages and disadvantages of the various existing (or proposed) measuring systems for tritium.

We shall concentrate upon measurements in the air and water around nuclear reactors and their fuel-reprocessing plants. We do not intend to imply that tritium is not now a significant problem outside of the reactor industry. Its use is increasing in medical and biological research as a tracer, and in industrial applications. A recent study (Ref. 14) of tritium's use in luminous watch dials is interesting, if only because of its historical connection with the problem of the radium dial painters of several decades ago. A recent study of the United Kingdom (Ref. 15) indicates that tritium is widely used in industry and that "the radiation



doses received by some workers are not inconsiderable." What was found was that a few percent of the workers studied received doses less than but comparable with MPC levels. Background information on the use of tritium in biology and medicine can be found in the recent book by Feinendegen (Ref. 16).

We will consider four distinct measurement problems: determinations in water, in air as HTO, in air as HT, and in urine. In turn, any of these (except that in urine) can be made in either environmental samples or process samples (water flow or gas stack). We shall attempt to discuss these various classes separately, where appropriate. Measurements in urine, however, will only be touched on briefly here. A fuller discussion can be found in Volume 4 ("Biomedical") of this Survey.

Various interferences must be considered depending on where, when, and how the tritium concentration is being sampled or measured. In the gas phase around fuel reprocessing plants, the principal beta emitter besides tritium is  $^{85}\text{Kr}$ , whose higher energy beta ( $E_{\text{max}} = 672 \text{ keV}$ ) can provide a significant interfering background. In the liquid phase, strontium-90 ( $E_{\text{max}} = 546 \text{ keV}$ ) is present and may interfere.

The literature on tritium measurements is voluminous, more so than perhaps for any other radionuclide of interest in environmental measurements. The IAEA held a Symposium in 1961, the proceedings of which (Ref. 17) are a basic reference in this field. A Tritium Symposium was held in Las Vegas in September, 1971 (Ref. 18), in which the major research groups in the field summarized the current status of instrumentation at that time. Also, a monograph by Jacobs (Ref. 4), part of the A.E.C. Critical Review Series, is an excellent reference on the sources of tritium and its behavior in the environment. Two recent bibliographies by Hannahs and Kershner (Ref. 19) and by Rudolph, Carroll, and Davidson (Ref. 20) are also valuable.

In the literature on tritium, one frequently encounters the use of the Tritium Unit (TU) or Tritium Ratio (TR), defined as a tritium concentration of 1 tritium atom in  $10^{18}$  normal hydrogen atoms. A concentration of 1 TU in water corresponds to  $3.23 \times 10^{-3} \text{ pCi/cm}^3$  of water, or alternatively 0.12 dps/liter.

Another notation which often appears is the designation 'protium' for normal hydrogen ( $^1\text{H}$ ), to distinguish it from deuterium ( $^2\text{H}$ ) and tritium ( $^3\text{H}$ ). The three isotopes are also commonly denoted by H, D, and T.

## 2. TRITIUM IN WATER

In water, tritium occurs primarily as HTO (tritiated water). It is also occasionally found in tritiated organic compounds dissolved or suspended in water, but we shall not discuss the measurement of these cases except when their measurement is contained within the measurement of HTO itself.

As mentioned above, the maximum permissible concentration (MPC)<sub>w</sub> of tritium as HTO in liquid water for an individual in the general population is  $3000 \text{ pCi/cm}^3$ . This MPC corresponds to 110 disintegrations per second per  $\text{cm}^3$  (dps/ $\text{cm}^3$ ). The main measurement problem is the low energy and range of the beta (the maximum range in water being 6 microns).

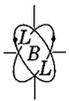
### A. Liquid Scintillation Counting

Efforts to obtain sensitive, reliable measurements have essentially all relied on one technique: *liquid scintillation counting*. There are many liquid scintillation systems in the literature and several are available commercially. Here we shall try to summarize the properties of the various liquid scintillation systems. Two recent books are excellent references on this subject: one edited by Horrocks and Peng (Ref. 21), and the other edited by Bransome (Ref. 22).

Liquid scintillation counting relies on the fluorescence of scintillator molecules excited by the deposition of ionization from the beta. Energy given to solute molecules by the  $\beta$  is transferred to a fluorescent molecule which it excites. A photon is emitted when the excited molecule returns to its ground state. For most common scintillator solutes, the fraction of excited molecules which de-excite by photon emission is about 90%. The photons emitted are detected by a multiplier phototube (MPT), or in the case of tritium nearly always by a pair of MPT's in time coincidence.

A good discussion of this technique has been given by Horrocks (Ref. 23):

"The main advantage obtained from the use of liquid scintillation counting as opposed to other conventional counting methods has to do with the fact that the sample containing the radioactivity is in intimate contact with, and in most cases dissolved in the detector. This produces the ideal  $4\pi$  geometry. A limiting factor in the liquid scintillation counting system is the MPT. Even the best MPTs have photocathode efficiencies for conversion of photons



into electrons of only 28%. . . .

"The liquid scintillator solution consists of three main parts. The bulk of the solution is the solvent, usually an alkyl benzene (i.e., toluene) or dioxane. To the solvent an organic compound (or compounds) is added which is an efficient photon emitter and which emits photons in the wavelength region which is easily measured by the MPT. And finally the liquid scintillator solution contains the sample; although there are some uses where the excitations are generated by an external sample. The most troublesome problem encountered with liquid scintillation counting involves those problems associated with the introduction of the sample into the scintillator solution in a homogeneous (or near homogeneous) form without causing a drastic reduction in photon emission due to so called quenching processes. . . .

"Quenching is defined as any of those mechanisms which lead to a decrease in the number of photons emitted by a scintillator solution. There are two basic types of quenching; chemical and color. Chemical quenching involves the action of foreign materials upon the excited molecules; either solvent or solute molecules. Energy is removed from the excited molecules by a process which does not involve the emission of photons. Color quenching involves the re-absorption of emitted photons by foreign material which may be present in the liquid scintillator solution or sample. The re-absorbed photons will produce excited molecules which deactivate by non-radiative processes.

"The overall effect of these processes is the decrease in the average number of photons emitted per unit energy of exciting particle stopped in the solution. It is important to keep in mind that a 50% quenching of photon yield does not mean a 50% decrease in counting efficiency. Indeed the same amount of quenching will produce a greater change in counting efficiency for tritium than for  $^{14}\text{C}$ ."

Recent papers by Moghissi et al. have studied the significant factors which affect low-level liquid scintillation counting of tritium, both in homogeneous liquid systems such as solutions (Ref. 24), and in emulsions (Ref. 25). From the paper on homogeneous solutions (Ref. 24), several conclusions are important:

- (a) Plastic vials are preferable to glass vials; "glass seems to decrease the counting efficiency for tritium counting and increase the background as compared to plastic."
- (b) The solution which optimizes counting efficiency is 6 to 7 g of PPO, 1.2 to 1.5 g of *bis*-MSB, and 120 g of naphthalene per liter of dioxane. [PPO is 2,5 - diphenyl-oxazole; *bis*-MSB is *p-bis*-(*o*-methylstyryl)-benzene.] Various studies of concentrations were conducted to determine this optimum.
- (c) In this solution, the optimum water content is about 1 ml water/4 ml solution. There are two competing processes here, namely that additional water increases quenching at the same time that it increases the proportion of water which contains the activity.
- (d) Phosphorescence is severe, and "in considering the long-lived component of phosphorescence, it is desirable to avoid any light which might excite the scintillation solution" (Ref. 24).
- (e) A constant check of possible instrument drift is required for precise measurements. "The recommended order of counting is: background, sample, standard calibration sample, duplicate of sample, background."
- (f) "The preparation of water samples is simple and consists of a single distillation." (In fact, some samples require no distillation at all.)
- (g) The Y-value (defined as minimum detectable activity at a confidence level of one sigma of statistical error for one-minute counting time) is about 1 pCi/cm<sup>3</sup>.

Even more sensitive systems are available, using emulsions. Lieberman and Moghissi (Ref. 25) report a Y-value of 0.7 pCi/cm<sup>3</sup> using the detergent Triton N101 (Rohm & Haas Co.) in scintillation solution at a ratio of 1:2.75. This mixture can hold 40 to 50% water, when the scintillation solution is *p*-xylene containing 7 g PPO and 1.5 g *bis*-MSB per liter. This cocktail has a counting efficiency of ~23% for the tritium beta. By using three photomultipliers viewing a large-volume (250 ml) vial, Moghissi (Ref. 26) has achieved a Y-value of 0.2 pCi/cm<sup>3</sup>.

Calibration and standardization of systems as sensitive as these are difficult problems. The National Bureau of Standards has tritiated-water and tritiated-toluene standards (Ref. 27). The most frequently used calibration method is the 'internal standard', in which a standard activity (e.g., from NBS) is used interchangeably with the unknown samples. The problem is that the addition of a standard can change



the counting efficiency, due to differences in quenching. This problem can be overcome by the addition of a very small (microliter) amount of a high-activity standard to a previously counted sample, but accurate measurements of small size samples, usually made gravimetrically, are difficult and tedious.

Moghissi and Carter (Ref. 28) have used two samples which differ only in that one employs normal and the other tritiated naphthalene. This minimizes most of the errors due to chemical differences.

Another method is to use one or more radioactive gamma sources placed outside of the counting vial (the so-called 'external standard method'). These can be used to adjust the gain on the photomultiplier tubes, or to partially compensate for differences in quenching characteristics among several samples in a train. Systems with automatic feedback control, using radioactive sources, are available from some of the commercial liquid-scintillation manufacturers. Both the American Public Health Association's Standard Methods (Ref. 29) and the American Society for Testing and Materials compilation (Ref. 30) contain standard methods using liquid scintillation counting for tritium analysis.

#### B. Tritium in Flowing Water

Measurements in flowing water can, of course, be made by taking grab samples for batch counting. A continuous flow-system with continuous measurements would be more desirable for many measurement situations. The requirements of such a system are that it be sufficiently sensitive and precise; that it respond in time comparable with the characteristic time for a change in tritium concentration in the flowing water; that it sample properly; and that it not be too expensive nor require excessive maintenance.

Ting and Little (Ref. 31, 32) have reported on a continuous flow system, developed at Beckman Instruments for use in the secondary coolant of a pressurized water reactor. When the device was field-tested, the tritium levels in this coolant at the San Onofre Nuclear Generating Station (Ref. 32) were in the region of 1500 pCi/cm<sup>3</sup> (about half of the MPC), measured with a few-percent precision.

The system works by mixing toluene-based liquid scintillator (or dioxane-based scintillator if very rapid mixing was required) with the liquid water sample in a mixing chamber; the mixed solution flows through a light trap, a liquid scintillation counter, and then out as waste. The difficult part of the system is the mixing chamber. Liquid scintillator flows continuously into a vial; the water sample enters the vial by dripping gravity flow,

and is stirred for thorough mixing twice before exit. Time response is remarkably good: Figure 3 (from Ref. 31) shows the response to 5 tritium samples of 10, 20, 30, 40, and 50  $\mu$ l, injected instantaneously with a scintillator flow rate of 7.0 ml/min. Residence time was less than two minutes in the test, and the total response peak is over in about 4 minutes, as seen in Figure 3. The efficiency for detection of the tritium beta is 32%. The sample flow rate is 0.5 ml/minute, and the minimum detectable tritium concentration is about 25 pCi/ml. This system shows great promise for the continuous monitoring of tritium in flowing water. Its main drawback is the high operating cost from consumption of liquid scintillator: at 7 ml/minute ( $\sim$ 10 liters/day), the annual cost of liquid scintillator can be in the range of \$9000 to \$15,000 (\$3 to \$5 per liter).

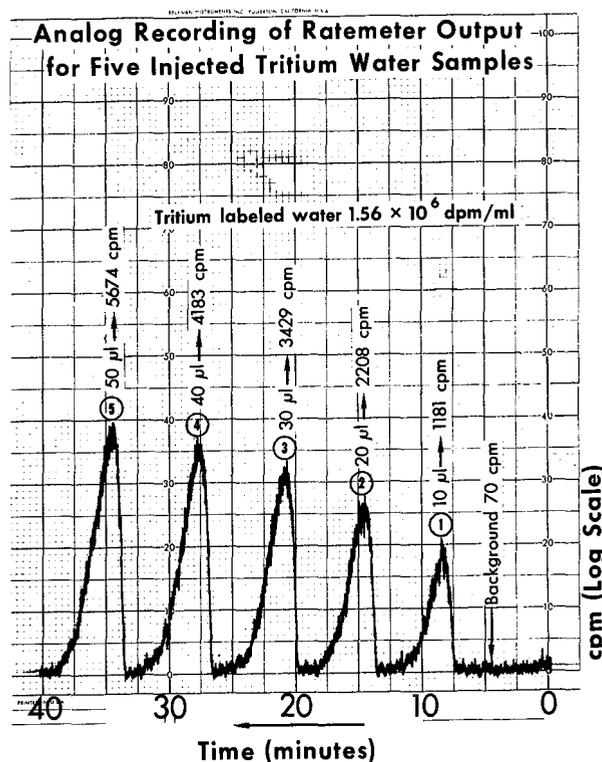


FIGURE 3. Response of the flowmeter described in Ref. 31. Analog recording of ratemeter output for five injected tritium water samples.

#### C. Urinalysis

Tritium measurements in urine are discussed in detail in Volume 4 ("Biomedical") of this Survey.

The most common procedure for the determination of tritium body burden is radioassay of tritium (as HTO) in urine. Chronic exposure of workers to air containing HTO vapor at the



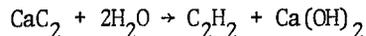
A.E.C. recommended occupational MPC level will produce tritium activity of about 30,000 pCi/cm<sup>3</sup> of urine (Ref. 33, 34).

Levels down to the 10 pCi/cm<sup>3</sup> region can be measured routinely with liquid-scintillation counting (Ref. 35), using an emulsion technique similar to that described earlier (Ref. 25). Thus sensitivity is not a problem for analysis in the laboratory. For rapid measurements, which are sometimes required after possible large accidental exposures, Osborne has developed an automatic liquid-scintillation urine analyzer (Ref. 36). This system provides a measurement two minutes after the sample is introduced, with a minimum sensitivity of about 1000 pCi/cm<sup>3</sup> and measurements up to 10<sup>7</sup> pCi/cm<sup>3</sup> with precisions of about ±10%. This device is particularly useful to assess a possible HTO exposure immediately after an accident.

One difficulty with urinalysis is the possible quenching due to foreign substances in the sample. This is the reason that the *p*-xylene-based emulsion technique is preferred over the homogeneous-solution method (Ref. 35). Alternatively, one can distill the uranium before determining the tritium activity level in any of the conventional dioxane-based systems.

Block, Hodgekins and Barlow (Ref. 37) have developed a proportional counter tech-

nique which can detect 1000 pCi/cm<sup>3</sup> and is relatively simple. The system is based on the reaction of water and calcium carbide to form acetylene as follows:



Tritiated water yields tritiated acetylene (C<sub>2</sub>HT), used as a counting gas in the proportional counter. This system is attractive particularly for its adaptability for use in the field, despite the fact that its sensitivity is much less than that of the liquid scintillation systems.

Sandia Laboratories (Ref. 38) have developed a urinalysis capability using metallic calcium to reduce the water in urine; a sealed ion chamber is filled with the evolved hydrogen. The main advantage of the instrument (the T-449) is an immediate analysis capability for use in emergency accident situations. The calcium cartridge is disposable, and the instrument makes determinations in five minutes, with ±15% accuracy (2σ) down to levels of 10,000 pCi/cm<sup>3</sup>. This instrument is portable (18 lb), and is useful for field use in case large exposures occur in a remote location.

#### D. Solid Scintillation Counting

Although liquid scintillation counting has been the most commonly used technique for tritiated water measurements, plastic scintillation

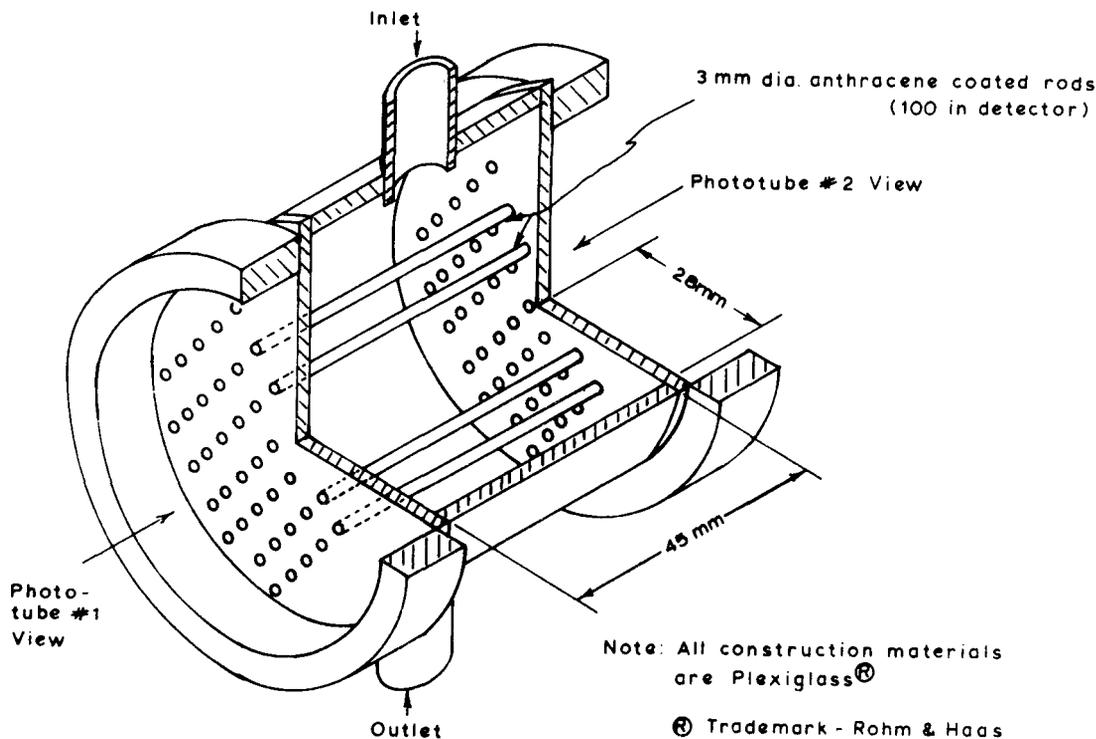


FIGURE 4. Schematic diagram of the scintillation detector (from Ref. 39).



systems are also used. These systems have several drawbacks compared to the liquid-scintillation systems, but two advantages: first, plastic scintillation systems provide faster response than the liquid methods; and second, the liquid system can be expensive due to consumption of large amounts of the organic solvents and solutes, which must be disposed of after use.

One needs a high surface-to-volume ratio, and the early systems used small crystals of anthracene. More recently, various kinds of detectors using thin sheets or layers of anthracene or plastic scintillator spread over lucite (for light piping) have been described. Figure 4 shows one such detector (Ref. 39), in which 100 rods of lucite are coated with anthracene powder. The counter can detect 1000 pCi/ml of water at a rate of 10 cpm. To measure urine, the sample must be decolorized by passage over a charcoal column. The response to tritium in air is 20 cpm for 1 pCi/cm<sup>3</sup> of air. In an unshielded laboratory environment, the normal external background gamma levels were at about 30 cpm.

Osborne (Ref. 40) has developed a plastic detector, shown in Figure 5, with the sensitivity of 24 cpm for 1000 pCi/cm<sup>3</sup> of water. At this counting rate, small fractions of an (MPC)<sub>w</sub> can be detected easily. For tritium in air, about 0.01 pCi/cm<sup>3</sup> can be detected. The background is about 40 cpm.

The main problem with plastic detectors is that over a period of time the plastic surfaces tend to degrade because of sedimentation, bacterial growth, and discoloration. Any of these can produce large decreases in counting efficiency because of the difficulty in light collection through the thin light paths. The detector can sometimes be cleaned, but not always. At present, this technique does not compare favorably with liquid scintillation counting for effluent water monitoring, although when specific activity is high and the water sample is pure it can be used satisfactorily.

### 3. TRITIUM IN AIR

In air, tritium occurs primarily in two forms: as water vapor (HTO) and as hydrogen gas (HT). Tritiated organic compounds in the vapor phase and on particulate matter also occur occasionally.

There are two common approaches to measurements in air: first, one can use an ionization chamber or gas proportional counter with filtered air introduced for internal counting; the tritium betas produce ionization within the sensitive chamber volume. This method is sensitive to tritium in all of its

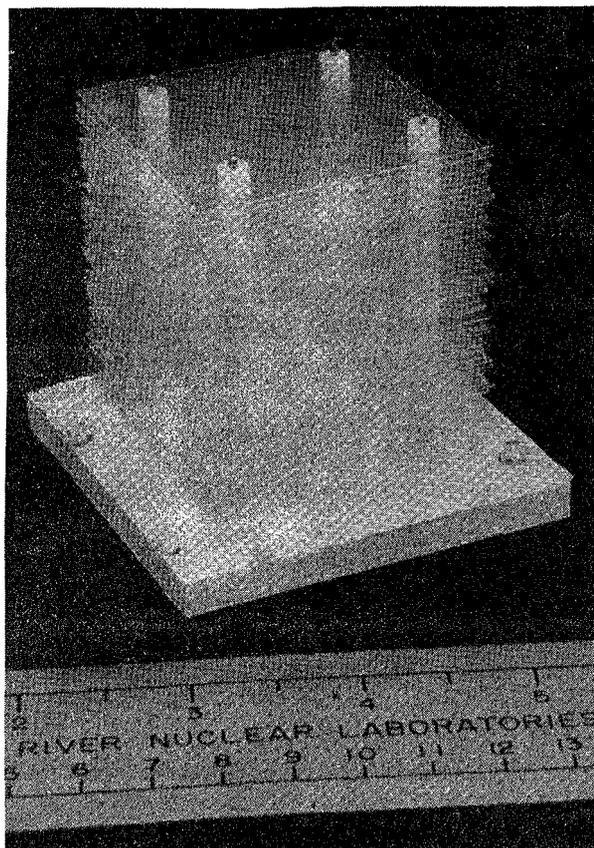
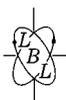


FIGURE 5. Detector for tritium in water. The scintillator sheets fit into a cubic lucite cell. The lucite plate shown here, on which the sheets are mounted, forms one vertical side of the cell (from Ref. 40).

gaseous chemical forms. Secondly, HTO vapor can be removed from the air with a bubbler or with some dessicant such as silica gel or a molecular sieve, and the resulting tritiated water counted using one of the liquid or plastic scintillation techniques. To measure tritium as HT or in tritiated organics, it is possible to burn the gas, converting the tritium to HTO before dessication and counting. We shall discuss each of these methods in turn.

#### A. Ionization Chambers

The earliest tritium detector for stack effluent measurements was the Kanne chamber (Ref. 41, 42, 43), developed in the early 1940's. The gas to be measured is drawn continuously through an electrostatic precipitator or ion trap, then into a large ionization chamber. The resulting current is measured with a picoammeter. Large chambers of volumes 18.5 and 51.5 liters are in common use at Savannah River Laboratory (Ref. 44); they have minimum sensitivity limits ( $2\sigma$ ) of 7 and 2 pCi/cm<sup>3</sup> in air, respectively.



To quote from Ref. 44:

"Although these instruments are sensitive to unusual particulate air loading, external gamma radiation, and other radioactive gases, these problems are reduced by filtering the air intake, by locating the chambers in areas of low gamma background, and by using moisture traps between two chambers in series, respectively."

One important problem with ionization chambers is the background in the presence of external gamma fields. . . . there is no way for a single ion chamber to differentiate between external gammas and internal betas (or internal  $\gamma$ 's or  $\alpha$ 's). The background is such that 1 mR/hr of gammas is about equivalent to 100 pCi/cm<sup>3</sup> of tritium in air (Ref. 45). To reduce this problem, one common procedure is to place two identical chambers side-by-side, one sealed and the other open to the ambient air (Ref. 46). This reduces the interference by a factor of 50 to 100 (Ref. 45). The limitation is in maintaining identical air masses in the two chambers in the presence of fluctuations in ambient pressure, temperature, and humidity.

If the gamma flux is directional, chambers adjacent to one another may not sample the same flux. This problem is reduced in a design by Osborne and Cowper (Ref. 47) in which the compensating chamber is placed inside the active chamber to provide equal response in all directions. A 40-liter chamber of this design has a minimum detectable limit ( $2\sigma$ ) for tritium at levels as low as about 1 pCi/cm<sup>3</sup> of air. Instruments with 0.3 and 1.2 liter volumes have also been constructed (Ref. 48).

Interference by other gaseous radionuclides is also a major problem for ion chambers. Particulates may be eliminated with filters, but radioactive noble gases (e.g., <sup>41</sup>Ar, <sup>85</sup>Kr) are inert and difficult to discriminate against. Each disintegration of <sup>85</sup>Kr produces many times the ionization of a tritium disintegration. This major disadvantage is what makes an ionization chamber inappropriate unless tritium is known to be the only significant radionuclide present. To overcome this, Osborne has reported the development of two chambers identical except that one has "a self-renewing drying system so that continuous subtraction (tritium + noble gases minus noble gases) can be accomplished" (Ref. 49).

Waters (Ref. 50), in a review of general problems with ion chambers, reports that factors such as cigarette smoke, ions, aerosols, humidity, and memory effects can also cause

erroneous readings in ion chambers. He suggests the use of a micron pore size filter followed by an electrostatic precipitator. Also, materials such as rubber and some plastics (Ref. 51) are particularly susceptible to absorption of tritiated water vapor, causing a low reading; later re-emission or exchange with ordinary water vapor then produces an error in subsequent measurements. Waters reports that: "Teflon is satisfactory but is difficult to use because of its tendency to cold-flow under pressure."

For monitoring tritium in air, the ionization chamber is undoubtedly the most commonly used instrument. Portable flow-through instruments of various sizes (typically with about 1 liter chamber volumes) have been a mainstay in health physics despite the problems discussed.

Perhaps the most sophisticated instrument of this type is the Sandia T-446 (Ref. 38), now available from Bendix Corporation (Ref. 52). This instrument is expensive (over \$5000) but provides a collection of features which makes it almost ideal for monitoring in those applications where accident-warning instrumentation is needed. The ion chamber output current is fed to a vibrating reed electrometer, then amplified and measured. All electronics are solid-state, on printed circuit boards. It is sensitive down to about 5 pCi/cm<sup>3</sup> of air and up to 10  $\mu$ Ci/cm<sup>3</sup> in seven decade ranges with automatic range changing. It employs an electrostatic precipitator, and has a 200  $\mu$ Ci <sup>63</sup>Ni source plated to one electrode of the precipitator. Ionization created by the source is used for self-contained calibration when the precipitator is turned off. There are two adjustable alarm levels which can be used to distinguish between immediate hazards and levels in the region of a few occupational (MPC)<sub>a</sub>. The instrument is either portable (batteries) or alternatively the detector can be used in remote-sensing situations; it has a fail-safe failure indicator, automatic range changing, a filter to eliminate ionizing aerosols, and other features. The T-446's principal disadvantages besides the high cost are a sensitivity to external gamma fields and radioactive gases which is common to all ion chambers.

## B. Proportional Counters

The gas proportional counter differs from the ionization chamber mainly in two respects: it is a *pulse* counting instrument, and it requires a particular filler gas to which the sample of (tritiated) air is added. It offers an advantage over the ion chamber in that *energy discrimination* is possible by electronic discriminating on the output pulse height, whose size is proportional to the energy deposited. The essential disadvantage of the gas proportional counter is the continual consumption of counting gas, which is discarded



after being mixed with the tritiated sample and counted.

Driver (Ref. 53) built a tritium counter of this type as early as 1956. More recently, Ehret (Ref. 54) has achieved a minimum detectable concentration ( $2\sigma$ ) of 0.1 pCi/cm<sup>3</sup> of air, using a chamber of about 1 liter volume with methane as the counting gas, to which is added 20 to 30% air.

"Air is pumped through the counter by means of a small membrane pump. At the counter inlet methane is added to the air within a small mixing chamber" (Ref. 54).

In this design, there must be a compromise between speed of response and inordinate consumption of methane. Using a methane flow rate of 25 liters/hour, a response time of a few minutes can be achieved. Ehret's design also includes an anti-coincidence proportional counter surrounding the tritium-sensitive counter, to discriminate against the longer-range betas of <sup>41</sup>Ar and <sup>85</sup>Kr. This counter suffered from "memory effects" due to absorption of tritiated water vapor (HTO) in plastic tubes, an effect mentioned earlier. Use of Teflon should eliminate this problem.

Block, Hodgekins and Barlow (Ref. 37) have built a large-area, thin window gas-flow proportional counter system: two counters are identical except that one has an aluminized mylar window opaque to tritium betas while the other has a Formvar window thin enough to allow detection of tritium. This system can thus compensate not only for gamma background but also for the beta activities of <sup>41</sup>Ar and <sup>85</sup>Kr. The window is porous to the counting gas (propane), and the flow occurs because of a positive pressure inside the counter. The counting volume is 0.2 liters. The window is permeable to T<sub>2</sub>O and HTO (but not HT or T<sub>2</sub>). In a gamma flux of 3 mR/hr, the counter had a minimum sensitivity in the region of about 1 pCi/cm<sup>3</sup>.

#### C. Silica Gel/Liquid Scintillation

The use of silica gel as a desiccant to remove moisture (H<sub>2</sub>O, HTO) from air before liquid scintillation counting is a common technique for tritium determinations in air. In fact, it has been proposed as a "Tentative Method for Analysis for Tritium Content of the Atmosphere" by the Intersociety Committee for a Manual of Methods for Ambient Air Sampling and Analysis (Ref. 55). Its main advantage is its extremely high sensitivity.

In the Intersociety Committee procedure, a 12" x 1.25" diam. aluminum cylinder is filled with silica gel (180 g). Silica gel can absorb moisture up to 40% of its own

weight. Air is pumped at about 100 to 150 cm<sup>3</sup>/minute through the silica gel column, which collects essentially all of the moisture. With this much gel, a continuous sample can be taken over as long as a two-week period; of course, a smaller set-up can be used for shorter sample-collection times.

Following the sample collection, the silica gel is heated in a distilling flask to remove the moisture. The distillate is then counted using the liquid scintillation technique of Moghissi et al. (Ref. 24). A sample of air at 30°C (86°F) and 100% relative humidity will yield 6600 pCi/cm<sup>3</sup> of moisture when tritium (as HTO) is present at the (MPC)<sub>a</sub> concentration of 0.2 pCi/cm<sup>3</sup> for individuals in the general public (Ref. 55). Also, the system has negligible interference from other radioactive nuclides, because they do not absorb on silica gel and further are eliminated in the distillation process. The largest source of systematic error is the uncertainty in the volume of air sampled and the prevailing temperature, pressure, and relative humidity of the air. Also, tritium as HT gas is completely excluded by this method.

Osloond et al. (Ref. 56) have reported a detection limit ( $2\sigma$ ) of  $2 \times 10^{-5}$  pCi/cm<sup>3</sup> of air by counting the silica gel directly (without distillation) in the liquid scintillation system.

The chief drawback of this system is the fact that concentrations are integrated over a long time period (hours to days), and that each batch analysis requires a separate procedure in the laboratory. The advantages are that its sensitivity and precision are excellent, and it suffers from almost no interferences. Another drawback is that the desiccant must be renewed periodically, and this must be done before it becomes saturated.

#### D. Bubbler Exchange

Osborne (Ref. 45) has removed tritiated water vapor (HTO) from air using a water-filled gas washing bottle ("bubbler"). Moist air is bubbled through water, and exchange occurs between the vapor and liquid stages. It is easy to sample a known volume of air so that the overall collection efficiency is within a few percent of 100%. For example, Osborne cites sampling for 10 hours at 10 cm<sup>3</sup>/sec (or for 1 hour at 100 cm<sup>3</sup>/sec) through 200 cm<sup>3</sup> of water with an overall collection efficiency of 96-97%. One advantage of this method is that tritiated hydrogen gas (HT) is selectively eliminated: in a bubbler system studied at Los Alamos, the collection efficiency for HT was <0.1% under conditions in which HTO collection efficiency was 90 to 95% (Ref. 57). Also selectively eliminated are noble gases: in Osborne's example (Ref. 45), "the ratio of noble gas activity to tritium activity in the water would



be  $<0.013\%$  of the average of the ratio in the sampled air". A typical bubbler system is somewhat less sensitive than the desiccant approach, with minimum detectable activity being in the region of  $0.1 \text{ pCi/cm}^3$ ; however, this sensitivity may be acceptable in many situations, given the convenience of the method.

Osborne (Ref. 45) has also described a method in which the bubbler can be tied onto a tritium-in-water liquid scintillation detector for continuous monitoring. Many elements of this system would then be very similar to that described above in the discussion on "Tritium in Flowing Water".

#### E. Combustion for HT Determinations

Techniques have been developed for the separate determination of tritium in air as HTO (water vapor) and HT gas. We shall discuss two different techniques here.

Griffin et al. (Ref. 58) have determined HTO and HT separately, using a technique whose flow train is shown in Figure 6. Because HT (or  $\text{H}_2$ ) in the gas to be measured is present at very small levels, tritium-free  $\text{H}_2$  gas is added as a carrier. The moisture (including HTO) is then removed in Air Dryer #1, consisting of Drierite-brand anhydrous calcium sulfate in plexiglass cylinders (Ref. 59). The dry gas is combusted in a catalytic burner: all combustible compounds

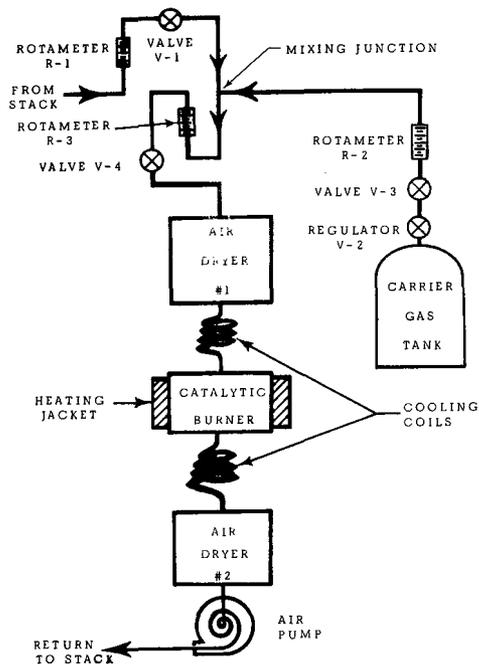


FIGURE 6. Sampling train for total airborne tritium (from Ref. 58).

are oxidized with the help of a 0.5% platinum on  $1/8$ " diameter aluminum oxide pellets, at a temperature of  $400^\circ\text{C}$  maintained by a heating jacket. Not only HT, but also tritiated organics (such as  $\text{CH}_3\text{T}$ , etc.), are converted to HTO in this step. The HTO is then removed from the flow by Air Dryer #2. Both HTO (Drierite) samples are analyzed by liquid scintillation counting after the Drierite is dehydrated and the water vapor distilled under vacuum. Griffin has indicated (Ref. 60) that a palladium catalyst, operating at lower temperature, would be much easier to use than his platinum catalyst. Levels of tritium (in stack gas) in the  $0.2$  to  $2.0 \text{ pCi/cm}^3$  region were measured with accuracies of  $\pm 8.5\%$ .

Ostlund (Ref. 61) has developed another technique. His procedure also converts HT and other tritiated compounds into HTO (water vapor) by a catalytic combustion process; palladium metal carried on a molecular sieve is used to combust hydrogen in air to better than 98%. The resulting water is absorbed in the same catalyst. Sampling times as short as 10 minutes can be used to measure tritium levels in the  $1 \text{ pCi/cm}^3$  region, with errors in the 3 to 5% region.

One difference between the two techniques is that Ostlund's is better adapted to single measurements, while Griffin's is designed for continuous, long-term sampling followed by batch analysis of the integrated activity. Also, because in Ostlund's technique the catalyst base molecular sieve is also the desiccant, there is not a problem with contamination and "memory" which might exist in the other technique. Either technique can perform independent measurements of HTO and HT in air, a significant advantage.

#### 4. ENRICHMENT TECHNIQUES

Most measurements of tritium in environmental samples ordinarily do not require an enrichment step prior to counting. However, techniques for tritium enrichment have been developed for those special cases in which tritium levels are extremely small. One example is in geological research, where studies have been made of underground natural waters whose ages are so great (many thousands of years) that tritium levels are minute ( $\leq 10^{-3} \text{ pCi/cm}^3$ ). Here we will not attempt to do more than summarize some of the important work in the field of enrichment.

Among the procedures used to enrich tritium in water samples are electrolysis, thermal diffusion, and chromatography. Hoy (Ref. 62) has developed a chromatographic system in which a palladium column is heated; tritium diffuses through the column faster than does normal hydrogen, and the first gas evolved is



tritium-enriched. Tritium recovery of about 60% was obtained in the first 500 ml of gas evolved, and with a 30-minute counting period a T:H ratio of  $10^{-17}$  could be measured with  $\pm 41\%$  ( $1\sigma$ ) error. Perschke (Ref. 63) was able to enrich tritium up to 40 times with 100% tritium recovery, using gas solid chromatography on metal sieves.

Another approach is electrolysis. When water is electrolysed, hydrogen is given off more readily than tritium; the relative loss rate can be as high as 35:1, varying with electrode material, surface conditions, electrolyte, and temperature. Hartley (Ref. 64) has done a detailed analysis of the parameters involved in electrolytic procedures. Ostlund (Ref. 65) has achieved enrichments in the  $10^2$  region by a series of successive electrolytic stages.

Thermal diffusion is another method used for tritium-hydrogen separation (Ref. 66) but it is complicated and time-consuming.

For further information on enrichment techniques, the interested reader is referred to the references cited.

## 5. SUMMARY AND CONCLUSIONS

Three important numbers determine the sensitivity required of instruments for measuring tritium in environmental media. These are the "Maximum Permissible Concentrations" (MPC) for an individual in the general population (Ref. 9):

- 3000 pCi/cm<sup>3</sup> (as HTO in water)
- 40 pCi/cm<sup>3</sup> (as HT gas in air)
- 0.2 pCi/cm<sup>3</sup> (as HTO vapor in air).

The ideal instrument is one capable of measuring down to fractions of MPC levels in the appropriate medium. There are two main classes of measurements in each medium, that is laboratory and field measurements. The requirements are quite different in the two cases, since field instruments generally must be more rugged, require less maintenance, be less sensitive to temperature and humidity effects, and have some method of automatic readout (whether the measurements are continuous or batched). We shall summarize the situation in each of the various measurement areas separately:

a) *Laboratory measurements of HTO in water.* The liquid scintillation technique is the established method for this task. It is extremely sensitive (sensitivities below 1 pCi/cm<sup>3</sup> of water), has excellent reproducibility, suffers from few interferences (especially if distillation precedes counting), and is now commercially available in a variety

of sophisticated, automatic systems. The disadvantages of liquid scintillation counting include both the high capital cost of equipment and high operating costs. Plastic scintillator systems are less expensive and have faster time response, but this is at the expense of perhaps two orders of magnitude in sensitivity. An important disadvantage of these plastic techniques is that the surface tends to degrade in time.

b) *Urinalysis.* This is now a well-developed technique, using liquid scintillation counting methods similar to those for HTO measurements in water. Automatic analysis equipment can reach sensitivities in the range of 10 pCi/cm<sup>3</sup> of urine. A proportional counter technique sensitive in the 1000 pCi/cm<sup>3</sup> range has also been developed for portable field use.

c) *Field measurements of HTO in (flowing) water.* A system for continuous measurements using a flow/mixing system and liquid scintillation counting has recently been developed. Refinements of this technique seem capable of ultimately providing reliable, sensitive in-stream monitoring capability for routine reactor effluent monitoring. Present sensitivities in the 25 pCi/cm<sup>3</sup> range seem adequate for effluent measurements, but can only be improved at present by using more scintillator at higher cost.

d) *Continuous field measurements of total tritium in air.* Flow-through ionization chambers and proportional counters have been a mainstay for tritium-in-air monitoring for nearly 25 years. Several commercially-available instruments now exist. The ion chamber suffers from various types of interferences, most notably from external gamma fields; however, well-developed compensation systems now seem capable of measuring down to about 1 pCi/cm<sup>3</sup> in the presence of gamma fluxes in the few-mrad/hour range. Ion chambers also cannot distinguish tritium from other gaseous emitters (e.g., <sup>41</sup>Ar, <sup>85</sup>Kr), and are sensitive to cigarette smoke, ions, and aerosols unless suitably protected (this protection is now available in some commercial instruments). Gas proportional counters can partially discriminate against other radionuclides, and are less sensitive to aerosols. Their sensitivity (in the 1 pCi/cm<sup>3</sup> range) is comparable to that of the ion chambers. To detect levels much below about 1 pCi/cm<sup>3</sup> in air, it is necessary to remove tritiated water vapor from the air for liquid scintillation counting, or use combustion techniques.

e) *HTO in air by removal & liquid scintillation counting.* There are two well-developed methods for HTO (vapor in air) determinations at very low levels: the use of silica gel, and bubblers, for removal of vapor. In each



case, liquid scintillation counting measures HTO in the resulting water. With silica gel, sensitivities well below  $0.01 \text{ pCi/cm}^3$  of air are achievable; with bubblers the sensitivities are typically in the region of about  $0.1 \text{ pCi/cm}^3$ . These are both integrating techniques, capable of field sampling, but the analysis is by batch counting in the laboratory. Interferences are few, and precision is high.

f) *HT in air (combustion)*. Combustion techniques followed by HTO counting can determine tritium as tritiated hydrogen gas (HT) at the level of a fraction of one  $\text{pCi/cm}^3$  of air. If HTO vapor is removed by dessication first, separate HT and HTO measurements can be made.

#### CONCLUSIONS

At present, measurements at levels well below MPC values for individuals in the general public are possible in each of the important measurement situations. In particular, laboratory liquid-scintillation analyses of batch

water samples (HTO in water, HTO vapor extracted from air) are extremely sensitive and suffer few interferences. Unfortunately, these measurements are time-consuming and expensive. These same comments apply to the HTO-in-water system for in-stream effluent measurements.

In air samples, continuous measurements with ion chambers and proportional counters are only sensitive at about the  $1 \text{ pCi/cm}^3$  level, which is well above the MPC value of  $0.2 \text{ pCi/cm}^3$  for general public exposure. More sensitive measurements in air require vapor removal and laboratory counting.

Clearly, advances are required to perform continuous, low-level monitoring of tritium in air. Also, cost-reduction in liquid (HTO) measurements is very desirable. Finally, the ruggedness, reliability, and maintainability of many present field instruments is of the highest priority.

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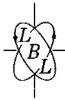


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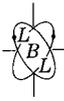
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KRYPTON-85

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2. Radiation Protection Guide
3. Measurement Considerations
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    - (i) Plastic Scintillator
    - (ii) Liquid Scintillator
  - b) Field Instruments
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## 1. INTRODUCTION

Krypton-85 is one of the main radioactive fission products normally released to the atmosphere in the reprocessing of nuclear reactor fuels. In fuel ready for reprocessing (150 days of cool-down) it is present in the amount of about 11,200 curies/metric ton of fuel (Ref. 1). In all reprocessing plants presently operating or under construction, the entire  $^{85}\text{Kr}$  content of the fuel is released to the atmosphere. Since these plants operate at a through-put of from one to six tons per day,  $^{85}\text{Kr}$  is a potential environmental pollutant, requiring measurement both in the plant stack during release and in the environment after dispersion. With the rise in nuclear power production, the world-wide  $^{85}\text{Kr}$  inventory is expected to increase dramatically. One projection is shown in Figure 1 (From Ref. 1).

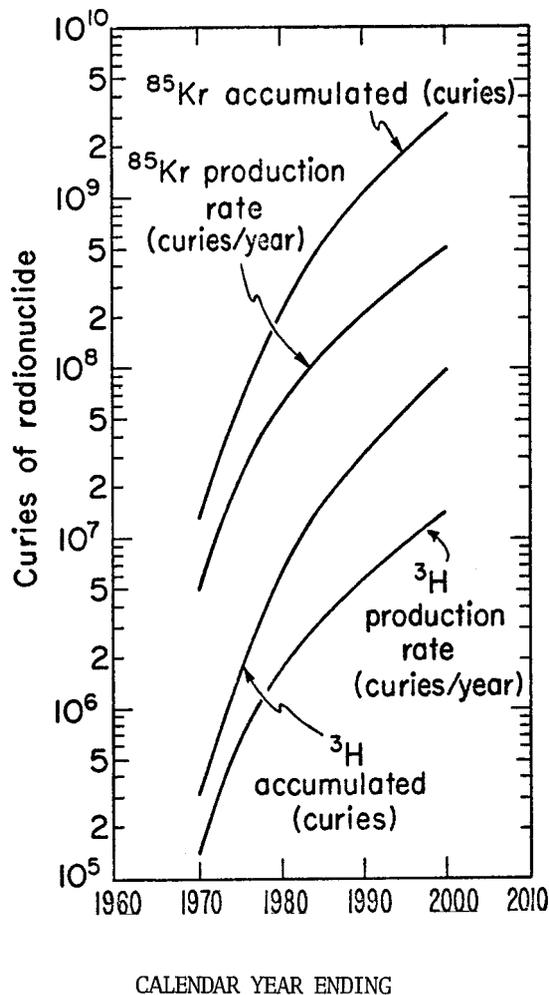


Figure 1. Estimated Production of Krypton-85 and  $^3\text{H}$  (Tritium) from the Nuclear Power Industry of the Free World (from Ref. 1).

A recent study (Ref. 2) indicates that present world-wide dose to man averages about  $10^{-3}$  mrad/year to the bare skin, and about  $10^{-4}$  mrad/year to the whole body. Figure 1 can be used to determine approximately how these doses might increase in the next few decades.

Krypton is a noble gas, chemically inert, which is present (as stable Krypton-78, 80, 82, 84, 86) in normal air at a concentration of 1.14 ppm = 1.14 ml/m<sup>3</sup> (Ref. 3). The radioactive decay of  $^{85}\text{Kr}$  has a half-life of 10.76 years. The principal decay mode is  $\beta^-$  emission, with a maximum beta energy of 672 keV. The beta has a broad energy spectrum shown in Figure 2 (from Ref. 4). The range for the maximum-energy beta is 0.3 g/cm<sup>2</sup> (3 mm in water, 250 cm in air). A small (0.41%) branching ratio also exists for emission of a 514-keV gamma.

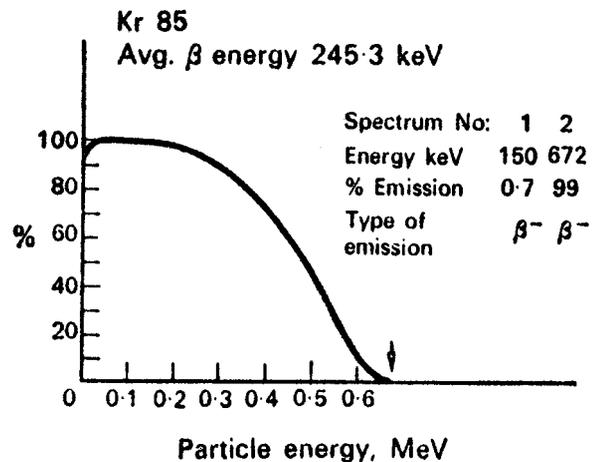


Figure 2.

## 2. RADIATION PROTECTION GUIDE

For 168-hour occupational exposures, the International Commission on Radiological Protection (Ref. 5) has established a maximum permissible concentration in air, (MPC). This occupational limit is  $3 \times 10^6$  picocuries per cubic meter (pCi/m<sup>3</sup>). For individuals in the general public, the applicable MPC values are a factor of 10 smaller ( $300,000$  pCi/m<sup>3</sup>) and for exposure to a suitably large sample of the general public, another factor of 3 smaller still ( $100,000$  pCi/m<sup>3</sup>). The U.S. Atomic Energy Commission's MPC values agree exactly with the ICRP MPC's just quoted (Ref. 6). Throughout this discussion, the MPC will be taken as  $300,000$  pCi/m<sup>3</sup>, since exposure to a few individuals in the general public at the 300,000 level is undoubtedly more likely than to a larger population at the 100,000 level.



This MPC is equivalent to fractional krypton-85 atomic concentrations of  $2.0 \times 10^{-7}$  ( $^{85}\text{Kr}/\text{total Kr}$ ) or  $2.3 \times 10^{-11}$  ( $^{85}\text{Kr}/\text{total air}$ ). At the fence around a typical fuel reprocessing plant, concentrations of a few percent of MPC are expected to occur periodically.

### 3. MEASUREMENT CONSIDERATIONS

In environmental samples, the measurement of  $^{85}\text{Kr}$  is essentially always based upon detecting the decay  $\beta^-$ , with any of a number of detectors sensitive to ionization energy loss. Several detection systems exist, including liquid scintillators, plastic scintillators, Geiger-Muller counters and ionization chambers. Each of the methods has its advantages and disadvantages; the discussion below is intended to point out some of the factors involved.

We shall concentrate upon measurements in the air around nuclear reactors and their fuel-reprocessing plants. Here the principal beta emitter besides  $^{85}\text{Kr}$  is tritium ( $^3\text{H}$ ); however, the tritium  $\beta^-$  has such a low energy ( $E_{\text{max}} = 18.6 \text{ keV}$ ) that its interference in the  $^{85}\text{Kr}$  measurement is usually not significant. (*The reverse is not true, as discussed in the tritium section elsewhere in this volume.*)

There are two quite different classes of  $^{85}\text{Kr}$  measurement systems: those which operate in the field, designed to run unattended except for periodic maintenance; and those which require laboratory procedures. The latter are invariably the more sensitive and the more precise. We shall discuss these two classes separately.

#### a) Laboratory Techniques

There are a number of  $^{85}\text{Kr}$  techniques which have been developed for use in the laboratory. They all involve one substantial handicap, namely, that the volume of gas which can be sampled at some remote location and then transported to the laboratory for analysis is not large. This disadvantage limits sensitivity, but is counterbalanced by the ability to concentrate the krypton before counting and to perform other analytical procedures which are impossible to carry out in the field.

The two most common laboratory techniques use plastic-scintillator shavings contained within a gas-tight vial, and liquid scintillator as a solvent for the  $^{85}\text{Kr}$ -containing gas.

##### (i) Plastic Scintillator

We shall discuss the technique of Sax, Denny, and Reeves (Ref. 3), which is typical of the plastic scintillator methods.

Plastic scintillator shavings of 20-40 mesh (an inexpensive by-product from any

of several scintillator manufacturers) are encased in a glass vial of volume about 4 ml. A sample of pure krypton gas is prepared using vacuum-cryogenic techniques. The vial is evacuated and then filled to a known pressure (near 600 torr) with the krypton samples. A scintillation counter then measures the pulse-height spectrum.

Calibration of the counting efficiency (the fraction of the  $\beta$  spectrum above the counting threshold) is determined using a known  $^{85}\text{Kr}$  standard acquired from NBS (Ref. 7). The counting efficiency is about 95%. Given about 0.5 ml of nearly pure krypton (equivalent to that present in about  $0.5 \text{ m}^3$  of air), measurements can be made of normal  $^{85}\text{Kr}$  background levels in the  $10 \text{ pCi}/\text{m}^3$  range to about  $\pm 10\%$  (Ref. 3). The limit of detection is about  $1 \text{ pCi}/\text{m}^3$  of air sample, with a 100-minute counting time, at the two-standard-deviation level above background.

##### (ii) Liquid Scintillator

A general description of liquid scintillation counting can be found in the section on "Tritium" elsewhere in this volume. Here we shall discuss that aspect of liquid-scintillation counting which is unique to or particularly relevant to  $^{85}\text{Kr}$ .

The liquid scintillator method relies upon the high solubility of krypton in many of the commonly used liquid scintillators. Krypton is soluble (Ref. 8, 9) in aromatic solvents to  $\sim (1 \text{ ml Kr})/(\text{ml solvent})$ .

The earliest descriptions of liquid scintillators used for  $^{85}\text{Kr}$  counting are those of Horrocks and Studier (Ref. 10), and Curtis, Ness, and Bentz (Ref. 11). Those early methods suffered from a limit on the amount of krypton which could be introduced into the solution.

Shuping, Phillips, and Moghissi (Ref. 9) have reported a method in which about 25 ml of toluene-based liquid scintillator acts as solvent for about 10 ml of gas. The gas is introduced into an evacuated glass vial, filled with deaerated scintillation solution. "If the  $^{85}\text{Kr}$  concentration in air is sufficiently high, the sample may be introduced directly into the solution and successfully counted" (Ref. 9). More commonly, krypton is concentrated cryogenically. A rate of 1 cpm above background corresponds to  $0.025 \text{ pCi } ^{85}\text{Kr}/\text{m}^3$  of air, so that levels of that order of magnitude are detectable with the method. Accuracies in the  $\pm 4\%$  range are achieved, when  $^{85}\text{Kr}$  activities of  $\sim 10 \text{ pCi}/\text{m}^3$  of air are present.

One of the more difficult problems in the procedure just described is the separation of krypton gas from the main air sample. To overcome this problem, a separation system for krypton, by Cummings, Shearin, and Porter



(Ref. 12) employs  $^{83m}\text{Kr}$  as a spike in the air sample, in order to determine directly the  $^{85}\text{Kr}$  yield after a complicated separation procedure involving "charcoal and molecular sieve cold traps, calcium sulfate, ascarite, and a titanium furnace (900°C) for the removal and separation of other air constituents from krypton" (Ref. 12). A description of a method for generating  $^{83m}\text{Kr}$  (2-hour half-life) from  $^{83}\text{Rb}$  in the laboratory has been given by Moghissi and Hupf (Ref. 13).

Another  $^{85}\text{Kr}$  system, described by Stevenson and Johns (Ref. 14), uses a battery-operated air compressor to collect 1 m<sup>3</sup> of air in the field. After a series of cryogenic absorptions and elutions in the lab, the krypton is dissolved and counted in a liquid scintillator. The krypton recovery ranges from 50 to 70%, measured volumetrically. These workers report a minimum detectable sensitivity (three standard deviations above background) of about 2 pCi  $^{85}\text{Kr}/\text{m}^3$  of air, with a 4-hour counting time.

#### b) Field Instruments

For measurements in the field, the ideal goal is an instrument which can record continuously, with reasonably short time-integration periods, at sensitivity levels well below the current average background of about 10 pCi/m<sup>3</sup> of air. The ideal instrument must also be rugged enough to withstand temperature and humidity extremes and shock; and should require little maintenance.

A study (Ref. 15) of several possible field instruments was carried out in 1969 by the Northeastern Radiological Health Laboratory, U.S. Public Health Service. Four ionization chambers and four Geiger-Muller counters were studied. Calibrations were performed in the laboratory, followed by determinations of the minimum detectable concentrations. Some of the instruments were then used in the field around a fuel-reprocessing plant, to determine performances under actual field conditions.

None of the field instruments came close to the sensitivity required for background measurements in the pCi/m<sup>3</sup> range. However, all were sensitive enough to measure fractions of the MPC for individuals in the general public (300,000 pCi/m<sup>3</sup>).

##### (i) Flow-through Ionization Chambers

The four flow-through ionization chambers were tested in the laboratory using dry, radon-free air. They ranged in size from 0.5 to 4.3 liters in volume. After calibration against an NBS standard (Ref. 7), the minimum detectable concentrations (MDC, defined in Reference 15 as twice the standard deviation in counting) were determined: they ranged from 130,000 to 40,000 pCi/m<sup>3</sup>. The calibrations were performed with errors (2σ) in the ±7% to

±17% range. These ionization chambers have several undesirable properties: first, they must be used downstream of a radon holdup trap and filter (or equivalent). Second, because of the use of the radon trap, the averaging time is in the 30-minute range, which is much longer than the few-minute time for a significant change in  $^{85}\text{Kr}$  concentration when a fuel-reprocessing-plant plume passes directly overhead. Third, it is necessary to make appropriate measurements of pressure, temperature, humidity, and external background variation to obtain reasonable accuracy. Finally, maintenance requirements are high. To quote from Reference 15:

"It should be noted that although the ionization chamber systems can be made to operate in the field, the degree of care, number of precautions, and amount of operator training required to obtain usable data for environmental levels, and the questionable nature of the data obtained when the field levels passing the instrument are fluctuating rapidly, all weight heavily in favor of using simpler systems for this purpose."

##### (ii) Geiger-Muller Counters

In Ref. 15, several Geiger-Muller (G-M) tubes were also studied. These were calibrated against the flow-through ionization chambers described above, and then minimum detectable concentrations were determined. The calibration errors (2σ) were in the ±13% region for all of the G-M detectors. The most sensitive G-M detector was found to be the Eon 8008H, a double end-window pancake detector, of 2" diameter with 3.5 mg/cm<sup>2</sup> mica windows. It had an MDC of 12,000 pCi/m<sup>3</sup> with a counting time of 10 minutes (sample) and 10 minutes (background). Unfortunately, this detector was so fragile that its use in the field was precluded.

The other G-M detectors all had MDC's in the region of about 25,000 pCi/m<sup>3</sup>. These instruments were an Amperex 18546 and an Eon 8001T (both one-window pancake detectors of 2" diameter); and an LND 719 and an Eon 5108E (both cylindrical probes, 5-1/2" long, 0.6" diameter, 30 mg/cm<sup>2</sup> wall thickness). The cylindrical instruments were found to be durable against mild shock and rain in the field; the single-window detectors were less durable against shock. However, all performed in the field with sensitivities nearly identical to those measured in the laboratory. The conclusion can be taken directly from Ref. 15:

"Long-term (weeks to months) environmental monitoring would demand the most sensitive detectors and systems that could endure the



environment with a minimum of attention. The choice at present is between the single windowed pancake tube (Amperex 18546) and the thicker walled (30mg/cm<sup>2</sup>) cylindrical probes (Eon 5108E or LND 719) which, though slightly less sensitive, are certainly the most durable of the detectors evaluated."

#### 4. SUMMARY AND CONCLUSIONS

Two important numbers determine the sensitivity required of instruments for measuring <sup>85</sup>Kr in environmental air:

1. the maximum permissible concentration for individuals in the general public is 300,000 pCi/m<sup>3</sup> (Ref. 5, 6).
2. the typical "background" level of <sup>85</sup>Kr in air today is in the range 10 to 15 pCi/m<sup>3</sup> (Ref. 3).

The ideal instrument is one capable of measuring the background level to a small fractional accuracy; several of the laboratory techniques described have this sensitivity. These techniques, which use liquid or plastic scintillation counting, all suffer from a common problem: a complicated series of laboratory steps is required to concentrate the krypton from the original gas sample.

In contrast, the field monitoring instruments (Ref. 15) are all several orders of magnitude less sensitive, with MDC's ranging from 12,000 to 28,000 pCi/m<sup>3</sup> (for the G-M detectors) and from 39,000 to 190,000 (for the ion chambers). All of these instruments are thus capable of detecting <sup>85</sup>Kr at levels well below the 300,000 pCi/m<sup>3</sup> MPC for individuals in the general public, but cannot measure <sup>85</sup>Kr in "background" samples.

At present, there does not exist (to our knowledge) a commercially available *Krypton-85 Monitor* which can be purchased off-the-shelf. The G-M detectors described in Ref. 15 are worthy of commercial development, to meet the need for monitoring in the field around fuel-reprocessing plants and nuclear power reactors. Any commercial instrument would have to satisfy several requirements besides sensitivity:

- (1) Rugged construction, insensitivity to changes in temperature and humidity.
- (2) Some method of continuous recording of the data: at least a strip-recorder, or preferably a direct magnetic-tape record.
- (3) Some method of calibration: perhaps a frequent, rapid check

with a high energy source, and a less-frequent check in a controlled chamber of known <sup>85</sup>Kr concentration.

For the more sensitive measurements of levels below the natural background of a few pCi/m<sup>3</sup>, field instruments are presently out of the question. Some kind of cryogenic concentration method is required to increase the specific activity to the point where reasonable counting times (λ a few hours) are possible.

One key problem here is with the accuracy and reproducibility of the concentration mechanism. Accurate determinations of the fractional yield through a multi-stage sequence are notoriously difficult. This is true especially if a known volume of "ordinary" krypton is introduced as a carrier, since all krypton today unfortunately contains <sup>85</sup>Kr at levels in the ppm region. One solution is the use of <sup>83m</sup>Kr as a tracer to determine the yield.

There are a few technological developments on the horizon which will provide some increased sensitivity. The most important of these is the recent development of substantially better photomultiplier tubes (Ref. 16). These tubes have higher quantum efficiency, lower noise, and generally superior operating characteristics. This will improve the scintillation counting method considerably.

The incorporation of integrated circuit electronics into the G-M detector systems will also be important, helping to increase the ruggedness and reliability of the technique. Also, read-out systems are being developed which will enable the data to be collected, recorded, and analyzed remotely, bypassing the strip-chart-recorder step entirely.

Finally, it should be noted that some of the instruments designed to measure tritium in environmental samples are also capable (or adaptable) for the <sup>85</sup>Kr problem, with some modifications and changes in procedure. The reader is referred to the "Tritium" section elsewhere in this volume for details.



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STRONTIUM-90 AND -89

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## 1. INTRODUCTION

There are only two radioactive isotopes of strontium of significance in radiological measurements in the environment: strontium-89 and strontium-90.

*Strontium-90* is the more significant from the point of view of environmental impact. This is due mostly to its long half-life (28.1 years). It is a pure beta emitter with only one decay mode, leading to yttrium-90 by emission of a negative beta with  $E_{\max} = 546$  keV. Subsequently, the  $^{90}\text{Y}$  daughter nucleus almost always decays by emitting a  $\beta^-$  ( $E_{\max} = 2.27$  MeV) with a 64 hour half-life, leading to stable zirconium-90. The beta energy spectra are shown in Figure 1 and Figure 2 (from Ref. 1).

*Strontium-89* has a 52 day half-life, much shorter than that of  $^{90}\text{Sr}$  but still quite long in its own right. It decays by nearly pure  $\beta^-$  emission to stable yttrium-89, the maximum beta energy being 1.463 MeV.

The main radiological impact of radiostrontium in man is due to its biochemical resemblance to calcium. When ingested, some of the strontium becomes lodged in the bones, where it remains for a long time (almost permanently). Its subsequent decay produces radiation dose directly at the site (the 2.27 MeV highest-energy beta of the daughter  $^{90}\text{Y}$  has a maximum range of only about 1.0 g/cm<sup>2</sup>). Because of this, the recommended maximum permissible concentrations (MPC) in air and water are quite small. For occupational exposure, the ICRP (Ref. 2) has recommended the following MPC's on a 168-hour-per-week basis for soluble strontium:

Isotope	168-hour-week Occupational MPC in air (pCi/m <sup>3</sup> )	168-hour-week Occupational MPC in water (pCi/liter)
Strontium-90	100	1,000
Strontium-89	10,000	100,000

The maximum permissible body burdens (MPBB) are 2  $\mu\text{Ci}$  (for  $^{90}\text{Sr}$ ) and 4  $\mu\text{Ci}$  (for  $^{89}\text{Sr}$ ), both for burden in the bone (Ref. 2). For individuals of the general public, the MPC and MPBB levels are a factor of 10 smaller; and for a suitable average sample of a large general population, another factor of 3 smaller still.

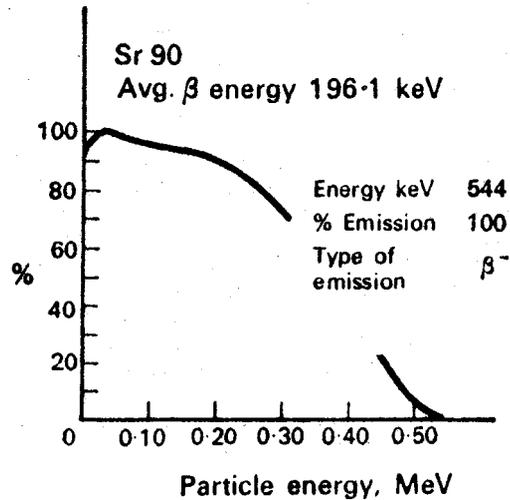


FIGURE 1. Spectrum of beta energies from decay of strontium-90 (from Ref. 1).

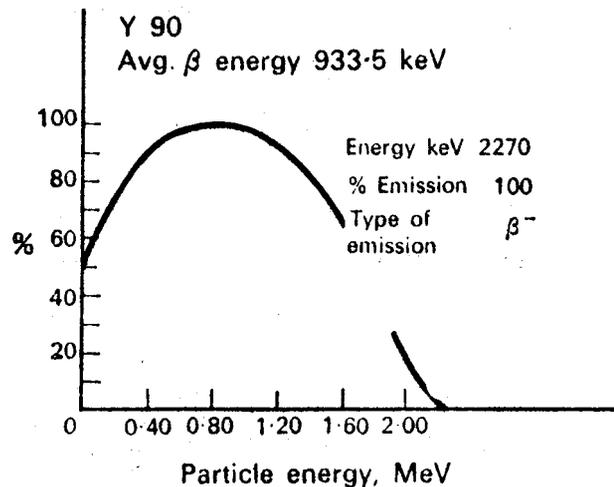


FIGURE 2. Spectrum of beta energies from decay of yttrium-90 (from Ref. 1).

## 2. SOURCES OF ENVIRONMENTAL RADIOSTRONTIUM

The two main sources of radiostrontium are from (i) nuclear reactors and their fuel reprocessing plants; and (ii) from fallout after atmospheric nuclear bomb tests. Both strontium radioisotopes are among the important fission products.



In *nuclear reactors*, there is a significant production of both  $^{90}\text{Sr}$  and  $^{89}\text{Sr}$ . Reference 3 projects that the *total* amounts of activity produced annually will increase dramatically in the coming decades because of growth in the nuclear power industry. In 1970 the total strontium-90 annual production rate was 4.0 megacuries/year; the projected figures for the years 1980, 1990, and 2000 are 227, 410, and 460 MCi/year, respectively. The integral (accumulated) strontium-90 activities in those three years are projected to be 960, 4600, and 9550 MCi, respectively. This is based upon an Oak Ridge National Laboratory projection whose detailed input is subject to revision, but the trend is clear. The strontium-89 production rates will grow similarly, although there is no long-term build-up of strontium-89 (because of its 52-day half-life).

Reactors can be designed and built so that most of the activities in the reactor fuel do not escape into the environment. At a reactor site itself, the average release rates of fission products are only very small fractions of the total fuel inventories. For example, at the Dresden Nuclear Power Station in Illinois, the ratios of released to produced activities in 1968 (Ref. 4) were  $4 \times 10^{-9}$  (for strontium-90) and  $6 \times 10^{-8}$  (for strontium-89). The average release rates in stack effluent were 700 pCi/second (strontium-89) and 3 pCi/second (strontium-90), while in liquid effluent the numbers were 8000 and 900 pCi/second, respectively.

The potential source for environmental release is mainly the nuclear fuel-reprocessing plants (see "Fuel Reprocessing" elsewhere in this Volume), where the fuel rods are dissolved and the radiostrontium is released from the cladding. However, sufficient precautions have been designed into the more recent plants to assure practically complete containment in normal operation. For example, the 150-day storage of fuel elements before reprocessing allows 86% of the  $^{89}\text{Sr}$  to decay (but only 1% of the  $^{90}\text{Sr}$ ). At the Midwest Fuel Recovery Plant in Morris, Illinois, the design calls for *no* environmental release of radioactive waste in either solid or liquid form. A limit on emissions of gaseous gross beta (particulates in air) at 7.0  $\mu\text{Ci}/\text{second}$  has been suggested (Ref. 5), based upon strontium-90 as the limiting radionuclide. From this it can be seen that the total environmental emissions should be small indeed.

*Fallout* is, of course, the source of radiostrontium that has gained the widest cognizance in the minds of the general public. Indeed, strontium-90 was a household word in the early 1960's. This widespread concern in the era of large-scale atmospheric nuclear bomb tests led to a great deal of research into the environmental behavior of radiostrontium in its pathways from the nuclear fireball to man. Fallout concentrations were measured extensively, and are still being measured routinely (Ref. 6). The deposition was considerable. We quote from the Federal Radiation Council (Ref. 7), "On January 1, 1963, the accumulated levels of Strontium-90 deposited over the United States varied from about 100 to 125 millicuries per square mile in the 'wet' areas (areas of greatest annual precipitation) to 40 to 50 millicuries per square mile in the 'dry' regions." The recent French and Chinese nuclear tests have again deposited strontium-90 as fallout.

Studies of the fate of strontium-90 from fallout have revealed that an important measure of its impact on man is the ratio of strontium to calcium in the total diet. "The (Sr/Ca) ratio in new bone being formed is one-fourth of that in the average diet because the body selectively discriminates against strontium" (Ref. 8). Typical  $^{90}\text{Sr}/\text{Ca}$  ratios in total diet in 1963 (Ref. 8) were 10 to 50 pCi/gram Ca (1 pCi of strontium-90 weighs about  $7 \times 10^{-15}$  grams). Activities in milk were 10 to 30 pCi/liter; and in flour about 40 pCi/kg. Strontium-89 in milk averaged 40 to 55 pCi/liter in 1963.

These figures indicate the range of activities requiring measurement. In biological and food samples, fallout measurements must be sensitive to a few pCi/liter of milk, and perhaps fractions of a pCi/kg of other foods such as vegetables and baked foods. In air-particulate samples from reactors or fuel-reprocessing plants, average release rates in the range from a few pCi/second to perhaps a few  $\mu\text{Ci}/\text{sec}$  need measurement.

Finally, the standard "gross beta" measurements in both drinking water and respirable air are designed mainly to place upper limits on strontium-90 in these media. For example, the standard method for gross-beta in drinking water (Ref. 9) explicitly states that a limit of 10 pCi/liter is allowed for strontium-90, but if strontium-90 is known to be absent, then from the point of view of beta activity "the water supply would usually be regarded as radiologically acceptable provided that the gross beta concentration does not exceed 1,000 pCi/liter" (Ref. 9).



### 3. MEASUREMENT CONSIDERATIONS

#### A. Introduction

We shall discuss methods for strontium-90 and strontium-89 determination in *air, water, milk, and other media*. Either of two types of measurements is commonly required: In one type, a quantitative measurement of strontium-90 specific activity (e.g., pCi/g) is made, while in the second type it is merely necessary to determine an upper limit of activity to assure safety. In the second type, a gross-beta measurement is usually performed as a first screening step, since it is easier than elaborate strontium-isolation chemistry. Only when the gross-beta measurement exceeds guidelines, or when a strong suspicion of strontium contamination is present, is specific strontium analysis called for.

Here we shall concern ourselves predominantly with the *quantitative analysis* of radiostrontium specific activity in various media. Because strontium must be determined at activity levels well below the permissible gross-beta levels in strontium's absence, radiochemical separation techniques are invariably required.

There is an extremely wide variety of methods for radiostrontium, but all follow a similar outline: *chemical concentration*, followed by *purification* and *sample preparation*, followed by *counting*.

Chemical techniques for radiostrontium have been tailored to the requirements of the various media to be measured. Since the chemical properties of yttrium play nearly as important a role as those of strontium itself, the procedures have generally depended upon multiple-step separations of yttrium as well as strontium from the bulk material. Also, media containing the radiostrontiums are frequently contaminated with radioactive cesium-137 (also a long-lived beta emitter), iodine-131, and perhaps other radionuclides as well, so that the radiochemistry can be complicated.

Here we shall give separate treatments to the various media: *air, water, milk, and other media* (principally foodstuffs and soil). The separate treatment is necessary because of the differences in the chemical techniques: for example, in both milk and urine yttrium forms an anionic complex which makes possible its separation by direct anion exchange. In air filters, the chemical techniques are quite different.

We shall *not* treat here measurements of strontium-90 activities in biomedical samples

such as urine, bone, or soft tissue. These are discussed in Volume 4 of this Survey ("Biomedical Instrumentation").

#### B. Counting

Beta counting is ultimately required in any of the methods. The final strontium sample has to be prepared properly for counting: for most of the counting instruments it must be a thin sample distributed reasonably uniformly over an area compatible with the instrument.

Counting is done with any of a number of instruments. Typical instruments include internal (or external thin-window) gas proportional counters; G-M tubes; semiconductor detectors; and less frequently liquid or plastic scintillation counters, which require different sample-preparation methods. The problems and differences which characterize the various beta counting techniques have been discussed elsewhere in this Volume ("Beta Particle Instrumentation") and will not be repeated here.

#### C. Strontium-90/Strontium-89 Separation

The chemical separation techniques cannot, of course, separate the two strontium radioisotopes from each other or from stable strontium. Consequently the end product of the chemical separation contains all strontium isotopes, including carrier strontium and in some cases strontium-85 added as a tracer for chemical yield.

Separation of strontium-89 from strontium-90 is performed using one of two methods. The first is *beta spectrometry*, which relies upon the differences in the maximum energies of the beta spectra ( $E_{\max} = 1.463$  MeV for strontium-89;  $E_{\max} = 546$  keV for strontium-90; and  $E_{\max} = 2.27$  MeV for yttrium-90). This type of measurement is not well suited to the cases in which the strontium-89/strontium-90 ratio is either very large or very small.

The most commonly used spectrometry technique employs a succession of different absorbers to count selectively the highest-energy betas (yttrium-90). If yttrium has been separated out chemically, the strontium-89 betas are counted. Detection efficiencies *must* be calibrated using known activities of the individual radionuclide species (or for yttrium-90 an equilibrium mixture of strontium-90 and yttrium-90); since geometrical effects can produce quite wide variations, it is not safe to rely upon calculations of the fractional detection efficiency. Although this problem is most important in beta spectrometry, it holds as well for the

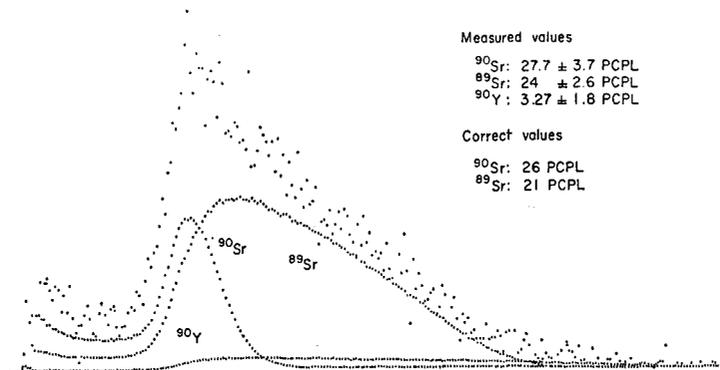


FIGURE 3. Beta pulse height spectra (from Ref. 11).  
PCPL = pCi/liter

determination of beta spectral efficiency in any of the methods being described.

The beta spectroscopy technique which shows the most promise is liquid scintillation spectroscopy, since after the sample is dissolved or suspended in a scintillator the geometrical efficiency factors are well understood. Direct counting of a sample with liquid scintillators suffers from severe limitations (Ref. 10). However, Piltingsrud and Stencel (Ref. 11) have described a technique which first separates strontium chemically as carbonate, using methods to be described later in this discussion. The  $\text{SrCO}_3$  is then suspended in liquid scintillator using a gelling agent, and counted, typically for 2 to 12 hours, using a Packard 'Tri-Carb' system and a 512-channel analyzer. (See 'Tritium' elsewhere in this volume for details.) For a 12-hour counting period, the minimum detectable activity is about 1 pCi. A typical spectrum is shown in Figure 3; the unfolding of the three isotopic spectral shapes ( $^{90}\text{Sr}$ ,  $^{90}\text{Y}$ ,  $^{89}\text{Sr}$ ) is done by computer. Another application of this technique is that of Kamada et al. (Ref. 12), who give detailed descriptions of the way the beta spectrum is unfolded.

The second, and today the more common technique for strontium-89/strontium-90 separation uses the ingrowth of the daughter yttrium-90. Figure 4 (from Ref. 9) shows the manner in which yttrium-90 and (yttrium-90 + strontium-90) activities grow in time from a pure strontium-90 sample. Counting rates (corrected for efficiency) can be compared from two measurements, the first immediately after strontium chemical separation (within a few hours, say) and the second 3 to 6 days later. Alternatively, chemical separation of the yttrium daughter can be performed after allowing for ingrowth. Both techniques are frequently used.

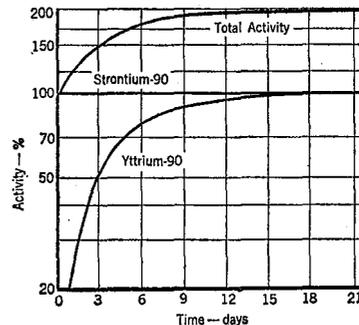


FIGURE 4. Yttrium-90 vs. strontium-90 activity as a function of time. (from Ref. 9)

Measurements of "total radiostrontium" usually refer to the total activity of a strontium fraction immediately after yttrium is separated chemically. The time elapsed before counting is critical, of course, because after strontium-90 and yttrium-90 reach parent/daughter equilibrium the "gross beta" counting rate represents exactly twice the activity of the parent strontium-90 itself.

#### 4. CHEMICAL TECHNIQUES

A good summary of the various types of chemical separation techniques has been given by Bowen (Ref. 13). We shall quote from his work extensively below. In the quotation, Bowen refers to sea-water measurements, but his comments should apply to other liquids. He begins by discussing two attempts to extract yttrium-90 directly from environmental samples, using thenoyltrifluoroacetone (TTA) in benzene. He continues as follows:



"With the exception of these two ventures, all the procedures known to us for strontium-90 analysis in the oceans depend on the separation of most of the strontium, together with more or less of the calcium, followed by removal of all or most of the calcium and then by decontamination of the strontium fraction from other natural or artificial radionuclides by a series of ferric hydroxide and barium chromate scavenges; the cleaned strontium, usually after precipitation with ammonium carbonate for convenience, is then stored for enough time to allow yttrium-90 to come again to equilibrium with its parent, two weeks being a convenient time; yttrium-90 is separated with a minimum amount of iron, samarium or yttrium carrier, often with some additional scavenging steps, finally filtered, weighed and mounted for beta counting. Counting has been done using a variety of gas-filled Geiger-Müller or proportional counters, or with solid plastic beta scintillator. The amount of data obtained to confirm the shape of the expected 64-hr half-life decay curve has varied widely, in response largely to economic factors; although 'good radiochemical technique' dictates the use of repeated milkings of yttrium-90 and insistence on superimposable decay curves, this has been a nicety often omitted.

"In such a separation scheme, the number of variations available is not large:

"(a) Strontium may be initially precipitated as carbonate or as oxalate [Ref. 13b,c] and for the carbonate precipitation, sodium carbonate [Ref. 13d, e], sodium carbonate plus ammonium chloride [Ref. 13f], or ammonium carbonate [Ref. 13g] may be used. Although other reagents have been proposed: manganese dioxide by Shipman [Ref. 13h] or coprecipitation with rhodizonic acid, we know of no data reported to have been obtained by these or any other separation processes than those cited above.

"(b) The oxalate precipitate may be decomposed by dry ignition [Ref. 13e] or by wet combustion with nitric acid.

"(c) Strontium may be separated from such calcium and magnesium as accompany it in the initial precipitation by:

"(1) Selective complexation precipitation, followed by fuming nitric acid precipitations to obtain pure strontium nitrate [Ref. 13d, f].

"(2) Oxalate reprecipitation, followed by fuming nitric separations (Shvedov and Ivonova, referred to in [13f]).

"(3) Nitric acid precipitations without other treatment (cf. Ref. [13c, e]; many others).

"(4) Selective elution serially, from cation exchange columns [Ref. 13a].

"(d) The overall chemical yield of strontium in the separation procedure may be determined gravimetrically [Ref. 13b], by counting tracer strontium-85 (e.g., Ref. [13c]) by flame photometry or atomic absorption spectrophotometry, or volumetrically [Ref. 13a].

"(e) The final separation of daughter yttrium-90 may be made using ferric hydroxide [Ref. 13d, e, f] or yttrium hydroxide [Ref. 13c], or samarium hydroxide. In each case the choice has been based on the availability - or ease of preparation - of radiochemically pure carrier and of the physical properties of its dried hydroxide as a counting medium."

#### A. Air

In air, measurements of radiostrontium are nearly always made by examining particulate matter. The sampling technique is relatively straightforward: particulate matter is collected on a high-efficiency filter in a Hi-Vol sampler. Air sampling rates of 1000 m<sup>3</sup>/day are typical.

Chemical separation is required because of the normal presence of other beta-emitting radionuclides in the particulate matter: for example, naturally-occurring radon daughters can adhere to particulate matter and produce sizeable beta activities. The compilation of *Methods of Air Sampling and Analysis* (Ref. 14) recently issued by the Intersociety Committee contains tentative methods for measuring both strontium-90 and strontium-89 in air. The chemical separation is the same for both radiostrontium isotopes, and this method is likely to be adopted as a "standard Method" in the near future.



The filter is ashed in a muffle furnace prior to the chemical procedure. We quote from Ref. 14:

"Strontium is separated from calcium, other fission products, and natural radioactive elements. Nitric acid separations remove the calcium and most of the other interfering ions. Radium, lead, and barium are removed with barium chromate. Traces of other fission products are scavenged with iron hydroxide. After the strontium-90 + yttrium-90 equilibrium has been attained, the yttrium-90 is precipitated as the hydroxide and converted to the oxalate for counting. Strontium chemical yield is determined gravimetrically as strontium carbonate. 0.5 pCi strontium-90 can be determined by this method with an error of less than  $\pm 10\%$  at the 95% confidence level under usual conditions (counter background of less than 1 cpm and a 60-minute count)."

Another technique for air filters has been developed at Atomic Energy of Canada Limited (A.E.C.L.) (Ref. 15). The first stages of the separation are described as follows:

"The ashed air filter is fumed with hydrofluoric and perchloric acids to remove silica. After extraction with nitric acid and fusion with sodium carbonate, an anion exchange column is used to separate plutonium from iron and other elements. The column effluent is retained for the estimation of barium (140), strontium (89, 90) and cesium (137)."

An alternative separation method, also described in the A.E.C.L. Manual (Ref. 15), is applicable when the ashed air filters contain little acid-insoluble material. In this method, "plutonium is separated from barium, strontium, and cesium by hexone extraction from a saturated ammonium nitrate solution" (Ref. 15). Hexone is methyl isobutyl ketone. This method is shorter and easier than either of the others mentioned, making it attractive where the conditions for its use are applicable. Use of either A.E.C.L. separation method must be followed by another stage to isolate radiostrontium from barium and cesium.

#### B. Water

Two types of water are of most concern: Drinking water and waste water. We have already referred above to the U.S. Public Health Service Drinking Water Standards (Ref. 16). The limit for strontium-90 is 10 pCi/liter, when other sources of intake are not considered. No

explicit limit is given for strontium-89, but the maximum permissible level for it would be significantly higher: for example, the ICRP's maximum permissible concentration for strontium-89 in water is a factor of 100 higher than for strontium-90.

In *Standard Methods* (Ref. 9) the method for total radiostrontium and strontium-90 in water applies to either drinking water or filtered raw water. We quote from Ref. 9:

"It is applicable to sewage and industrial wastes, provided that steps are taken to destroy organic matter and eliminate other interfering ions. In this analysis, a known amount of inactive strontium ions, in the form of strontium nitrate, is added as a "carrier". The carrier, alkaline earths and rare earths are precipitated as the carbonate to concentrate the radiostrontium. The carrier, along with the radionuclides of strontium, is separated from other radioactive elements and inactive sample solids by precipitation as strontium nitrate from fuming nitric acid solution. The strontium carrier, together with the radionuclides of strontium, is finally precipitated as strontium carbonate, which is dried, weighed to determine recovery of carrier, and then measured for radioactivity. The activity in the final precipitate is due to radioactive strontium only, because all other radioactive elements have been removed...."

"Radioactive barium ( $^{140}\text{Ba}$ ,  $^{140}\text{La}$ ) interferes in the determination of radioactive strontium inasmuch as it precipitates along with the radioactive strontium. This interference is eliminated by adding inactive barium nitrate carrier and separating this from the strontium by precipitating barium chromate in acetate buffer solution. Radium isotopes are also eliminated by this treatment...."

"Two alternate procedures are given for the separation of  $^{90}\text{Y}$ . In the first method,  $^{90}\text{Y}$  is separated by extraction into tributyl phosphate from concentrated nitric acid solution. It is back-extracted into dilute nitric acid and evaporated to dryness for beta counting. The second method consists of adding yttrium carrier, separating by precipitation as yttrium hydroxide, and finally precipitating yttrium oxalate for counting."

A similar technique, not identical in all respects but following the same general lines, is described in the A.E.C.L. Manual (Ref. 15).



### C. Milk

Milk is one of the most important pathways to man for environmental strontium. The U.S.P.H.S. Pasteurized Milk Network has developed a standard analytical procedure (Ref. 18-22) which will be discussed here as typical of several other methods. The method removes strontium (and other alkaline and alkaline earth ions) by passing a liter of milk through a cation exchange resin, followed by removal of yttrium in an anion exchange resin.

We quote from Ref. 20:

"The procedure consists of storing the milk samples with formaldehyde preservative for the ingrowth of the yttrium-90 daughter of strontium-90, adding yttrium carrier, and then passing the milk consecutively through cation and anion exchange resin columns. The alkaline earth ions in milk are replaced by sodium ions in the cation exchange column, after which the yttrium is retained as an anionic complex - probably of citrate - in the anion exchange column. The effluent milk is discarded and the yttrium complex on the anion exchange resin is destroyed with hydrochloric acid. The yttrium, eluted with the acid, is precipitated as the oxalate, and the radio-yttrium is measured with a low-background beta counter....

Initially, the maximum yttrium yield was 65% and average yields were 55%. Yttrium retention on the resin was improved from 80 to 90% by adding sodium citrate to the milk. The per cent eluted in 35 ml of hydrochloric acid was increased from 81 to 96% by decreasing the amount of anion exchange resin and by thoroughly stirring the resin during elution. Other losses were minimized by small procedural changes, so that the overall average yield was increased to 86%."

The U.S. Public Health Service manual (Ref. 21) describes other methods for milk analysis. Their "reference method", used mainly for quality control, is not applicable for routine analysis. The method uses strontium-85 as a tracer; strontium is separated by successive carbonate precipitation and HNO<sub>3</sub> washing, and the yttrium-90 ingrowth occurs in solution. The special feature is separation of ingrown yttrium from contaminating lanthanum-140 using tributyl phosphate. A similar method is described in the *Procedures Manual* of the USAEC Health and Safety Laboratory (Ref. 32).

Another method in the U.S.P.H.S. manual (Ref. 21) removes milk proteins by precipitation with trichloroacetic acid (TCA); precipitates the alkaline earths as oxalates, subsequently converted to nitrates; and separates strontium from calcium by solubility. When working in humid climates, care is required since the sample is counted as strontium nitrate on a stainless steel planchet, which can corrode the counting chamber if nitric acid is formed at high humidities. The *Handbook* of the Environmental Protection Agency's National Environmental Research Center (Las Vegas) (Ref. 24) describes a method very similar to this last (TCA) technique.

### D. Other Media

Chemical separation techniques have been developed for a wide variety of media besides air, water, and milk. The techniques are as varied as the problems, since media such as food or biological samples can have any of a large number of (radioactive or stable) impurities. Here we shall only mention briefly a few methods.

In most biological samples, the first procedure is to convert the sample to ash in a muffle furnace (500° to 800°C). This eliminates the large organic substrate, which might originally contain strontium or yttrium in complexed form. Ashing is also performed on milk and bone samples. For some samples, which are volatile at dry-ashing temperatures, it is preferable to perform *wet ashing* with nitric acid, followed by an equal-volume mixture of perchloric and nitric acids (Ref. 25). The U.S.P.H.S. manual (Ref. 21) contains good outlines of the procedures for both dry and wet ashing.

*Fusion* is used for those samples (e.g., soil, silt) which are not easily dissolved by acid digestion because of their high silica content. The fusion procedure in the U.S.P.H.S. manual is typical: the sample, with carrier plus 5 g of NaOH pellets per gram of sample, is fused over a Meeker burner for 20-30 minutes. Then 0.5 g of anhydrous Na<sub>2</sub>CO<sub>3</sub> per sample gram is slowly stirred in and heated again. The melt is cooled, taken up in water, centrifuged, heated again after 3N Na<sub>2</sub>CO<sub>3</sub> is added and centrifuged again. The precipitate is dissolved in 6N HCl, ready for analysis (Ref. 21).

Once ashing or fusion has provided a dissolved sample, the radiochemical methods are quite similar to those already described. There are some differences, but we refer the reader to the appropriate references for details.

The U.S.P.H.S. manual (Ref. 21) contains radiostrontium techniques for urine, bone, bone ash, tobacco, vegetation, soil, and silt.



The A.E.C.L. manual (Ref. 15) has techniques for animal and fish flesh, cereals, eggs, fats, and various vegetation as well as bone. The HASL *Procedures Manual* (Ref. 22) contains methods for vegetation, tissue, soil, and urine. The British A.E.R.E. has published radiostrontium methods in bone ash, milk ash, vegetable matter, human teeth, and soil (Ref. 26). Porter et al. (Ref. 27) have developed a method using excess ethylenediaminetetraacetic acid (EDTA) which is applicable in samples with very large calcium/strontium ratios, yet still permits strontium yield (as carbonate) to be determined gravimetrically. It has been used for a wide variety of organic media. Finally, Baratta and Reavey (Ref. 28) have described a method used by the U.S. Food and Drug Administration for food and other biological samples, which uses tributyl phosphate extraction and analysis of yttrium-90.

This is only a partial list of the many radiochemical analysis techniques in the literature; for more complete coverage, the reader is referred to the literature cited. Also, as mentioned above, radiostrontium assay in biomedical samples is treated in Volume 4 ("Biomedical") in this Survey.

#### E. Yttrium Recovery after Ingrowth

The chemical separation of radiostrontium is often followed by ingrowth of yttrium-90, yttrium separation, and counting of yttrium-90. This method is especially useful to determine strontium-90 in the presence of strontium-89.

One critical parameter is the specificity of this yttrium-90-separation scheme for strontium-90. This specificity has been studied by Goldin et al. (Ref. 25), whose procedure, after purifying  $\text{SrCO}_3$  precipitate, is described as follows:

"After ingrowth of yttrium, the strontium carbonate precipitate is dissolved with hydrochloric acid, neutralized to methyl orange with ammonia, and buffered at pH 5. The yttrium is extracted into 2-thenoyltrifluoroacetone solution. After the organic phase is washed with water buffered at pH 5, the yttrium-90 is stripped by extraction with 1N hydrochloric acid. The hydrochloric acid extract is evaporated on copper planchets, heated thoroughly to destroy residual organic matter, and counted, correcting for decay of the yttrium-90 between its extraction and the counting time."

The yield of strontium-89 is *very small* in this procedure: only  $4 \times 10^{-4}$  of the

strontium-89 is counted as yttrium-90, while the strontium-90 yields are typically 85% and the yttrium yield in the solvent extraction is greater than 95%.

Proper analytical procedure demands that calibrations be taken periodically to check the *64-hour half-life* of yttrium-90 measured from the final yttrium sample. This half-life is in the range where such measurements are easy, yet they are omitted in some laboratories for reasons of economics or inconvenience. This check is one of the easiest ways to exercise internal quality control on a modest scale.

#### F. Interferences

When chemical separation is performed, the main interferences in environmental samples are *stable calcium, barium-140, and stable phosphates*. The chemical considerations have been well discussed by Goldin, Velten and Frishkorn (Ref. 25), from whom we quote:

"*Calcium.* Strontium as an alkaline earth element is chemically similar to the other members of this family. Its resemblance to calcium, barium, and radium is especially marked. Calcium, which is widely distributed in nature, is one of the inert elements whose separation from radiostrontium has been most studied. Such separation is necessary if the strontium radiations are to be measured directly in samples of moderate or high calcium content, or if the final precipitate is weighed to determine the chemical recovery of strontium. Occasionally calcium is present in such large quantities as to give unmanageable amounts of precipitate unless it is removed.

"The precipitation of strontium (and barium) nitrate in concentrated nitric acid also separates strontium from calcium. The concentration of nitric acid is critical, as too dilute a nitric acid solution will solubilize too much strontium, while too strong a nitric acid will precipitate calcium with the strontium. The most favorable range is from 65 to 75% nitric acid. Concentrated nitric acid will suffice. If the sample is in aqueous solution, fuming nitric acid probably will be needed to take care of the dilution...."

"*Barium and Radium.* The fission product barium-140, also a member of the alkaline earth group, is carried along with strontium as carbonate.



It must be removed whether the strontium or its daughter is to be counted, as there is a radioactive daughter, lanthanum-140, which will contaminate the yttrium. It is precipitated, with carrier barium, as chromate from a solution buffered at pH 5. Radium, which follows barium in this procedure, can be determined, although rather crudely, by alpha-counting the barium chromate precipitate....

"*Phosphates.* Phosphates interfere by causing precipitation of strontium when the solution is made basic in the hydroxide scavenging step. Although the phosphate content of water is usually too low to be of consequence, this interference is very troublesome in the analysis of biological samples, whose ash is largely calcium phosphate. Phosphates are removed in the nitric acid treatment already described, being converted to phosphoric acid, which is removed with the supernate. Several treatments usually are necessary to ensure complete removal."

#### G. Calibration Techniques

There are a number of problems in performing accurate measurements using the chemical techniques discussed above. Among them are losses of radiostrontium, losses of yttrium, and counting errors.

The use of standard radiostrontium preparations can help to reduce counting efficiency errors. The detection efficiency over the spectrum of strontium-90 (and its daughter yttrium-90) must be understood, and for this purpose low-level solid  $\text{SrCO}_3$  sources have been prepared by the USAEC Health and Safety Laboratory (Ref. 29) by dilution of primary sources from the National Bureau of Standards. Standard sources for strontium-90 in milk have also been prepared (Ref. 30).

Chemical yields can be studied using strontium-85 as a chemical tracer: it decays ( $T_{1/2} = 64$  days) by electron capture followed by emission of a 514-keV gamma ray. Its routine use is uncommon partly because of its own intrinsic radiological hazard, but it serves a valuable function in periodic quality control. Other methods for determining chemical yield rely on classical chemical techniques, such as gravimetric measurements of carrier strontium or of possible interfering radionuclides.

Another quality control measurement, mentioned above, is the observation of yttrium-90's 64-hour half-life in counting of the yttrium fraction.

As in any analytical procedure, much of the success of radiostrontium assay depends upon the care of the analyst ... an obvious point but one which cannot be overemphasized.

#### H. Quality Control

Quality control is one of the most important elements of any radionuclide-analysis program. Unfortunately, quality-control checks have usually revealed significant variations among laboratories analyzing for  $^{90}\text{Sr}$  at low levels. These variations are a cause for concern since they demonstrate the fundamental difficulty in making routine, reliable, accurate  $^{90}\text{Sr}$  measurements in environmental media.

We shall discuss one recent study as an example ... others show comparable results. In 1971, the EPA Office of Radiation Programs carried out a study of analysis capabilities in *milk* (Ref. 31). Thirty-five participating laboratories made triplicate determinations of unknown concentrations of  $^{89}\text{Sr}$ ,  $^{90}\text{Sr}$ ,  $^{131}\text{I}$ , and  $^{137}\text{Cs}$  at activity levels ( $\pm 2\%$ ) of 31, 42, 69, and 52 pCi/liter, respectively. Unfortunately, *nine* laboratories reported results whose mean exceeded the 'control level' of about  $\pm 10\%$  for  $^{90}\text{Sr}$ . [For  $^{89}\text{Sr}$ ,  $^{131}\text{I}$ , and  $^{137}\text{Cs}$  the outlying laboratories numbered 4, 6, and 2 respectively.] For many of the outlying laboratories in the  $^{90}\text{Sr}$  study, the range of reported results for the triplicate analyses exceeded 25%. This study seems to show that, while many laboratories can perform  $^{90}\text{Sr}$  analyses well, a significant minority cannot.

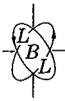
#### 5. SUMMARY AND CONCLUSIONS

In this section we have attempted to give an overview of the present status of radiostrontium measurements in environmental media.

Perhaps the key point to make first is that, at the low levels which concern us here, the only methods for *specific radiostrontium analysis* involve *analytical chemistry* followed by *beta counting*. We have therefore emphasized the considerations characteristic of the chemical techniques in common use.

One difficulty in comparing different chemical techniques is that so much rests in the hands (and mind) of the analyst himself. A choice among the various 'commonly used' techniques may depend more on the particular equipment and skills available than on the differences among the techniques themselves.

This point has been made extremely well by Bowen, from whom we quote (Ref. 13):



"In the extensive series of studies of strontium-90 in sea-water choice among these various possibilities has been made partly on the basis of prejudice, partly on economics, partly on the basis of which other radionuclides were to be sought on the same samples, partly on the basis of which alternative used fewest reagents known or suspected to be significant contributors to the radiochemical blank and partly on the basis of safety. In most of these variations the decisions have been legitimate ones, but we feel of even more importance have been questions which one can still only describe as those of art: in the hands even of very competent and experienced analysts, no one method, whether of trace element or of low-level radiochemical analysis, appears ever to have been uniformly superior in quality of results. For this reason, we strongly resist the suggestion that a single 'standard' method be selected and recommended."

In support of his viewpoint, Bowen cites interlaboratory calibration analyses in which nearly the full spectrum of possible radiochemical procedures was represented. We quote Bowen again:

"... it is clear, we believe, that in suitable hands each method was capable of delivering data of high quality; it is also clear from some of the entries ... that these well-established methods were capable in unsympathetic hands of yielding some pretty bizarre sets of numbers, indeed."

We agree with this general viewpoint. Apparently, none of the techniques suffers from critical intrinsic flaws; yet equally apparently, the results obtained with a particular method seem to depend upon a specific skill of the analyst rather than upon his general competence. This situation is *not* satisfactory or necessary. Much can be done in the development of these or other methods to make them less dependent upon individual talents.

We can summarize by stating first that sensitivity is adequate in the methods commonly used; second, that further research and development in this field is most needed in simplifying, standardizing, and automating the methods; but finally, that the market for a fully automated system (even if available) is probably insufficient to justify its development.

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## 1. INTRODUCTION

There are two main iodine isotopes of significance in radiological measurements in the environment: iodine-129 and iodine-131. Their properties are sufficiently different that they will be treated separately below. The difference is due to the physical characteristics of the decays.

*Iodine-131* has a half-life of 8.05 days. It decays by emission of one of several betas leading to excited states of xenon-131; the excited states then decay to the xenon-131 (stable) ground state by gamma emission. The most prominent beta (85% branching ratio) has  $E_{max} = 606$  keV, followed by a 364 keV gamma. This gamma is the emission which is principally used for detection. About 15% of the decays proceed by a weaker  $\beta^-$  ( $E_{max} = 315$  keV) followed by a more energetic gamma (638 keV).

*Iodine-129* is a beta-emitter with a 17 million year half-life. All of the decays proceed by emission of a beta ( $E_{max} = 150$  keV), followed by a 40-keV gamma, leading to stable xenon-129. Neither the  $\beta^-$  nor the gamma is very energetic, which leads to difficulties in detection. On the other hand, even small activities represent sizeable weights of iodine-129 in a material: one curie of pure iodine-129 would weigh about 6000 grams. It is therefore possible to use a technique such as neutron activation analysis on some iodine-129 samples.

## 2. RADIATION PROTECTION GUIDES

The ICRP's recommended maximum permissible concentrations (MPC) of these nuclides for occupational exposure are as follows on a 168 hour-per-week basis (Ref. 1):

Isotope	168-hour Occupational MPC in Air (pCi/m <sup>3</sup> )	168-hour Occupational MPC in Water (pCi/liter)
Iodine-131	3,000	20,000
Iodine-129	600	4,000

With the thyroid as the critical organ, the maximum permissible body burdens (MPBB) are 0.7  $\mu$ Ci (<sup>131</sup>I) and 3  $\mu$ Ci (<sup>129</sup>I), according to the ICRP (Ref. 1).

For individuals in the general public, the MPC and MPBB levels are a factor of 10 smaller; and for a suitable average sample of a large general population, another factor of 3 smaller still.

These MPC and MPBB values are based upon direct ingestion of water or inhalation of air, leading to dose to the thyroid (the principal site for iodine concentration in the body). The biological half-life of iodine in the thyroid is about 180 days (Ref. 2). The dose from iodine-131 decaying there is mostly (~90%) from the beta emissions, not from the gammas.

The U.S. Atomic Energy Commission's regulations are set forth in the Code of Federal Regulations, 10CFR20 (Ref. 3). All of the AEC occupational limits are identical to those recommended by ICRP. Recently, however, the AEC has proposed revised design guidelines which would govern the emissions from light-water power reactors under its jurisdiction (Ref. 4). If promulgated, these guidelines would, among other changes, decrease the MPC<sub>a</sub> for <sup>131</sup>I in the air around these reactors by a factor of 100,000. The motivation for this large decrease seems to be the desire to limit the <sup>131</sup>I dose to an infant thyroid to 5 mrem/year, assuming as a critical pathway the following chain: reactor/air/grass/cow/milk/infant. The cow used in calculating the pathway would eat grass exposed to <sup>131</sup>I from the reactor, and the infant would drink 1 liter of milk per day from that cow.

Although there is still not total agreement on the concentration factors in this pathway, it is possible to reproduce the reasoning leading to the factor of 100,000. Two elements in the chain have been studied: air-to-milk and milk-to-thyroid.

Burnett (Ref. 5) has deduced an iodine-131 concentration of 56,000 pCi/liter of milk, for an air concentration of 100 pCi/m<sup>3</sup> (which is the MPC<sub>a</sub> for the general public and equal to 1/30 of the occupational MPC<sub>a</sub> cited above). Recent measurements by VanAs and Vlegaar (Ref. 6) find a similar value for dry deposition and about twice as high a milk concentration when iodine is washed out of the air by rain.

Morley and Bryant (Ref. 7) have studied the case of a 6-month-old infant consuming 0.7 liters of milk per day. An iodine-131 concentration in milk of 400 pCi/liter is found to produce a dose of 1500 mrem/year to the child's thyroid. Using 1 liter/day consumption, this dose would be about 2100 mrem/year.

If an infant thyroid dose of only 5 mrem/year is desired, and assuming 1 liter/day consumption, the milk concentration must be only about 1 pCi/liter using Morley and Bryant's calculation. This then implies an air concentration limit of about 0.002 pCi/m<sup>3</sup> using



Burnett's calculation or about  $0.001 \text{ pCi/m}^3$  using VanAs and Vlegaar's measurements for iodine washed out of the air by rain. This latter figure ( $0.001 \text{ pCi/m}^3$ ) is precisely a factor of 100,000 less than the original MPC<sub>a</sub> of  $100 \text{ pCi/m}^3$ . The calculation just performed therefore seems to be the approximate reasoning behind the AEC's proposed reduction factor (Ref. 4).

### 3. SOURCES OF ENVIRONMENTAL RADIOIODINE

Both  $^{131}\text{I}$  and  $^{129}\text{I}$  are important fission products in the nuclear reactor industry: the fission yields (iodine atoms per fission) from uranium-235 in power reactors are 2.9% for  $^{131}\text{I}$  (Ref. 8) and 1.0% for  $^{129}\text{I}$  (Ref. 9). Because of its long half-life, there will be a long-term buildup of  $^{129}\text{I}$  as the nuclear power industry grows. One projection, from Oak Ridge National Laboratory (Ref. 8), indicates that the total annual U.S. generation of  $^{129}\text{I}$  in the years 1970, 1980, 1990, and 2000 will be 2, 110, 440, and 670 curies/year, respectively. On this time scale essentially none of the  $^{129}\text{I}$  decays, and the total accumulated activities up to those four years will be 2, 480, 2700, and 7600 curies, respectively. A good discussion of  $^{129}\text{I}$  and its role in the nuclear power industry can be found in the article by Russell and Hahn (Ref. 9).

With its eight-day half-life, there is no long-term buildup of iodine-131. Of course, the inventory of  $^{131}\text{I}$  will grow as the nuclear power industry grows. The routine releases of  $^{131}\text{I}$  from operating nuclear power plants are small, but  $^{131}\text{I}$  could be a serious problem near fuel-reprocessing plants, especially when future breeder-reactor fuel is being reprocessed. . . . this is discussed in the section on "Fuel Reprocessing Plants" elsewhere in this Volume. The requirement of 150-day hold-up of spent fuel before present reprocessing is governed mainly by the need to eliminate  $^{131}\text{I}$  by decay. Under present normal operating procedure, essentially no  $^{129}\text{I}$  or  $^{131}\text{I}$  is released to the environment from the nuclear reactor itself. At the reprocessing plant a very large fraction of the total inventory is released into the stack and must be removed to prevent its release to the environment. Fortunately, the need for iodine removal (mainly because of  $^{131}\text{I}$ ) was recognized early; this has brought about the introduction of scrubbing techniques, filters, and other methods, so that today  $\lesssim 1/2\%$  of the fuel inventory is actually released (Ref. 8).

Fallout is another source of radioiodine. During the large scale atmospheric bomb tests of the early 1960's (and to a lesser extent during recent French and Chinese atmospheric tests), iodine-131 was present in fallout

throughout the northern hemisphere. Its principal route to humans is through the grass-cow-milk sequence, on which considerable research has been conducted.

There are thus three main *potential* sources for fission-product iodine-129 and iodine-131 release to the environment:

- (a) fallout from nuclear bomb tests in the atmosphere;
- (b) release from the normal operations of nuclear power reactors;
- (c) release from nuclear fuel-reprocessing plants.

Other sources of lesser importance, but nevertheless requiring measurement capability, are from fuel transportation accidents, from leakage out of storage facilities, and from medical and industrial uses.

The media which require measurement are varied: biological samples (urine, feces); whole-body or external (thyroid) counting; food, principally milk; soil samples; air filters; and water effluent streams. To further complicate the measurement problem, the radioiodines can occur in any of several chemical forms: as elemental iodine, protein-bound iodine, ionic iodates and iodides, or various simple organic compounds such as methyl iodides.

The measurement of radioiodines in bioassay samples (whole body; urine; feces) will not be discussed here; the reader is referred to Volume 4 ("Biomedical") in this Survey.

Radioiodines are now in widespread use in nuclear medicine: for example, about 750,000 individuals received iodine-131 for thyroid uptake or thyroid function in 1966, with an average activity of 35 microcuries per application (Ref. 10). This adds up to about 30 curies of  $^{131}\text{I}$  nationwide, and although some of it decays in the body a sizable fraction ( $\lesssim 50\%$ ) is expelled as human waste into municipal wastewater systems.

The recent development of diagnostic procedures using a different isotope, *iodine-123*, has enabled the individual dose per application to be reduced considerably: the average iodine-123 thyroid dose is only 1 to 3% of the dose from a similar procedure using iodine-131 (Ref. 11). The probable spread of iodine-123 usage will require measurements of its specific activity in various media (e.g., urine, wastewater). Iodine-123 has a 13.3 hour half-life, and its principal emission is a 159-keV gamma. It is thus probable that iodine-123 can be



measured in environmental media with sufficient sensitivity by gamma spectroscopy.

A discussion of the present and potential uses of several of the radioiodine isotopes can be found in an article by Wellman and Anger (Ref. 12).

There is a considerable range in the specific activity levels which require measurement, depending upon the medium sampled. For example, in a study of the Nuclear Fuel Services (NFS) reprocessing plant (Ref. 13), the activity released during reprocessing of one metric ton of fuel (20,000 megawatt days burnup, aged 150 days) was about 1.8 curies of iodine-131 and 0.022 curies of iodine-129. Normal use of a silver-reacting scrubber reduces these activities by a factor of 200 before release to the stack, so that the released activities should be 9 and 0.11 millicuries, respectively. Since levels such as this are liberated daily at the (design) rate of 1 ton of fuel per day, it can be seen that activities in the millicurie range would require measurement; this turns out to correspond to specific activities of a few  $\times 10^{-3}$  pCi/cubic meter of stack gas, for both iodine-131 and iodine-129.

In deer thyroids taken near the NFS reprocessing plant, iodine-129 concentrations in the range of 0 to 210 pCi/gram have been observed (Ref. 9). A study of bovine thyroids revealed iodine-131 concentrations in the range up to about 100 pCi/gram, observed shortly after the detonation of nuclear weapons by the Chinese in 1965 (Ref. 14). Concentrations at other times were at least an order of magnitude smaller.

Iodine-131 concentrations are typically below one pCi/liter in milk samples routinely measured by the U.S. Public Health Service Pasteurized Milk Network (Ref. 15). This is also the level at which it is necessary to measure iodine-131 in drinking water or other (liquids in) foodstuffs.

#### 4. MEASUREMENT CONSIDERATIONS: IODINE-131

The measurement of iodine-131 in environmental media is nearly always performed by detecting its 364-keV gamma (85% branching ratio); sometimes also by its 638-keV gamma (15%); and less often by its betas. Because of its relatively high toxicity (reconcentration in the thyroid), the MPC levels for it are low. Methods for assessing iodine-131 concentrations in milk have received considerable attention, because the early fall-out data seemed to indicate that this was the principal pathway for iodine-131 exposure to man. The U.S. Public Health Service has maintained its Pasteurized Milk Network since

1957: about sixty stations throughout the U.S. collect samples which are analyzed by the P.H.S. for iodine-131, cesium-137, barium-140, strontium-89, and strontium-90 (Ref. 15).

The use of low-background gamma spectrometer systems is discussed elsewhere in this Volume (see "Gamma Spectrometry") and will not be treated in detail here. The two dominant types of instruments are sodium-iodide crystal spectrometers and lithium-drifted-germanium semiconductor spectrometers. The key to achieving great sensitivity for  $^{131}\text{I}$  measurements is the preparation of samples for the gamma-spectrometry measurements. This usually consists of chemical concentration procedures, which we shall discuss for each of the various media in the next paragraphs.

The minimum detectable concentration (MDC) of  $^{131}\text{I}$  using gamma spectroscopy varies considerably. For example, we shall quote below (from the Intersociety Committee, Ref. 16) an MDC of about 10 pCi; and we will indicate that an MDC of about 1 pCi is very nearly as well as can be done using very-low-background gamma systems and a 1000-minute counting period.

Using specially designed systems which detect beta-gamma coincidences, lower MDC values have been achieved. As an example, Brauer et al. (Ref. 36) have used a 5"  $\times$  5" NaI(Tl) well-type spectrometer, inside of which is placed a 0.75"  $\times$  0.020" plastic scintillator. Betas detected in the plastic phosphor are recorded in time coincidence with gammas detected in the NaI(Tl) well crystal; this arrangement lowers backgrounds considerably. Figure 1 shows the set-up schematically; a typical spectrum for  $^{131}\text{I}$  is shown in Figure 2. Brauer et al. quote a remarkably small MDC of about 0.01 pCi using this system (Ref. 36).

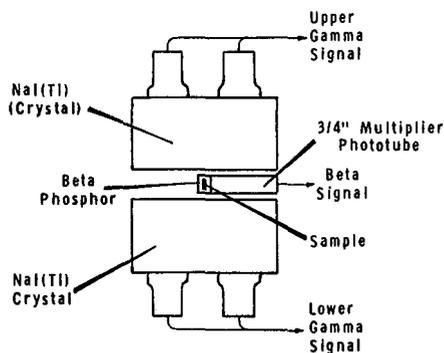


FIGURE 1. Low-level beta-gamma detector, showing beta phosphor inside well-type NaI(Tl) system (from Ref. 36).

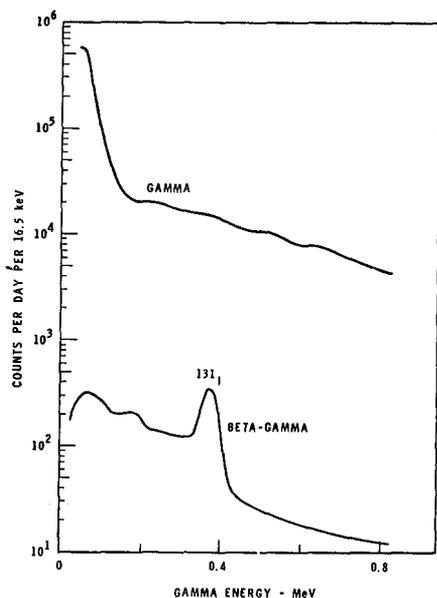


FIGURE 2. Gamma spectra of separated iodine sample taken in a 5" x 5" NaI(Tl) well system, with beta detection with small plastic phosphor inside well (see Figure 1). Gamma and beta-gamma coincidence spectra shown separately (from Ref. 36).

#### A. <sup>131</sup>I in Air

The task of measuring iodine-131 activity in the atmosphere is complicated by the multiplicity of possible chemical forms, such as gaseous I<sub>2</sub>, organic gaseous iodides and iodates, vapors adsorbed on particulate matter, and particulate iodine compounds. Recently, the Intersociety Committee published a "Tentative Method for Analysis for Iodine-131 Content of the Atmosphere" (Ref. 16), and we shall discuss this technique as an example of a number of similar air-sampling methods. The following is taken from Ref. 16:

"The proposed method involves sampling iodine in its solid and gaseous states with an 'absolute' particulate filter in series with an activated charcoal cartridge. The iodine-131 activity is then quantitatively determined by gamma spectroscopy. Iodine adsorbed on solid particles and possibly some vaporous iodine will be deposited on the 'absolute' filter while the remaining vaporous iodine will be adsorbed on the activated charcoal. . . ."

"Minimum detectable quantity of iodine-131 by gamma spectroscopy is approximately 10 pCi. At a sampling ratio of 0.01 m<sup>3</sup>/min the practical minimum detectable quantity for a 7 day sample is 0.1 pCi/m<sup>3</sup> (1.0 x 10<sup>-13</sup> μCi/cc). . . ."

"The minimum detectable quantity (10 pCi of iodine-131) can be determined with a precision of 10% at the 95% confidence level by employment of a well-shielded counting system and a sufficiently long counting time. . . ."

"Although collection of elemental iodine vapor on charcoal under the described conditions will approximate 100%, other vaporous iodine compounds such as organic iodides or hydrogen iodide may have much lower efficiencies."

The "Tentative Method" relies on an accurate flow measurement of sampled air through the system. The recommended charcoal cartridges are 5/8" in diameter and 1-1/2" deep, containing 3 grams of 12-30 mesh activated charcoal. After collection, the sample is counted in a NaI(Tl) scintillation counting well-type detector coupled to a multichannel analyzer. In the presence of other gamma emitters, decay measurements of the iodine-131 photopeak can be made over several days to observe the 8-day half-life. Calibration is by the use of a standard iodine-131 source in the same geometry as the unknown sample.

One of the difficulties with this 'standard method' is that it employs a NaI(Tl) scintillation detector. In the presence of other (higher-energy gamma) emitters a substantial interference can occur, making quantitative measurement difficult due to backgrounds under the iodine-131 peak. The use of a semiconductor-detector/spectrometer would improve this because of its vastly superior resolution.

In the presence of extremely large gamma background, the Intersociety Committee (Ref. 16) indicates that it is necessary to employ chemical-separation techniques before counting.

A specially-prepared filter for collection of iodine has been described by Baratta et al. (Ref. 17). The filter is impregnated with barium hydroxide-iodide-iodate, and after sample collection the iodine is concentrated chemically by dissolution and exchange between the radioiodine and the stable iodine.

Concentrations as low as 10<sup>-14</sup> μCi/cm<sup>3</sup> (0.01 pCi/m<sup>3</sup> of air) can probably be measured, using the ultimate in state-of-the-art techniques. For example, using the sampling rate of 0.01 m<sup>3</sup>/minute mentioned by the Intersociety Committee, a week's run will sample about 100 m<sup>3</sup> of air and collect 1 pCi of <sup>131</sup>I (at a concentration of 0.01 pCi/m<sup>3</sup>). This is just about the minimum detectable activity which one can expect to measure in



a very-low-background gamma-spectroscopy system, even counting for a 1000-minute period (see "Gamma Spectrometry" elsewhere in this Volume). Increasing the sampling rate by a factor of, say, 3 would yield 3 pCi of  $^{131}\text{I}$  or alternatively reduce the minimum detectable air concentration by the same factor of 3. However, too high a sampling rate reduces the collection efficiency of the charcoal filters, especially for the non-elemental forms of iodine.

#### B. $^{131}\text{I}$ in Milk

The Public Health Service Pasteurized Milk Network has developed a routine, standard analysis procedure for iodine-131 determinations in milk (Ref. 18, 19). After a composite is made from each sampling station in the network, the following simple procedure is carried out: with a *gamma-ray scintillation spectrometer* used for a 50-minute counting time, the three major gamma-emitters commonly found in milk (iodine-131, cesium-137, and barium-140) are all determined with minimum detectable concentrations in the range of 10 pCi/liter. A description of the gamma spectroscopy technique with NaI(Tl) scintillator, as developed by the PHS, can be found in the article by Hagee, Karches, and Goldin (Ref. 20).

Methods for iodine-131 in milk without the use of gamma spectroscopy must, of course, rely on chemical separation techniques to isolate individual radionuclides. Various separation techniques have been developed, mostly using an anion exchange resin followed by gamma counting. This method relies upon two facts: that "iodine-131 in market milk is mostly in the inorganic form as the iodide, very little being bound by proteins" (Ref. 19); and that "of the several fission products . . . , iodine-131 is the only anionic radionuclide present in milk" (Ref. 20).

The method described by Murthy (Ref. 19) is typical: 50 ml of anionic resin (strongly basic quaternary type, 20-50 mesh size, in chloride form) with 1 gallon of milk flowing through the column at 40-50 ml/min will recover all but the protein-bound iodine. Single-channel counting with a 2" NaI(Tl) crystal is used, and concentrations in the range of 50 pCi/liter can be determined to  $\pm 4$  pCi/liter.

In another similar method described by Porter and Carter (Ref. 21), the anion exchange resin retains  $96 \pm 2$  percent of the radioiodine in the first liter of milk, with a loss of an additional 2% per liter up to four liters. Porter and Carter's method is designed for sample collection in the field, at a very low (1965) cost of about \$3 for the complete field sampling unit, plus \$0.50 per

(non-reusable) vial with resin. The analytical and counting work is subsequently performed in the laboratory.

Both of the above methods are termed "rapid", meaning that determinations may be made in 2-3 hours.

Another method for fresh whole milk samples is described in the U.S.A.E.C. Health and Safety Laboratory's Manual (Ref. 22):

"Iodine-131 in whole milk samples is collected by the coagulation of protein-bound iodine with trichloroacetic acid and the simultaneous precipitation of ionic iodine as silver iodide. The filtrate from this separation and an aliquot of the untreated milk are then gamma counted on a NaI(Tl) scintillation crystal assembly."

This procedure has the advantage of measuring any protein-bound iodine-131. However, the subtraction of the corrected counting rate with and without treatment is necessary. Of course (quoting from Ref. 22 again), "the analysis may be made more specific by measuring with a gamma spectrometer."

Concentrations as low as fractions of one pCi/liter of milk can be measured if the chemical analysis procedure can remove enough iodine from enough milk. For example, using 10 liters of milk at 0.1 pCi/liter, one can by brute force obtain 1 pCi of  $^{131}\text{I}$  for determination by gamma spectroscopy. This 1 pCi of activity is very close to the ultimate detection limits of the better low-background gamma-spectroscopy systems, counting for 1000 minutes. Alternatively, one can use a commercially-available low-background beta counter of the type designed for tritium. Any of these low-level analyses, if performed commercially, will cost on the order of \$75 to \$150 per sample (Ref. 23).

It should be noted here that methods have been developed for *removing* iodine-131 from contaminated milk samples. It is beyond the scope of this work to discuss them; the reader is referred to a recent Environmental Protection Agency summary (Ref. 24) for details.

#### C. $^{131}\text{I}$ in Water

Two types of waters are of most concern: drinking water and waste water. Measurements of iodine-131 in drinking water are not made routinely as a part of public health practice, and no Standard Method is found in the American Public Health Association compilation (Ref. 25). The U.S. Public Health Service "Drinking Water Standards" pointed out in 1962 that "iodine-131 is not found in significant quantities in



public water supplies frequently enough to call for routine monitoring" (Ref. 26). The reason for this is that fallout iodine-131 does not significantly contaminate surface waters because of dilution, nor ground waters because of both dilution and the 8-day half-life; furthermore, in the event of massive fallout reconcentration in the grass-cow-milk system would produce significant iodine-131 contamination in milk of far greater consequence than for drinking water.

Contamination of waste waters is of great concern, however, because of increasing use of the radioiodines in the medical and industrial fields. The American Society for Testing and Materials (Ref. 27) has published three different Standard Methods for radioiodines in waste water. The methods are all designed for concentrations greater than 100 pCi/liter.

The first ASTM method (Ref. 28) employs "cation exchange followed by heterogenous isotopic exchange of iodide with silver iodide." Only silver isotopes might interfere, and these are rarely encountered.

The second ASTM method

". . . is based on the separation of iodine from other activities by distillation of elemental iodine into cold carbon tetrachloride. To assure chemical interchange with the iodine carrier an oxidation-reduction cycle is made. The iodine is oxidized to iodate with permanganate, reduced to iodide with bisulfite, and distilled over as free iodine in the presence of nitrite."

The third ASTM method (Ref. 22)

". . . is based on the separation of iodine isotopes from other radioisotopes by extraction into carbon tetrachloride . . . The iodine is oxidized to periodate in the presence of sodium hypochlorite and then reduced to iodide with sodium bisulfite in acid medium."

In each ASTM method, silver iodide (precipitate) is counted either with a Geiger-Muller or proportional counter, or preferably with a gamma spectrometer of the NaI(Tl) or Ge(Li) type.

#### D. <sup>131</sup>I in Other Media

The measurement of iodine-131 in media other than air, milk, and water is usually carried out by routine gamma spectroscopic analysis. For that reason, and also because very little in the way of concern has arisen to motivate specialized techniques in other media, we shall not discuss the measurements

here. The reader is referred to the section "Gamma Spectrometry" elsewhere in this Volume.

Measurements in bioassay media (urine, whole body, etc.) are discussed in Volume 4 ("Biomedical") of this Survey.

#### E. Quality Control

Quality control is one of the most important steps in the entire measurement process.

The calibration of gamma spectroscopic systems has been discussed in "Gamma Spectrometry" elsewhere in this Volume and will not be treated here. The other key part of the iodine measurement system is the chemical-separation process.

Unfortunately, quality-control checks have usually revealed significant variations among laboratories analyzing for <sup>131</sup>I at low levels. These variations are a cause for concern since they demonstrate the fundamental difficulty in making routine, reliable, accurate <sup>131</sup>I measurements in environmental media.

We shall discuss one recent study as an example. . . . others show comparable results. In 1971, the EPA Office of Radiation Programs carried out a study of analysis capabilities in *milk* (Ref. 30). 35 participating laboratories made triplicate determinations of unknown concentrations of <sup>131</sup>I, <sup>89</sup>Sr, <sup>90</sup>Sr, and <sup>137</sup>Cs at activity levels ( $\pm 2\%$ ) of 69, 31, 42 and 52 pCi/liter, respectively. Unfortunately, *six* laboratories reported results whose mean exceeded the 'control level' of about  $\pm 15\%$  for <sup>131</sup>I. [For <sup>89</sup>Sr, <sup>90</sup>Sr, and <sup>137</sup>Cs the outlying laboratories numbered 4, 9, and 2, respectively.] For many of the outlying laboratories in the <sup>131</sup>I study, the range of reported results for the triplicate analyses exceeded 30%. This study seems to show that, while many laboratories can perform <sup>131</sup>I analyses well, a significant minority cannot.

#### 5. MEASUREMENT CONSIDERATIONS: IODINE-129

Iodine-129 decays with a 17-million-year half-life by emitting a low-energy beta ( $E_{max} = 150$  keV) followed by a 40-keV gamma. Both the beta and gamma are of relatively low energy, and consequently difficult to detect. The MPC values for <sup>129</sup>I are all about a factor of 5 more restrictive than those for <sup>131</sup>I (because after ingestion or inhalation the <sup>131</sup>I decays relatively rapidly with its 8-day physical half-life while the <sup>129</sup>I lasts for the biological half-life of about 180 days). Combining these considerations, the measurement problem is seen to be fairly difficult.

Fortunately, however, the long half-life and low specific activity can be used to



advantage. Since 1 curie of  $^{129}\text{I}$  weighs about 6000 grams, even minute activities represent sizeable numbers of atoms amenable to techniques besides direct radioactive counting.

We shall discuss two measurement techniques: *liquid scintillation counting* and *neutron activation analysis*. The former is useful when the activity to be measured is known to exceed a few pCi of  $^{129}\text{I}$ , while the latter is recommended for the ultimate in sensitivity. In either case, the iodine must be isolated chemically to free it from the sample matrix in which it is found. We shall not discuss here any of the several radiochemical techniques already mentioned for iodine-131.

#### A. Liquid Scintillation Counting

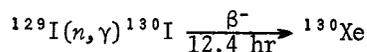
This method relies upon the ability of a liquid scintillation system to detect the low-energy beta particles from  $^{129}\text{I}$ . As just mentioned, the iodine must be isolated chemically before introduction to the scintillation system, to provide a small sample free of the many impurities which might cause quenching (self-absorption by the sample of the scintillation light).

Liquid scintillation counting has been discussed in depth in the section "Tritium" elsewhere in this Volume, and that discussion will not be repeated here. The best systems have minimum detectable activities ( $3\sigma$ ) for tritium in the range of about 1 pCi/cm<sup>3</sup> of water, and the performance for  $^{129}\text{I}$  should be comparable. In the liquid scintillation method described by Magno et al. (Ref. 31) and Gabay et al. (Ref. 32), the only feature which is different for  $^{129}\text{I}$  than for tritium is that the sample is placed under a fluorescent light for about 2 hours to decolorize the iodine before counting. A possible source of error is that the counting efficiency for  $^{129}\text{I}$  must be known as a function of energy, since the wide beta energy spectrum extends from the maximum (150 keV) down to zero energy. A standard  $^{129}\text{I}$  solution is useful for this purpose. The U.S. National Bureau of Standards has recently prepared such a standard (Ref. 33). The calibration should, if possible, be performed experimentally rather than determined by calculation, since the  $^{129}\text{I}$  beta is followed essentially immediately by a 40-keV gamma ray which will also be detected in the scintillator some of the time, distorting the apparent beta-energy spectrum.

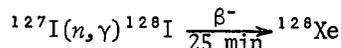
Using a solvent extraction procedure on 4 liters of milk, followed by liquid scintillation counting, Gabay et al. (Ref. 32) report a  $^{129}\text{I}$  detection limit (95% confidence) of about 0.3 pCi/liter, counting for 100 minutes at average background of 25 cpm.

#### B. Neutron Activation Analysis

Iodine-129 determination by neutron activation analysis relies upon the transformation of  $^{129}\text{I}$  to  $^{130}\text{I}$  by a neutron-capture reaction:

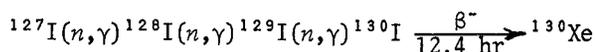


The  $^{130}\text{I}$  is counted in a gamma spectrometer by measuring its own decay to  $^{130}\text{Xe}$  (12.4 hour half-life; one of several betas is followed in each case by a series of gamma rays, including two easily measurable ones at 538 and 669 keV). Typically,  $^{129}\text{I}$  is accompanied in any medium by much larger concentrations of stable, naturally-occurring  $^{127}\text{I}$ . This also is determined by a similar ( $n,\gamma$ ) process:



The  $^{128}\text{I}$  decays to  $^{128}\text{Xe}$  with a 25-minute half-life; the most easily detected emission is a 441-keV gamma ray (13%).

The intrinsic limit on sensitivity of this method is caused by problems from the multiple activation of  $^{127}\text{I}$  up to  $^{130}\text{I}$ , where it masks  $^{129}\text{I}$ :



Typically, neutron activation analysis is capable of measuring  $^{129}\text{I}/^{127}\text{I}$  ratios to much greater precision than the determination of the  $^{129}\text{I}$  activity itself. This ratio may then be used together with other information on  $^{127}\text{I}$  content to determine  $^{129}\text{I}$  very well.

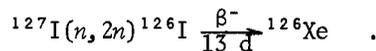
Neutron activation analysis for  $^{129}\text{I}$  was first suggested by Edwards (Ref. 34), who pointed out the possible use of  $^{129}\text{I}$  as an environmental tracer. More recent investigations by Keisch et al. (Ref. 35), Cochran (Ref. 13), and Magno et al. (Ref. 31) show that with chemical separation the technique suffers few interferences and may be used down to very low concentrations. As mentioned above, iodine-isolation chemistry is required to obtain the ultimate in sensitivity; for example, an important interference can result if stable cesium-133 is present, since the reaction  $^{133}\text{Cs}(n,\alpha)^{130}\text{I}$  produces the same  $^{130}\text{I}$  isotope as that from ( $n,\gamma$ ) activation of  $^{129}\text{I}$ .

Because of the multiple-capture production of  $^{130}\text{I}$  from  $^{127}\text{I}$ , there is an optimum neutron flux for the ultimate sensitivity. Keisch et al. (Ref. 35) have shown this to be about  $10^{13}$  thermal neutrons cm<sup>-2</sup>sec<sup>-1</sup>. These investigators report that the theoretical limitation on the  $^{129}\text{I}/^{127}\text{I}$  ratio would be about  $3 \times 10^{-13}$  in a sample containing one



gram of iodine. This corresponds to an incredibly small  $^{129}\text{I}$  activity of about  $5 \times 10^{-17}$  Ci!

Ordinary gamma-spectrometry measurements after activation are performed using NaI(Tl) or Ge(Li) systems. Keisch et al. (Ref. 35) also report use of a beta-gamma coincidence arrangement for  $^{130}\text{I}$  to reduce backgrounds. This in turn may suffer interferences in the presence of the isotope  $^{126}\text{I}$ , which is also a beta-gamma coincidence emitter.  $^{126}\text{I}$  is produced from  $^{127}\text{I}$  by the reaction



This occurs when the beam contains neutrons of higher than thermal energy. Use of a well-thermalized neutron flux suppresses this background.

Boulos et al. (Ref. 36) report a variation on activation analysis, in which after irradiation the sample is measured in a mass-spectrometry system for the xenon isotopes which are the end-products of  $^{130}\text{I}$  and  $^{126}\text{I}$  decay. Both the  $^{130}\text{Xe}/^{128}\text{Xe}$  and the  $^{130}\text{Xe}/^{126}\text{Xe}$  ratios should be indicative of the  $^{129}\text{I}/^{127}\text{I}$  ratio. Detection limits on the ratio in the range of  $\sim 10^{-10}$  are reported.

## 6. SUMMARY AND CONCLUSIONS

The intent of this section has been to discuss the sources and principal measurement techniques for radioiodine (iodine-131 and iodine-129) in the environment. Here we shall summarize the present situation by delineating the various individual measurement problems:

a. *Iodine-131 in milk.* Milk is the medium to which the greatest effort has been devoted. There are several rapid laboratory methods for milk analysis at minimum detectable concentrations of about 10 pCi/liter, and with the very best equipment and long counting periods fractions of 1 pCi/liter can be determined. Field sampling methods for milk are well in hand, but laboratory analysis is still required and 2-3 hour determinations are considered "rapid". It appears that instrumentation is now adequate to cover this measurement problem.

b. *Iodine-131 in air.* The "Tentative Method" of the Intersociety Committee can determine levels in the region of 0.1 pCi/m<sup>3</sup>; the MPC level for the general public is 10 pCi/m<sup>3</sup>. However, this is for a seven-day sample; for a one day sample the minimum detectable level would be seven times higher.

Detection limits perhaps one order-of-magnitude lower can be obtained using the very best low-background equipment. This method is not adequate when vaporous iodine compounds such

as organic iodides or hydrogen iodide are present since they are not efficiently collected. Clearly, some improvement is needed here, since precise measurement on *part* of a potential problem is not adequate when one cannot be sure that other parts are also well below levels of concern.

c. *Iodine-131 in other media.* Gamma spectroscopy appears to be generally adequate for activity determinations in most solid media, such as biological samples.

d. *Iodine-129.* The problem of sampling iodine-129 in air is similar to that for iodine-131 (see above). However, iodine-129 is not now often a problem of human radiological significance (even locally near nuclear fuel reprocessing plants), and measurements of it are principally to assure that it is not released to the environment in the first place. The restrictions on release, in turn, are to prevent long-term environmental build-up. Both neutron activation analysis and liquid scintillation spectrometry appear adequate to determine  $^{129}\text{I}$  in the media requiring measurement.

## CONCLUSIONS

Measurements of iodine-131 in air, milk, and solid biological samples can (in the opinion of the authors) all be made with existing instrumentation to sufficient accuracy and low enough concentration. The only exception is for those vaporous iodine compounds such as light organic iodides or HI which are not presently collected with adequate efficiency in the established method for determining iodine-131 in air. For iodine-129 the situation is also reasonably well developed.

## 7. ACKNOWLEDGMENT

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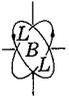


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RADON-222 AND ITS DAUGHTERS

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## 1. INTRODUCTION

In the *uranium mining* industry, radiological exposures to radon-222 daughters are the most significant hazard. Radon gas also appears in the *natural radiation environment* wherever natural uranium exists in surface rocks and ores. Because of their importance, and because instrumentation for their measurement is unique, radon and its daughters are discussed here in a separate section of this Survey.

We shall begin by outlining the physical characteristics of the radon decay chain, and follow with discussions of the radiation protection guides and the special unit (Working Level) developed for measuring these activities. Finally, we will discuss the types of measurement capabilities required of instrumentation, and the instrumentation itself. We shall deal both with measurements in the natural environment, and also with measurements for occupational radiation exposure monitoring and control, mainly in uranium mines.

## 2. PHYSICAL CONSIDERATIONS

### a. Radon-222 and Its Decay Chain

The decay chain of which radon-222 is a part is one of the oldest and best-studied phenomena in nuclear physics. It begins with the naturally occurring isotope uranium-238 and ends with the stable lead-206. In between there are 8 alpha decays and 6 beta decays. The first part of the series (uranium-238 to radon-222) is shown in Figure 1, while Figure 2 shows the second part (radon-222 to lead-206). More detailed data are given in Table 1 (from Ref. 1). Note that many of the chain members are still often referred to by their historical names, and we shall often use this nomenclature here also.

Radon-222 provides a natural division for our purposes, because we are interested mainly here in radon and its daughters. However, there is also an important physical basis for the separation: that is, *radon is an inert gas*. As  $^{238}\text{U}$  decays through five steps to  $^{226}\text{Ra}$ , the heavy nucleus remains fixed in place (in rock, for example). However, there is diffusion of radon gas from the local site of production, and in any particular (surface) rock some of the radon will escape into the surrounding atmosphere. This is the source of the historical name for  $^{222}\text{Rn}$ , "Emanation Radon".

Thus for many practical purposes one can think of uranium-laden ore as a "source" of radon gas, as well as a "source" of the many  $\alpha$ ,  $\beta$ , and  $\gamma$  radiations being emitted by the constituent nuclei themselves.

The properties of the chain radium  $\rightarrow$  radon-daughters concern us here. Note that radium-226 has a half-life of 1602 years. Since that is much longer than the half-life of any subsequent daughter,  $^{226}\text{Ra}$  can be viewed (for short times) essentially as a fixed-rate generator for  $^{222}\text{Rn}$  (whose half-life is 3.82 days). The equilibrium build-up of the succeeding daughters (RaA, RaB, RaC) is shown in Figure 3 (from Ref. 2). Note that after RaC decays, RaC' follows nearly immediately (164  $\mu\text{sec}$ ) and that RaD ( $^{210}\text{Pb}$ ) has a 22-year half-life, which for short times effectively blocks the decay chain. Hence we shall deal here almost exclusively with the first four daughters, RaA, RaB, RaC, and RaC'.

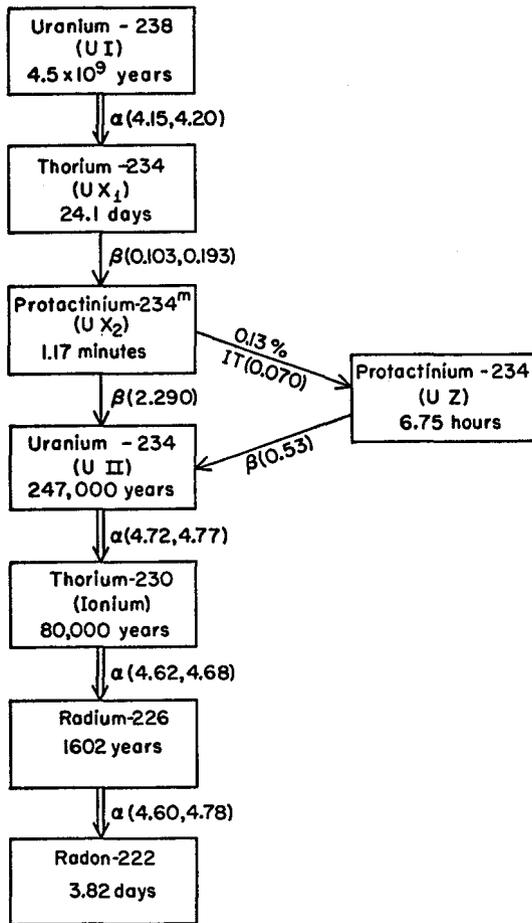
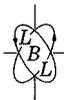
Evans (Ref. 2) has given a very useful summary of the physical and engineering considerations concerning the radon daughters. The chemical properties of the daughters are crucial. Although radon is inert, the three elements immediately below it in the periodic table (polonium, bismuth, and lead) are all chemically active. In particular, when radon (gas) decays, most of the newly created  $^{218}\text{Po}$  (RaA) atoms, typically ionized, tend to attach almost immediately to any particulate matter in the atmosphere. Most RaB, RaC, and RaC' atoms are so attached at birth. In dust-laden mine air, the consequence is that *the particulate matter becomes radioactive* by adsorption. Whether or not they are attached to particulate matter, the daughters cause important radiological consequences when inhaled, since they tend to lodge in the lung mucosa.

The alpha-emitting daughters are RaA and RaC'. The alpha decay energies are as follows:

$^{222}\text{Rn}$	$\rightarrow$	$^{218}\text{Po}$ (RaA)	5.48 MeV
$^{218}\text{Po}$ (RaA)	$\rightarrow$	$^{214}\text{Pb}$ (RaB)	6.00 MeV
$^{214}\text{Po}$ (RaC')	$\rightarrow$	$^{210}\text{Pb}$ (RaD)	7.69 MeV

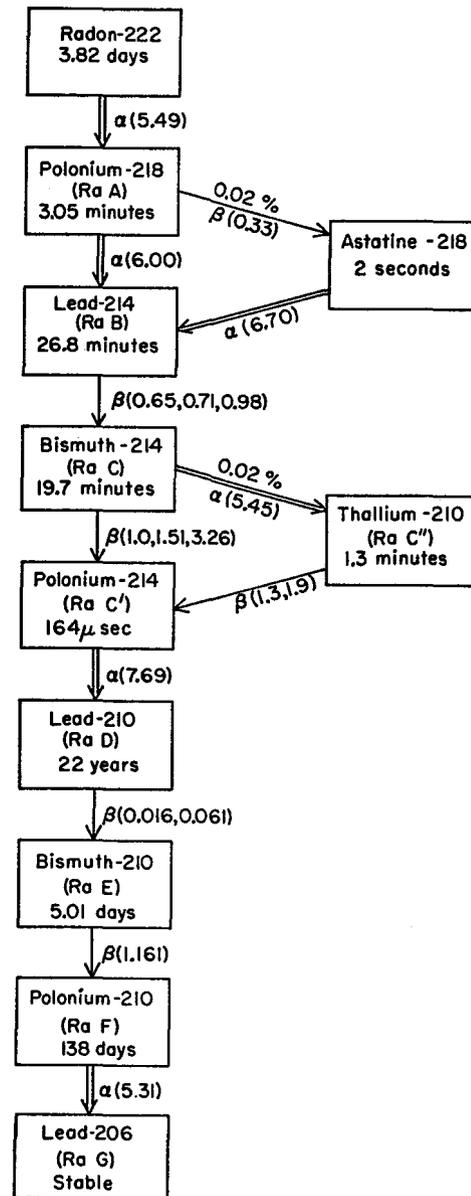
The energy differences can, of course, provide a basis for distinguishing the individual components by alpha spectroscopy.

The two important beta/gamma-emitting daughters are RaB and RaC. Each has a rather complicated set of emissions, with three important betas and several gamma lines each (see Table 1). Because of this, beta detection as a means of measurement is complicated: the spectral efficiency of any beta detector is very hard to determine. The RaB and RaC gamma lines could be detected by gamma-spectroscopic methods but this is not now commonly done.



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FIGURE 1. Decay Chain,  
Uranium-238 to Radon-222  
(α, β energies in MeV)



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FIGURE 2. Decay Chain,  
Radon-222 to Lead-206  
(α, β energies in MeV)



TABLE 1.

*Main sequence of decays from uranium-238 to lead-206. Three very weak collateral branch disintegrations, all with branching ratios less than 0.1%, are omitted.  
(from Ref. 1)*

Common name or symbol	Isotope	Half life	Principal radiations	Alpha energy (MeV)	Beta maximum energy (MeV)	Gamma-ray quanta per disintegration	Average gamma-ray energy (MeV)
Uranium I	Uranium <sup>238</sup>	4.49 x 10 <sup>9</sup> years	α	4.18			
Uranium X <sub>1</sub>	Thorium <sup>234</sup>	24.1 days	β		.205 (80%) .111 (20%)		
Uranium X <sub>2</sub>	Protactinium <sup>234</sup>	1.17 minutes	β		2.32 (80%) 1.5 (13%) .6 (7%)		
Uranium II	Uranium <sup>234</sup>	248,000 years	α	4.76			
Ionium	Thorium <sup>230</sup>	80,000 years	α	4.68 (75%) 4.61 (25%)			
Radium	Radium <sup>226</sup>	1,602 years	α	4.78 (94.3%) 4.69 (5.7%)			
Radon	Radon <sup>222</sup>	3.825 days	α	5.486			
Radium A	Polonium <sup>218</sup>	3.05 minutes	α	5.998			
Radium B	Lead <sup>214</sup>	26.8 minutes	β γ		.65	.82	.295
Radium C	Bismuth <sup>214</sup>	19.7 minutes	β γ		3.13 (23%) 1.67 (77%)	1.45	1.050
Radium C'	Polonium <sup>214</sup>	164 μsec	α	7.68			
Radium D	Lead <sup>210</sup>	22 years	β γ		.018	1.	.047
Radium E	Bismuth <sup>210</sup>	5.02 days	β		1.17		
Radium F	Polonium <sup>210</sup>	138.3 days	α	5.298			
Radium G	Lead <sup>206</sup>	Stable	Stable				

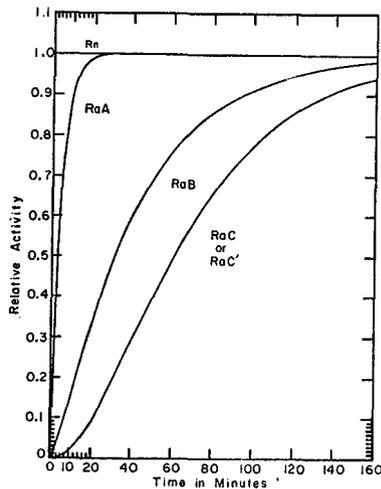
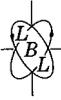


FIGURE 3. Growth of activity of the individual short-lived decay products in a constant source of radon having unit activity (from Ref. 2).

b. Radon-220 (Thoron) and Radon-219 (Actinon)

In situations where radon-222 must be measured, two other isotopes of radon are potential interferences. The most important is radon-220 (historically known as 'thoron'), a member of the decay chain which originates with naturally-occurring thorium-232. The other isotope is radon-219 (historically, 'actinon'), from the chain originating with uranium-235. Both of these gases behave as does radon-222: they emanate from the radioactive ores after their birth by the alpha decays of their immediate parents ( $^{224}\text{Ra} \rightarrow ^{220}\text{Rn}$ ,  $^{223}\text{Ra} \rightarrow ^{219}\text{Rn}$ ).

In the uranium mines, thorium content of the ores is low (typically < 1% of that of  $^{238}\text{U}$ ), and the  $^{235}\text{U}/^{238}\text{U}$  ratio is uniform at about 0.72%. Thus the equilibrium production of both thoron and actinon gases is relatively small. Equally important, the half-lives are quite short (55 sec for thoron, 4.0 sec for actinon). Because of their limited abundance and short half-lives, their direct radiological impacts are in both cases much less significant than that of radon-222.

Some of thoron's daughters can produce possible backgrounds when radon-222 daughters are collected on air filters for measurement:  $^{212}\text{Pb}$  and  $^{212}\text{Bi}$  have half-lives of 10.6 and 1.01 hours, respectively. Because of actinon's short half-life, its ability to diffuse out of earth and rock is so limited that it is seldom, if ever, present at levels requiring measurement in the environment.

Throughout this section, the term 'radon' will be used to denote 'radon-222' unless specifically stated otherwise.

3. RADIATION PROTECTION GUIDES

For the purposes of radiological protection of the lungs in the uranium mining industry, a specialized unit of exposure to radon-222 has been developed. This is the Working Level (WL), defined as "any combination of radon daughters in one liter of air that will result in the ultimate emission of  $1.3 \times 10^5$  MeV of potential alpha energy" (Ref. 3). This value is derived from alpha energies released by the total decay of the short-lived daughters (RaA, RaB, RaC, RaC') at radioactive equilibrium with 100 pCi of  $^{222}\text{Rn}$ /liter of air. Note that the WL considers only the alphas from radon-222 daughters and not from radon gas itself.

The reason for the specialized unit is mainly operational: the WL is a concept having validity in any mixed concentration of radon and its daughters, whether or not they are in equilibrium. Just as important, it lends itself to practical measurements in the mines.

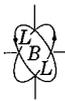
An extension of the WL concept is the "Working Level Month" (WLM), which expresses a cumulative exposure. It is defined as follows: "Inhalation of air containing a radon daughter concentration of one WL for 170 working hours results in an exposure of one WLM" (Ref. 3).

The Secretary of Labor, acting under provisions of the Walsh Healy Act, promulgated the following standard in late 1968 (Ref. 4):

"Occupational exposure to radon daughters in mines shall be controlled so that no individual will receive an exposure of more than 2 WLM in any consecutive 3-month period and no more than 4 WLM in any consecutive 12-month period. Actual exposures shall be kept as far below these values as practicable."

In early 1969, the Department of the Interior issued the following standard calling for action on the basis of individual concentrations (Ref. 5):

"If samples show an atmospheric concentration of radon daughters of more than 1 WL but less than 2 WL, immediate corrective action shall be taken or the men shall be withdrawn. When concentrations higher than 2 WL are indicated, the men shall be withdrawn from the area until corrective action is taken and the radon-daughter atmospheric concentrations are reduced to 1 WL or less. . . . Smoking shall be prohibited where uranium is mined."



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The above two standards are now considered the operational guidelines for exposure of miners to radon daughters.

The absorbed dose (rad) and dose equivalent (rem) resulting from 1 WL of exposure depend upon the exact nature of the mine atmosphere. With modern ventilation procedures, the equilibrium condition is never reached for any but the first daughter, RaA. Consider as an example the "reference atmosphere" discussed by Altshuler (Ref. 6), which contains in one liter of air the following activities totaling 200 pCi: 94 pCi of  $^{218}\text{Po}$ , 62 pCi of  $^{214}\text{Pb}$ , 44 pCi of  $^{214}\text{Bi}$ . This yields  $7.4 \times 10^4$  MeV or 57% of a WL. We quote from the Federal Radiation Council (Ref. 7):

"Altshuler's reference atmosphere results in 60% of a 'WL dose' to the bronchi, and has a value between 10 and 30 rads per year. In view of the ambiguities of conversion, Altshuler's reference atmosphere will be considered, with important reservations, to produce 20 rads in a normal working year. . . Exposure for one year at . . . one WL would give 33 rads. The mean organ dose would be lower. In the absence of an appropriate factor for the RBE, the calculated dose cannot be converted to rem."

It should be emphasized that the WL standards are based on epidemiological evidence, rather than on calculated dose equivalent to the lungs (Ref. 7).

Exposure to radiation other than the inhalation of radon daughters must be considered separately. The usual occupational limits apply: that is, 5 rem/year for whole body external exposure, and so on. These limits have been discussed in detail in the introductory section of this volume (see "Radiation Guides").

A limit for *radon-222 gas* itself is not considered separately by the Federal Radiation Council, because of the general recognition that the impact of the radon daughters is the more important consideration.

The International Commission on Radiological Protection (ICRP) explicitly considers the impact of RaA in its recommendation. We quote from ICRP Report No. 6, written in 1959 (Ref. 8):

"Recent studies have indicated that when radon and its daughters are present in ordinary air the free ions of RaA constitute only about 10 per cent of the total number of RaA atoms that would be present at equilibrium and these unattached atoms

deliver all but a small fraction of the dose to the bronchi. Based on these measured dose rates the (MPC)<sub>a</sub> for exposure to radon and daughter products is found to be

$$(\text{MPC})_a = \frac{3000}{(1 + 1000f)} \text{ pCi } ^{222}\text{Rn/liter of air}$$

where  $f$  is the fraction of the equilibrium amount of RaA ions which are unattached to nuclei."

If we set  $f = 10\%$ , as the ICRP indicates might be typical, then the occupational (MPC)<sub>a</sub> is 30 pCi/liter.

There is no explicit guideline for exposure of the general public to radon and its daughters. However, both the ICRP (Ref. 9) and the NCRP (Ref. 10) have recommended in their general overviews that individuals in the general public be limited to exposures at levels one-tenth as high as those for occupational exposure. Also, for a suitably large sample of the general population, the general guideline is another factor of 3 smaller still.

#### 4. SOURCES OF RADON AND ITS DAUGHTERS

From the standpoint of radiological impact, the most important potential problem from radon and its daughters is occupational exposure in the *uranium mining* industry. This has been discussed in detail in the section "Mining and Milling of Uranium Ores," elsewhere in this volume. Here we shall summarize by noting that in the U.S. a few thousand miners now require routine radiological monitoring.

Another source of radon is mill tailings. In some of the Rocky Mountain states, this has been a public health problem in recent years (Ref. 11, 12). A 1969 study indicated that background radon-222 concentrations in four study sites in Colorado and Utah were in the range of 0.4 to 0.8 pCi/liter. Directly over tailings, levels higher by a factor of about 10 were reported, with typical spatial distributions such that the area beyond about one-half-mile radius was not directly affected.

In late 1972, the AEC published proposed 'remedial action criteria' (Ref. 13) for the area around Grand Junction, Colorado. These are aimed at eliminating some of the more important sources of exposure to the general public from tailings. Remedial action would consist of removal of tailings, ventilation, shielding or use of sealants; such action is "suggested" when a radon daughter concentration exceeding 0.01 WL (or a level of 0.05 mR/hour external gamma radiation) is measured, and is "indicated" when 0.05 WL (or 0.10 mR/hour) is exceeded.



Radon occurs *naturally* in air wherever uranium-laden soils or rocks occur. Rates of emanation and radon concentrations have been measured by Pearson (Ref. 14). Near Chicago, radon was present one meter above the ground at levels which varied diurnally from about 0.1 to 1.4 pCi/liter. Emanation rates in regions where  $^{238}\text{U}$  mining is commercially feasible were found to be larger by as much as two orders of magnitude than rates near Chicago. (Ref. 14). However, there is not necessarily a direct correlation between emanation rates and radon concentrations near the ground.

## 5. MEASUREMENT TECHNIQUES

There are four distinct classes of measurement which we shall discuss here:

- i) Measurements of radon (gas) concentrations
- ii) Measurements of Working Level
- iii) Measurements of individual radon daughter concentrations.
- iv) Working Level Dosimetry

A fifth class of measurement, discussed in detail in Volume 4 ("Biomedical Instrumentation") of this Survey, is:

- v) Bioassay measurements (e.g., urine, breath) as indicators of exposure or body burden.

Some of the discussion here relies heavily on a 1972 summary of radon instrumentation written by A.J. Breslin of the U.S.A.E.C. Health and Safety Laboratory (Ref. 15). Another useful reference concerned with instrumentation is the 1963 IAEA Symposium (Ref. 16).

### a. Measurements of Radon (Gas)

Here we shall discuss methods for measuring radon gas activity as distinct from the activity of its daughters. This type of measurement is made by some of the more sophisticated ventilation engineers in studying mine-air quality; by those concerned with natural levels of radon gas in the air; to measure radium-226 from the emanation of its daughter radon-222; and when radon-222 is used as a tracer in atmospheric studies. Methods for measuring individual daughters will be discussed in a later sub-section.

There are two quite different approaches to the specific measurement of radon-222. In the first, equilibrium can be assumed to have become established between radon and its daughters; in the second, the daughters are removed

from the sample gas, after which either the decay of radon itself is detected or the daughters are allowed to ingrow again for counting.

Measurements of radon activity can be made with rather simple instruments, for routine monitoring measurements, or with quite complex instruments, usually used in research applications. We shall begin by discussing the two simplest techniques.

The most common simple instrument for measuring radon gas is the Lucas chamber (Ref. 17). The chamber itself is a small metal or glass cell with a flat glass bottom. Its shape is usually spherical but can be cylindrical or conical as well. The inside is lined with zinc-sulfide scintillator, and the scintillations are viewed and counted through the flat window by a photomultiplier tube. Filtered air samples can be drawn into the chamber using a pump, or alternatively by evacuating the chamber and then admitting a filtered sample through a valve. Typical chamber volumes are in the 100-200 ml range, and are thus small enough to be easily portable. Figure 4 shows the original chamber of Lucas.

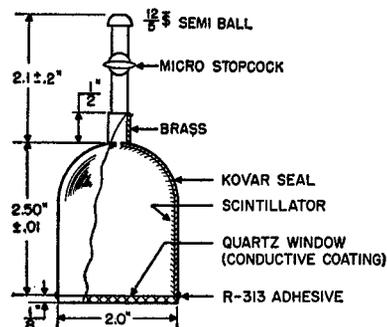


FIGURE 4. Lucas chamber. Photomultiplier tube, which views the  $\text{ZnS}(\text{Ag})$  scintillations through the quartz window, is not shown (Ref. 17).

The filter removes the radon daughters so that only the parent radon gas is admitted to the chamber. The radon gas subsequently decays and reaches equilibrium with its daughters. A difficulty in the original design was optimizing and stabilizing the detection efficiency for the daughters, which are usually electrically charged. When a conducting layer on the inside of the window is used, the charged daughters turn out to distribute themselves uniformly on the window and the  $\text{ZnS}(\text{Ag})$  walls. (According to a recent study (Ref. 18), this conducting layer is unnecessary if the photomultiplier is operated with a grounded photocathode.) The alpha radiation detected by the  $\text{ZnS}(\text{Ag})$  phosphor is a measure of the activity of  $^{222}\text{Rn}$ ,  $\text{RaA}$ , and  $\text{RaC}'$ . Interferences from thoron gas ( $^{220}\text{Rn}$ ) or even actinon gas ( $^{219}\text{Rn}$ ) are possible, since they are both also  $\alpha$ -emitters; these are small because of the short half-lives



(55 sec for thoron and 4 sec for actinon). There are no other significant interferences. The sensitivity of the method as usually used in the mines is about 10 pCi/liter (Ref. 15), although sensitivities an order of magnitude better are achievable with much longer counting times.

A detailed description of a technique for constructing Lucas chambers has been given in the Handbook of the EPA's Las Vegas National Environmental Research Center (Ref. 19). Also, detailed calibration procedures for Lucas chambers have been described in the APHA Standard Methods (Ref. 20), under the section on determining radium-226 in water by a radon-222 emanation technique.

An alternative to the Lucas chamber is the two-filter method (Ref. 21). A metal cylinder with a filter on each end is the basic sampling unit. Sample air is pumped through the cylinder (typically for 5 minutes at about 10 liters/minute). The upstream filter removes all particles, most importantly all radon daughters. Radon gas passes through, and inside the cylinder a small fraction decays to  $^{218}\text{Po}$  (RaA). Some of the RaA is deposited on the downstream filter, which is immediately counted for alpha activity (RaA's half-life being only about 3 minutes). Counting can be done with any of a number of instruments:

Thomas and LeClare (Ref. 21) used a ZnS(Ag)/photomultiplier system. The radon concentration must be calculated using an algebraic expression developed by Thomas and LeClare, which depends upon geometrical sizes, sampling rate, and counting time. Using a 52-in. long, 3.6-in. diameter tube, this technique is sensitive at the level of a few pCi/liter; calculations indicate that sensitivities as low as 0.1 pCi/liter could be achieved with a larger chamber, higher flow rate, and other small changes (Ref. 21). A variation of the method has been used for measurements of environmental radon concentrations <0.1 pCi/liter, and sensitivities as low as 0.01 pCi/liter are claimed (Ref. 59).

One disadvantage of this technique is the degree of care needed in sampling, because of the small amount of activity present on the downstream filter. Another is that at relative humidities below about 25% the method has been found to yield results up to almost 20% too high (Ref. 21). The reproducibility of the method has been studied by Breslin (Ref. 15), who finds 10% to 20% replication errors.

Figure 5 (from Ref. 22) shows a study of the accuracy of the two filter method, in which it is compared to measurements of flask samples analyzed in the laboratory using a pulse-type ionization chamber. The average precision of this method was about  $\pm 20\%$ .

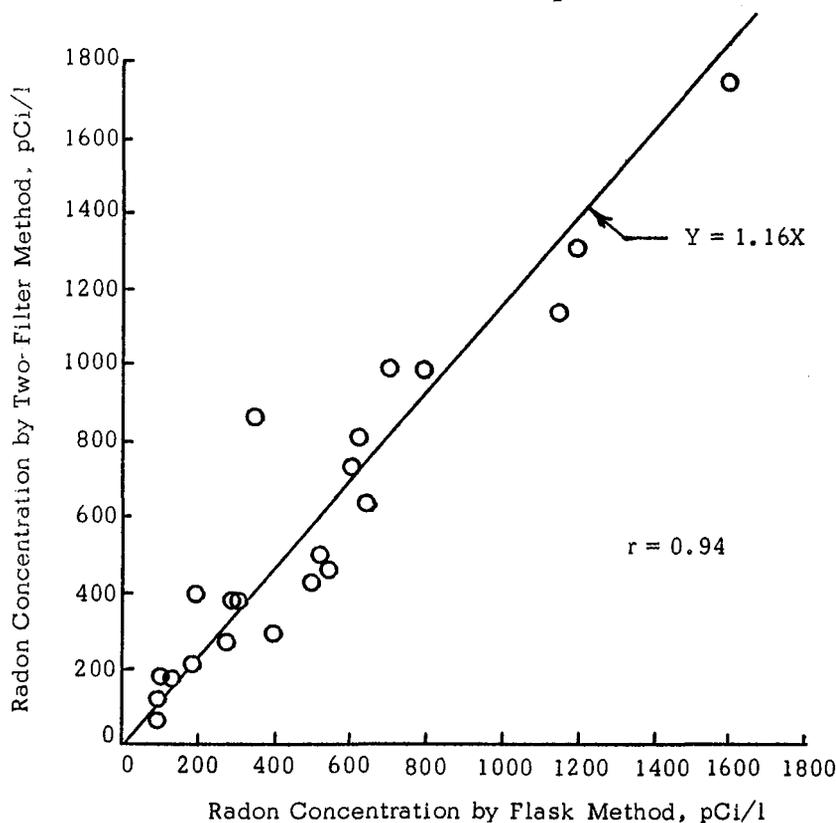


FIGURE 5. Comparison of Field Measurements by the Two-Filter Method and the Flask Method for Radon Determination (Ref. 22).



Both the Lucas-chamber and two-filter methods appear to be satisfactory for routine use in the uranium mines. Although the standard Lucas chamber technique is simpler, it does not have the intrinsic sensitivity of the two-filter system for research-type studies.

Another technique, with sensitivity similar to that of these methods, is the use of a pulse-type ionization chamber. A sample of gas, collected through a filter to eliminate radon daughters, is admitted to the ion chamber and allowed to come to equilibrium before counting. This technique is described in detail in the Procedures Manual of the U.S.A.E.C. Health and Safety Laboratory (Ref. 23).

More elaborate techniques have also been developed for high-sensitivity studies, and these will be discussed next.

Sensitive methods for radon sampling with cooled activated charcoal have been used for many years. At low temperatures such as that of dry ice ( $-78^{\circ}\text{C}$ ), gaseous radon rapidly adsorbs on activated charcoal; radon can be subsequently de-emanated at temperatures of about  $300^{\circ}\text{C}$ , collected, and counted.

The Intersociety Committee's compilation of Tentative Methods for radon (Ref. 24) describes one application of this technique. Figure 6 shows the collection apparatus. Filtered air is dehumidified in a drying column (e.g., Drierite) and any remaining water is trapped before the gas flows through the cooled activated charcoal. The cooling mixture is dry ice mixed with 1:1 chloroform and carbon tetrachloride (Ref. 25). Radon is transferred to the counting chamber along with helium carrier gas.

For counting of the gas, two different techniques are described by the Intersociety Committee (Ref. 24). In the first, the radon is transferred to a Lucas chamber, allowed to reach equilibrium with its daughters after a 4-hour wait, and counted. In the second, the collection trap is counted directly with a NaI(Tl) crystal, a photomultiplier, and a multi-channel analyzer. The 0.61 MeV and 1.76 MeV gammas from  $^{214}\text{Bi}$  (RaC) are counted, after equilibrium has been established.

The methods are both quite sensitive. The NaI(Tl) system can detect  $\sim 0.12$  pCi/liter of radon with a  $\pm 10\%$  error at 95% confidence (Ref. 24), while the Lucas chamber method has been used to measure levels as low as about 0.010 pCi/liter with errors in the  $\pm 0.005$  pCi/liter range (Ref. 25). The main disadvantage is, of course, that the apparatus is relatively sophisticated, expensive, and not very 'portable'. The NaI(Tl) gamma spectroscopy technique suffers from essentially no interferences, and the interferences in the Lucas chamber approach are also small.

Two other highly-sensitive methods using air filters are also recommended by the Intersociety Committee (Ref. 24). Neither is applicable for uranium-mine measurements. Each relies upon the existence of radioactive equilibrium between radon and its daughters in the sampled air, and as such each is only useful for approximate environmental measurements.

The first of these methods uses *alpha counting*. The short-lived radon daughters are collected on a 0.8 micron pore size filter, which is nearly 100% efficient and also has very little self-absorption during counting.

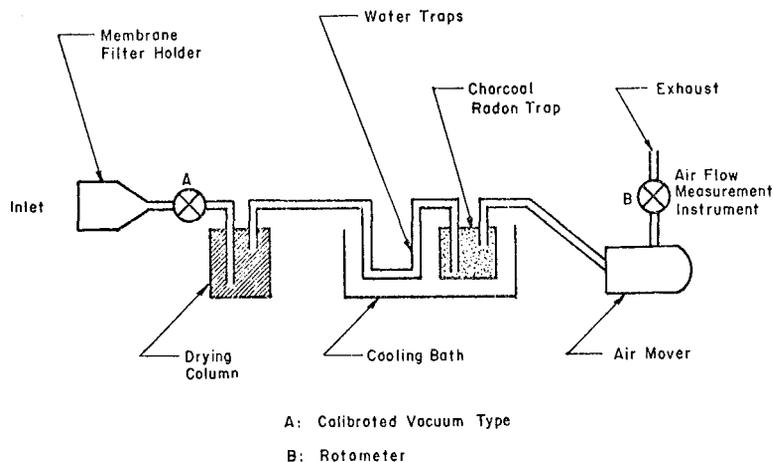


FIGURE 6. Sampling apparatus for collection of gaseous airborne radon.  
(from Ref. 24)



Because the half-lives in the RaA  $\rightarrow$  RaC' chain are short, the collection and decay rates reach equilibrium after a few hours. A 4-hour sampling time is recommended, at about 10 liters/minute. The filter is counted with a zinc-sulfide alpha scintillation counter and a 2" photomultiplier tube, or alternatively with an internal gas proportional counter. The alphas from the two daughters RaA and RaC' are counted. Using a counting time of 10 minutes, the lower detection limit is quoted as being about 0.030 pCi/liter, with counting error of  $\pm 10\%$  at 95% confidence. Interferences can occur from the long-lived daughters of thoron ( $^{220}\text{Rn}$ ), but these can be corrected for by re-counting the filter after a delay of 4 hours. Other alpha-emitters which might collect on the filter are plutonium-239, 240, 238; uranium-238 and 235, and radium-226, but none of these is usually present with sufficient activity to constitute an important background. Of course, the most important uncertainty in this type of measurement is the assumption that equilibrium exists in the air being sampled. In most environmental sampling situations, this is not true and difficult to establish.

In the second Intersociety Committee air-filter method, beta activity is counted on the filter (Ref. 24). Because betas are easier than alphas to count, one can use a thicker

filter and greater sampling volumes without problems of self-absorption or dust-loading. A positive-displacement blower forces air at 500 liter/minute through a 2" glass fiber filter. The activities to be counted are the betas from RaB and RaC. After 20 minutes of sampling and 1 minute for transfer, a 10 minute beta count is taken through 75 mg/cm<sup>2</sup> of absorber using an internal gas proportional counter or a GM counter. A concentration of 0.001 pCi/liter of radon in equilibrium with its daughters will yield 332.2 disintegrations of RaB and 416.9 of RaC in this time interval. A second count after 5 hours permits subtraction of possible thoron daughters. A more complex analysis procedure uses another 10-minute count after 1 hour to determine the RaB/RaC ratio, and hence the possible extent of disequilibrium in the original sample (Ref. 26).

Another, quite different monitor using Polaroid film and ZnS(Ag) phosphors has been developed by Bedrosian (Ref. 27). Figure 7 shows the device, consisting of fast Polaroid film in a holder containing two ZnS(Ag) disks, one covered by filter paper and the other not covered. The uncovered ZnS(Ag) phosphor responds to alphas from radon gas, RaA, and RaC' in the ambient air; the covered phosphor responds only to alphas from radon gas, which diffuses through the filter. The images on

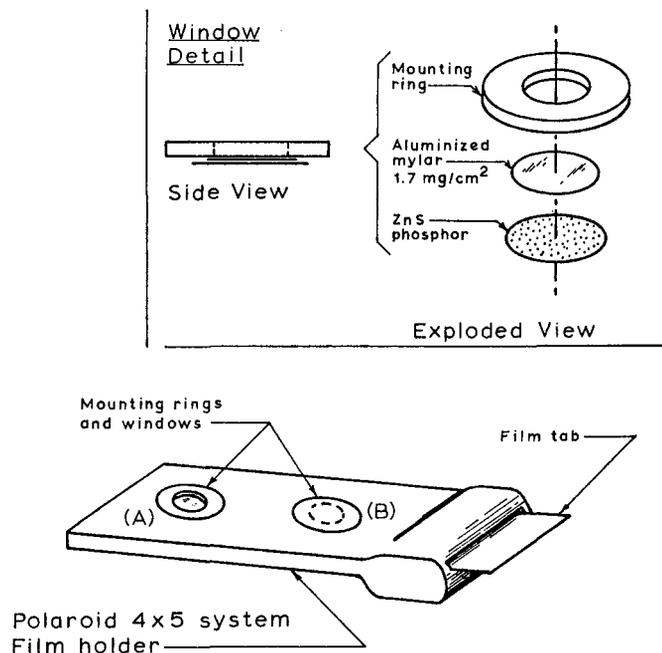


FIGURE 7. Schematic of Polaroid Land 4 x 5 film system used as a radon monitor. The film holder contains two aluminized mylar-covered ZnS(Ag) windows, one of which (A) is exposed and the other (B) covered by filter paper. Except for the filter paper, both windows are identical and consist of elements indicated in the inset (from Bedrosian, Ref. 27).



the film, exposed by the scintillations from the ZnS(Ag), are measured with a reflection densitometer. Bedrosian claims a lower limit of sensitivity for radon gas of 200 pCi/liter after 15 hours of exposure, and for radon daughters of less than 1 WL after 30 hours of exposure. This technique shows promise where levels are high, because the results are available within a few minutes after the end of an exposure, and the system is very inexpensive. Of course, it yields no information about the extent to which the ambient air being sampled is in equilibrium.

#### b. Working Level

Two types of Working Level (WL) measurements are required in uranium mines:

- i) Measurements of WL at a given time and place
- ii) Measurements of integrated exposure either for a worker or for a working area; these are usually expressed as Working Level Hours (WLH) or Working Level Months (WLM).

We shall discuss (i) here, leaving (ii) for a later sub-section.

The WL is defined as "any combination of radon daughters in one liter of air that will result in the ultimate emission of  $1.3 \times 10^5$  MeV of potential alpha energy" (Ref. 3). Of the daughters, RaB and RaC are beta/gamma emitters, so only the alphas from RaA and RaC' decay need be measured. However, it must be emphasized that the sum energy considered in the WL contains contributions from the RaC' alphas which arise from decay of the RaA, RaB, and RaC in the sample air. Figure 8 (from Ref. 2) shows the growth of Working Levels in initially pure radon.

The use of an air filter is a feature common to many of the methods which we shall discuss. The filter collects the three daughters (RaA, RaB, and RaC), after which the decays are counted. If only alphas are counted, then what matters is the activity of RaA and RaC', the alpha-emitting daughters. Figure 9 (from Ref. 1) shows the build-up and decay of alpha activity from individually isolated radon daughter isotopes, each with an initial decay rate of 10 dpm (= 4.5 pCi).

The most common method for measuring WL is the *Kusnetz method*. Originally developed in 1956 (Ref. 28), it has been the mainstay of WL monitoring in the uranium mines ever since, and is now recommended by ANSI as the "standard method" (Ref. 29). We shall begin by discussing it, before going on to recent or proposed improvements.

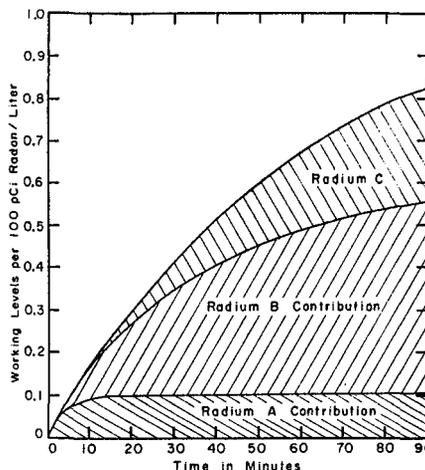


FIGURE 8. Growth of Working Levels in initially pure radon (e.g., freshly filtered air). Note that the earliest contribution to WL is from radium A, then radium B, and still later radium C (from Evans, Ref. 2).

The Kusnetz method employs an air sampler (pump and filter) and an alpha counter (usually of the ionization chamber or zinc-sulfide-scintillation type).

The American National Standards Institute standard method (Ref. 29) specifies sampling at 5 to 20 liters per minute ( $\ell$ pm) for five minutes; 10  $\ell$ pm was the most common rate at the time ANSI 7.1 was written, producing a total sample volume of 50 liters. Today,  $\sim 2$   $\ell$ pm is more commonly used, as we shall mention below. After a delay of from 40 to 90 minutes (most commonly 40), the *count rate* in counts per minute is measured, using a *rate meter*. After determining disintegrations per minute (dpm) by correcting for the efficiency of the alpha detector, a tabulated scale factor is used to relate dpm to the WL in the original sampled air.

The main feature which commends the Kusnetz method is its relative insensitivity to the concentration ratios of the three daughters RaA, RaB, and RaC. The intrinsic error from not knowing the concentration ratios is at most  $\leq 25\%$ . For example, suppose a 40-minute delay before counting; if RaA:RaB:RaC concentrations are in the ratios of 100:100:100, 100:90:80, 100:45:35, and 100:15:6, the intrinsic error in determining WL from the data is only +7%, +8%, +2%, and -7%, respectively (Ref. 31). Groer (Ref. 32) has shown that in very "young" air, such as air in which only RaA has had much chance to grow in from the parent  $^{222}\text{Rn}$ , the Kusnetz method underestimates the true WL by as much as 25%. Even so, since these errors are smaller than typical uncertainties in the way sampling represents true



concentration, the Kusnetz method can be said to be intrinsically accurate enough for most purposes.

The minimum sensitivity of the Kusnetz method as just described has been studied by Breslin et al. (Ref. 22) and shown in Figure 10.

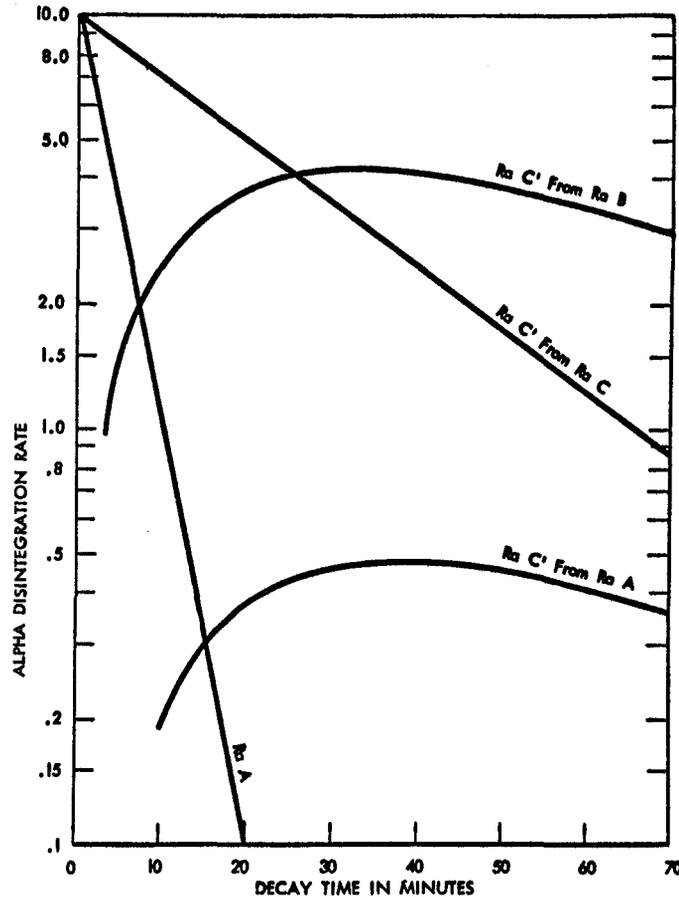


FIGURE 9. Build-up and decay of alpha activity from individual initially isolated radon daughter isotopes, RaA through RaC, with an initial disintegration rate for each isolated isotope of 10 disintegrations per minute (from Ref. 1).

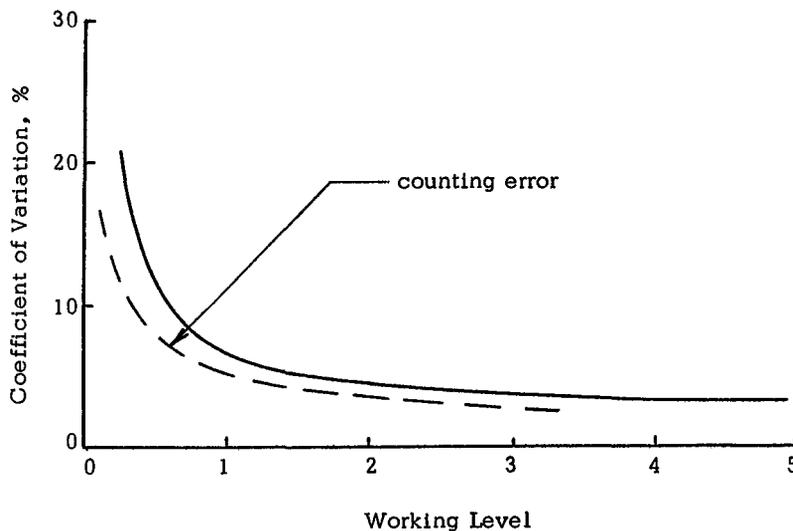


FIGURE 10. Precision for Measurements of Working Level by Kusnetz Method (from Ref. 22).



The precision is about  $\pm 15\%$  at 0.3 WL but de-generates rapidly at lower concentrations. This increased error is mainly dominated by statistical fluctuations in the rate measurement (Ref. 15) and is hence unavoidable. A full discussion by Loysen (Ref. 33) of the various sources of error in the Kusnetz method indicates that with appropriate care, errors from sampling can be kept smaller than the counting (statistical) fluctuation.

The rate-meter measurement is one source of error which can be improved upon, by counting for a fixed period instead of by measuring rate. To improve the method's sensitivity, one must either sample more air or count for a longer time. In a laboratory environment, Breslin (Ref. 15) counted for four minutes (from +38 to +42 minutes) using an alpha scintillation/scaler instrument. Using this "modified Kusnetz method," reproducibility was found to be  $\pm 4.2\%$ ,  $\pm 14\%$ , and  $\pm 35\%$  at mean levels of .041, .0029, and .00046 WL, respectively. This indicates that this modified method is intrinsically sensitive enough for almost any application.

There are problems with the Kusnetz method, however. The 10-lpm air pumps typically used are heavy and cumbersome, and there has been a recent switch to smaller, lightweight 2-lpm pumps (Ref. 31). At concentrations above about 0.3 WL, the 2-lpm and 10-lpm pumps, both sampling for five minutes, gave reasonably reproducible and comparable results (Ref. 31). Of course, at low WL ranges the lower volume of air sampled with the 2-lpm pump will seriously degrade the sensitivity.

Membrane filters have been most widely used because of their 99+% retention of submicron particles and because the particles are mostly deposited right at the filter surface, minimizing self-absorption of the alphas during counting (Ref. 30). Glass fiber filters, equally efficient as collectors, can suffer from more penetration and hence more important self-absorption corrections; but recent tests indicate that most commercial glass filters now have little problem with self-absorption (Ref. 31). Direct moisture on the filter face can cause absorption problems for any of the filters as well as pressure problems in pumping.

Air pump flowmeters are an especially tricky problem because their calibration is density-dependent, and hence will vary if an instrument calibrated at sea level is used at elevations well above sea level, where many radon measurements are made. The details of this problem are discussed in the Bureau of Mines Handbook (Ref. 31).

An advantage of the Kusnetz method is that the alpha-detection system, which must respond only to RaC' alphas (7.69 MeV), need

not have an energy-independent response. The detection efficiency at that one energy must be known, of course. The Bureau of Mines Handbook (Ref. 31) discusses both laboratory and field calibration procedures.

One need still awaiting a solution is that of a good, light-weight electronic scaler. We quote from Breslin (Ref. 15): "Commercial scalers employed by mine operators have not been found to be satisfactory, either being too slow in the case of mechanical counters or too bulky in the case of electronic counters." Breslin recommends the commercial development of an alpha counter weighing less than ten pounds, with 4-decade scaler display, 8-hour battery lifetime before recharge, and a variable preset timer.

Perhaps the biggest drawback of the Kusnetz method is the minimum 45-minute delay from start to finish. This inherent difficulty has stimulated the development of other WL techniques.

A method developed by Rolle in 1969 (Ref. 34) and described further in a 1972 paper (Ref. 35) makes possible much more refined measurements, using equipment identical to that of the Kusnetz method. Rolle describes how the choice of counting time affects systematic error, and indicates that intrinsic errors can be kept below about  $\pm 12\%$  with counting for 10 minutes after about a five minute wait. Rolle also discusses in detail the way volumetric and radiometric errors limit ultimate uncertainties to the  $\pm 20\%$  range.

The methods just described all require a considerable time delay between the start of sampling and the end of counting: the Rolle method takes about 20 minutes, the Kusnetz upwards of 45. This has motivated the development of several prototype "Instant Working Level Meters" (IWLM's).

Two different versions of such an instrument were developed in 1968-69 to provide for the rapid, automatic measurement of WL. The manufacturers were GeoCon Corp. (Ref. 36) and Bedford Engineering Corp. (Ref. 37), the latter working with an MIT group (Ref. 38). Neither unit is now commercially available.

The idea of the IWLM is to count the accumulating activity as it collects on a membrane filter. The RaA and RaC' alphas, and also the total beta activity, are counted with separate detectors right next to the filter. Electronic circuitry is used to calculate WL, which is displayed directly on a meter. The total time required for one measurement is about 4-5 minutes.

Unfortunately, Breslin (Ref. 15) indicates that "reproducibility, calculated from paired measurements, was about  $\pm 50\%$  for the GeoCon, and about  $\pm 100\%$  for the MIT-Bedford. . . .



Based on these tests, neither instrument has sufficient reliability for use in mines." The instruments discussed were also bulky, heavy (15-20 lb) and awkward to operate.

Further development work on an improved IWLM is being carried on by Groer (Ref. 39) at Argonne National Laboratory. The signal/background ratio will be improved by increasing the pumping rate to 12  $\mu\text{pm}$  (for a 4-minute sample time), and by using a very thin ( $\sim 0.003$  inch) plastic scintillator for beta detection, to decrease background from external gamma radiation. The proposed device will calculate not only WL, but also the three individual daughter concentrations, using a small digital calculator. The hoped-for sensitivities are about 0.01 WL and about 1 pCi/liter for each of RaA, RaB, and RaC. If successfully developed, this instrument will be of major importance for uranium-mine measurement.

Another development project is being carried out at Colorado State University by a group under Schiager (Ref. 40). A membrane filter sample is taken by manually turning a pump (one liter/stroke), and WL is determined approximately by measuring RaA and RaC' separately with a surface barrier detector. No beta radiation is measured, but WL is 'calculated' by an electronic weighting procedure: the sum ( $\text{RaA} + 8\text{RaC}'$ ) or ( $\text{RaA} + 10\text{RaC}'$ ) is used as a measure of WL. Figure 11 shows how such a procedure yields results within  $\pm 10$  to  $\pm 15\%$  of WL over most of the range from equilibrium to complete disequilibrium. Prototypes are now being field-tested (early 1973). The chief advantages, if successful, will be light weight, ruggedness, low unit cost, and rapidity of measurement (about two minutes).

### c. Measurements of Individual Radon Daughters in Air

The ability to isolate the relative activities of individual radon daughters in an unknown atmosphere is of great use. In an equilibrium atmosphere, of course, all of the short lived daughters have activities equal to that of the parent; the relative activities are used to determine the degree to which equilibrium has been achieved. Another measurement problem is the determination of the "uncombined RaA fraction," that is, the fraction of RaA in the sample atmosphere which is free and not combined with particulate matter.

Even with its short (3-minute) half-life, the first radon daughter, RaA ( $^{218}\text{Po}$ ), is not always in equilibrium with the parent radon. The subsequent daughters are nearly always only partially ingrown, especially in well-circulated air. This is true in environmental as well as in uranium-mine atmospheres.

We have already discussed one technique (Ref. 26) for measuring the RaB/RaC ratio using the air filter/beta counting technique. This is a complex technique which is not very sensitive unless the RaB/RaC ratio is very large or very small. It is not often used for these reasons.

Two more useful approaches are the *Tsvoglou method*, involving several counting intervals, and the use of *alpha spectroscopy*.

The Tsvoglou method (Ref. 41) is one of the oldest techniques for radon daughter determinations. It employs an air filter and

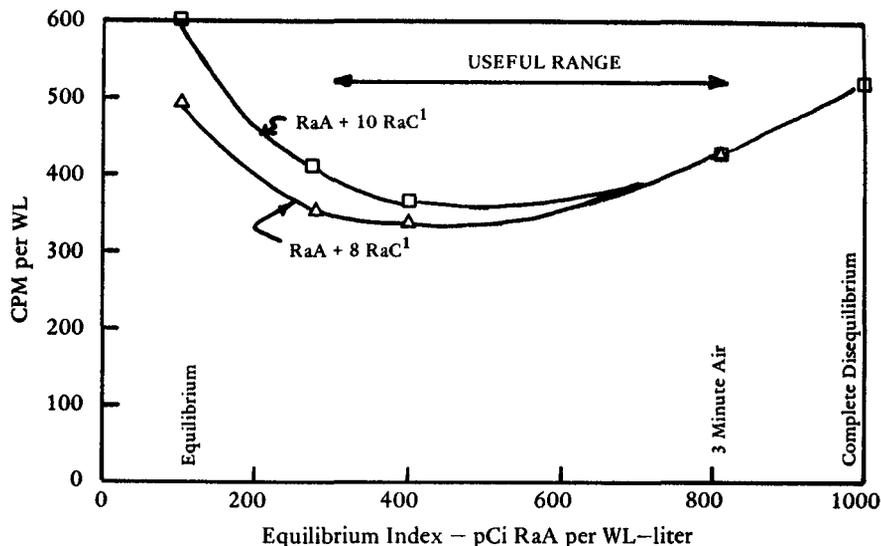


FIGURE 11. Theoretical survey meter response (cpm per WL) as a function of the degree of equilibrium of airborne radon progeny (pCi RaA per WL-liter) (from Ref. 40).



an alpha counter/rate-meter, apparatus similar to that required by the Kusnetz method for WL, and the technical considerations required to obtain good data are similar also. One key difference is that the alpha counter response must be energy-independent, which is not required in the Kusnetz method. (The Bureau of Mines Handbook [Ref. 31] discusses calibration procedures in detail.) After the air sample is taken (typically for 10 minutes at 5 lpm), the alpha count rate is measured at three later times: after delays of 5, 15, and 30 minutes. RaA, RaB, and RaC concentrations are then determined by solving three simultaneous equations.

Unfortunately, studies by Breslin et al. (Ref. 22), with 100-liter sample volumes at concentrations of 0.5 to 5 WL, indicate that reproducibility is poor for identical repeat measurements: RaB/RaA ratios and RaC/RaA ratios had replication errors in the 15-25% and 25-35% ranges, respectively.

This problem has motivated work on improvements in the method. The simplest improvement is similar to the way the original Kusnetz method can be improved: use of count totals rather than count rates. Thomas (Ref. 42), using a five minute sampling time at 10 lpm, has taken count totals in three time intervals: 2 to 5, 6 to 20, and 21 to 30 minutes. He is able to express RaA, RaB, and RaC concentrations directly in terms of the three count totals: essentially, the simultaneous equations are solved and the matrix inverted. The precision of the determinations is good according to calculations. Quoting Breslin (Ref. 15): "At . . . 0.3 WL and a radon daughter ratio of 100:30:10, . . . the calculated precisions for RaA, RaB, and RaC measurements are 4%, 4%, and 12%, respectively.

This modified-Tsivoglou technique can also be used for WL measurements, and the results are comparable to those obtained with the modified Kusnetz method. However, because of the added complication, the method is probably not to be preferred when WL measurements alone are required, unless accuracy is at a high premium.

A theoretical analysis of the Tsivoglou method, extending the treatment to any number of general counting times, has been given by Martz et al. (Ref. 43).

*Alpha spectroscopy* is another useful approach for measuring individual radon daughter concentrations. Instead of using the air filter/alpha counting technique, one can substitute an alpha spectrometer for the alpha detector system. Several types of alpha spectrometers have been developed, all of which have been described elsewhere in this volume (see "Alpha Particle Instrumentation"). The

best resolutions are now obtainable with solid state detector systems. The alpha lines requiring resolution and measurement have energies of 6.00 MeV (RaA) and 7.69 MeV (RaC').

The method developed by Martz et al. (Ref. 43) uses a solid-state detector and multichannel analyzer, with a 4.3-lpm collection rate onto a membrane filter. Separate determinations of the RaA and RaC' alpha activities are made at two times, 5 and 30 minutes after sampling. This leaves only two simultaneous equations (compared to the three required in the Tsivoglou method) to determine the relative daughter concentrations.

The main advantage of this method is its improved accuracy in determining the short-lived (3-min) RaA: the RaA alpha is counted directly. In experimental comparisons with the Tsivoglou method, Martz et al. (Ref. 43) found that the spectroscopic method was significantly more precise for both RaA (8% standard deviation compared to 29%) and RaC (14% compared to 27%), and comparable for RaB (12%). This approach thus appears to be promising, albeit one requiring more expensive and elaborate instrumentation. It is obviously possible to use alpha spectroscopy in uranium mines, but this is probably difficult because of the inconvenience. The better accuracy for RaA would tend to be nullified by the delay in taking mine samples out to surface counting equipment, and hence this technique is probably most applicable to environmental samples.

A more complicated analysis technique has been developed by Raabe and Wrenn (Ref. 44), who generalize the Tsivoglou method by performing a regression analysis to fit mathematically the observed count totals during various counting periods. Thus one would not be limited to three count periods and three simultaneous equations. Also, simultaneous determinations of thoron daughters are possible. Measurements with the system over 7 time intervals (the last 3 hours after taking a one-minute, 7.5-liter sample) are clearly more accurate than those of the Tsivoglou method, but the complicated analysis is probably only useful when research work requires high sophistication and accuracy.

Before leaving the subject of individual radon daughter measurements, a brief discussion of "uncombined RaA fraction" determinations will be given. The ICRP (Ref. 8) has noted that the fraction of RaA which is not combined with particulate matter seems to play a major role in the radiobiological impact of the radon daughters. One possible explanation for this is that they are exceedingly active, owing to their high diffusion velocity. This has motivated attempts to measure this uncombined fraction (f). These measurements are mostly performed for research purposes rather than in routine monitoring.



Duggan and Howell (Ref. 45) describe a system in which two filters sample the atmosphere side by side. One of them is "preceded by a diffusion battery designed to remove most of the unattached daughters but hardly any of the attached ones" (Ref. 45). RaA is distinguished from RaC' by alpha spectroscopy. Another method has been described by Fusamura and Kurosawa (Ref. 46), in which gas is passed through a diffusion tube, and the differing diffusion coefficients are relied upon to bring about a partial separation. In this way, f-values in the range 6 to 25% were measured to within better than a factor of 2. Another device based on diffusion has been described by Mercer and Stowe (Ref. 47). It is shown in Figure 12 (from Ref. 48). "Air enters through an orifice in the center of the upper of two discs and flows radially outward between the discs and down past the edge of the lower disc" (Ref. 48). As much as 80% of unattached RaA atoms (and typically 60 to 70%) can be collected on the discs (Ref. 47). Using this system, George and Hinchliffe (Ref. 48) measured f-values below 0.10 with precisions in the range of  $\pm 0.007$  to  $\pm 0.014$ .

#### d. Personnel Dosimeters

The usefulness of *Working Level dosimetry* for occupational workers is obvious: a portable instrument which could accurately integrate WL exposure over time would help to provide for the radiological protection of any occupationally exposed individuals (e.g., miners). Present dosimetry is done by measuring WL and correlating with the amount of time spent in each of the various working areas by a worker. This present method has, of course, served a valuable purpose over a long period, and has certain advantages over dosimeters, among which are that measurements are made by trained personnel with less inconvenience for the miner. In any event, such WL measurements will always be required to supplement a personnel dosimetry system even if it achieved wide acceptance.

The requirements of a personnel dosimeter are that it should be sensitive down to an integrated exposure of, say, less than about 1 WL-hour; that it be capable of weekly or bi-weekly readout; that it properly sample the air being breathed; and that it be light, rugged and failsafe.

A number of development efforts in recent years have been directed toward this problem. White of the U.S.A.E.C. Health and Safety Laboratory performed three sets of evaluations of several of the dosimeters (Ref. 49, 50, 51), which Breslin (Ref. 15) has summarized. The various dosimeters employed different kinds of detectors: some had pumps and filters and some sampled passively; some were sensitive

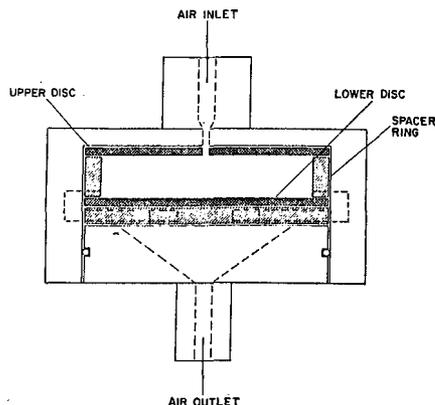


FIGURE 12. Diffusion sampler for uncombined radon daughters (from Ref. 48).

to radon gas, some to the radon daughters, and one to both. Their properties are given in Table 2.

White's tests of the two dosimeters designed to measure *radon gas* showed that neither performed satisfactorily in laboratory standardization tests (Ref. 49). One of these was an alpha-track-count film detector from Eberline (Ref. 52), the other a ZnS(Ag) scintillator with film recording from NYU (Ref. 53). One gave responses which varied by factors of as much as 10 when repeat runs were taken; the response of the other varied by a factor of as much as 3, but the dosimeter only increased its response by a factor of about 3 when radon concentrations increased by a factor of 10.

Of the six dosimeters responsive to *radon daughters* (or radon gas plus daughters), the responses of 5 were judged to be much poorer than 'satisfactory' (Ref. 49, 50, 51). The best performance was that of the HASL dosimeter, called the 'MOD'. The reason for this is partly that the MOD's design occurred later than, and was able to profit from, the designs of some of the others.

There were a variety of reasons for the poor performances of the other dosimeters. These are the units from Oak Ridge (Ref. 54), MIT, Colorado State University (Ref. 55), and General Electric (Ref. 56, 57). In the mine tests, the harsh conditions of use (mechanical abuse, high humidity, corrosion problems, mud on the detectors and filters, pump failure) caused many problems. For most of the unsatisfactory units, reproducibility was poor; and the results of a comparison study (in which two similar units were worn together by the same miner) were also poor.



**TABLE 2.**  
*Radon Dosimeters*

Source	Ref.	Atmospheric Component Detected	Type of Detector	Location		Pump Air Flow (liters/min)	Effective Range of Measurement	Total Wgt. (oz)	Period of Operation (hours)
				Pump	Detector				
Health & Safety Lab.	51	Rn dtrs	TLD (LiF)	belt	hat	0.100	3 - 1000 WL-hr	28	> 9
Oak Ridge Natl. Lab.	51, 54	Rn dtrs	$\alpha$ track etch	hat	hat	0.021	1 - 400 WL-hr	2	>13
Mass. Inst. of Tech.	50	Rn dtrs	TLD (CaF <sub>2</sub> :Dy)	belt	hat	0.100	.026 - 2 x 10 <sup>5</sup> WL-hr	7	9
Colo. State Univ.	49, 55	Rn dtrs	TLD (LiF)	belt	hat	0.170	.025 - 6 x 10 <sup>6</sup> WL-hr	24	40
Eberline Inst. Co.	49	Rn dtrs	$\alpha$ track count	belt	belt	0.015	4 - 600 WL-hr	12	10
General Elect. Co.	49, 56	Rn and Rn dtrs	$\alpha$ track etch	—	hat	passive	5 - 100 WL-hr	<1	unlimited
Eberline Inst. Co.	49, 52	Rn	$\alpha$ track count	—	hat	passive	4 - 1000 $\frac{\text{pCi-hr}}{\text{cc}}$	1	unlimited
New York Univ.	49, 53	Rn	scint. +film	—	—	passive	10 - 200 $\frac{\text{pCi-hr}}{\text{cc}}$	4	unlimited



Unfortunately, the performance even of the MOD dosimeter was still less than fully satisfactory in White's last test (Ref. 51). Figure 13 (from Ref. 51) shows White's mine data, comparing response to WL-hours of exposure. The paired measurements joined by vertical lines denote data from duplicate dosimeters worn by the same miner. Among the problems was the pump, which was specially redesigned after the experience with the earlier dosimeters, but which still suffered from occasional leakage of dirt into the pump casing. Also, dosimeter response to external (beta and gamma) radiation was a problem. Both of these can be corrected, the latter by use of a normal TLD dosimeter to measure the beta-

gamma background. The replication error of the MOD dosimeter was about 15%. Work is now underway at HASL (Ref. 15) to improve the pump still further.

A research program on radon dosimetry based on an alpha track-etch method is now going on under Benton at the University of San Francisco (Ref. 58). This approach is similar to that used in the General Electric dosimeter (Ref. 56), one of those tested and found unsatisfactory by White (Ref. 49). Small plastic polymers have been successfully used for dosimetry in space applications, and Benton's project hopes to improve their applicability in uranium-mine dosimetry.

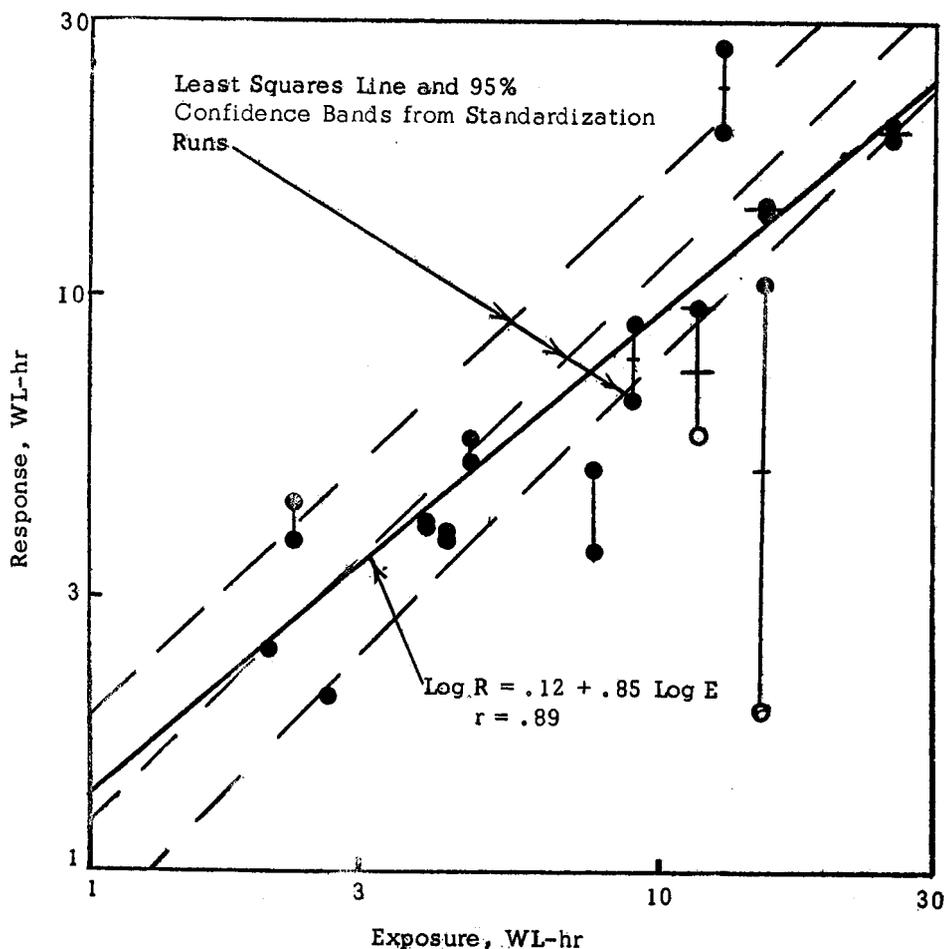


FIGURE 13. MOD Dosimeter - Mine Data (from Ref. 51). The paired measurements joined by vertical lines denote data from duplicate dosimeters worn by the same miner. The solid line shows the least squares best fit. The dashed lines show the mean and 95%-confidence level from laboratory standardization runs.



## 6. SUMMARY AND CONCLUSIONS

In this section, we have attempted to outline the various techniques for measuring radon-222 and its daughters in uranium mines and in environmental media. There are several quite different measurement problems, and we shall summarize the situation in each area separately:

### a) Radon as Gas

Two simple methods exist for radon measurements in the concentration range down to below 10 pCi/liter. These are the Lucas chamber (Ref. 17) and two-filter method (Ref. 21). Both appear to be satisfactory for routine use in the uranium mines, and each can be modified for sensitivities below 1 pCi/liter. For environmental measurements, where sensitivities well below 0.1 pCi/liter are sometimes required, several more elaborate methods have been developed. The situation appears to be satisfactory, since the elaborate methods, usually used only in research applications, probably do not merit significant improvement effort at this time.

### b) Working Level Measurement

The Kusnetz method (Ref. 28), used for many years as a standard technique in the uranium mines, is not sensitive enough to measure WL in the range below about 0.3 WL. The 'modified Kusnetz method' (Ref. 15) and the method developed by Rolle (Ref. 34, 35) are both sensitive enough ( $\leq 0.01$  WL) to meet almost any need. Unfortunately, although the other components of the measurement system are available, there does not yet exist an alpha-counter-with-scaler adequate to the task for use in the mines. Furthermore, neither of the methods gives an immediate answer, the delays being about 45 minutes (modified-Kusnetz) and 20 minutes (Rolle).

A more rapid WL monitor has still not been successfully developed, although some progress in this regard has occurred recently. In particular, the development project under Schiager (Ref. 40) may soon produce an instrument which is portable, simple, and inexpensive (although not as accurate as might ultimately be desired). Also, the development project under Groer (Ref. 39) shows great promise for the combined measurement of WL and individual daughter concentrations.

One point which must be borne in mind is that the total market even for an excellent WL meter is probably quite limited, so that commercial exploitation might not occur once an instrument is developed.

On the other hand, the assessment of occupational integrated exposure will probably be performed for some time to come by combining WL area measurements with worker area-time records. Thus the premium on a rugged, reliable, accurate and rapid instrument is still important.

### c) Individual Radon Daughters

A number of methods, all variations upon the long-established Tsivoglou technique (Ref. 41), rely upon measuring activity collected on an air filter. To determine the activity ratios of several daughters, several measurements at different times (after collection) are required. The most precise of these techniques, that of Martz et al. (Ref. 43), uses alpha spectroscopy instead of alpha counting.

The usefulness of instruments of this type is undisputed, but their value is to mining ventilation engineers rather than for broad-based radiological monitoring. Unfortunately, this is a small market, which would probably not merit the commercial development of a fully-automatic instrument. Such an instrument might have a pre-set clock to measure the air sample at the appropriate times; and it might calculate the RaA/RaC and RaA/RaB ratios automatically, as well as the WL value. This seems like an instrument easy to design and build but unlikely to be developed in the near future. Alternatively, the development project under Groer seems possibly capable of filling this need, since the instrument is designed to yield individual radon-daughter concentrations down to the level of 1 pCi/liter.

### d) Personnel Dosimeters

The MOD dosimeter under development at HASL (Ref. 51) seems to be on the verge of success, in which case it should be given extensive field tests in the mines. The fact that several other prototype dosimeters did not operate satisfactorily should not discourage further attempts, since some of the other techniques deserve another try. In particular, the polymer alpha-etch technique being studied by Benton (Ref. 58) seems capable of possible application, as does the Oak Ridge alpha track count technique (Ref. 54).



INSTRUMENTATION  
FOR ENVIRONMENTAL  
MONITORING

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In conclusion, much instrumentation in this area of measurement is less than satisfactory. We have attempted to illustrate those techniques which show the greatest promise for further exploitation, but ultimately the problem with all of the instruments is that their expense will limit their use, and their limited potential use will discourage commercial development -- unless use of an instrument is mandated by the Bureau of Mines. Finally, the need for instruments which are simple to operate in the mines must be emphasized, since the general lack of skill among many mine monitoring personnel is recognized by all.

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University



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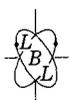
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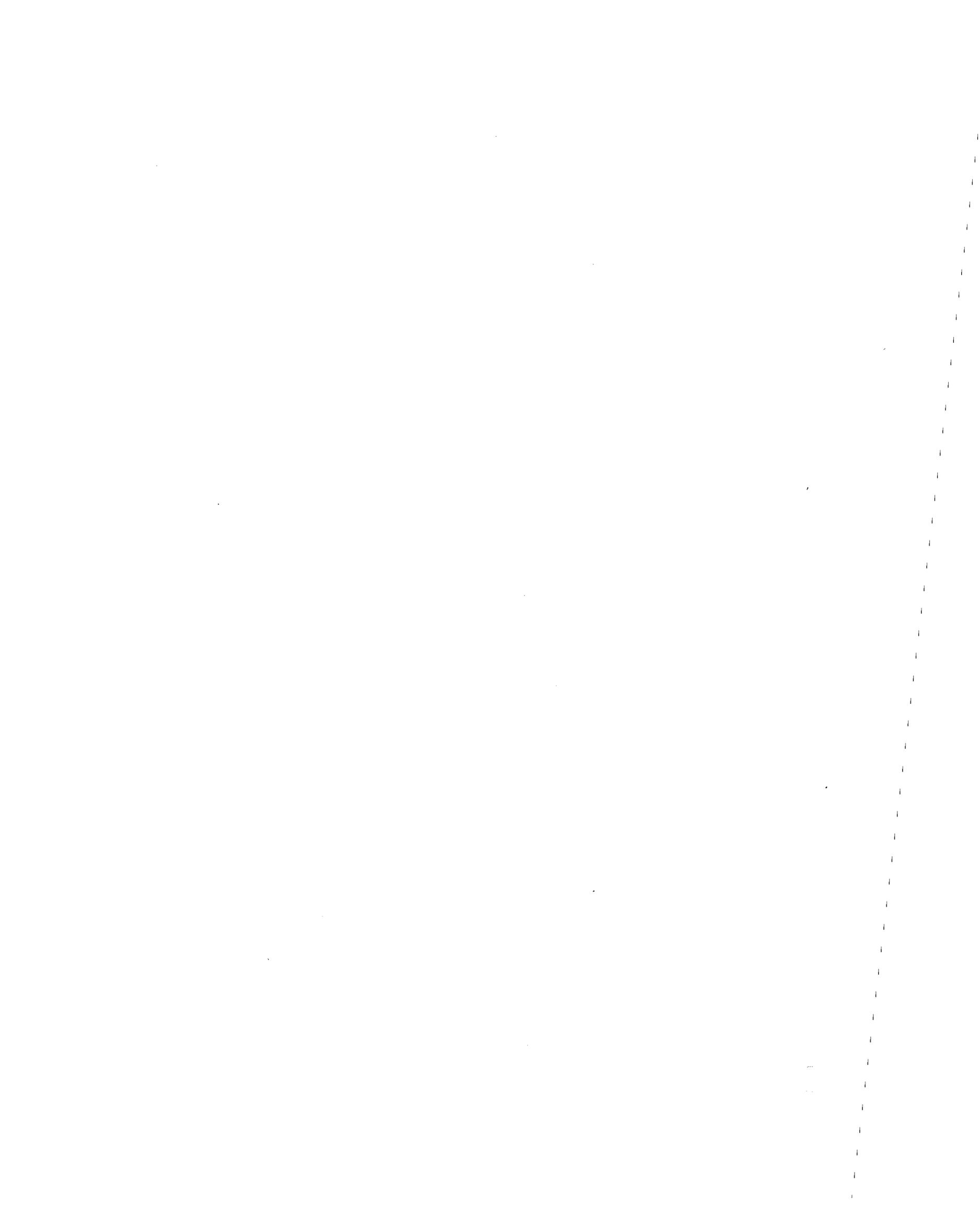


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## RADIUM

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## 1. INTRODUCTION

Radium is, of course, perhaps the most popularly familiar of all the radioactive elements. Its use in cancer radiotherapy was one of the earliest "miracles" among the applications of nuclear physics, and its toxicity when taken internally is equally well-known, from the case of the radium watch-dial painters.

In this section a discussion will be given of the measurement of radium concentrations in environmental media, primarily for environmental radiological protection. We shall concentrate on methods for determining low levels of radium in water and certain food-chain media, but shall not treat in detail bioassay methods, which are covered in Volume 4 ("Biomedical Instrumentation") of this Survey. While the main emphasis will be on radium-226, the other important isotopes (223, 224, 228) will be discussed where appropriate.

First a review of the physical characteristics of the important radium isotopes will be given; next, the radiation protection guides for radium concentrations will be discussed, and the ranges of radium-226 concentrations in the natural environment will be presented. Various radium measurement techniques will then be discussed.

Discussions which are important for a more complete picture of the measurement techniques dealt with in this section are found in two other sections of this Volume. These sections are "Radon-222 and Its Daughters," in which the uranium-238 decay chain and the instrumentation for radon-222 measurements are discussed; and "Alpha Particle Instrumentation," in which instruments for detecting alphas are treated in detail.

## 2. PHYSICAL CHARACTERISTICS

Radium occurs naturally in four isotopic states, each of which is a member of a naturally-occurring decay chain:

(1) Radium-226 (half-life = 1602 years) is a member of the uranium-238 chain. Its immediate parent is the alpha-emitter thorium-230 (80,000 years) and its daughter is the inert gas radon-222 (3.82 days). The  $^{226}\text{Ra}$  decay to  $^{222}\text{Rn}$  is by alpha-emission (94.6% of the decays by a 4.78-MeV  $\alpha$ ; the remaining 5.4% by a 4.60-MeV  $\alpha$  followed by a 187-keV gamma).

(2) Radium-223 (half-life = 11.4 days) is a member of the uranium-235 chain. Its immediate daughter is radon-219 (half-life = 4.0 seconds), an inert gas known historically as 'actinon'. The  $^{223}\text{Ra}$  decay to  $^{219}\text{Rn}$  is by alpha emission. The most important  $\alpha$  energies and branching ratios are: 5.75-MeV (9%); 5.71-MeV (52%); 5.61-MeV (25%); 5.54-MeV (9%). Each  $\alpha$  is accompanied by a gamma before the ground state of  $^{219}\text{Rn}$  is reached.

(3) Radium-228 (half-life = 6.7 years) is a beta-emitting member of the thorium-232 chain. Its low energy beta ( $E_{\text{max}} = 55\text{-keV}$ ) leads to actinium-228 (6.1 hours), which in turn beta-decays to thorium-228 by one of a large number of beta energy spectra, the most energetic of which is 2.15 MeV. Thorium-228 (1.9 years) then decays by  $\alpha$  emission (5.4 MeV) to radium-224.

(4) Radium-224 (half-life = 3.64 days) decays by  $\alpha$  emission to the gas radon-220, historically 'thoron' (55 sec). The decay products are a 5.68-MeV alpha (94%) or a 5.44-MeV  $\alpha$  followed by a 241-keV gamma (6%).

This information is summarized in Table 1 (information taken from Ref. 1)

TABLE 1.

*Some Physical Properties of Radium Isotopes (from Ref. 1)*

Isotope	Half-Life	Member of Which Decay Chain	Immediate Parent	Immediate Daughter	Main Decay Products and Energies (for $\beta$ , $E_{\text{max}}$ )	
					Branching Ratio	Decay Products
Radium-223	11.4 days	U-235 chain	Thorium-227 (18.2 days)	Radon-219 (4.0 sec)	( 9%) (52%) (25%) ( 9%)	5.75-MeV $\alpha$ + 338-keV $\gamma$ 5.71-MeV $\alpha$ + 270-keV $\gamma$ 5.61-MeV $\alpha$ + 154-keV $\gamma$ 5.54-MeV $\alpha$ + 122-keV $\gamma$
Radium-224	3.64 days	Th-232 chain	Thorium-228 (1.9 years)	Radon-220 (55 sec)	(94%) ( 6%)	5.68-MeV $\alpha$ 5.44-MeV $\alpha$ + 241-keV $\gamma$
Radium-226	1602 years	U-238 chain	Thorium-230 (80,000 years)	Radon-222 (3.82 days)	(94.6%) ( 5.4%)	4.78-MeV $\alpha$ 4.60-MeV $\alpha$ + 187-keV $\gamma$
Radium-228	6.7 years	Th-232 chain	Thorium-232 (14 billion years)	Actinium-228 (6.1 hours)	(100%)	55-keV (max) $\beta$



Chemically, radium is an alkaline earth element in the same family with magnesium, calcium, strontium, and barium. Its chemical similarity to calcium leads to its principal toxic effect, the preferential lodging in bone upon ingestion.

An interesting historical note is that 1 curie was originally defined as the activity of 1 gram of  $^{226}\text{Ra}$ . Small shifts in the definitions have occurred since then, so that now radium-226's specific activity is about 0.988 Ci/gram. However, today radium-226 is still sometimes measured in, for example, pg/liter instead of pCi/liter . . . these are numerically almost equal.

### 3. RADIATION PROTECTION GUIDES

Radium's toxicity depends upon whether its form is soluble or insoluble. When radium-226 is soluble, for example, it is a much more severe problem upon ingestion by mouth. These differences have been taken into account in the establishment of the radiation protection guides.

For 168-hour occupational exposures, the International Commission on Radiological

Protection has established maximum permissible concentrations in air and water  $(\text{MPC})_a$  and  $(\text{MPC})_w$ ; these are given in Table 2, from ICRP Publication 2 (Ref. 2). The maximum permissible body burdens (MPBB) are also given. Note that for individuals in the general public, the applicable  $(\text{MPC})$  values are a factor of 10 smaller than those tabulated in Table 2; and for exposure to a suitably large sample of the general public, another factor of 3 smaller still.

With that factor of 30 reduction, the  $(\text{MPC})_w$  for soluble  $^{226}\text{Ra}$  in drinking water for a large population would be  $100/30 = 3.3$  pCi/liter. The 1962 U.S. Public Health Service Drinking Water Standard (Ref. 3) adopts that very value (actually, 3.0 pCi/liter) as its recommended limit. The P.H.S. recommends "gross-alpha" measurements for routine screening of water samples; specific analyses for radium is only required if gross-alpha exceeds 3 pCi/liter. Gross-alpha measurements, which can be performed using any of several well-established methods, are discussed elsewhere in this Volume ("Alpha Particle Instrumentation").

TABLE 2.

*168-Hour/Week Occupational Maximum Permissible Concentrations in Air and Water, and Maximum Permissible Body Burdens (from ICRP Pub. 2, Ref. 2)*

Isotope	Soluble or Insoluble	Critical Organ	$(\text{MPC})_w$ pCi/liter	$(\text{MPC})_a$ pCi/liter	MPBB pCi
Radium-223	soluble	bone	7,000	0.6	50,000
Radium-224	soluble	bone	20,000	2.0	60,000
Radium-226	soluble	bone	100	0.01	100,000
Radium-228	soluble	bone	300	0.02	60,000
Radium-223	insoluble	lung GI(LLI)	-- 40,000	0.08 --	-- --
Radium-224	insoluble	lung GI(LLI)	-- 50,000	0.2 --	-- --
Radium-226	insoluble	lung GI(LLI)	-- 300,000	* --	-- --
Radium-228	insoluble	lung GI(LLI)	-- 300,000	0.01 --	-- --

\* Bernard and Ford (Ref. 4) recommend a value of 0.02 pCi/liter.  
GI(LLI) refers to gastrointestinal tract (lower large intestine).



#### 4. SOURCES AND LEVELS OF ENVIRONMENTAL RADIUM

Wherever uranium or thorium occurs naturally in the environment, the various radium isotopes will inevitably be found as well. Generally, the decay of the parents in the chain will proceed down through several steps, during all of which time the heavy nucleus will usually remain immobile in the original location. Each of the decay chains ultimately leads to one of the radons (222, 220, 219), and these gases can and do migrate. They emanate from surface layers or liquids, and are one of the more prominent external manifestations of the natural radioactivity present.

Another important migration mechanism is leaching: Water-soluble or reactive members of the chain, including radium, can be leached out into the ground water. This leaching is a significant radiological effect in some areas. For example, although radium-226 concentrations in domestic tap water are typically in the range from 0.01 to 0.2 pCi/liter (Ref. 5), concentrations in the range of 1 to 5 pCi/liter occur in several areas and concentrations in the 10 to 50 range are occasionally found (Ref. 7). Because of the shorter half-lives of the other radium isotopes, the long-lived  $^{226}\text{Ra}$  is the dominant isotope. However,  $^{228}\text{Ra}$  and  $^{224}\text{Ra}$  are sometimes found at levels comparable to those of  $^{226}\text{Ra}$  (Ref. 8).

Radium enters the human food chain in many ways. Plant uptake of radium directly from the soil, or from the water, produces small but measurable radium concentrations in foods. For example, a 1969 hospital diet study in California (Ref. 9) found daily per capita intakes of  $^{226}\text{Ra}$  in food in the range of 0.5 to 2 pCi/day. Studies in Italy (Ref. 10) of food from sources not known to be artificially contaminated showed the following approximate specific activities for radium-226: 0.4 pCi/kg in fruits; 0.4 to 0.7 pCi/kg in potatoes; 0.7 to 1.2 pCi/kg in green vegetables and lettuce; 3.6 to 3.8 pCi/kg in flour; and 5.8 to 6.9 pCi/kg in eggs.

In air, concentrations of radium itself are usually minute. Mechanisms which might produce airborne (particulate) radium concentrations would probably include such local effects as duststorms. Studies of airborne radioactivity (Ref. 11, 12) do not mention radium at all. Of course, the emanating gases (radon-222, 220, 219) and their daughters are widely distributed in measurable concentrations.

In soils, radium concentrations vary in a way correlated with the natural uranium/thorium content. Pearson (Ref. 13) found  $^{226}\text{Ra}$  concentrations in the range of 0.1 to 0.8 pCi/gram (dry soil), in several samples

scattered through the U.S., but as high as 15 pCi/g in uranium mining areas. In uranium ores, the  $^{238}\text{U}/^{235}\text{U}$  atomic ratio is constant at about 138.5/1 (Ref. 1). In equilibrium, the  $^{226}\text{Ra}/^{223}\text{Ra}$  atomic ratio is about 20/1 (Ref. 14), and one would thus expect a comparable ratio from uranium-mill effluents. This is seldom the case, however, because during the sulfuric acid leaching process used in most mills, radium is barely soluble but actinium-227 and thorium-227 dissolve quantitatively. These nuclides then give rise to their daughter  $^{223}\text{Ra}$  in the aqueous phase (Ref. 14). The bulk of mill effluent radium activity turns out to be radium-223.

This brief summary is intended only to point out the range of concentrations of radium in typical sample media. Most of the levels mentioned are 'natural,' in that they are not enhanced by man's use of radioactivity . . . which is not to imply that man has not by his actions increased radium activities in the environment. However, for the most part man's activities have not significantly increased the natural levels of radium in various media. Obvious exceptions are in the uranium mining/milling industry; and in the disposal or accidental release of radium used in industrial or medical applications.

#### 5. MEASUREMENT CONSIDERATIONS

Because radium concentrations in most samples are small, and because the decay alphas have such short ranges in matter ( $\sim 5$  to  $10$  mg/cm<sup>2</sup>), direct alpha counting of environmental samples is seldom practical.

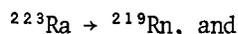
The techniques which have been developed fall into two separate categories, which we shall treat separately below. In the first, chemical methods are used to separate radium from the matrix in which it is found, followed by alpha counting (or sometimes alpha spectroscopy). In the second, radon gas (usually radon-222) is allowed to emanate from the radium-containing medium, followed by measurement of the activity of the radon or its daughters. Sometimes the two approaches are combined: after chemical separation, the prepared sample may be stored for radon ingrowth, followed by radon measurements.

##### A. Radon Emanation Techniques

Radon emanation is by far the most widely used method for measuring radium-226. Here we shall give an overview of the various techniques developed for  $^{226}\text{Ra}$  which use radon emanation followed by measurements of  $^{222}\text{Rn}$ . However, for detailed discussions of  $^{222}\text{Rn}$  measurements the reader is referred to the section "Radon-222 and Its Daughters" elsewhere in this Volume.



There are three radium → radon decays of possible interest:

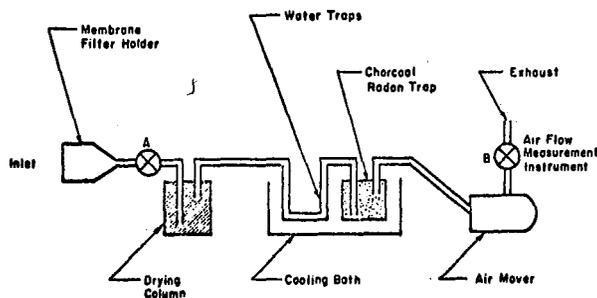


However, two of the three radon isotopes, radon-219 and radon-220, are so short-lived (4 second and 55 second half-lives, respectively) that measurements of them using emanation are unsatisfactory. Hence, the emanation technique is used exclusively with radon-222 (to determine its parent radium-226). Indeed, since the emanated gas is usually stored for radon-222 decay, the interference from the other isotopes can be made unimportant.

*Direct de-emanation* is an old technique. A sample is placed in a vessel filled with helium or "old air" (e.g., air stored long enough to allow decay of natural radon) and put aside to allow  $^{222}\text{Rn}$  ingrowth. For example, two half-lives (7.65 days) will give ~75% of the maximum buildup. The simplest procedure is transfer of the ingrown radon, plus carrier gas, through a particulate filter to a Lucas chamber for alpha counting. The Lucas chamber (Ref. 15) is a small (~100 to 200 cm<sup>3</sup> volume) sphere coated on the inside with ZnS(Ag) scintillator; through a flat window on one side, the scintillation light is viewed and counted by a photomultiplier. This instrument is described in more detail in "Radon-222 and Its Daughters."

There are several difficulties with direct de-emanation. Foremost is uncertainty in knowledge of the quantitative emanation rate. Emanation from a solid medium is a process intimately dependent upon geometrical idiosyncrasies: for example, radon from radium locked in crystal structures far from a surface simply does not get out with a definite measurable efficiency. Other problems can occur because direct transfer to the Lucas chamber (even through a high-efficiency filter, which removes radon's daughters) can result in errors at high humidity, as well as in uncertainties in the electrostatic properties within the chamber, which can seriously degrade counting efficiency.

To solve this latter set of problems, one common procedure today is to pass the radon through a *purifying system* before introduction to the Lucas chamber. The system (Ref. 16, 17), described in "Radon-222 and Its Daughters" and shown in Figure 1,



- A. Calibrated Vacuum Type
- B. Rotameter

FIGURE 1. Sampling apparatus for collection of gaseous airborne radon (Ref. 17).

sends the radon through a Drierite trap and a cooled water trap into a pre-treated activated-charcoal radon trap at  $-78^{\circ}\text{C}$  (dry ice). The charcoal is subsequently de-emanated at  $300^{\circ}\text{C}$  and the radon gas transferred to a Lucas chamber with helium carrier. This transfer/purification system is quantitative, nearly 100% efficient, and is almost error-free when properly operated.

To solve the problem of uncertainties in the emanation of radon from the original (solid) medium, the commonest procedure today is chemical dissolution into the liquid state. Once radium is in solution, radon de-emanation occurs with essentially 100% efficiency. During radon ingrowth, some of the radon gas collects in the (helium) gas above the liquid. The remainder is flushed from the liquid at the end by forcing helium through the solution. The helium-carried radon is then run directly into a Lucas chamber, or alternatively through the Drierite/water-trap/activated-charcoal/Lucas chamber system.

Special flasks have been designed for de-emanation of the radium-in-solution. One typical design is shown in Figure 2 (from Ref. 18): a 500 ml Pyrex flask is altered by insertion of a 10-mm Pyrex tube (for the helium). This flask can be easily decontaminated before introduction of the liquid sample.

Rushing, Garcia, and Clark (Ref. 20) describe a system in which, after chemical dissolution and de-emanation, the helium-radon mixture is run directly into a Lucas chamber. After transfer, 3-4 hours are allowed to elapse before counting, to permit  $^{222}\text{Rn}$  to come to transient equilibrium with its daughters. These investigators achieved minimum detectable radium activities in the range of 0.03 pCi (in 300 ml) or about 0.10 pCi/liter. An interference from radon-220 may occur in this

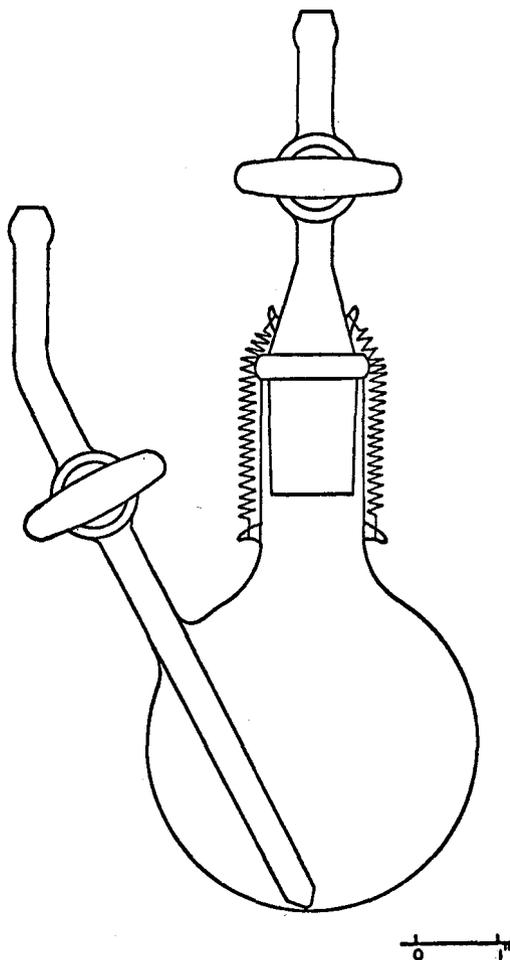
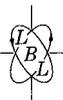


FIGURE 2. The de-emanation flask (from Ref. 18).

method, but can be overcome by interim storage of the gas before introduction to the Lucas chamber. Even if storage is not used, a 1:1 ratio of  $^{224}\text{Ra}$ : $^{226}\text{Ra}$  in the original sample only produces a radium-226 overestimate in the 0.5% range, while the effect of  $^{223}\text{Ra}$  is an order of magnitude smaller still.

Blanchard (Ref. 18) has described a complete system, including the Drierite/activated-charcoal purification procedure. With care to flush remanant radon from various parts of the system, and using a 300-ml radium-containing sample with a one-hour radon-emanation period, Blanchard measured an NBS standard (1.00 pCi of radium) to within  $\pm 0.04$  pCi, with an average recovery of  $100 \pm 4\%$  for 9 trials. The limit of detection (95% confidence) is calculated to be 0.016 pCi, which (in 300 ml) corresponds to about 0.050 pCi/liter.

Another emanation technique, similar to Blanchard's, is included as a Standard Method

in the APHA compilation (Ref. 21). The sensitivity is comparable, and this Reference also reports that "neither radium-224 at an activity equal to that of the radium-226 nor dissolved solids up to 610 mg/liter produced a detectable error in the results." This Standard Method satisfies nearly all requirements for sensitivity, accuracy, and versatility which might be desired for environmental monitoring. Its drawback is the expense and difficulty of operating and maintaining the equipment.

The importance of running reagent blanks through the entire system must be emphasized: minute radium contamination in any of the reagents can sometimes occur, and a correction must be applied (or the contaminant eliminated) to achieve the ultimate in sensitivity.

#### B. Sample Preparation for Radon Emanation

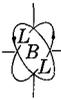
The radon emanation technique as used today requires that the radium be in solution. Here we shall outline briefly some of the various processes developed for radium dissolution from the more important sample media. We shall discuss only a few methods found in some of the main radiochemical procedures manuals.

Of course, in many media (e.g., water or urine) the radium is already in solution. However, even for some liquid media, such as raw sewage, it might be desirable to process the radium before emanation. Also, in some samples the dissolved radium is at too low a concentration to permit measurement by emanation, so a concentration step is necessary.

A method applicable to water, urine, or fecal samples is described in the USAEC Health and Safety Laboratory (HASL) Procedures Manual (Ref. 22). Radium is co-precipitated with barium sulfate; silica is removed with hydrofluoric acid and reprecipitation of the sulfate. The precipitate is then dissolved in alkaline EDTA (ethylenediaminetetraacetic acid), the solution used for emanation.

A method applicable to food, milk, vegetation, soil and bone is also described in the HASL Manual (Ref. 22). Some samples require a sodium carbonate fusion, after which the radium carbonate is dissolved in  $\text{HNO}_3$ .

"Successive fuming nitric acid separations remove the calcium and most other interfering ions. Traces of some fission products are scavenged with yttrium hydroxide. Finally, radium is coprecipitated with barium chromate, then dissolved in perchloric acid and water" (Ref. 22).



The Handbook of the EPA's Las Vegas laboratory (Ref. 23) describes a slightly different method for ashed samples using ammonium chromate, and also a method for soil, sludge, and air filters in which fusion is performed with Nicholson's flux, followed by  $H_2SO_4$  dissolution and a barium-carrier addition for co-precipitation of Ba-Ra phosphate. The U.S. Public Health Service's procedures compilation (Ref. 24) describes an ammonium dichromate dissolution method for ashed samples.

The APHA Standard Methods (Ref. 21) describes separate methods for determining dissolved and undissolved radium in water, using a membrane filter to separate the undissolved (suspended) matter for separate dissolution using Nicholson's flux.

### C. Radiochemistry with Radium Alpha Counting

Here a number of the techniques will be discussed which employ chemical separation of radium from the medium in which it is sampled, followed by counting or spectroscopy of the radium alphas.

Ordinary chemical techniques are intrinsically incapable of differentiating among the various radium isotopes. Thus if simple alpha counting is used, a single measurement can only yield total radium activity (curies). Alpha spectroscopy, on the other hand, enables the various alpha energies to be separately identified quantitatively. Another technique for separating isotopes is to make activity measurements at several different times, so that the build-up and decay of activities from the various radium/daughter series can be followed. The basis for this can be seen in Figure 3 (from Ref. 25), which shows the theoretical growth and decay of total alpha activity (parent plus daughters) into pure radium isotopes. This time variation can be used to determine ratios of  $^{226}Ra:^{224}Ra:^{223}Ra$ .

The section "Alpha Particle Instrumentation" elsewhere in this Volume contains detailed discussions of instruments for alpha counting and spectroscopy. We will summarize here by noting that among the common alpha instruments are internal gas proportional counters, solid state (surface-barrier) detectors, pulse ion chambers, and scintillators.

Because the range of alphas in matter is so small ( $\sim 5$  to  $10$  mg/cm<sup>2</sup>), one problem for the chemist is preparation of a sample thin enough to be counted without important self-absorption corrections. For alpha spectroscopy this is particularly difficult.

Some techniques for radiochemical determination of specific radium isotopes by measuring their daughters (lead, polonium, bismuth) will be discussed below.

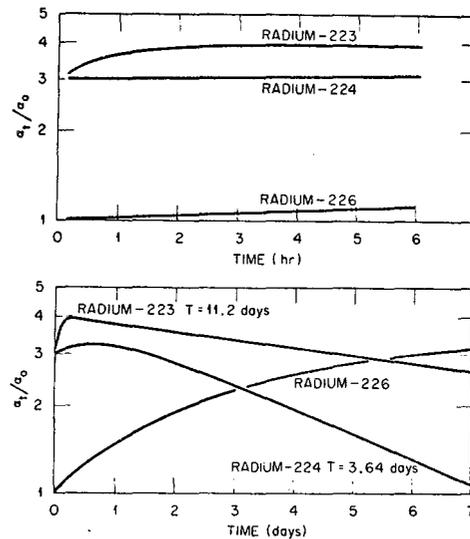


FIGURE 3. Growth of alpha activity into pure radium isotopes (Ref. 25).

The APHA Standard Methods (Ref. 21) describes a technique designed for screening of drinking water. The method is also applicable to sewage or industrial wastes, provided that organic matter is destroyed.

"Lead and barium carriers are added to the sample containing alkaline citrate, then sulfuric acid is added to precipitate radium, barium and lead as sulfates. The precipitate is purified by washing with nitric acid, dissolving in alkaline EDTA, and re-precipitating as radium-barium sulfate after adjusting the pH to 4.5. This slightly acidic EDTA keeps other naturally occurring alpha emitters and lead carrier in solution." (Ref. 21)

The precipitate is counted with an internal gas proportional counter, after a 3 to 18 hour wait. To determine isotopic mixtures,

"additional counting at 7, 14 or 28 days is suggested, depending on the number of isotopes in mixture. The amounts of the various isotopes can be determined by solving a set of simultaneous equations" (Ref. 21).

Average chemical recoveries are excellent ( $\sim 97\%$ ) in this method, and the technique is valuable as a screening method for routine radium-in-water analysis.

The ASTM compilation (Ref. 25) describes a very similar method. The lower detection limit is about 1 pCi/liter. Both the APHA and the ASTM methods are based on a technique



developed at Idaho Falls (Ref. 26) and also described by Goldin (Ref. 27).

Another chemical method is described in the U.S.P.H.S. compilation (Ref. 24). Here radium is coprecipitated with  $\text{BaSO}_4$  from solution containing EDTA; the precipitate is converted to carbonate and dissolved in  $\text{HNO}_3$ . Radium daughters are extracted into thenoyltrifluoroacetone, and  $\text{Ra-BaSO}_4$  is reprecipitated for counting.

A quite different method has been developed by Duquesne et al. (Ref. 28). Here the final sulfate precipitate is suspended in ethanol together with powdered scintillator,  $\text{ZnS(Ag)}$ , and this mixture is filtered and dried for counting with a photomultiplier. Sensitivity below 0.1 pCi/liter is claimed, and by studying the alpha activity as it evolves in time the  $^{224}\text{Ra}/^{226}\text{Ra}$  ratio can be roughly determined.

Kelkar and Joshi (Ref. 29) describe a system which is similar but even simpler. After a one-liter sample is filtered,  $\text{HCl}$  and  $\text{BaCl}_2$  are added. Radium, lead, and polonium are co-precipitated with 36 M  $\text{H}_2\text{SO}_4$ , and 400 mg of  $\text{ZnS(Ag)}$  are added while stirring. A Whatman 42 filter then picks up the sulfates and the  $\text{ZnS(Ag)}$  together; the filter is dried under an infrared lamp for two minutes, and counted for 30 minutes. Chemical yields exceed 95%, and concentrations as low as 1 to 2 pCi/liter can be determined. This method, though less sensitive than the more sophisticated techniques, is quick and inexpensive.

To summarize, there are a variety of chemical methods which take advantage of coprecipitating radium with barium as sulfate or carbonate, to separate it from other alpha-emitting elements (principally daughters of the various radiums). After successive purifications in one of several chemical media, the methods either count the radium salt in an internal gas proportional counter, or use  $\text{ZnS(Ag)}$  and a photomultiplier tube for scintillation counting.

In any of the methods which use alpha counting, the substitution of alpha spectroscopy can enable a direct determination of radium isotope ratios. However, resolution is a key consideration. Typical resolutions ( $\pm$  half-width) that can be obtained in various kinds of alpha spectrometers are about  $\pm 15$  keV with surface-barrier detectors;  $\pm 20$  keV with pulse ion chambers; and  $\pm 50$  keV with gas proportional counters. Liquid scintillators ( $\pm 150$  keV) and solid scintillators ( $\pm 200$  keV) are poorer. Any of these systems can resolve  $^{226}\text{Ra}$  (4.60, 4.78 MeV alphas) from  $^{223}\text{Ra}$  (5.54, 5.61, 5.71, 5.75 MeV) and  $^{224}\text{Ra}$  (5.44, 5.68 MeV). However, separating these latter

two from each other is more difficult because the energies are so intermingled.

#### D. Radiochemistry with Counting of Radium Daughters

Here we shall mention methods in which radium daughters (*other than radon gas*) are counted after radiochemical separation.

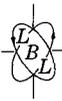
One technique, in the HASL Manual (Ref. 22) separates lead-212 in urine and measures the alphas from bismuth-212 and polonium-212 after equilibration. This technique is specific for *radium-224* (the precursor of  $^{212}\text{Pb}$ ). In another HASL technique applicable in urine or water (Ref. 22),  $\text{Ra-BaSO}_4$  precipitate is stored for buildup of  $\text{Rn-222}$ ,  $\text{Po-218}$ , and  $\text{Po-214}$ , which are measured in equilibrium using a  $\text{ZnS(Ag)}$  disk phosphor and a photomultiplier.

Petrow and Allen (Ref. 14) describe a specific determination of *radium-223*. After radium is separated from carrier lead, it is stored for 3 hours to allow ingrowth in solution of  $^{223}\text{Ra}$ 's daughter, lead-211 (36.1 minutes, 1.39 MeV (max)  $\beta$ ). The lead is then extracted from solution into dithizone. Interference from lead-214 (daughter of radium-226) is minimized by boiling the solution during the 3 hour period, in order to purge the 3.82-day radon-222. The selection occurs because some of the  $^{219}\text{Rn}$  (4 sec) in the  $^{223}\text{Ra} \rightarrow ^{211}\text{Pb}$  chain decays before it too is purged in the boiling. The 6.62 MeV  $\alpha$  from bismuth-211 (lead-211's daughter) is the activity which is counted.

A similar procedure can be used to determine *radium-224*. The separated radium sample is aged for 16 - 24 hours to enable lead-212 ingrowth from  $^{224}\text{Ra}$ . Lead-212 (10.6 hours) is separated from lead-211 (36.1 minutes) by aging of the lead extract for 3 - 4 hours before beta-counting.

Petrow and Allen (Ref. 14) also describe a method for *radium-228* by separation and counting of its daughter, actinium-228. To avoid an interference, the radium sample must be purified of  $^{227}\text{Ac}$  before  $^{228}\text{Ac}$  ingrowth. This is achieved by using lanthanum hold-back carrier prior to the radium-sulfate precipitation. After  $^{228}\text{Ac}$  ingrowth from  $^{228}\text{Ra}$ , the sample is beta-counted in an internal gas proportional counter.

Sanderson (Ref. 30) has described the determination of  $^{226}\text{Ra}$  in food, soil, or biological samples by coincidence  $\text{NaI(Tl)}$  gamma spectroscopy on the 0.61-MeV and 1.12-MeV gamma rays of bismuth-214. Thorium-228 can also be determined by the coincident 0.58-MeV and 2.61-MeV gammas of its daughter thallium-208. Sanderson reports  $^{226}\text{Ra}$  measurements in the range of 0.30 pCi/g with  $\pm 10\%$



errors, using 20 to 40 g samples. Minimum sensitivities in the range of 0.03 to 0.06 pCi/g are reported for both  $^{226}\text{Ra}$  and  $^{228}\text{Th}$  in food, soil, and fecal ash.

To summarize, we have briefly discussed several chemical methods which selectively determine one of the radium isotopes by the chemical or physical properties of its daughter(s). These analytical methods are useful only in restricted circumstances, but they are a valuable tool when detailed knowledge of the various radium isotopic ratios is required.

#### 6. CALIBRATION SOURCES

There is an obvious need for sources of known activity to be used for calibration purposes. To fulfill this need, the U.S. National Bureau of Standards had developed a set of standards with radium-226 content ranging from 0.1 to ~100 micrograms, in flame-sealed glass ampoules (Ref. 31). These 'emanation' standards, in liquid form, are for use in calibrating an entire counting system.

#### 7. SUMMARY AND CONCLUSIONS

This section has discussed the typical levels at which radium is found in environmental samples; the radiation protection guides; and the types of measurement techniques developed to measure radium in the environment. The emphasis has been placed on radium-226, although the other main isotopes (223, 224, 228) have also been discussed.

The activity levels requiring measurement are very small, and the measurement techniques are (correspondingly) sophisticated. Most routine measurements today use either an emanation technique or radiochemical separation followed by alpha counting. Each of these types of measurement can be performed sensitively and accurately enough to determine the radium isotopes at their naturally-occurring activity levels in the environment. Unfortunately, each suffers from drawbacks: The radiochemical methods require exacting analytical procedures; the emanation techniques are time-consuming and the apparatus clumsy and relatively expensive.

However, in the overall picture of radionuclide measurement capabilities in environmental media, radium measurements appear to be in a reasonably sound position.

The quick screening (gross-alpha) measurement techniques for public health protection are adequate, and radium levels can seldom be expected to increase suddenly to the danger point where both time and accuracy would be essential. Thus the present methods, though time-consuming, are probably adequate for the task of routinely monitoring environmental samples for radium content.

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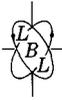


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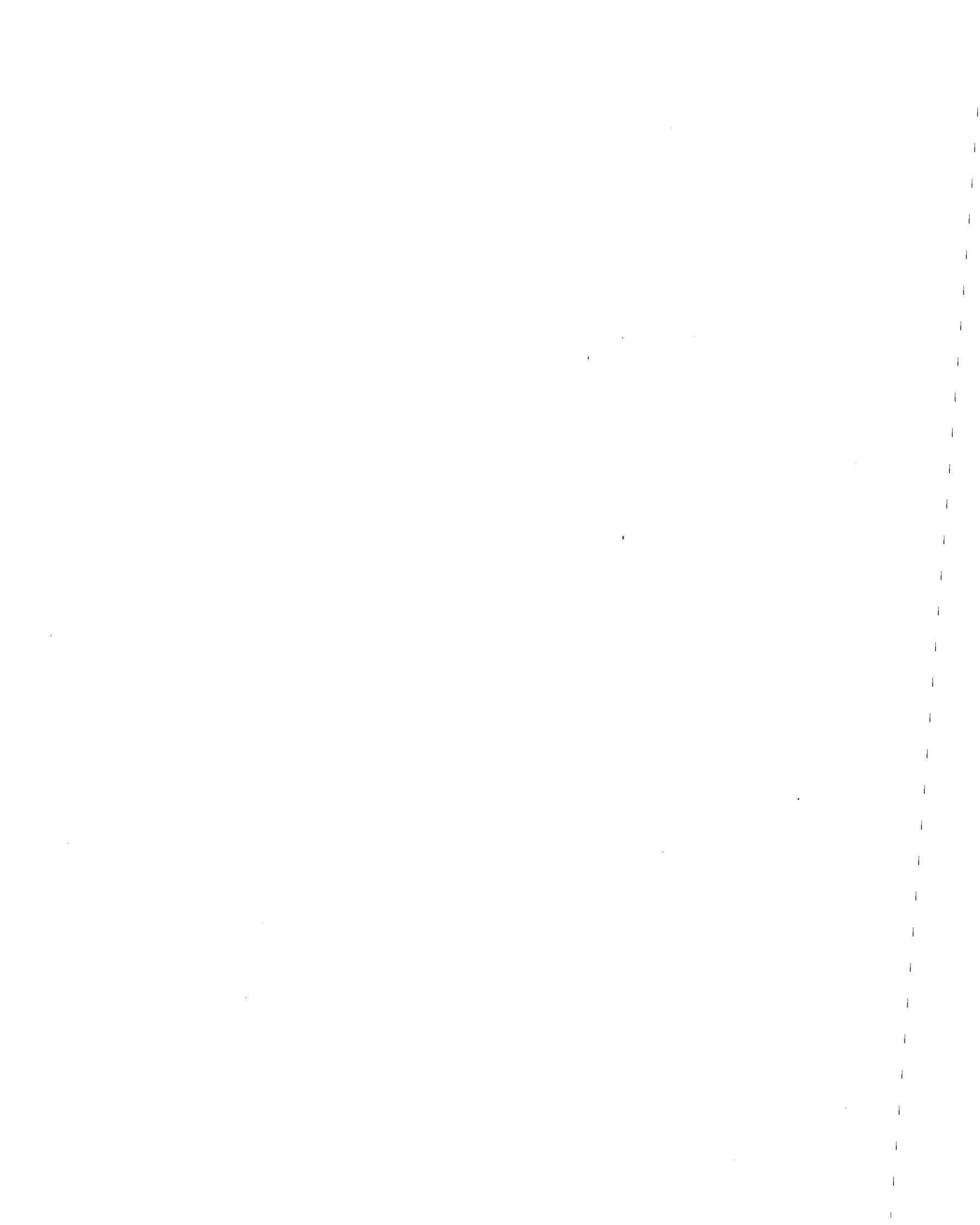


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## URANIUM

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## 1. INTRODUCTION

Uranium is widely familiar to the general public. Its use in atomic (fission) bombs brought it to the attention of the entire world nearly three decades ago. Today it is the fuel for the fastest growing energy source in our modern economy, the nuclear power reactor.

Here we shall deal with the measurement of uranium concentrations in environmental media. We shall concentrate on methods for determining low levels of uranium in air, water, and certain other environmental media; but we shall not treat bioassay methods, which are covered in Volume 4 ("Biomedical Instrumentation") of this Survey.

We shall first provide overviews of the physical characteristics of the important isotopes; of the radiation protection guides for uranium concentrations; and of the ranges of uranium concentrations in the natural environment. We shall then discuss the various measurement techniques for uranium.

Discussions which are important for a more complete picture of the measurement techniques treated in this section are found in several other sections of this volume. These sections are "Radon-222 and Its Daughters," in which the uranium-238 decay chain and the instrumentation for radon-222 measurements are discussed; "Alpha Particle Instrumentation," in which instruments for detecting alphas are treated in detail; "Radium"; and "Gamma Spectrometry".

## 2. PHYSICAL CHARACTERISTICS

Uranium occurs naturally in three isotopic states, uranium-234, 235, and 238. However,  $^{234}\text{U}$  is a daughter in the decay chain of  $^{238}\text{U}$ , and only the two heavier isotopes occur in natural, primordial material.

The  $^{235}\text{U}/^{238}\text{U}$  ratio is the same essentially everywhere on earth: 0.72%. The daughter  $^{234}\text{U}$  occurs at an abundance of 0.0057%, when it is in secular equilibrium with its precursor  $^{238}\text{U}$ .

Some physical properties of the three isotopes are shown in Table 1 (information from Ref. 1). Note that all three are alpha-emitters, and that gamma rays accompany some of the alpha decays. The half-lives of the three isotopes are all quite long. The physical property which has given uranium its unique role in modern technology is fission; indeed, it is the fission of uranium-235 which is used in nuclear reactors. However, for our purposes, the fission property has very little direct consequence as a radiological hazard. It finds occasional use in detection of uranium-235, as will be discussed below.

## 3. RADIATION PROTECTION GUIDES

There are differences in toxicity depending upon whether uranium, when ingested, is soluble or insoluble. These differences have been taken into account in the establishment of radiation protection guides. The guides for "natural uranium", which is the isotopic mixture as tabulated in Table 1, will be discussed here.

For 168-hour occupational exposures, the maximum permissible concentrations in air and water (MPC)<sub>a</sub> and (MPC)<sub>w</sub> are given in Table 2, from ICRP Publication 2 (Ref. 2). The maximum permissible body burdens (MPBB) are also given. For individuals in the general public, the applicable (MPC) values are a factor of 10 smaller than those tabulated in Table 2; and for exposure to a suitably large sample of the general public, another factor of 3 smaller still.

TABLE 1.

*Some Properties of the Naturally Occurring Uranium Isotopes*

Isotope	Half-Life	Natural Abundance	Member of Which Decay Chain	Immediate Daughter	Main Decay Products and Energies	
Uranium-238	4.5 billion years	99.27%	U-238 Chain	Thorium-234 (24.1 days)	(77%) 4.20-MeV $\alpha$ (23%) 4.15-MeV $\alpha$ +	48-keV $\gamma$
Uranium-235	0.71 billion years	0.72%	U-235 Chain	Thorium-231 (25.5 hours)	(18%) 4.37-MeV $\alpha$ + (57%) 4.40-MeV $\alpha$ + (25%) others	234-keV $\gamma$ 204,163-keV $\gamma$ 's
Uranium-234	247,000 years	0.0057%	U-238 Chain	Thorium-230 (80,000 years)	(72%) 4.77-MeV $\alpha$ (28%) 4.72-MeV $\alpha$ +	53-keV $\gamma$

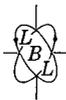


TABLE 2.

*Natural Uranium and Radium-226 MPC's for Critical Organs.  
168-hour/week Occupational Maximum Permissible Concentrations in Air  
and Water, and Maximum Permissible Body Burdens (from ICRP Pub. 2, Ref. 2).*

Isotope	Soluble or Insoluble	Critical Organ	(MPC) <sub>w</sub> pCi/liter	(MPC) <sub>a</sub> pCi/liter	MPBB pCi
Natural Uranium	soluble	G.I. tract	200,000	—	—
Natural Uranium	soluble	Kidney	—	0.03	5,000
Natural Uranium	insoluble	G.I. tract	200,000	—	—
Natural Uranium	insoluble	Lung	—	0.02	—
Radium-226	soluble	Bone	100	0.01	100,000
Radium-226	insoluble	Lung	—	0.02	—

The comparison with radium-226 in Table 2 is especially revealing. The (MPC)<sub>w</sub> is a factor of 2000 less restrictive for uranium than for radium, while for (MPC)<sub>a</sub> the values are quite comparable. Neither uranium nor radium-226 is a problem in airborne suspension, except in rare circumstances involving duststorms. Uranium is also almost never a public health problem in drinking water, although radium occasionally is found in natural concentrations which exceed its much smaller (MPC)<sub>w</sub>.

#### 4. SOURCES AND LEVELS OF ENVIRONMENTAL URANIUM

Uranium occurs widely throughout the environment, as a trace constituent of soils and rock. The isotopic  $^{235}\text{U}/^{238}\text{U}$  ratio of 0.72% is constant almost everywhere, as we have mentioned.  $^{238}\text{U}$ 's presence is manifest by the universal emanation of radon-222 from rocks and soils (see "Radon-222 and Its Daughters").

Uranium content of rock is usually given in parts per million (ppm), by weight. One microgram of pure  $^{238}\text{U}$  has an activity of 0.33 pCi. Hence one ppm of pure  $^{238}\text{U}$  corresponds to 0.33 pCi/gram; however, if  $^{234}\text{U}$  is in equilibrium, then it adds another 0.33 pCi/g, as does every one of the other dozen or so members of the  $^{238}\text{U}$  chain.

In most rocks, uranium composition is in the range from 0.5 to 4 ppm, with thorium-232 about 4 to 5 times more abundant (Ref. 4). True granite is the most uranium-rich of the common rocks, having typically 5 ppm of

uranium. In uranium-mining areas, concentrations of  $\sim 30$  ppm are sometimes found (Ref. 4). Volcanic rocks have uranium at about 0.1 - 0.4 ppm (Ref. 5), and Atlantic beach sands about 0.1 - 1.5 ppm (Ref. 6). River waters in Japan (Ref. 7) have uranium concentrations at the level of a fraction of 1  $\mu\text{g}/\text{liter}$  ( $\sim 0.33$  pCi/liter).

In material that has remained totally undisturbed,  $^{234}\text{U}$  should be in equilibrium (equal activities) with its precursor,  $^{238}\text{U}$ . However, this is not always the case, perhaps because of the fact that  $^{234}\text{U}$  seems to exist preferentially in the hexavalent rather than the quadrivalent state (Ref. 8). Disequilibrium has been reported at about the -15% level in the upper several meters of a rocky granitic outcropping (Ref. 9). Several investigators have found enhanced  $^{234}\text{U}$  in deep underground waters, in some cases with more than 7:1 for the  $^{234}\text{U}:^{238}\text{U}$  activity ratio (Ref. 10). Soils, conversely, have been found to be slightly depleted of  $^{234}\text{U}$  on the average. Enhancements have also been observed in normal fecal samples (Ref. 8).

Studies have been made of the uranium content of typical diets. In the United Kingdom (Ref. 11), the concentrations in nearly all foods were found to be in the range from 0.0002 to 0.001 ppm. The intake of uranium in food was found to be about 1.0  $\mu\text{g}/\text{day}$ , which is similar to the value of 1.3 to 1.4  $\mu\text{g}/\text{day}$  found by Welford and Baird for the U.S. (Ref. 12). Domestic drinking water in the U.S. contains uranium nearly always at levels well below 0.1  $\mu\text{g}/\text{liter}$  (Ref. 11), but some drinking supplies have concentrations as high as 5 to 8  $\mu\text{g}/\text{liter}$ . Since the (MPC)<sub>w</sub> for



individuals in the general public is 0.02  $\mu\text{Ci/liter}$  ( $= 60,000 \mu\text{g/liter}$ ), it can be seen that even the highest concentrations are about 4 orders of magnitude below the (MPC)<sub>w</sub>.

Air concentrations have been found to be in the range of 20 pg/liter in England; a mean of 4 pg/liter over the North Atlantic Ocean; and a mean of 3 pg/liter over Antarctica (Ref. 13). Again, the (MPC)<sub>a</sub> for individuals in the general public is 0.002 or 0.003 pCi/liter ( $\approx 6000$  to  $9000 \mu\text{g/liter}$ ), which is 2 to 3 orders of magnitude greater than the reported concentrations.

Uranium levels are typically not much affected by man's activities. The most important exceptions are in and around uranium mining/milling facilities, and at nuclear fuel reprocessing plants. Even there, the main radiological and environmental effects are seldom from uranium itself. Discussions of the radiological problems in these areas are found in other Sections of this Volume.

From the numbers just presented, it can be seen uranium is not often a radiologically important nuclide. This does not imply that uranium measurements are unimportant: In situations such as around uranium mills, environmental measurements are required to monitor any possible long-term build-up trends in the environment. Also, the radiological significance of radium, radon, and the radon daughters is widespread.

## 5. MEASUREMENT CONSIDERATIONS

Among the environmental media requiring measurement of uranium activity are rocks, silt and soil; natural and waste waters; food; and air (air filters). Because the uranium concentration in most samples is small, and because the decay alphas have such short ranges in matter ( $\sim 5$  to  $10 \text{ mg/cm}^2$ ), direct alpha counting of environmental samples is seldom practical.

The techniques which have been developed fall into several categories. We shall discuss them separately below. These categories are:

- a. gamma spectroscopy
- b. neutron-induced fission methods
- c. radiochemical analysis followed by alpha counting
- d. fluorometric determinations
- e. other methods

The number and variety of uranium-determination methods is so large that to survey them all would be impossible. For example, the index to Nuclear Science Abstracts for the

year 1971 lists about 200 entries under the subject heading "Uranium Determination. . ." . We shall not attempt here to do more than mention those techniques which are found in the procedures manuals of some of the larger analytical laboratories; or which are worthy of mention as representative of a large class of techniques.

### A. Gamma Spectroscopy

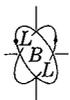
Table 1 indicates that several of the uranium alpha decays are accompanied by gamma emissions. Also, when  $^{238}\text{U}$  or  $^{235}\text{U}$  is in equilibrium with its daughters in the long decay chains, several gamma-emitting nuclides can be detected. The unique signature of a chain can make possible increased sensitivity in the presence of higher backgrounds, compared to the measurement of only a single gamma.

We shall not discuss here the detailed considerations involved in gamma spectroscopic measurements: These are treated in "Gamma Spectrometry" elsewhere in this Volume. It suffices to mention here that the two main techniques are NaI(Tl) scintillation systems and Ge(Li) solid state detector systems, the latter with significantly better energy resolution but poorer detection efficiency than the former.

Of course, the farther down the chain a nuclide is, the greater the chance for disequilibrium with the parent; this is especially true for measurements of radon-222 daughters as indicators of uranium-238, since radon's emanation can destroy secular equilibrium.

The most prominent gamma in the  $^{238}\text{U}$  chain is the 1.76-MeV gamma from  $^{214}\text{Bi}$ . A study of the  $^{238}\text{U}$  chain reveals that among the nuclides prior to radium-226, there are no gamma rays which combine a significant branching ratio ( $\approx 1\%$ ) and an energy above 100 keV. Only  $^{214}\text{Pb}$  and  $^{214}\text{Bi}$  have gammas with energies above 300 keV. The  $^{226}\text{Ra}$  gamma (4%) has an energy of 185.7 keV. The  $^{235}\text{U}$  chain has several more prominent gamma rays, but their relative intensity is limited by the  $^{235}\text{U}/^{238}\text{U}$  isotopic ratio of 0.72%. Indeed,  $^{235}\text{U}$  itself has a 185-keV gamma (54%), but it is very difficult to resolve in the presence of the  $^{226}\text{Ra}$  gamma at 185.7 keV.

When NaI(Tl) is used, the gamma emission most commonly analyzed is the 1.76-MeV  $\gamma$  of bismuth-214, since it stands out at high energy above nearly all of the other naturally-occurring gamma lines. Similarly, the 2.62-MeV  $\gamma$  from thallium-208 is commonly used as a signature for the thorium-232 chain. Adams and Fryer (Ref. 14) describe a portable NaI(Tl) spectrometer used for field surveys of natural U, Th, and  $^{40}\text{K}$  content in soils and rocks, using which accuracies in the few percent



range are obtained for uranium concentrations in the range of several ppm.

Wollenberg and Smith (Ref. 15) have also applied NaI(Tl) techniques to geological assays. They point out the possible difficulties with disequilibrium in the  $^{238}\text{U}$  series:

"Uranium can be leached from its host material, there are long half-lived decay members, and the gaseous member can be lost to the air -- three conditions which may lead to lack of equilibrium. Most of the uranium-series  $\gamma$  rays come from daughters of radium; therefore, some . . . assays may really reflect only radium content. This situation would seem most likely to occur with soil samples." (Ref. 15)

Smith has indicated (Ref. 16) that disequilibrium is seldom a serious problem in solid rocky material, since neither leaching nor radon emanation is significant.

Of course, Ge(Li) detectors, with their vastly superior resolution, are able to distinguish gamma lines which could never be resolved by a NaI(Tl) system. Figures 1 and 2 (from Ref. 18) show a comparison of NaI(Tl) and Ge(Li) spectra measured on the same sample. Note that the Ge(Li) spectrum, of which only the region below 1 MeV is shown, reveals fine structure not even suggested in the NaI(Tl) spectrum. Nevertheless, NaI(Tl) instruments have competed reasonably with Ge(Li) systems for field uranium assay, especially at the lowest levels of sensitivity. There are several reasons for this, among which are that the  $^{214}\text{Bi}$  line stands out prominently even with NaI(Tl); that NaI(Tl) has greater detection efficiency; and that the lines which might be studied with Ge(Li) are mainly at low energy, where backgrounds are higher, especially in the field.

Both NaI(Tl) and Ge(Li) systems have sensitivities in the range of a fraction of one ppm, more than adequate for most uranium assay studies. An intrinsic advantage of gamma-spectroscopic methods is the ability to measure large (many-gram) samples non-destructively. Adams and Gasparini (Ref. 19) have written a good overview of the entire field of gamma-spectral studies of rocks.

#### B. Neutron-Induced Fission Methods

The fact that  $^{235}\text{U}$  fissions in the presence of thermal neutrons is exploited in nuclear reactor technology. ( $^{238}\text{U}$  requires fast neutrons.) This is also the basis for

several measurement techniques for extremely low concentrations of  $^{235}\text{U}$  in various media.

When a heavy nuclide ( $Z \approx 90$ ) undergoes fission, a small fraction of the fission products are neutron-rich and subsequently decay by neutron emission. This "delayed neutron emission" is unique to the fission process: There is hardly any other process which produces neutron emission. The only known exceptions are 4.2-second nitrogen-17 and 0.17-second lithium-9, neither of which causes a background in uranium measurements of this kind. Measurement of the delayed neutrons can be used as an indicator of the concentration of uranium-235 (or other fissionable radionuclides).

Amiel (Ref. 20), irradiating samples in a slow-neutron flux of  $4 \times 10^{12}$  n/cm<sup>2</sup>sec, reports sensitivities to less than 0.1  $\mu\text{g}$  of uranium. With a (maximum) sample size of 200 g, this sensitivity approaches 1 ppb. For thorium this method is about one order-of-magnitude less sensitive. The system, while expensive, suffers from practically no interferences. Hamilton has reported an even lower detection limit, about 0.01  $\mu\text{g}$  U with  $\pm 10\%$  precision (Ref. 13).

Hamilton has also reported a technique of much greater sensitivity (Ref. 13): Uranium-laden dust (on an air filter or in solution) is placed on a polycarbonate detector which is irradiated for 1 hour in a neutron flux of  $5 \times 10^{11}$  n/cm<sup>2</sup>sec. After 24-hour storage to allow for the decay of short-lived activity, the polycarbonate is etched in 6 N NaOH and examined under a conventional X450 microscope. The total number of tracks is simply counted, each track being the record of a fission event in the detector. Sensitivities in the range of  $1 \times 10^{-14}$  g U are reported. Hamilton shows that this method is practically unique for  $^{235}\text{U}$ , except in circumstances where very large amounts of plutonium-239 are present.

#### C. Radiochemical Analysis Followed by Alpha Counting

Chemical analysis techniques are a mainstay for the determination of extremely low levels of activity. The chemical separation of uranium produces enrichment with respect to the background matrix in which the sample is found, and makes possible alpha-counting or alpha-spectroscopic determinations of very great sensitivity.

The difficulty with alpha-counting and spectroscopy is that the short range of alphas (5-10 mg/cm<sup>2</sup>) requires the preparation of a nearly 'weightless' sample. This in turn leads to the main disadvantages with these

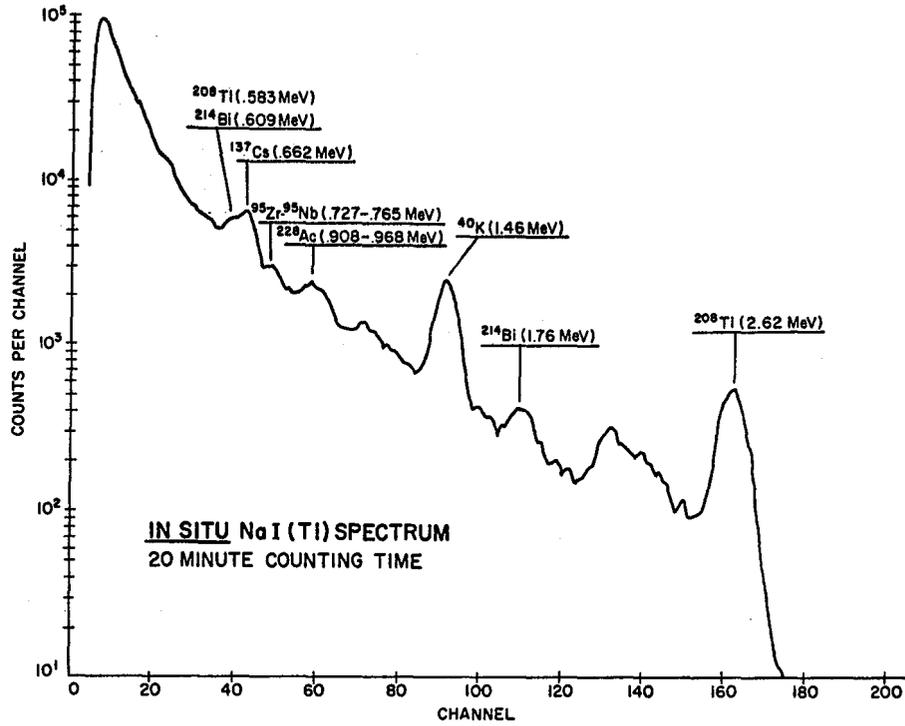


FIGURE 1. In situ spectrum, northeastern U.S. location, taken in 1971 with 10 cm x 10 cm NaI(Tl) crystal, 20-minute counting time (from Ref. 18).

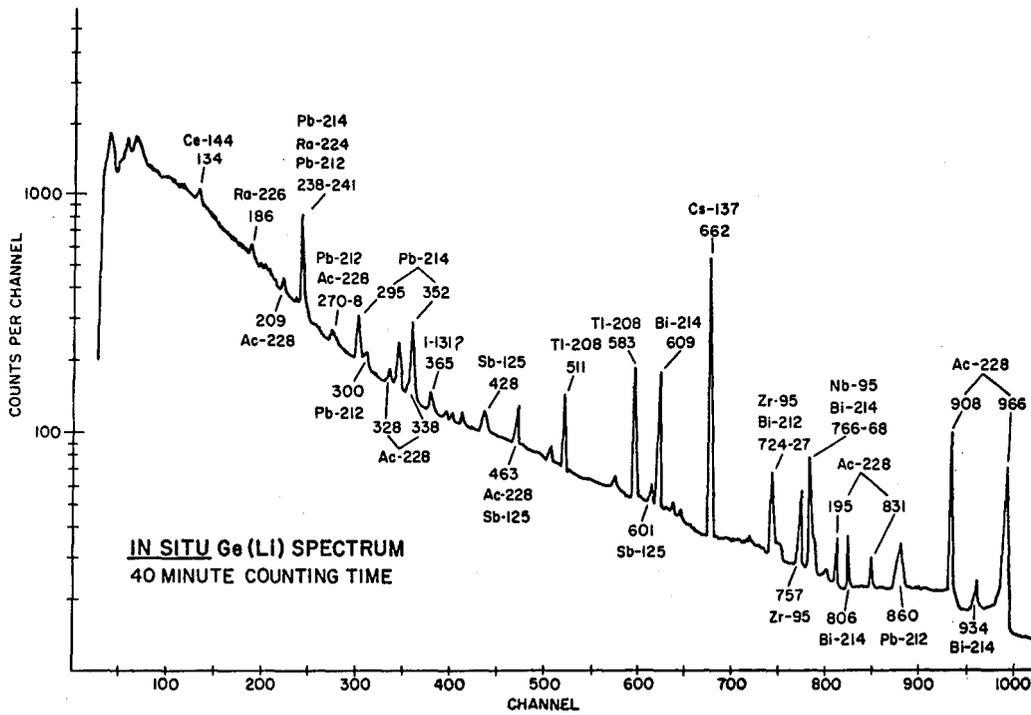
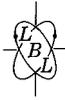


FIGURE 2. In situ spectrum, taken with that in Fig. 1 with 60-cm<sup>3</sup> Ge(Li) detector, 40-minute counting time. Photon energies in keV (from Ref. 18).



techniques, which are the difficulty and inconvenience of the analytical procedures, and the liability for error. (See "Alpha Particle Instrumentation" elsewhere in this Volume.)

*Electrodeposition* is one commonly-used technique for preparing a uranium sample for counting. The discussion here will be limited to only a few of the very many electrodeposition techniques in the literature. The U.S. Public Health Service procedures manual (Ref. 21) describes a procedure in which uranium nitrate, dissolved in dilute ammonium oxalate, is electrodeposited on a nickel-plated copper disk. Iron, commonly present with uranium, does not interfere. Another U.S.P.H.S. technique, by Levine and Lamanna (Ref. 22), is based on solvent extraction of uranium with ethyl acetate, using aluminum nitrate as a salting agent. This procedure is simple, but less sensitive than the other. Talvitie (Ref. 23) describes still another method for electrodeposition of Pu, Th, U and Am as hydrous ions. Separation of the various activities is performed using an alpha spectrometer, with a claimed detection limit of about 0.02 pCi of activity.

Another class of chemical techniques is *coprecipitation*. This type of technique, widely used for radium determination (see "Radium"), can be used for simultaneous determinations of several alpha activities when followed by alpha spectroscopy. For example, Lieberman and Moghissi (Ref. 24) describe a technique suitable for environmental samples including ashed food, vegetation, and soil. Uranium coprecipitation with lanthanum fluoride is followed by alpha spectroscopy using a gas-flow proportional counter. Samples containing a few  $\times 10^{-15}$  Ci/liter of uranium and thorium have been analyzed quantitatively (Ref. 24). This type of procedure is simpler and less expensive than electrodeposition, but more liable to error either from interferences or from inability to obtain complete dissolution of the ashed sample prior to the precipitation. An oxidation-reduction procedure can help to avoid this problem, but is tricky.

Sill (Ref. 25) describes a coprecipitation method using barium sulfate, in which all of the elements from radium through californium, *except uranium*, are precipitated by  $\text{BaSO}_4$ . Uranium can be included if it is all reduced to the quadrivalent state. Alpha spectroscopy can be used to distinguish among the several alpha emitters, or alternatively, "because of their markedly different oxidation potentials, uranium and the first four trans-uranium elements can be separated efficiently from each other by proper choice of oxidant" (Ref. 25). Sill's separated fraction can either be alpha-counted directly, or electrodeposited for precise alpha spectroscopy.

#### D. Fluorometric Determinations

The use of *fluorometry* is one of the oldest techniques for uranium determination. Here only a few of the methods in common use will be described.

The Manual of the EPA's Las Vegas laboratory (Ref. 26) describes a fluorometry technique in which, after total dissolution, a uranium/uranium-fluoride complex (which fluoresces under ultraviolet light) is formed. The principal interference is from iron, which unfortunately accompanies uranium in many sample media and which must be eliminated by careful chemical analysis.

The U.S.A.E.C. Health and Safety Laboratory (HASL) Procedures Manual (Ref. 27) describes a method in which uranium is fused with sodium fluoride for UV fluorescence. This method, developed for urinalysis, is equally applicable to most other dissolved media. A sample-preparation method for bone, soil, food, and tissue is also described in the HASL Manual. Uranium is leached from a dry-ashed residue and isolated by anion exchange chromatography plus mercury cathode electrolysis before fusion with sodium fluoride.

Rushing, Garcia, and Clark (Ref. 28) present a fluorometric method similar to that in the HASL Manual. Their description goes into detail about some of the important problems with the technique. For example,

" . . . in order to obtain the maximum fluorescence . . . , it is necessary that the uranium exist in the hexavalent condition in the final melt. While the fusion process will oxidize the uranium, it is best that reducing substances, such as carbon, be oxidized before the fusion step is undertaken " (Ref. 28).

One drawback with fluorometric methods is the possibility that a standard radiochemical laboratory might not have the UV fluorometric equipment available. Another problem, already mentioned, is interference from iron, especially in iron-rich media. Centanni, Ross and DeSesa (Ref. 29) have discussed some of the detailed considerations required for accurate fluorometric measurements, and Akaishi and Yabe (Ref. 30) have pointed out some of the cautionary points with the technique.

#### E. Other Methods

There are numerous other methods for uranium determination, which shall not be



discussed here at all. Among these are polarography (Ref. 31), spectrophotometry (Ref. 32), coulometry (Ref. 33, 34), and x-ray fluorescence (Ref. 35). The recent book by Jeffrey (Ref. 36) is a general reference on rock chemical analysis. The reader is referred to the appropriate references for details.

## 6. SUMMARY AND CONCLUSIONS

This section has briefly discussed the typical levels at which uranium is found in the environmental media; the radiation protection guides; and some of the types of techniques which have been developed to measure uranium in the environment.

Although the activity levels requiring measurement are small, the radiological hazard from uranium in the environment is also nearly always small.

The techniques for uranium measurement are varied, and among them they offer sufficient sensitivity to perform nearly all of the required measurements. Each of the classes of methods has significant disadvantages, and the choice of method should undoubtedly be based as much on the skill of the analyst and the facilities available to him as on the intrinsic merit of the particular method. However, some general comments can be made in summary:

- a. The *gamma spectroscopy* methods, while relying on equilibrium in the long decay chains, are otherwise perhaps the easiest and cleanest of the techniques.
- b. The *neutron-induced-fission* techniques offer high selectivity and sensitivity, but are unfortunately not very accessible to many analytical laboratories.
- c. *Alpha spectrometry* after radiochemical separation allows the simultaneous determination of several alpha-emitting activities, which is advantageous in many measurement situations. However, care must be taken to obtain known chemical yields in the presence of possible unknown interferences.

- d. Several *chemical-analysis* techniques (fluorometry, coulometry, spectrophotometry) probably require more analytical care than the other techniques, but otherwise seem to be acceptable. The existence of multivalent uranium compounds in a complex medium such as soil requires special consideration.

In conclusion, the analysis of uranium in environmental samples can now be performed with any of several methods, and is probably not one of the problems which requires extensive new research and development in the next few years. This is especially true given the small radiological importance of uranium and also the several other more pressing problems in the area of environmental radiation measurements.

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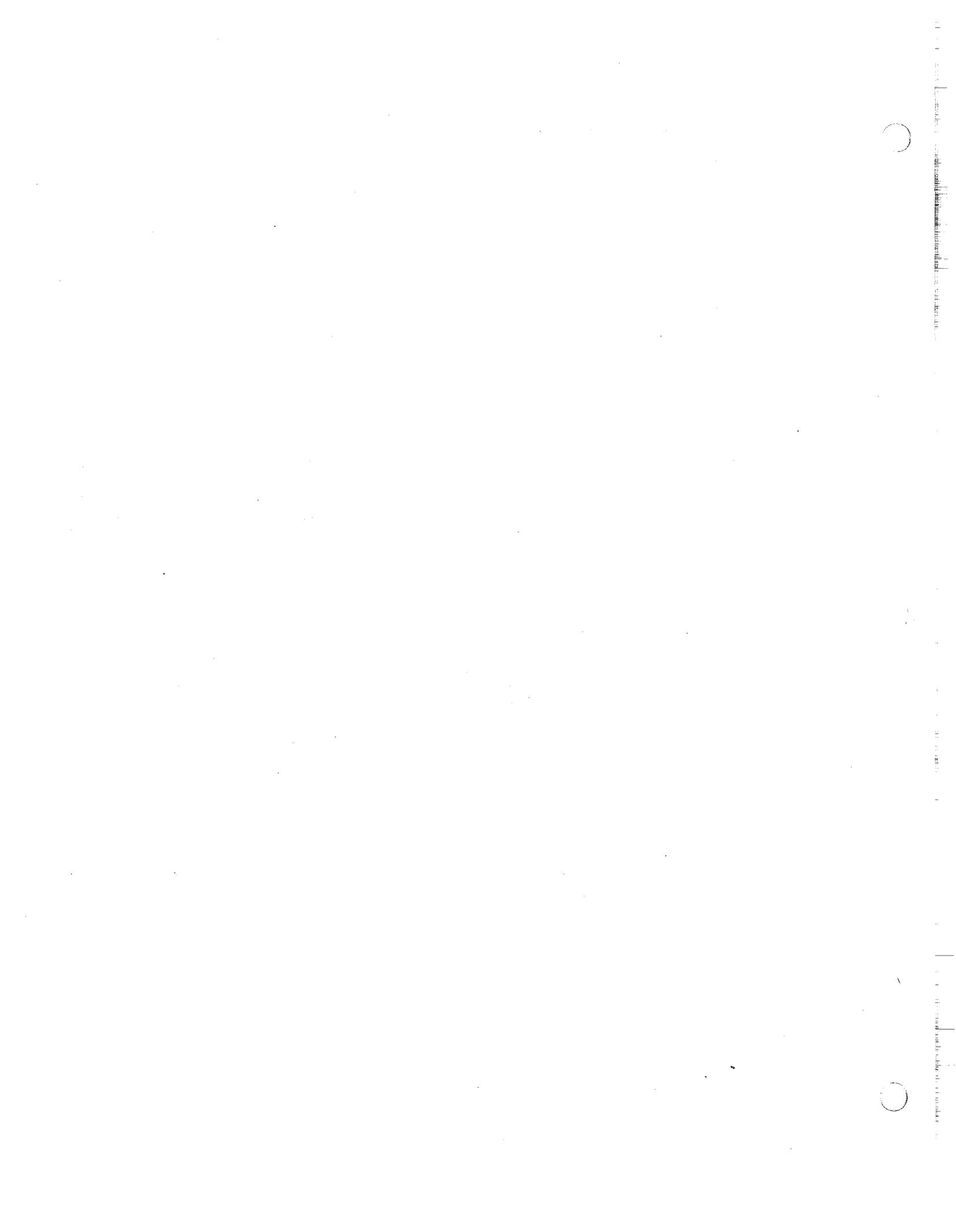


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## PLUTONIUM

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## 1. INTRODUCTION

Plutonium is by far the most important of the transuranic elements (elements heavier than uranium). Its use in atomic (fission) weapons has made it a strategically important material, and it is the cornerstone in the fast-breeder-reactor system which is now under development. Plutonium is also highly toxic, a fact recognized very early in its history. This toxicity makes measurements of plutonium in environmental media very important.

We shall not dwell on those many facets of plutonium which are not directly relevant to our immediate task. Rather, we shall deal with the measurement of plutonium concentrations in environmental media. We shall concentrate on methods for determining low levels of plutonium in air, water, soil, and certain other environmental media; but bioassay methods will not be discussed; these are covered in Volume 4 ("Biomedical Instrumentation") of the Survey.

We will first provide overviews of the physical characteristics used in the detection of the important isotopes; of the radiation protection guides for plutonium concentrations; and of the sources of and typical ranges of plutonium concentrations in the natural environment. We shall then discuss the various measurement techniques for plutonium. This discussion will concentrate on the various broad categories of instruments and techniques for plutonium measurements, their sensitivities, areas of applicability, and limitations.

Discussions which are important for a more complete picture of the measurement techniques treated in this section are found in several other sections of this volume; these sections are "Alpha Particle Instrumentation," in which instruments for detecting alphas are treated in detail; "Gamma Spectrometry"; "Radium"; and "Uranium".

Four recent publications provide detailed information on a number of subjects concerning plutonium. The first is the "Proceedings of the Rocky Flats Symposium on Safety in Plutonium Handling Facilities, 1971" (Ref. 1). The second is *Radiobiology of Plutonium*, a 1972 compendium edited by Stover and Jee (Ref. 2). The third is a 1971 "Symposium on the Biological Implications of the Transuranium Elements", containing much valuable information (Ref. 3). Finally, a 1971 *Environmental Plutonium Symposium* discusses pathways to man and measurement methods (Ref. 4).

## 2. PHYSICAL CHARACTERISTICS

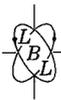
The isotopic states which concern us are plutonium-238 through -242. Some physical properties of these isotopes are shown in Table 1 (information from Ref. 5). The isotope americium-241, the immediate daughter of  $^{241}\text{Pu}$ , is also listed in Table 1 because its properties are used in some plutonium measurements.

The half-lives of the three most important isotopes are all relatively long:  $^{238}\text{Pu}$  (86 years);  $^{239}\text{Pu}$  (24,400 years); and  $^{240}\text{Pu}$  (6580 years). The alpha energies are grouped such that present alpha spectrometry systems cannot separate  $^{239}\text{Pu}$  from  $^{240}\text{Pu}$ , but can cleanly resolve all of the other plutonium isotopes.

TABLE 1.

*Some Properties of the Important Plutonium Isotopes, and of Americium-241  
(from Ref. 5)*

Isotope	Half-Life (years)	Main Decay Products and Energies (MeV)
Plutonium-236	2.85	$\alpha$ (5.72, 5.77)
Plutonium-237	0.12	electron capture
Plutonium-238	86.	$\alpha$ (5.46, 5.50)
Plutonium-239	24,400.	$\alpha$ (5.16, 5.15, 5.11) x-rays (0.017, 3.8%)
Plutonium-240	6,580.	$\alpha$ (5.12, 5.17)
Plutonium-241	13.2	$\beta^-$ (0.021 maximum)
Plutonium-242	379,000.	$\alpha$ (4.86, 4.90)
-----		
Americium-241	433.	$\alpha$ (5.44, 5.49) $\gamma$ -ray (0.060, 36%) x-ray (0.014, 0.017, others)



The fact that  $^{239}\text{Pu}$  fissions under the impact of slow neutrons is the physical property which has made plutonium so important. However, the fission property has for our purposes very little direct consequence, either as a hazard or as a means for measuring plutonium in the environment.

### 3. RADIATION PROTECTION GUIDES

Plutonium is relatively hazardous, compared to many of its neighboring elements in the periodic table. The various hazards have been taken into account in the establishment of the radiation protection guides. The toxicity depends on the medium, of course: airborne soluble plutonium transfers across the lung mucosa and ultimately is largely deposited in bone, where its tumor-forming hazard is greater than that of radium-226. As soluble plutonium in water, it is absorbed by the GI tract and again deposits preferentially in bone. When plutonium is insoluble, the critical organs are the sites of direct exposure, the lungs and GI tract. Recently, concern has also been expressed that the limiting effect for insoluble airborne plutonium might be the dose to the pulmonary lymph nodes (Ref. 6), autopsies having shown significant plutonium burdens there (Ref. 7).

For 168-hour occupational exposures, the maximum permissible concentrations in air and water,  $(\text{MPC})_a$  and  $(\text{MPC})_w$ , are given in Table 2, from ICRP Publication 2 (Ref. 8). The maximum permissible body burdens (MPBB) are also given. Note that for individuals in the general public, the applicable MPC values are a factor of 10 smaller than those tabulated in Table 2; and for exposure to a suitably large sample of the general public, another factor of 3 smaller still (Ref. 9).

Note the comparisons in Table 2 with radium-226 and natural uranium. In insoluble form Pu, U, and Ra have comparable MPC values. However, the MPC for soluble plutonium in air is a factor of 15 more restrictive than that for  $^{226}\text{Ra}$ . In water, on the other hand, the MPC is 500 times less restrictive than for radium. These differences are direct manifestations of the biomedical-pathway toxicities.

### 4. SOURCES AND LEVELS OF ENVIRONMENTAL PLUTONIUM

Plutonium is produced by certain nuclear reactions, mainly in nuclear reactors. Its isotopic composition is variable, depending upon the method and conditions of production. The most important production mechanism creates

TABLE 2.

*168-Hour/Week Occupational Maximum Permissible Concentrations in Air  
and Water, and Maximum Permissible Body Burdens  
(from ICRP Pub. 2, Ref. 8)*

Isotope	Soluble or Insoluble	Critical Organ	$(\text{MPC})_w$ $\mu\text{Ci/liter}$	$(\text{MPC})_a$ $\text{pCi/liter}$	MPBB ( $\mu\text{Ci}$ )
Plutonium (238, 239, 240, 242)	Soluble	Bone	0.05	0.0006	0.04
Radium (226)	Soluble	Bone	0.0001	0.01	0.10
Plutonium (238, 239, 240, 242)	Insoluble	Lung	---	0.01	---
Radium (226)	Insoluble	Lung	---	0.02	---
Natural Uranium	Insoluble	Lung	---	0.02	---
Plutonium (238, 239, 240, 242)	Insoluble	GI tract	0.30	---	---
Natural Uranium	Insoluble	GI tract	0.20	---	---



$^{239}\text{Pu}$  by neutron capture on  $^{238}\text{U}$  to form  $^{239}\text{U}$ , which  $\beta$ -decays (23.5 minutes) to  $^{239}\text{Np}$ , which in turn  $\beta$ -decays (2.3 days) to  $^{239}\text{Pu}$ . Plutonium-240, the next most abundant isotope after  $^{239}\text{Pu}$ , is formed by a second neutron capture on one of the 239 isotopes. In plutonium being produced for weapons, the  $^{240}\text{Pu}$  concentration must be kept below the few-percent level; otherwise, too many spontaneous neutrons from  $^{240}\text{Pu}$  will either set off a premature chain reaction or decrease the energy yield of the explosion (Ref. 10). In plutonium for commercial power-reactor use, on the other hand,  $^{240}\text{Pu}$  is expected to be present at about 30% abundance (Ref. 10). At Rocky Flats (where weapons-grade plutonium is produced), a typical mixture contains the following isotopic fractions (Ref. 11):

$^{238}\text{Pu}$	0.04 $\pm$ 0.01%
$^{239}\text{Pu}$	93.34 $\pm$ 0.5 %
$^{240}\text{Pu}$	6.00 $\pm$ 0.5 %
$^{241}\text{Pu}$	$\sim$ 0.58 %
$^{242}\text{Pu}$	$\sim$ 0.04 %

The concentration of americium-241 is dependent upon the time for ingrowth from its parent,  $^{241}\text{Pu}$  (13.2 years).

Nearly all of the plutonium in the world today exists at only a few types of locations: in nuclear weapons; in the nuclear-reactor fuel cycle; and in stockpiles. The most probable sources of plutonium in the environment are equally easy to classify: nuclear weapons already exploded; wastes or leaks from the reactor fuel cycle, most likely around plutonium production facilities or fuel reprocessing plants; and occasional unique events, such as the burnup of a plutonium-fuelled power cell in a satellite accident (Ref. 12, 13).

Studies of plutonium in surface air have been carried out for several years by the U.S.A.E.C.'s Health and Safety Laboratory. A recent compilation (Ref. 14) indicates that in 1969, typical  $^{239}\text{Pu}$  world-wide concentrations were in the range from 1 to  $50 \times 10^{-8}$  pCi/liter of air. The higher number is a factor of 40 lower than the  $(\text{MPC})_a$  value for exposure to large general populations; the smaller number is 2000 times lower. In 1966, the average  $^{238}\text{Pu}/^{239}\text{Pu}$  ratio in air was 0.04 (Ref. 15), and the monthly surface deposition of  $^{239}\text{Pu}$  from precipitation ranged from about 0.5 to about 1.0 pCi/m<sup>2</sup> (Ref. 15). Studies of plutonium in surface soil in the Livermore Valley of California have found  $^{239}\text{Pu}$  surface levels of 200 to 2000 pCi/m<sup>2</sup> (Ref. 16). Children's diets in 1966 showed  $^{239}\text{Pu}$  concentrations of about 0.002 to 0.006 pCi/kg (Ref. 15). This is more than 5 orders of magnitude smaller than the general-population  $(\text{MPC})_w$ .

These world-wide distributions are mostly due to fallout from nuclear weapons tests, plus the aforementioned satellite accident, which deposited 17,000 curies of  $^{238}\text{Pu}$  in the stratosphere over the Indian Ocean in 1964 (Ref. 12, 13). As the numbers reveal, the world-wide levels are not now significant fractions of the MPC limits for general-population exposure; and since fallout hopefully will not increase dramatically, the levels should also not increase dramatically. The main radiological hazard is thus more likely local than worldwide in scope. Local concentrations can range up to values much higher than the worldwide averages just mentioned. One example is at the Nevada Test Site, where many locations have  $^{239}\text{Pu}$  in soils at levels of 100's to 1000's of pCi/gram over several acres (Ref. 17). Another example is near the Rocky Flats plutonium production facility, where plutonium air concentrations have occasionally reached as much as 30% of the soluble  $(\text{MPC})_a$  for individuals in the general public (Ref. 18).

One important point which must be borne in mind is that plutonium deposited on soil is not a significant direct public health hazard. One prime pathway to man is resuspension from the ground into the air, which can result in widespread distribution of an originally well-localized source, and which in turn may lead to more widespread public health exposures. This possibility has motivated a program within the U.S.A.E.C. to study the problems of resuspension (Ref. 19).

## 5. MEASUREMENT CONSIDERATIONS

Among the environmental media requiring measurement of plutonium activity are silt and soil; natural and waste waters; food; and air. The techniques which have been developed fall into several categories. We shall discuss them separately below. These categories are:

- Radiochemical Analysis Followed by Counting or Spectroscopy
- Air Filter Collection Followed by Counting
- Gamma and X-Ray Spectroscopy
- Other Methods

Because plutonium concentrations in most samples are small, and because the decay alphas have such short ranges in matter ( $\sim 5$  to 10 mg/cm<sup>2</sup>), direct alpha counting of environmental samples is seldom practical.

### A. Radiochemical Analysis Followed by Counting or Spectroscopy

A wide variety of techniques for radiochemical analysis have been developed for



plutonium determinations in various media. Chemical separation produces enrichment with respect to the matrix in which the plutonium is found, and makes possible alpha counting or alpha spectroscopy with increased sensitivities.

The main difficulty in alpha detection is the extremely short range of the alphas (5 to 10 mg/cm<sup>2</sup>), which requires preparation of a nearly 'weightless' sample. An important chemical-analysis difficulty is that the weight of the plutonium in samples may be minute: 1 pCi of <sup>239</sup>Pu weighs only 16 pg.

Perhaps the greatest problem with plutonium radiochemistry is the insolubility of one common chemical form, plutonium oxide. We quote from Healy (Ref. 20):

"Plutonium oxide fired at high temperatures is a notoriously insoluble ceramic. If present at larger particle sizes it is exceedingly difficult to put into form for a chemical separation. However, very small particles such as are found in natural fallout or other materials, including oxide which has been formed at a lower temperature, are much more soluble and can be leached from the sample with relatively mild treatment."

The media requiring chemical analysis are usually of three general classes: water samples, air-filter samples, and soil samples. Other possible media are food, urine, and biological and fecal samples. The general problem of low solubility may be present in nearly all of these media.

Chemical methods can be divided into two general classes: those that bring the entire sample into solution, and those that use a leaching technique to extract the plutonium from the matrix. The leach methods are generally easier, but are also more liable to error ... hence, periodic total-dissolution and cross-comparisons are often used to check the yields from the leach methods.

In all of the radiochemical analyses, the use of either plutonium-236 or plutonium-242 as a tracer for chemical yield is common. Both <sup>236</sup>Pu (5.72, 5.77 MeV) and <sup>242</sup>Pu (4.86, 4.90 MeV) decay by emitting alphas of sufficiently different energy that they can be resolved from <sup>239</sup>Pu by using alpha spectroscopy. The tracer should be added at the earliest possible stage, of course. Unfortunately, the tracer technique cannot adequately correct for insolubility losses, one of the commonest problems, since soluble tracer does not exchange with insoluble <sup>239</sup>PuO<sub>2</sub> unless there is complete dissolution to soluble ionic forms (Ref. 22). Another consideration, empha-

sized by Sill (Ref. 22), is that *at least as much* tracer should be used as unknown Pu being determined. Otherwise, the statistical uncertainty in the determination is limited by the <sup>236</sup>Pu statistics. Lindeken (Ref. 23) has pointed out that one advantage of <sup>242</sup>Pu is that it has a lower energy, so that there are no contributions to the <sup>239</sup>Pu energy region from possible energy degradation of the tracer's alpha.

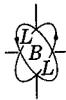
Sill (Ref. 24) describes a method for soils, in which "the sample is decomposed completely by a combination of potassium fluoride and silicon tetrafluoride. The cake is dissolved in dilute HCl and all alpha emitters are precipitated with barium sulfate." Sill claims recoveries of 94 ±3% for all elements from thorium through plutonium, and describes a method for separating and electrodepositing three fractions for separate alpha spectroscopy: (Am, Cm, and Cf), (Th), and (Pa, U, Np, and Pu). Sill also claims that the method is reliable "in soils in which complete dissolution of both the siliceous matrix and the most insoluble forms of plutonium oxide that can be produced can be guaranteed routinely" (Ref. 24).

Another total-dissolution method has been described by Talvitie (Ref. 25), in which adsorption from a hydrochloric acid medium is followed by a nitric acid wash for removal of iron. Anion exchange is then used for plutonium separation, followed by electrodeposition.

The leaching methods, which are commonly used for soil analysis, are claimed to be easier and faster than the total dissolution methods. One method prominently mentioned in the literature is that of the U.S.A.E.C.'s Health and Safety Laboratory (Ref. 26, 27). This method is designed for 100-gram soil samples. Soil is leached with a mixture of HNO<sub>3</sub> and HCl. Leached Pu is converted to Pu(IV) with sodium nitrite and absorbed from 8N HNO<sub>3</sub> for electroplating.

This method presents problems in the presence of highly insoluble PuO<sub>2</sub>, and the U.S.A.E.C.'s Nevada Applied Ecology Group (Ref. 28) has adopted instead a Los Alamos-modified technique, the so-called "modified HASL-LASL" method (Ref. 29), which uses HF as well as HCl in the leach process to improve the yield. After leaching, an anion exchange resin column is used for Pu separation followed by electroplating. The electroplating step may present an important roadblock, because it limits the number of samples which can be run in parallel unless many electroplating cells are available. Also, even this improved method may suffer from yield difficulties when very large samples are analyzed.

Levine and Lamanna (Ref. 30) describe methods in which "solubilized plutonium as



the chloride is reduced to the tri-valent state, coprecipitated with lanthanum carrier as the fluoride, dissolved in acid aluminum nitrate and extracted with thenoyltrifluoroacetone (TTA) for separation and removal of impurities and carrier. The plutonium is then back-extracted with strong acid, oxidized and electrodeposited on stainless steel for counting." Bowen et al. (Ref. 31) report a similar procedure.

#### B. Air Filter Collection Followed by Counting

The most common method for monitoring plutonium in air is by the collection of activity on an air filter, followed by direct counting of the collected activity. Unless an identification of the isotopes is possible from separate knowledge, the type of measurement is really a "gross-alpha" determination. The MPC<sub>a</sub> for soluble plutonium is  $6 \times 10^{-5}$  pCi/liter for individuals in the general public. The counting rate from such an activity level is extremely low:  $6 \times 10^{-5}$  pCi gives about 1.3 disintegrations per week. Measurement of such activities requires the sampling of many liters of air. There are two different classes of measurements: those which measure continuously while the activity accumulates, and those which measure the activity after a fixed collection period. The latter are the more sensitive, and the more frequently used.

In all of the methods to be described below for air filters, no distinction is made between soluble and insoluble plutonium. Thus the measurement must be considered as, and compared to, the soluble MPC<sub>a</sub> (which is the more restrictive).

One common collection system has been used by Lawrence Livermore Laboratory personnel at the Nevada Test Site (Ref. 32). An air sampler with a pumping speed of 10 ft<sup>3</sup>/min (1 ft<sup>3</sup>/min = 28.4 liters/min) pulls air through a convoluted fiberglass filter. Particles as small as 0.025 μm are collected at >99% efficiency, and 90% efficiency can be obtained even at sizes as large as 50 μm.

At Rocky Flats, sampling at about 1 ft<sup>3</sup>/min is used (Ref. 33). Using a rate of 10 ft<sup>3</sup>/min, and with an air concentration equal to one general-public-MPC<sub>a</sub>, it would take about 1 hour of sampling to collect 1 pCi (= 2.2 dpm) on the filter.

The most important background is usually from the daughters of radon-222. The MPC<sub>a</sub> for radon daughters is about 5 orders of magnitude higher than for plutonium, and radon occurs naturally in typical surface air at concentrations of 0.1 to 1 pCi/liter. In some locations (for example, in some areas in the

Rocky Mountains) even higher ambient concentrations are found. Radon-220 (thoron gas) and its daughters can also provide a similar type of background, but usually at much smaller concentrations. Since the filters are efficient collectors of radon and thoron daughters, any measurement system for plutonium must be able to discriminate against the much larger radon background.

There are four techniques which are commonly used to achieve the necessary discrimination:

- i. alpha spectroscopy
- ii. alpha-to-beta ratio method
- iii. radiochemical analysis
- iv. 4-day hold-up to allow for decay of the short-lived activities

#### i. Alpha Spectroscopy

Measurement of the alpha energy spectrum from the activity deposited on an air filter is very effective for discrimination against radon and thoron daughters. The alpha energies from decay of radon-222, polonium-218, and polonium-214 are 5.49, 6.00, and 7.69 MeV, respectively. The highest-energy alpha from an important plutonium isotope is <sup>238</sup>Pu's 5.50 MeV, which would be difficult to resolve in the presence of large amounts of <sup>222</sup>Rn. <sup>239</sup>Pu emits in the 5.15 MeV region, which is easily resolvable. The daughters of radon-220 (thoron) all emit alphas with energies above 6 MeV, and are therefore quite easily rejected. At Rocky Flats, Griep (Ref. 33) has performed alpha spectroscopy using a large surface-barrier detector, with an area of 350 mm<sup>2</sup>. An earlier device using a different solid-state detector was developed at Lawrence Livermore Laboratory (Ref. 34). These types of instruments are capable of discriminating between <sup>239</sup>Pu and the higher-energy backgrounds with plutonium activity of less than 1 pCi/m<sup>3</sup> of air, sampled for 8 hours. Both of the instruments just mentioned were designed primarily for occupational rather than environmental monitoring. The Livermore instrument can easily detect one 40-hr-occupational MPC<sub>a</sub> with an 8-hr sampling time at 1 ft<sup>3</sup>/min: this concentration yields about 15 counts/min in the detector, which has a 25% overall efficiency.

Figure 1 (from Ref. 34) shows the spectrum from the Livermore detector when radon daughters are present together with plutonium. The sampling parameters are 1 ft<sup>3</sup>/min for 8 hr. The radon daughters reach equilibrium very quickly: Ra A (<sup>218</sup>Po) has a 3.05 min half-life. Therefore their activity levels on the filter are smaller by orders of magnitude than their total collected activities.

Use of an internal gas proportional counter for alpha spectrometry yields resolutions nearly as good as with a solid-state detector, with the added ability to count larger filter areas.

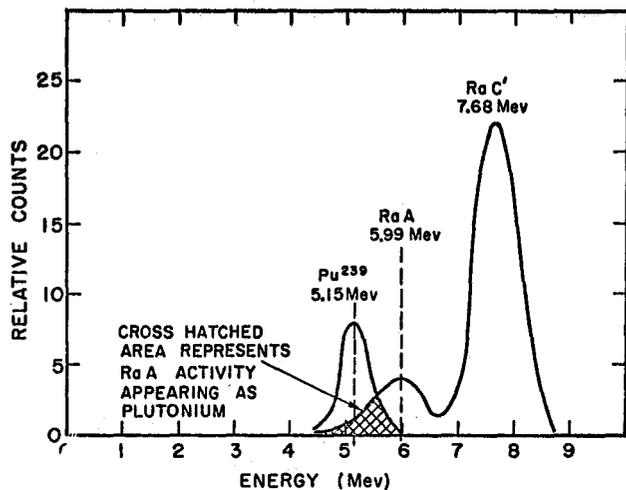


FIGURE 1. Radium A (Polonium-218) interference in plutonium detection (from Ref. 34).

#### ii. Alpha-to-Beta Ratio Method

In this method, plutonium is distinguished from the radon and thoron daughters using the fact that the important natural alpha emitters have a special signature using  $\alpha$   $\beta$  coincidence counting. Ra C ( $^{214}\text{Bi}$ ) emits a  $\beta^-$  almost immediately before Ra C' ( $^{214}\text{Po}$ ) emits its alpha ... the same is true of Th C ( $^{212}\text{Bi}$ ) and Th C' ( $^{212}\text{Po}$ ). This is because the half-lives of Ra C' and Th C' are so short: 164  $\mu\text{sec}$  and 0.30  $\mu\text{sec}$ , respectively. Thus  $\alpha$   $\beta$  coincidences can be used to measure the level of the radon/thoron-daughter activities. One difficulty with this technique is that it relies on comparing two (sometimes) large numbers to obtain the (sometimes small) plutonium activity.

When a filter is measured by two different (alpha- and beta-sensitive) detectors, lack of 100% detection efficiencies means that there cannot generally be a one-to-one subtraction of the  $\alpha$   $\beta$  coincidences. Rather, the  $\alpha$   $\beta$  coincidence rate must be electronically weighted before subtraction. Given a fixed resolving time and fixed detection efficiencies it is theoretically possible to cancel out both the radon and the thoron chains exactly, aside from statistical fluctuations (Ref. 35), provided the deposited daughters are in equilibrium. To do this subtraction properly, very careful calibrations are required. Because thoron daughters have a longer average half-life, they will dominate for collection periods in the 10-hr range, while for a 1-hr collection period radon daughters usually dominate (Ref. 36). However, for a given detector, one must take care to equalize the detection efficiencies for the Th C and Ra C betas, which

have very different energy spectra. This can be done by using an appropriate absorbing foil to properly compensate for the different energies.

Typical instruments of this design have been described by Spaa (Ref. 36), Rankin (Ref. 37), and Gupton (Ref. 38), among others. Spaa's design aims for continuous alarm capability. A filter paper is moved semi-continuously (stepwise) across a 70 liter/min pump. Continuous counting is performed during activity accumulation, using a ZnS(Ag)/photomultiplier alpha detector and an end-window G-M beta counter. Every 12 hours the filter is automatically moved stepwise, and is available for more precise measurements if required. In the presence of radon daughters at 10 pCi/liter of air, plutonium at 0.002 pCi/liter was the approximate limit of detection after about 1 hr of sampling.

Rankin (Ref. 37), using solid scintillation detectors for both  $\alpha$  and  $\beta$  counting, detected plutonium at a lower limit of about 800 pCi in the presence of 0.1 pCi/liter of radon-222. With a 24-hr sampling time at 10 ft<sup>3</sup>/min, this also corresponds to a plutonium concentration of 0.002 pCi/liter of air. Gupton's instrument (Ref. 38) is very similar, but also provides another G-M detector which is not exposed to the filter, for external gamma compensation.

Tanaka et al. (Ref. 35) describe an improvement of the technique in which an alpha spectrometer is used to further select the  $^{239}\text{Pu}$  signal. A CsI(Tl) scintillation spectrometer is used, and alpha particles are selected from beta particles by pulse-shape discrimination. In 24 hours, it can detect plutonium levels as low as  $10^{-4}$  pCi/liter in the presence of radon at about 0.1 pCi/liter.

#### iii. Radiochemical Analysis of Air Filters

A method for radiochemical analysis of air filters has been adopted as a tentative method by the Intersociety Committee (Ref. 39). This method is similar to one found in the HASL Manual (Ref. 27). "The plutonium is ... isolated by co-precipitation with cerium and yttrium fluorides ... further purified by a double anion-exchange column technique ... finally electro-deposited onto a platinum disk" (Ref. 39). The first ion-exchange process is from HCl, the second from an HNO<sub>3</sub> medium. A solid-state spectrometer determines the Pu isotopes;  $^{236}\text{Pu}$  is used as a tracer. This method claims a practical detection limit of about  $8 \times 10^{-8}$  pCi/liter with  $\pm 10\%$  error (1  $\sigma$ ), when air is collected at about 700 liters/min for about one week. For smaller volumes of air, sensitivity is reduced proportionately. The difficulty with this method is, as might be expected, that dissolution of high-temperature-fired PuO<sub>2</sub> is not necessarily complete, leading to possible underestimation of



the concentration. However, since the MPC<sub>a</sub> for soluble plutonium is smaller by a factor of 16 than that for insoluble plutonium, the problem of insolubility losses is usually not serious for air-filter measurements.

### C. Gamma and X-Ray Spectroscopy

Gamma and x-ray spectroscopy is an important plutonium measurement technique in some situations, and may be thought of as complementary to the radiochemical-analysis methods, since the applications of this type of spectroscopy are usually for measurements of an entirely different class. The reader is referred to a separate section of this Volume for a more detailed treatment of this class of instruments (See "Gamma Spectrometry").

The field survey of large surface areas for possible plutonium contamination is one such application: for example, much effort has gone into large-area surveys around the Nevada Test Site. Another application is for plutonium assay in media which are not accessible to radiochemical analysis: examples are large soil samples, bioassay of plutonium in lungs, or plutonium detection in a solid-waste container.

The method usually relies on detecting decay photons from one of two important phenomena. The first phenomenon is the 59.6-keV gamma ray emission which accompanies 36% of the decays of <sup>241</sup>Am, the daughter of <sup>241</sup>Pu. <sup>241</sup>Pu is seldom the main plutonium isotope present, and even if it were it is not usually the main hazard, being a beta-emitter. Therefore, this method is useful for gross-plutonium measurement only if one knows both the 'age' of the plutonium (to determine ingrowth of <sup>241</sup>Am) and the original isotopic ratios.

The second phenomenon is the 17-keV x-ray emission group which accompanies about 3.8% of the <sup>239</sup>Pu decays. Actually, the emissions are L x-rays from the daughter <sup>235</sup>U. The energies of the three most prominent lines are 13.6, 17.2, and 20.2 keV with relative intensities 100:120:25 (Ref. 40). Americium-241 decays are also accompanied by x-rays in the same 17-keV region (actually the L x-rays of its daughter, <sup>237</sup>Np). The energies of the americium lines (Ref. 41) are 14.0, 17.8, 20.8, and 26.3 keV, with relative intensities (photons/alpha) of 12%, 13%, 3%, and 2.5%. (Note again for comparison the 36% 59.6-keV americium gamma.)

The task of field-surveying for plutonium contamination on the ground has received considerable attention, although it actually has a rather narrow range of application such as at the Nevada Test Site, near plutonium

production facilities, and so on. Here the goal is to locate and quantify the surface activity ( $\mu\text{Ci}/\text{m}^2$ ), rapidly and conveniently. The presumption is made that detailed soil sampling must be relied on for more refined measurements.

One instrument type often used for this task is a thin NaI(Tl) crystal of large area, coupled to a photomultiplier tube. The archetype specifically for plutonium contamination is the FIDLER detector developed at Lawrence Livermore Laboratory (Ref. 42). It has a 5" diameter 1/16" thick NaI(Tl) crystal, and is designed to be carried by hand over the terrain at about one foot above ground level. The battery-operated electronic circuitry separates and counts two 'windows' in the energy spectrum, one around 17 keV and the other around 60 keV. These are shown, along with a spectrum from a pure <sup>241</sup>Am source, in Fig. 2 (from Ref. 42). An instrument of this type must be calibrated against several variables. First, its response is sensitive to activity distributed over a wide area. Further, the thickness and composition of any overburden material, as well as the plutonium isotopic distribution, will affect the interpretation of the results.

Assuming a 12-second time-constant for the count-rate meter and that the instrument is being carried slowly over the terrain, the minimum detectable <sup>241</sup>Am point source is about 9400 pCi,

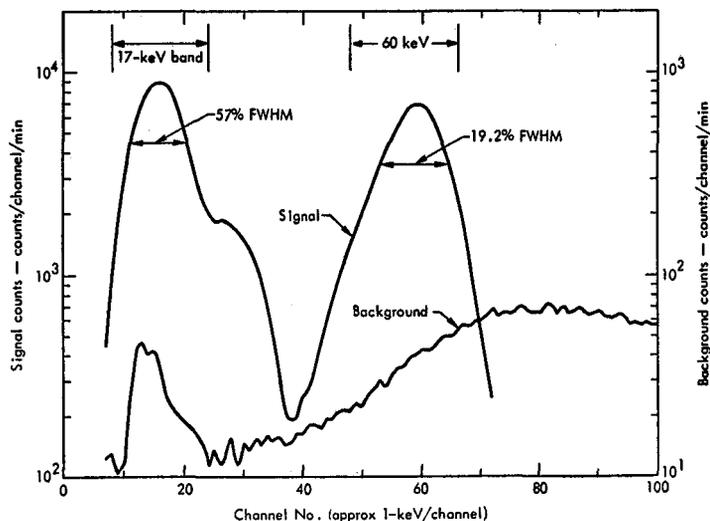


FIGURE 2. Spectra for <sup>241</sup>Am and background as measured with a 5-in.-diam by 1/16-in.-thick NaI(Tl) scintillation detector. Source was 10.7- $\mu\text{Ci}$  <sup>241</sup>Am, and source-to-detector distance was 30.5 cm. Optimum discriminator settings for 17-keV band and 60 keV are shown (from Ref. 42).



and the minimum detectable area-distributed source is 19,200 pCi/m<sup>2</sup>, both at 3.3  $\sigma$  above a natural background of about 200 counts/min.

An instrument similar to the FIDLER but using a 5"-diameter, 1/8"-thick CaF<sub>2</sub>(Eu) scintillator has been developed by Nuclear Chicago (Ref. 43). The claimed advantages of CaF<sub>2</sub>(Eu) are that it does not require the careful hermetic sealing characteristic of NaI(Tl); and that there is better optical coupling to a tapered conical light pipe, which partially compensates for the lower light output (40-50% of that for NaI(Tl)).

A similar instrument, but using an 8"-diameter, 0.2"-thick NaI(Tl) crystal, has been used for assay of the lung burden of exposed individuals (Ref. 44). For a 100-minute counting time, the minimum detectable activity (3  $\sigma$ ) was found to be 6200 pCi of <sup>239</sup>Pu. Counting efficiency must be determined with a phantom, because absorption and scattering of the low-energy photons is very large. The counting efficiency (counts per <sup>239</sup>Pu disintegration) was measured to be only about 4 x 10<sup>-4</sup>, the majority of the x-rays being absorbed or scattered. The principal irreducible background is from degraded gammas due to the decays of <sup>40</sup>K and <sup>137</sup>Cs in a normal body.

Use of twin large-area gas proportional counters is also common for Pu lung assays. With this technique Tomitani and Tanaka (Ref. 40) report 12.6% resolution (full width half maximum) on the 13.6 keV line; this is shown in Figure 3. With an outer ring of wires for anti-coincidence shielding to provide a veto whenever cosmic ray particles traverse the chamber, the minimum detectable activity (3  $\sigma$ ) is 6000 pCi of <sup>239</sup>Pu, using a 100-minute counting time.

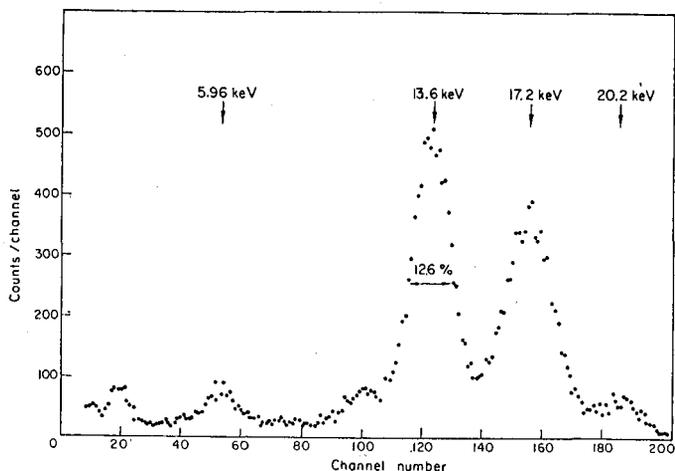


FIGURE 3. X-ray spectrum of <sup>239</sup>Pu. 13.6, 17.2 and 20.2 keV X-rays are L<sub>α</sub>, L<sub>β</sub> and L<sub>γ</sub> characteristic X-rays of <sup>235</sup>U, respectively. X-rays of energy about 5.96 keV are the fluorescent X-ray of the stainless steel housing (from Ref. 40).

The backgrounds are higher using NaI(Tl) for lung-counting applications, but the detection efficiencies are also higher. It seems that the proportional counters as described in Ref. 40 are more difficult to construct and operate. Overall, the two types of instruments have comparable sensitivities.

A solid-state Ge(Li) assay technique has been developed by Tyree and Bistline (Ref. 45). This system can resolve the 51.6-keV gamma ray from <sup>239</sup>Pu in the presence of the 59.6-keV <sup>241</sup>Am gamma. The 51.6-keV <sup>239</sup>Pu gamma occurs with an 8.4% branching ratio. Using this system and a 1-hr counting time, minimum detection (3  $\sigma$ ) of a few thousand pCi of <sup>239</sup>Pu was possible. These authors also report detection of the 14.0-keV <sup>241</sup>Am line with 3.7% resolution (full width, half maximum) using a Si(Li) detector, compared to 14% using a proportional counter (Ref. 46). This is shown in Fig. 4.

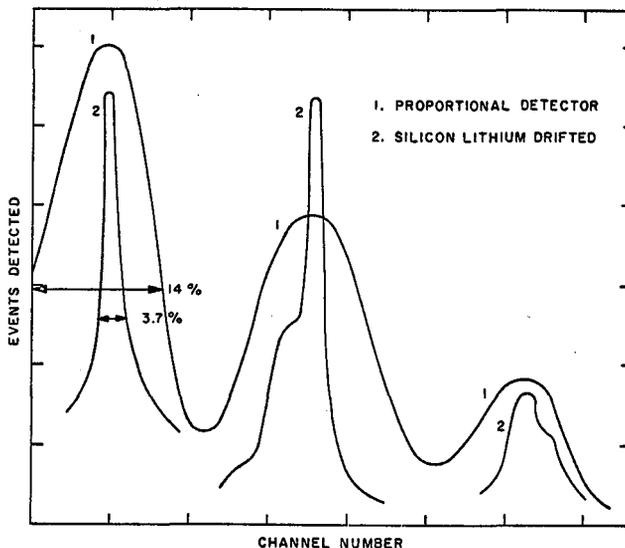
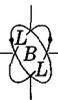


FIGURE 4. Proportional and Si(Li) detector spectra from americium-241. The three peaks shown have energies of 14.0, 17.8, and 20.8 keV, and are actually L x-rays from neptunium-237 (from Ref. 46).

Tyree and Bistline (Ref. 46) report use of the relative attenuation of the various x-ray lines to give a measurement of the depth of a contaminating source, such as plutonium in a wound. Several-mm depths are measured to fractions of one mm. The relative attenuations of the 13 and 20 keV uranium L x-ray lines with thickness are shown in Figure 5 (Ref. 47).



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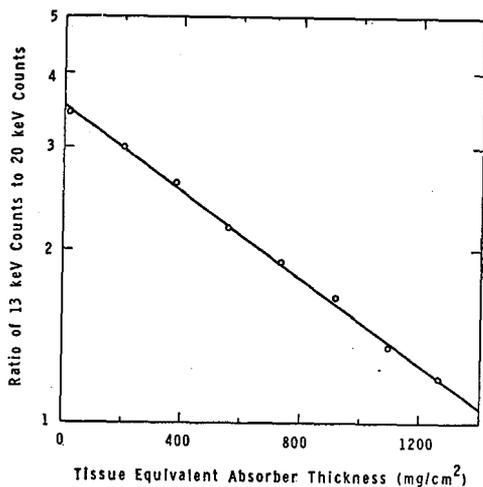


FIGURE 5. Plot of the ratio of the 13 and 20 keV uranium L x-ray intensities as a function of increasing absorber thickness (from Ref. 47).

Another detector type useful for this class of measurement consists of a thin NaI(Tl) crystal optically coupled to a much thicker CsI(Tl) crystal behind. This system is usually called a 'phoswich' detector. The low-energy x-rays are totally absorbed by the NaI(Tl), while higher-energy photons (or charged particles) count in the CsI(Tl). Both scintillators are viewed by the same photomultiplier tube (see Fig. 6), and the signals can be differentiated by pulse-shape analysis since NaI(Tl) has a 250-nsec optical decay time compared to 1100 nsec for CsI(Tl). This type of instrument can suppress high-energy backgrounds by factors of 3 to 10.

Koranda et al. (Ref. 48) have developed a semi-portable Ge(Li) spectrometry system for field surveys of soil. The system rides in a van, and is designed to combine the better resolution of a solid-state detector system with the needs for mobile surveying in the field. The system seems more valuable for the higher energy gammas from fission products and natural emitters, rather than for the low-energy plutonium photons.

#### D. Other Methods

Several other types of determinations, which do not ultimately rely on alpha or gamma counting for the plutonium measurement, have been developed. We shall not discuss these here at all, but shall merely refer the reader to some references for details. Generally these methods are not sufficiently sensitive for environmental measurements. (i) *Controlled-potential coulometry* (Ref. 49) is useful when Pu concentrations are high ( $\geq 0.1$   $\mu\text{Ci/liter}$ ) and accuracy is required. (ii) *Constant-current potentiometric titration* (Ref. 50) is also commonly used at these concentrations. (iii) *Amperometric techniques* (Ref. 51) can be used for accurate work with very large ( $\geq 10$   $\mu\text{Ci/liter}$ ) concentrations. (iv) *Spectrophotometry* (Ref. 52) is applicable above about 0.1  $\mu\text{Ci/liter}$ . (v) *Square wave polarography* (Ref. 53) is sensitive down to  $\sim 0.1$   $\mu\text{Ci/liter}$ .

#### 6. SUMMARY AND CONCLUSION

In this section we have given overviews of the typical levels at which plutonium is found in environmental media; of the radiation

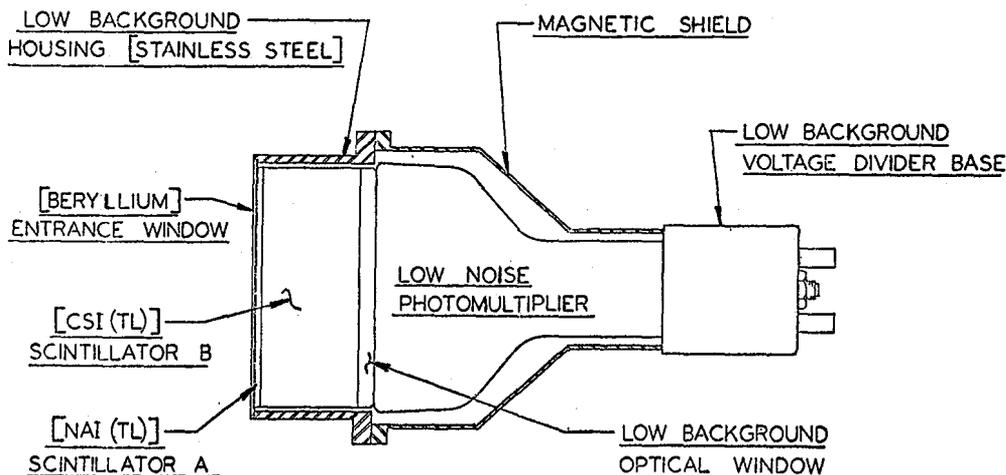


FIGURE 6. Harshaw phoswich detector: matched window type for x-ray and low energy gamma detection.



protection guides; and of some of the types of measurement techniques which have been developed for plutonium measurements in the environment.

The material is very dangerous and the radiation protection guides are necessarily very restrictive. Fortunately, important plutonium contamination in the environment is not widespread. A small level of world-wide plutonium from fallout is supplemented by higher concentrations in only a few well-localized areas, where plutonium contamination may be an important radiobiological factor.

The ideal instrument or technique for plutonium determinations should be capable of measuring concentrations down to fractions of MPC levels in the appropriate medium. There are two main classes of measurements, field and laboratory measurements. Of the types of measurements considered here, only two are of the field-type: surveys of plutonium deposition on the ground, and air concentration measurements. Even the latter are often performed by laboratory analysis of an air filter after field sampling.

We shall summarize the situation in each of the various measurement areas separately:

#### A. Measurements of Plutonium on and in Soil

There are two basically different techniques: first, field surveying; and second, radiochemical analysis followed by alpha counting or spectroscopy. The field-survey measurements demand less accuracy, and are now performed by carrying counting instruments above the terrain. The archetype of this class is the FIDLER, which has a thin, large-area NaI(Tl) scintillation detector. Instruments of this type seem adequate for most rough field-survey measurements. Radiochemical analysis techniques are intrinsically much more sensitive and accurate, but suffer from their own types of difficulties. The most precise and surefire techniques involve total dissolution ... but these are laborious, time-consuming, and expensive. There are a number of leaching methods which are intrinsically faster and cheaper. Many of these present difficulties in the presence of insoluble matter, such as high-temperature-fired PuO<sub>2</sub>. There certainly seems to be need for further improvement in the analytical chemical methods. In particular, the development of a faster and less expensive leach method possessing a high yield would be desirable.

#### B. Measurements of Plutonium in Air

Measurements of plutonium in air are nearly always made by collecting particulate

activity on an air filter for counting. The main background is usually due to daughters of radon, which must be discriminated against. One method is to perform radiochemical analysis, followed by spectroscopy. This technique, while intrinsically sensitive and accurate, is expensive, time-consuming, and liable to radiochemical-yield errors. The two most common methods which use direct counting of the filter are the alpha beta coincidence technique and alpha spectrometry. Both are sensitive and precise; perhaps  $\alpha$  spectrometers are to be preferred for the most precise and background-free requirements, but the  $\alpha$   $\beta$  coincidence technique is advantageous since it has been adapted for continuous measurements during sample collection. However, because of the very low MPC<sub>a</sub> value, none of the methods presently available combines rapid or continuous detection with sensitivity well below MPC<sub>a</sub>.

#### C. Measurements in Soil or Bioassay Using Gamma Spectroscopy

Bioassays of lung or other body burdens, and also measurements in bulky samples such as large amounts of soil, can be performed with gamma spectrometers, typically using either NaI(Tl) or Ge(Li) detectors. These methods all suffer from one intrinsic problem: it is necessary to trade off resolution for sensitivity. High resolution Ge(Li) detectors capable of high detection efficiency are available only at great expense; on the other hand, the much larger NaI(Tl) crystals suffer from poor resolution which is a handicap especially in the presence of higher-energy gamma fluxes. This problem is inherent to the method.

#### D. Measurements in Other Media

Measurements in media such as urine, plants, food, and tissue are typically made radiochemically. There is also a wide variety of chemical methods (coulometry, spectrophotometry, polarography, etc.), which are mainly useful at relatively high concentrations. The radiochemical analyses of various media usually consist of initial sample-treatment steps designed to bring all of the plutonium into solution, followed by analysis identical to that used in the analysis of soil or water.

It seems probable that more sensitive and faster methods for measurements of plutonium in man can be developed, using extensions of presently-available technology. The motivation for this is probably not great at the present time since its application would be limited largely to measurements in a few occupational-exposure situations.



E. Conclusions

At present, measurements at levels significantly below MPC values for individuals in the general public are possible in each of the main measurement situations where public health is involved or where pathways in the environment require study. In particular, radiochemical analyses in the laboratory followed by alpha spectroscopy are quite sensitive and suffer from few interferences. Unfortunately, the best of these methods are expensive and time-consuming.

Given the probable growth in use of plutonium as the nuclear power industry expands and as the breeder-reactor program grows, it seems important that methods be developed which enable sensitive but inexpensive determinations of plutonium in air and water effluents, as well as in soil and biomedical media.

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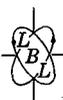


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RADIONUCLIDE INSTRUMENT NOTES

Included in the following pages are Instrument Notes for those instruments useful for radionuclide measurements of various kinds. The reader should be aware of the following facts, in order to find all Instrument Notes appropriate for radionuclide measurements.

- 1.) The filing system lists Instrument Notes alphabetically by manufacturer.
- 2.) Instruments for general spectrometry are found elsewhere: under RAD-ALP, RAD-BET, or RAD-GAM, as appropriate. These instruments are often similar to instruments found in this section (RAD-NUC), the distinction usually being based upon the manufacturer's advertised specifications. Thus if a manufacturer claims that an instrument is useful for a specific radionuclide, or for a broad class of nuclides (e.g., "gases"), it is found here; general spectrometry instruments are elsewhere.
- 3.) All liquid-scintillation systems are in this section, rather than under beta-detection (RAD-BET).



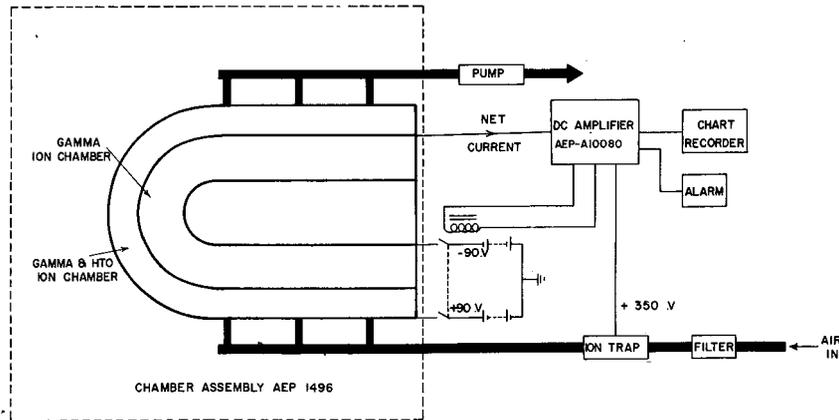
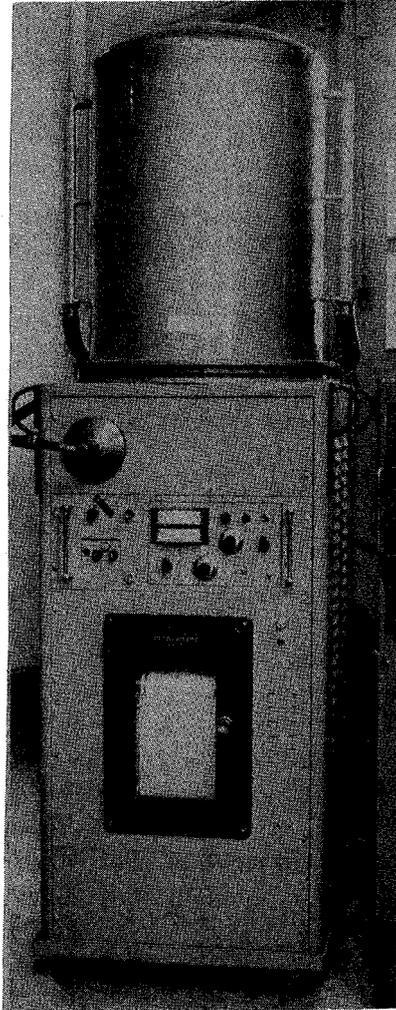
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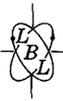


INSTRUMENTATION  
FOR ENVIRONMENTAL  
MONITORING

RAD-NUC  
Tritium  
AECL/Chalk River  
Aug. 1972

Tritium Air Monitor  
AECL/Chalk River





INSTRUMENTATION  
FOR ENVIRONMENTAL  
MONITORING

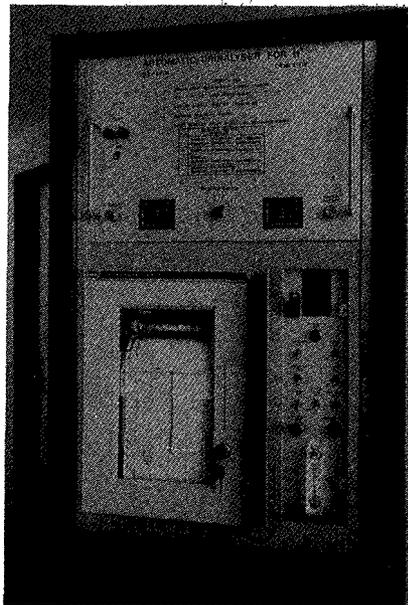
RAD-NUC  
Tritium  
AECL/Chalk River  
Page 2 Aug. 1972

Class	Laboratory area monitor
Stage of Development	Not commercially available
Principle of Operation	Flow-through ionization chamber, with compensating chamber for external gamma compensation. 40-liter volume chamber. Electrometer, dc amplifier, meter or chart recorder output.
Sensitivity and Range	Sensitivity: Down to 1 pCi/cm <sup>3</sup> of air; 6 ranges, up to 2500 pCi/cm <sup>3</sup> full scale
Sampling	Continuous; air, filtered through glass wool, is pumped through chamber at 36 liters/min; ion trap between filter and chamber
Requirements	Power: 115 V ac Size: 60" height (150 cm)
Features	1. Response time about two minutes to sudden increase in activity. 2. Very insensitive ( $\approx 10\%$ ) to humidity changes. 3. Gamma background of 7 mR/hr produces reading of 15 pCi/cm <sup>3</sup> .
References	R.V. Osborne and G. Cowper, <u>The Detection of Tritium in Air with Ionization Chambers</u> , Report AECL-2604, Chalk River Nuclear Laboratories.
Cost	Not commercially available
Remarks	Portable chambers of volumes 0.3 and 1.2 liters also developed by same group at Chalk River
Address	R.V. Osborne Atomic Energy of Canada Limited Chalk River Nuclear Laboratories Chalk River, Ontario, CANADA (613) 584-3311

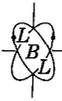


Tritium Urinalyser

AECL/Chalk River Model AEP-5216



Class	Laboratory urinalyzer
Stage of Development	Not commercially available
Principle of Operation	Sample is placed in input of instrument, then mixed with liquid scintillator, counted by two photomultipliers in coincidence.
Sensitivity and Range	Sensitivity: Minimum of 1000 pCi/cm <sup>3</sup> , maximum of 10 <sup>7</sup> pCi/cm <sup>3</sup> of urine
Sampling	Batch basis. Requires 20 cm <sup>3</sup> urine sample.
Performance	Accuracy: About ±10%, with about 1% carryover between samples Response: Two minutes between introducing sample and readout of activity. Background: About 1000 pCi/cm <sup>3</sup> Air Supply: 10 cm <sup>3</sup> /sec Output: Chart recorder, 4-decade logarithmic
Requirements	Power: 117 VAC, 250 watts Size: Fills most of one relay rack
Features	1) Automatic processing, with standard and background samples included 2) Also capable of assaying tritium activities in other samples
References	1) R.V. Osborne, "Performance of an Automatic Analyser for Tritium in Urine," Health Physics, 18, 87 (1970) 2) R.V. Osborne, "Automatic Urinalyser for Tritium AEP-5216," Report AECL-2702, Chalk River Nuclear Laboratories (1968)
Cost	Not commercially available

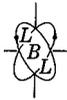


INSTRUMENTATION  
FOR ENVIRONMENTAL  
MONITORING

RAD-NUC  
Tritium in Urine  
AECL/Chalk River 2  
Page 2 Aug. 1972

Address

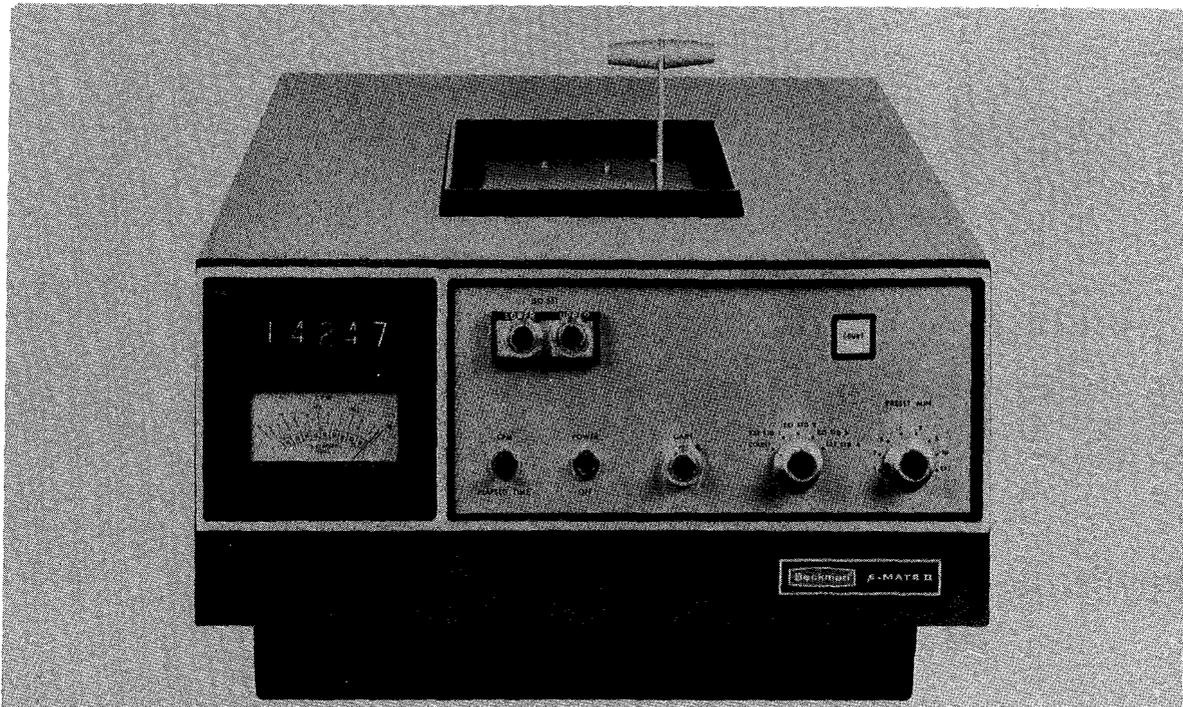
R.V. Osborne  
Atomic Energy of Canada Limited  
Chalk River Nuclear Laboratories  
Chalk River, Ontario, Canada  
(613) 584-3311



INSTRUMENTATION  
FOR ENVIRONMENTAL  
MONITORING

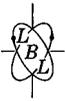
RAD-NUC  
Tritium, Carbon-14  
Beckman  
Nov. 1972

Isotope Counter  
Beckman  $\beta$ -Mate II



Class	Laboratory, Portable
Principle of Operation	Single-sample, liquid scintillation counter. External standard counts in two windows and the ratio of these counts is used for quench calibration.
Sensitivity and Range	Tritium Efficiency > 57% Carbon-14 Efficiency > 90% Carbon-14 spillover into tritium < 25%
Sampling	Accepts one standard 20 ml liquid scintillation vial (glass or polyethylene) or flow cell accessory
Performance	Volume: 8 to 18 ml Temperature: Ambient to 25°C Overall System Resolving Time: $\approx 0.5$ $\mu$ sec Coincidence Gate Resolving Time: $\approx 20$ nsec Dead Time Loss: 1% at system limit of $10^6$ cpm
Requirements	Power: 120 or 240 V ac, $\pm 10\%$ , 1 amp, 50 or 60 Hz Size: 45.7 cm W, 71.6 cm D, 72.5 cm H (18", 28.2", 28.5") Weight: 147.2 kg (325 lb)
References	Manufacturer's specifications
Cost	\$4150.
Address	Beckman Instruments, Inc. 2500 Harbor Boulevard Fullerton, CA 92634 (714) 521-3700

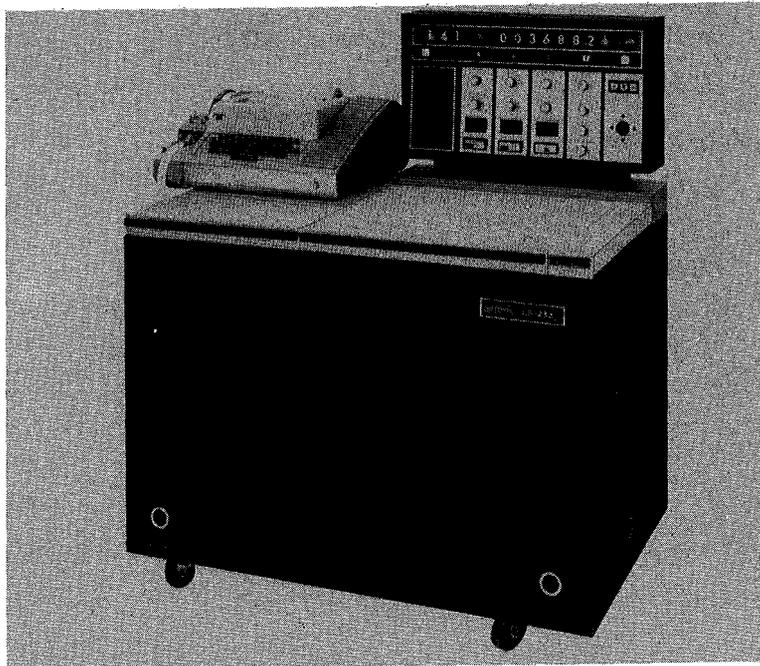
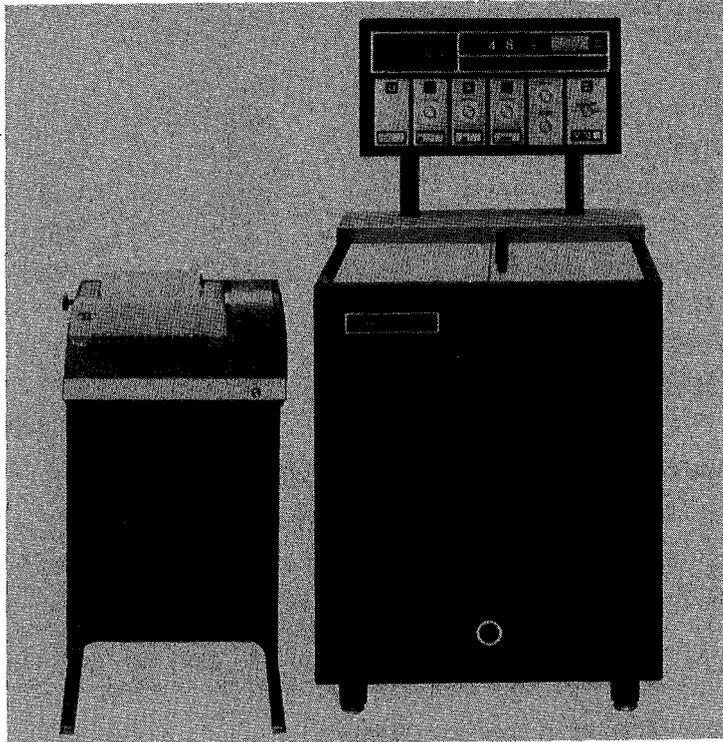
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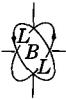


INSTRUMENTATION  
FOR ENVIRONMENTAL  
MONITORING

RAD-NUC  
Tritium, Carbon-14  
Beckman 2  
Jan. 1973

Tritium (Water) Monitor  
Beckman LS-100, LS-200, LS-300 Series





Class Laboratory, Automated Sample Counter, Printer.

Principle of Operation Liquid scintillation systems. Matched RCA photomultiplier tubes in coincidence detect light from sample. Output in cpm (digital display) and by log rate meter.

	<u>LS-100 Series</u>	<u>LS-200 Series</u>	<u>LS-300 Series</u>
Tritium Efficiency	60%	63%	63%
Carbon-14 Efficiency	90%	90%	90%
Carbon-14 Spillover into Tritium	12%	10%	10%
Current at 115 V ac	4.5 A	8 A	8 A
Width	27"	47"	47"
Depth	29"	33"	33"
Height	55"	58"	58"
Weight	600 lb	950 lb	1000 lb

Sampling Batch sampling capacity is 100, 200, or 300 samples.

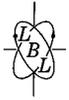
Features

1. Coincidence gate resolving time: 20 nsec.
2. Detector well: one standard 20 cm<sup>3</sup> vial or flow cell accessory.
3. Gain control feature.
4. External standardization by ratio of counts in two calibrated windows.
5. Automatic quench calibration (models LS-150, LS-250) compensates for varying quench by adjusting system gain for each sample.
6. Computer compatibility, many accessories and options available.

References Manufacturer's specifications

<u>Cost</u>	<u>Model</u>	<u>Price</u>
	LS-100 Series	\$ 6,900. - \$13,120.
	LS-200 Series	\$11,550. - \$19,200.
	LS-300 Series	\$12,550. - \$20,200.

Address Beckman Instruments Inc.  
2500 Harbor Boulevard  
Fullerton, CA 92634  
(714) 521-3700

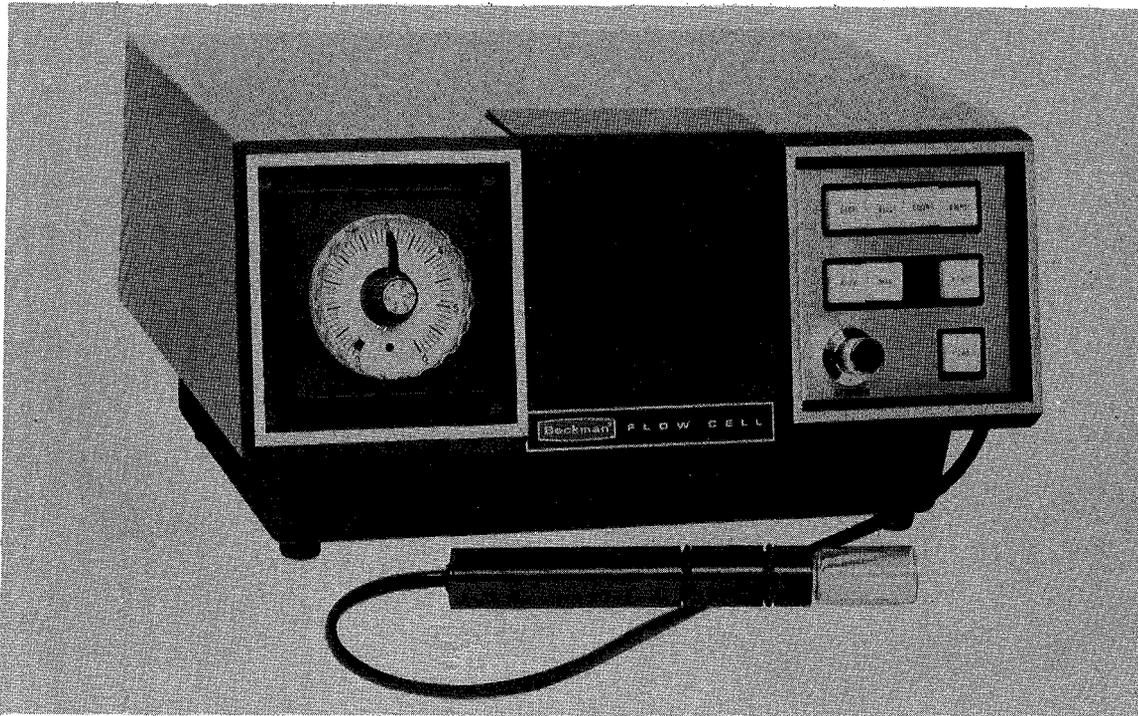


INSTRUMENTATION  
FOR ENVIRONMENTAL  
MONITORING

RAD-NUC  
Tritium  
Beckman 3  
Oct. 1972

### Tritium Flow Cell

#### Beckman Discrete Sampling Flow Cell



Class	Laboratory	
Principle of Operation	Liquid water sample is mixed with a dioxane-based liquid scintillator in a mixing chamber. The mixed solution is then pneumatically transferred to a probe-vial assembly, ready to insert into one of the Beckman liquid scintillation systems (see separate Instrument Note).	
Sensitivity and Range	Sensitivity: Minimum Detectable Tritium Concentration: 25 pCi/ml of water	
Sampling	Discrete sample from flowing stream	
Performance	Sample Flow Rate: 0.5 ml/minute Scintillator Flow Rate: 7 ml/minute	
Requirements	Power: Unknown Size: Unknown Weight: Unknown	
Features	See discussion in text on "Tritium Monitoring."	
References	1) Manufacturer's specifications 2) P. Ting and M.K. Sullivan, "Reactor Coolant Monitoring with a Discrete Sampling Flow Cell Liquid Scintillation System," Report No. 558, Beckman Instruments, Inc.	
Cost	For Use With Beckman $\beta$ -Mate II Interface If Used With LS-100 or LS-200 Series Instrument	\$1670 265
Address	Beckman Instruments Inc. 2500 Harbor Blvd. Fullerton, CA 92634 (714) 521-3700	



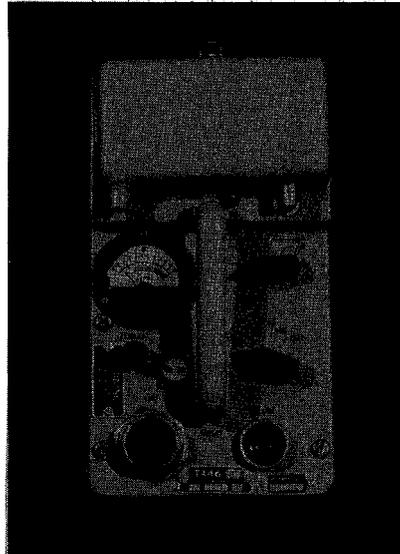


INSTRUMENTATION  
FOR ENVIRONMENTAL  
MONITORING

RAD-NUC  
Tritium in Air  
Bendix/Sandia  
Aug. 1972

Tritium in Air Monitor

Bendix/Sandia Model T446



Class	Portable
Principle of Operation	Flow-through ionization chamber, output feeds vibrating-reed electrometer. Signal is amplified. Electrostatic precipitator at input.
Sensitivity and Range	Lower limit 5 pCi/cm <sup>3</sup> of air; upper limit 10 $\mu$ Ci/cm <sup>3</sup> of air. Seven-decade ranges.
Sampling	Continuous
Performance	Accuracy: Linearity $\pm 5\%$ , maximum error $\pm 15\%$ of full scale on any range Temperature: 0° F to 135° F, with extended operation down to -40° F with NiCd batteries
Requirements	Power: 115 VAC, or 10 D cells (24-hour lifetime), or rechargeable NiCd F cells (75-hour lifetime) Size: 6.5" x 11.6" x 8.8" (16 x 29 x 22 cm) Weight: 20 to 23 lb (9 to 11 kg)
Features	1) Electronics all solid-state, on printed circuit boards 2) Electrostatic precipitator at input, and filter to eliminate ionizing aerosols 3) Adjustable alarm level 4) Either portable (batteries) or stationary (ac power) 5) Fail-safe failure indicator 6) Automatic range changing
References	1) Sandia Laboratories Report SC-M-68-245 2) Sandia Laboratories Report SC-M-67-244 3) R.P. Baker and R.D. Richards, "Reliable and Versatile Instrumentation for Detection of Tritium Hazard Designed for Military Application," available from Sandia Laboratories
Cost	\$8600.



INSTRUMENTATION  
FOR ENVIRONMENTAL  
MONITORING

RAD-NUC  
Tritium in Air  
Bendix/Sandia  
Page 2 Aug. 1972

Remarks

- 1) Extremely versatile instrument with multitude of features.
- 2) Built by Bendix on contract for the AEC. Available from Bendix on a contract basis only. Non-AEC customers must order through the AEC.

Address

Design Agency:

Sandia Laboratories  
Box 5800  
Albuquerque, NM 87115  
Attn: Reo De Pew  
(505) 264-8211

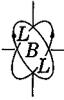
Production Contractor:

Bendix Corporation  
Kansas City Division  
Box 1159  
Kansas City, MO 64141  
Attn: Order Control Supervisor, D/213  
(816) 363-3211 - Extension 2465

AEC:

U.S. Atomic Energy Commission  
Albuquerque Operations Office  
P.O. Box 5400  
Albuquerque, NM 87115  
Attn: Director, Space and Special Programs Division  
(505) 264-8211

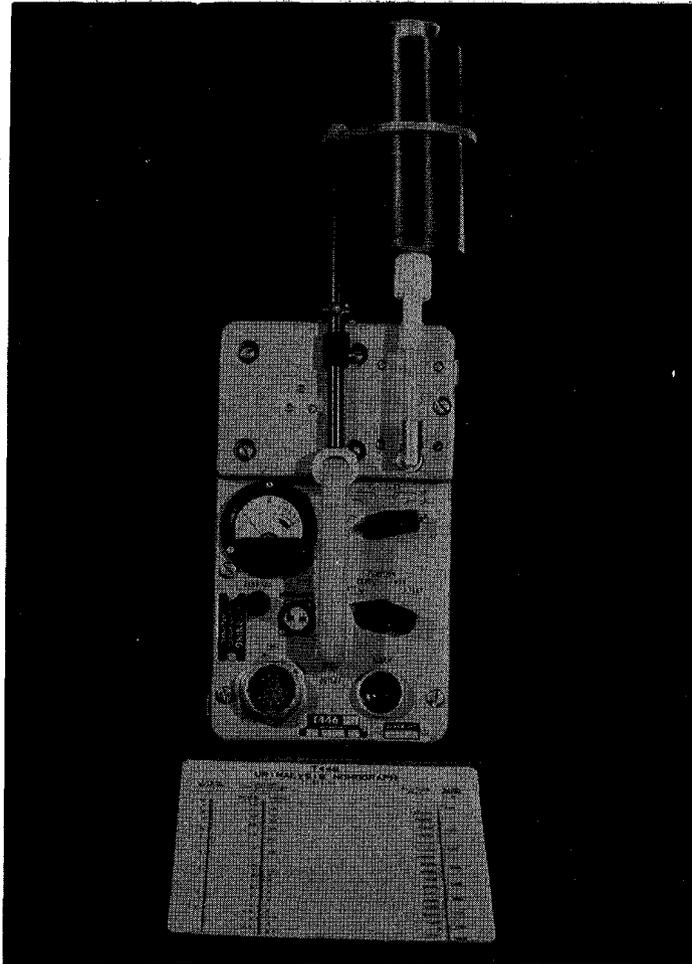
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INSTRUMENTATION  
FOR ENVIRONMENTAL  
MONITORING

RAD-NUC  
Tritium in Urine  
Bendix/Sandia 2  
Jan. 1973

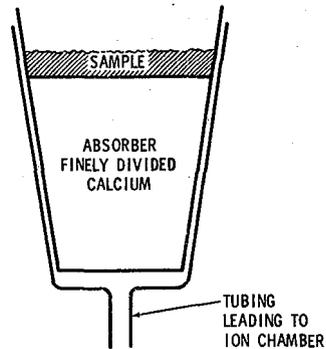
Tritium Urinalysis Monitor  
Bendix/Sandia Model T449

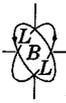


T449 Urinalyzer Instrument with T446 Tritium Monitor

URINALYSIS GAS GENERATION

CARTRIDGE CONFIGURATION

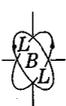




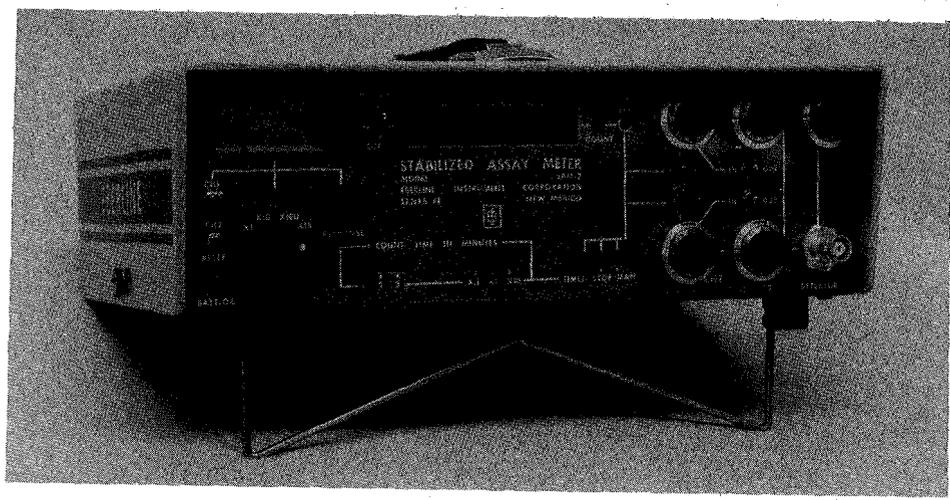
INSTRUMENTATION  
FOR ENVIRONMENTAL  
MONITORING

RAD-NUC  
Tritium in Urine  
Bendix/Sandia 2  
Page 2 Jan. 1973

Class	Portable
Principle of Operation	Metallic calcium (disposable cartridge) reduces water in urine. Evolved hydrogen gas fills an ionization chamber. For use with the T446 Tritium Monitor (see separate section).
Sensitivity and Range	Sensitivity: Down to 20,000 pCi/cm <sup>3</sup> of urine
Sampling	20 cm <sup>3</sup> batch sampling; requires five minutes per sample.
Performance	Accuracy: Maximum error ±25 of full scale, or ±20,000 pCi/cm <sup>3</sup> , whichever is greater Temperature: 35-135°F
Requirements	Size: 12.5" x 10" x 14" (32 x 25 x 35 cm) Weight: 18 lb (8 kg) including 12 cartridges Cartridge: 7" length, 1.5" diameter (17 x 37 cm)
Features	1. For use with T446 Tritium Monitor. 2. Useful in remote or emergency situations.
References	1. Sandia Laboratories Report SC-M-68-245. 2. Sandia Laboratories Report SC-M-67-244. 3. R.P. Baker and R.D. Richards, "Reliable and Versatile Instrumentation for Detection of Tritium Hazard Designed for Military Application"; available from Sandia Laboratories.
Remarks	Built by Bendix on contract for the AEC. Available from Bendix on a contract basis only. Non-AEC customers must order through the AEC.
Cost	By contract only
Address	Design Agency: Sandia Laboratories Box 5800 Albuquerque, NM 87115 Attn: Reo De Pew (505) 264-8211  Production Contractor: Bendix Corporation Kansas City Division Box 1159 Kansas City, MO 64141 Attn: Order Control Supervisor, D/213 (816) 363-3211 - Extension 2465  AEC: U.S. Atomic Energy Commission Albuquerque Operations Office P.O. Box 5400 Albuquerque, NM 87115 Attn: Director, Space and Special Programs Division (505) 264-8211



Uranium Enrichment Monitor  
Eberline Model SAM-2



Class	Portable
Principle of Operation	Scintillation crystal NaI(Tl) doped with $^{241}\text{Am}$ , photomultiplier tube, two single channel analyzers, amplifier, high voltage gain control circuit, scaler, timer, ratemeter and digital readout.
Sensitivity and Range	Energy Range: Optimized for the detection of the 185 keV gamma emitted by $^{235}\text{U}$ , may be used for other energies. Rate: 500, 5000, 50,000, and 500,000 cpm, linear Threshold: 0 to 1.0 volts, calibrated Window: 0 to 1.0 volts, calibrated
Performance	Pulse Height Resolution: (with RD-19) $\sim$ 15% FWHM at 185 keV Temperature: $0^{\circ}$ to $60^{\circ}\text{C}$ ( $32^{\circ}$ to $140^{\circ}\text{F}$ ) Timer: From 0.1 to 50 minutes Response Time: 1 to 13 sec continuously variable, reset switch Linearity: $\pm 5\%$ of full scale Paired Pulse Resolution: $\sim$ 4 $\mu\text{sec}$
Requirements	Power: 115 or 230 Vac switch selectable, 0.25 Amps Battery: 7.5 V to 14 V at 1.0 Amps Size: Detector -- 6.6 cm dia, 24.5 cm L (2.6" D, 9.6" L) Electronics -- 11.5 cm H, 24.5 cm D, 26.8 cm W (4.5" H, 9.6" D, 10.5" W) Weight: Detector -- 1.1 kg (2.5 lb) Electronics -- 3.2 kg (6.8 lb)
Features	Mode switch: Timed-stop-manual Test circuit, voltage stabilization circuit, read either channel, add both channels, subtract one channel, scaler and ratemeter displays, preset counting times. Accessories: Battery pack, lead shield, carrying case. Model RD-19 Stabilized Scintillation Detector: 2-inch diameter x 1/2 inch thick NaI(Tl) crystal doped with $^{241}\text{Am}$ , with 2-inch, 10 dynode, S11 response photomultiplier tube. Dynode resistor network and magnetic shield built into detector. Weighs 2-1/2 lb.



Features (Cont'd)

Model RD-20 Stabilized Scintillation Detector: Same as Model RD-19, but with thin window crystal and lower gamma equivalent energy of the  $^{241}\text{Am}$  for low energy gamma measurements. Weighs 2-1/4 lb.

Model RD-21 Stabilized Scintillation Detector: 5-inch diameter photomultiplier tube and crystal version of the RD-20, also for low energy gamma measurements.

Neutron Probe: Model RD-18 Directional Fast Neutron Detector (not stabilized).

Model BP-4 Battery Pack: 12 volt, 7.5 amp-hours, rechargeable battery in carrying case. Includes charger and connecting cables. Weighs 8 lb.

Lead Shield for RD-19: 3/16" thick aluminum encased lead shield with collet type retention to the detector. Has easily interchangeable end plates for varying collimation. Weighs 5-1/2 lb.

References

Manufacturer's Specifications

Cost

SAM-2 \$2,650.

Address

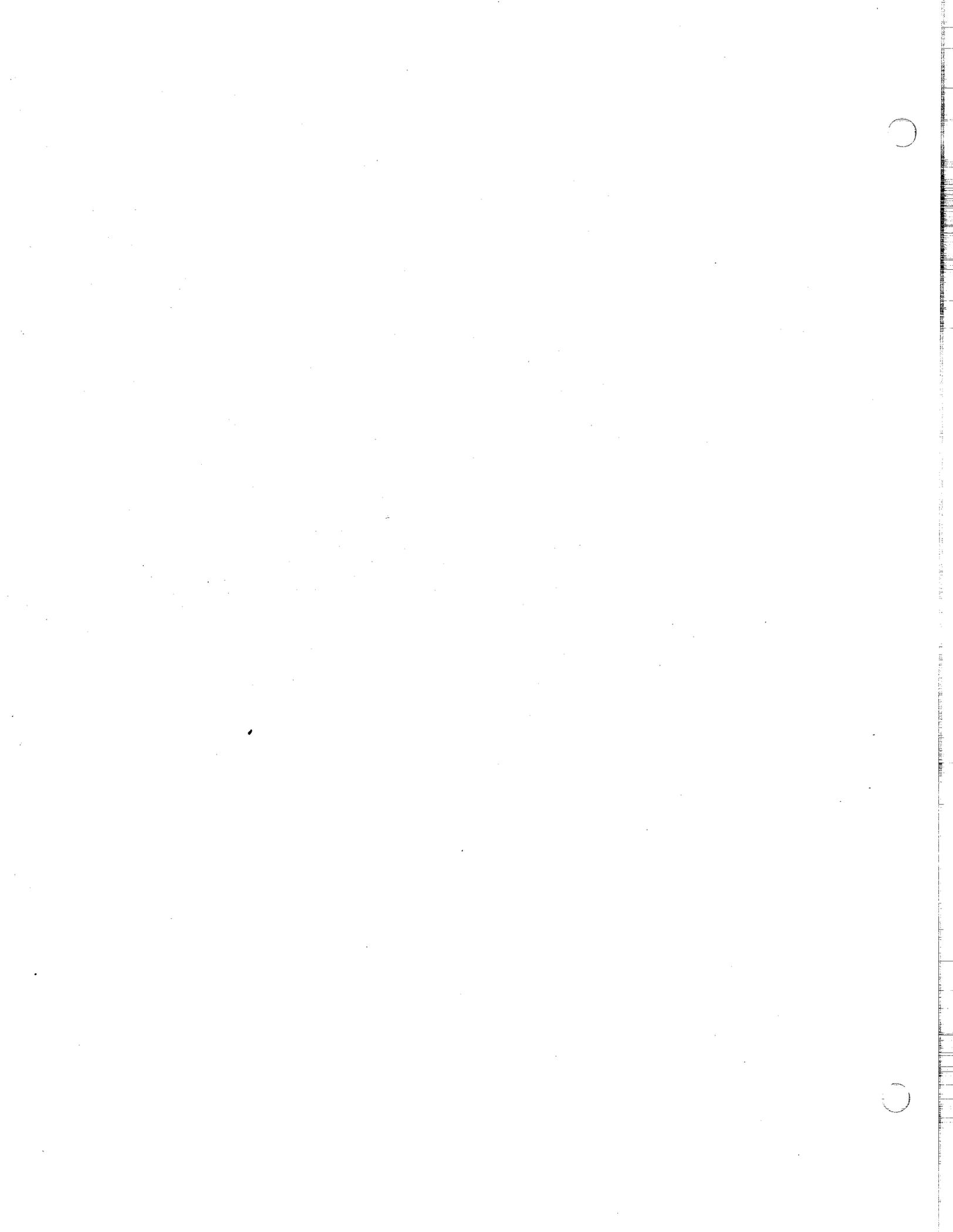
Eberline Instrument Corp.  
P.O. Box 2108  
Airport Road  
Santa Fe, NM 87501  
(505) 982-1881



0 3 0 0 3 0 0 2 4 8 5

<sup>85</sup>Kr Liquid Scintillation System  
EPA Eastern Environmental Radiation Lab

Class	Laboratory
Stage of Development	Not commercially available
Principle of Operation	Krypton gas is concentrated cryogenically, introduced at 500-600 torr pressure and dissolved into a 25-ml vial filled with toluene-based liquid scintillator, and counted by a photomultiplier system.
Sensitivity	Minimum detectable concentration = 1 pCi/m <sup>3</sup> of air
Sampling	Air can be introduced directly into solution at sufficiently high concentrations.
Performance	Accuracy: 1. Individual measurements of background <sup>85</sup> Kr (in the 10 pCi/m <sup>3</sup> range) are determined to about ±4%. 2. One cpm above background = 0.1 pCi/ml of air
References	1. R.E. Shuping, C.R. Phillips, A.A. Moghissi, "Low Level Counting of Environmental <sup>85</sup> Kr by Liquid Scintillation," Analytical Chem. <u>41</u> , 2082 (1969); this paper describes the method. 2. R.E. Shuping, C.R. Phillips, A.A. Moghissi, "Krypton-85 Levels in the Environment Determined from Dated Krypton Gas Samples," Rad. Health Data and Reports <u>11</u> , 671 (1970).
Address	R.E. Shuping EPA, Eastern Environmental Radiation Laboratory P.O. Box 61 Montgomery, AL 36101 (205) 272-3402





INSTRUMENTATION  
FOR ENVIRONMENTAL  
MONITORING

RAD-NUC  
Krypton-85  
EPA, EERL 2  
Aug. 1972

$^{85}\text{Kr}$  Liquid Scintillation System

EPA Eastern Environmental Radiation Lab

Class	Laboratory
Stage of Development	Not commercially available
Principle of Operation	Krypton is separated from gas samples with charcoal and molecular sieve cold traps, calcium sulfate, ascarite (NaOH preparation) and a titanium furnace (900°C). Sample is then dissolved in liquid scintillator and counted by a photomultiplier system.
Sensitivity	Minimum detectable concentration $\sim 1 \text{ pCi/m}^3$ air.
Performance	Accuracy: Individual measurements of background $^{85}\text{Kr}$ (in the $15 \text{ pCi/m}^3$ range) are determined to about $\pm 10\%$ .
Comments	$^{83}\text{mKr}$ is used as a tracer in the krypton separation process. Krypton recoveries are in the 80-90% range. About $1 \text{ m}^3$ of air can be processed in four hours.
References	S.L. Cummings, R.L. Shearin, C.R. Porter, "A Rapid Method for Determining $^{85}\text{Kr}$ in Environmental Air Samples," page 163 in <u>Rapid Methods for Measuring Radioactivity in the Environment</u> , International Atomic Energy Agency, Vienna (1971).
Address	S.L. Cummings EPA, Eastern Environmental Radiation Laboratory P.O. Box 61 Montgomery, AL 36101 (205) 272-3402





INSTRUMENTATION  
FOR ENVIRONMENTAL  
MONITORING

RAD-NUC  
Krypton-85  
EPA, NERHL  
Aug. 1972

Krypton-85 Ion Chamber Systems

EPA Northeastern Radiological Health Lab

Class	Field Instruments										
Stage of Development	Not commercially available										
Principle of Operation	Flow-through ionization chambers for $^{85}\text{Kr}$ measurements										
Sensitivity	<table><thead><tr><th><u>Chamber Volume</u></th><th><u>Minimum Detectable Concentration <math>^{85}\text{Kr}</math> (pCi/m<sup>3</sup>)</u></th></tr></thead><tbody><tr><td>0.5 liters</td><td>130,000</td></tr><tr><td>1.0 liters</td><td>190,000</td></tr><tr><td>2.8 liters</td><td>39,000</td></tr><tr><td>4.3 liters</td><td>39,000</td></tr></tbody></table>	<u>Chamber Volume</u>	<u>Minimum Detectable Concentration <math>^{85}\text{Kr}</math> (pCi/m<sup>3</sup>)</u>	0.5 liters	130,000	1.0 liters	190,000	2.8 liters	39,000	4.3 liters	39,000
<u>Chamber Volume</u>	<u>Minimum Detectable Concentration <math>^{85}\text{Kr}</math> (pCi/m<sup>3</sup>)</u>										
0.5 liters	130,000										
1.0 liters	190,000										
2.8 liters	39,000										
4.3 liters	39,000										
Sampling	Continuous										
Performance	Accuracy: $\pm 7\%$ to $\pm 17\%$ ( $2\sigma$ ) error on calibration										
Comments	<ol style="list-style-type: none"><li>1. Dry, radon-free air required.</li><li>2. Radon holdup trap needed.</li><li>3. Needs frequent maintenance.</li><li>4. <math>^{85}\text{Kr}</math> concentration is averaged over about 20-30 minute interval due to radon trap.</li></ol>										
References	D.G. Smith, J.A. Cochran, B. Shleien, "Calibration and Initial Field Testing of $^{85}\text{Kr}$ Detectors for Environmental Monitoring," Document BRH/NERHL 70-4, Northeastern Radiological Health Lab (1970).										
Address	EPA, Northeastern Radiological Health Laboratory 109 Holton Street Winchester, MA 01890 (617) 729-5700										





INSTRUMENTATION  
FOR ENVIRONMENTAL  
MONITORING

RAD-NUC  
Krypton-85  
EPA, NERHL 2  
Aug. 1972

### Krypton-85 G-M Detectors

EPA Northeastern Radiological Health Lab

Class                      Field Instruments

Stage of Development      Not commercially available

Principle of Operation      Five Geiger-Müller tubes for  $^{85}\text{Kr}$  measurements

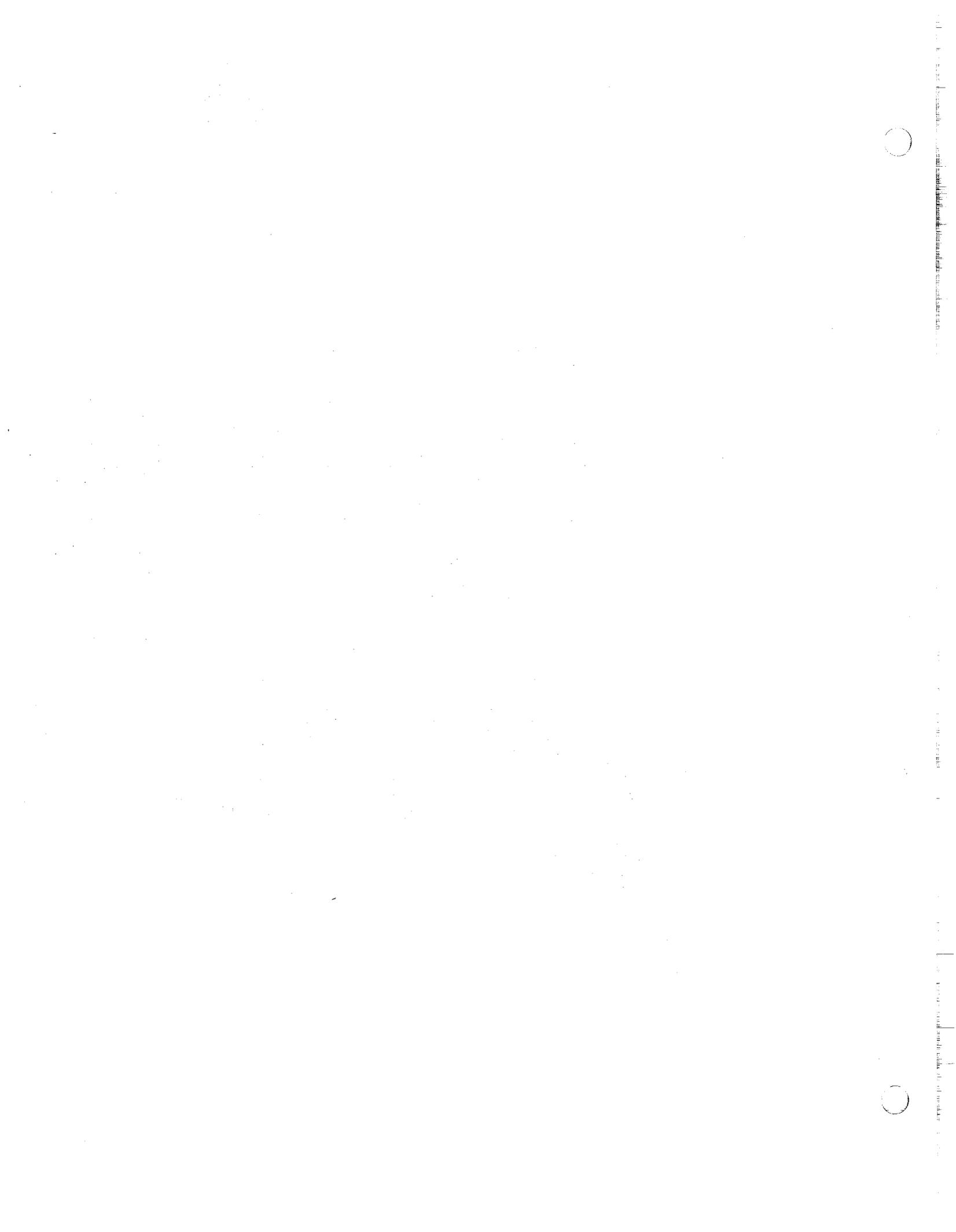
<u>Instrument (G-M Detector)</u>	<u>Geometry of G-M Detector</u>	<u>Dimensions</u>	<u>Window Thickness mg/cm<sup>2</sup></u>	<u>Minimum Detectable Concentration (pCi/m<sup>3</sup>) (10 min sample/- 10 min background)</u>
Eon 8008H	2 window pancake	2" diameter	3.5	12,000
Eon 8001T	1 window pancake	2" diameter	1.4	25,000
Amperex 18546	1 window pancake	2" diameter	3.5	27,000
Eon 5108E	cylindrical	5" length, 1/2" diameter	30.	28,000
LND 719	cylindrical	5" length, 1/2" diameter	30.	24,000

Accuracy                      All probes calibrated  $\pm 13\%$  against standard ion chamber.

Comments                      1. Cylindrical probes are the most rugged.  
2. Two-window pancake was too fragile for field use.  
3. All probes are sensitive enough for use near a fuel-reprocessing plant.  
4. Time-averaging is very short (a few minutes).

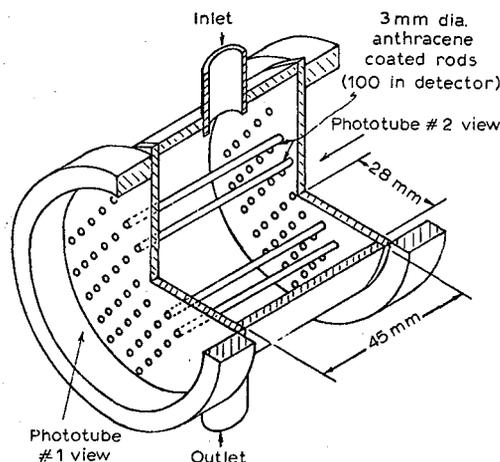
References                      D.G. Smith, J.A. Cochran, B. Shleien, "Calibration and Initial Field Testing of  $^{85}\text{Kr}$  Detectors for Environmental Monitoring," Document BRH/NERHL 70-4, Northeastern Radiological Health Lab (1970).

Address                        EPA, Northeastern Radiological Health Laboratory  
109 Holton St.  
Winchester, MA 01890  
(617) 729-5700





Tritium Monitor for Air, Water, Urine  
EPA Plastic Scintillation System



Schematic Diagram of the Detector

Class	Laboratory
Stage of Development	Not commercially available
Principle of Operation	Plastic scintillation system, two bialkali photomultiplier tubes in coincidence, single channel analyzer, ratemeter. Scintillator is about 100 3mm plexiglas rods coated with anthracene or plastic phosphor.
Sensitivity and Range	Sensitivity: Air: 20 cpm per pCi/cm <sup>3</sup> of air Water: 10 cpm per 1000 pCi/cm <sup>3</sup> of water Urine: Same as water Background: About 30 cpm from normal external gamma background
Sampling	Continuous
Requirements	Power: 115 VAC with HV power supply Size: Detector about 3" diameter (7.5 cm), 6" length (15 cm)
Features	1. Electronic system records only those events with coincidence in 30-50 nsec resolving time. 2. Audible and visible alarm feature, variable level.
References	A.A. Moghissi, H.L. Kelley, C.R. Phillips, J.E. Regnier, "A Tritium Monitor Based on Scintillation," Nuclear Instr. and Methods <u>68</u> , 159 (1969).
Cost	Not commercially available
Address	EPA National Environmental Research Center P.O. Box 15027 Las Vegas, NV 89114 Attn: A.A. Moghissi (702) 736-2969



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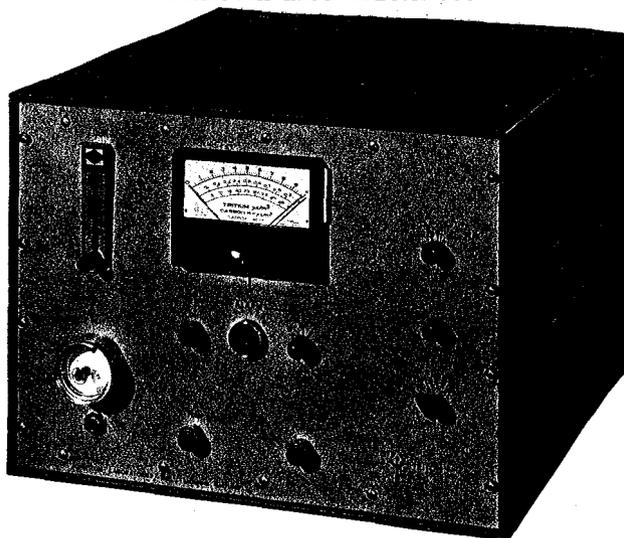


INSTRUMENTATION  
FOR ENVIRONMENTAL  
MONITORING

RAD-NUC  
Tritium, C-14, A-41, Kr-85  
Johnston Labs  
August 1973

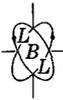
Radionuclide Gas Monitor

Johnston Labs Triton 955B



Class	Laboratory or stationary ionization chamber
Principle of Operation	Positive displacement pump draws air through submicron filter and electrostatic precipitator, then into two flow-through ionization chambers. Signal amplified in electrometer circuit. Two identical sealed chambers measure external gamma flux, which is subtracted electronically.
Sensitivity and Range	Tritium Ranges: 0-10, 100, 1000, 10,000 $\mu$ Ci/M <sup>3</sup> Carbon-14 Ranges: 0-2, 20, 200, 2000 $\mu$ Ci/M <sup>3</sup> Gamma Ranges: 0.05, 0.5, 5, 50 mr/hour
Performance	Accuracy: $\pm 10\%$ of full scale, reproducibility $\pm 2\%$ Zero Drift: After 15 minute warm-up less than 2% of FS in 24 hours Time Constants: 15 or 45 seconds Gamma Compensation: Up to 5 mr/hour for uniform external field Air Flow: 2 to 10 liters/minute, adjustable 0.5 micron filter
Requirements	Power: 115/230 Vac, 100 W, 50/60 Hz Size: 19.6" x 14.8" x 21.6" (50 x 37 x 55 cm) Weight: 67 lb (30 kg)
Features	Alarm for fail-safe operation, variable set point, fail-safe Cleanable chambers, recorder output, remote operation mode Solid-state circuitry
References	1) Manufacturer's specifications 2) J.R. Waters, Johnston Labs Internal Report #JLI-506, "Pitfalls and Errors in the Measurement of Tritium in Air"
Cost	Model 955B                      \$2990.
Remarks	Very high sensitivity
Address	Johnston Laboratories Inc. 3 Industry Lane Cockeysville, MD 21030 (301) 666-9500



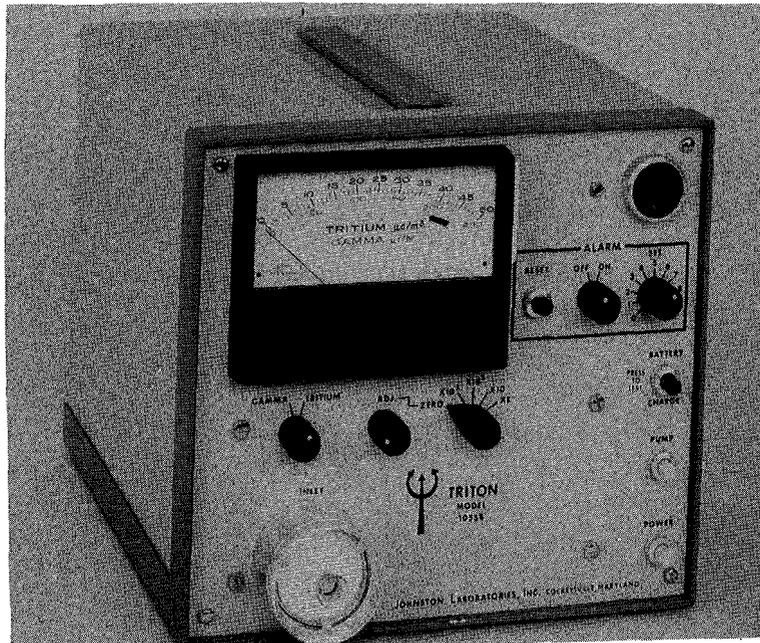


INSTRUMENTATION  
FOR ENVIRONMENTAL  
MONITORING

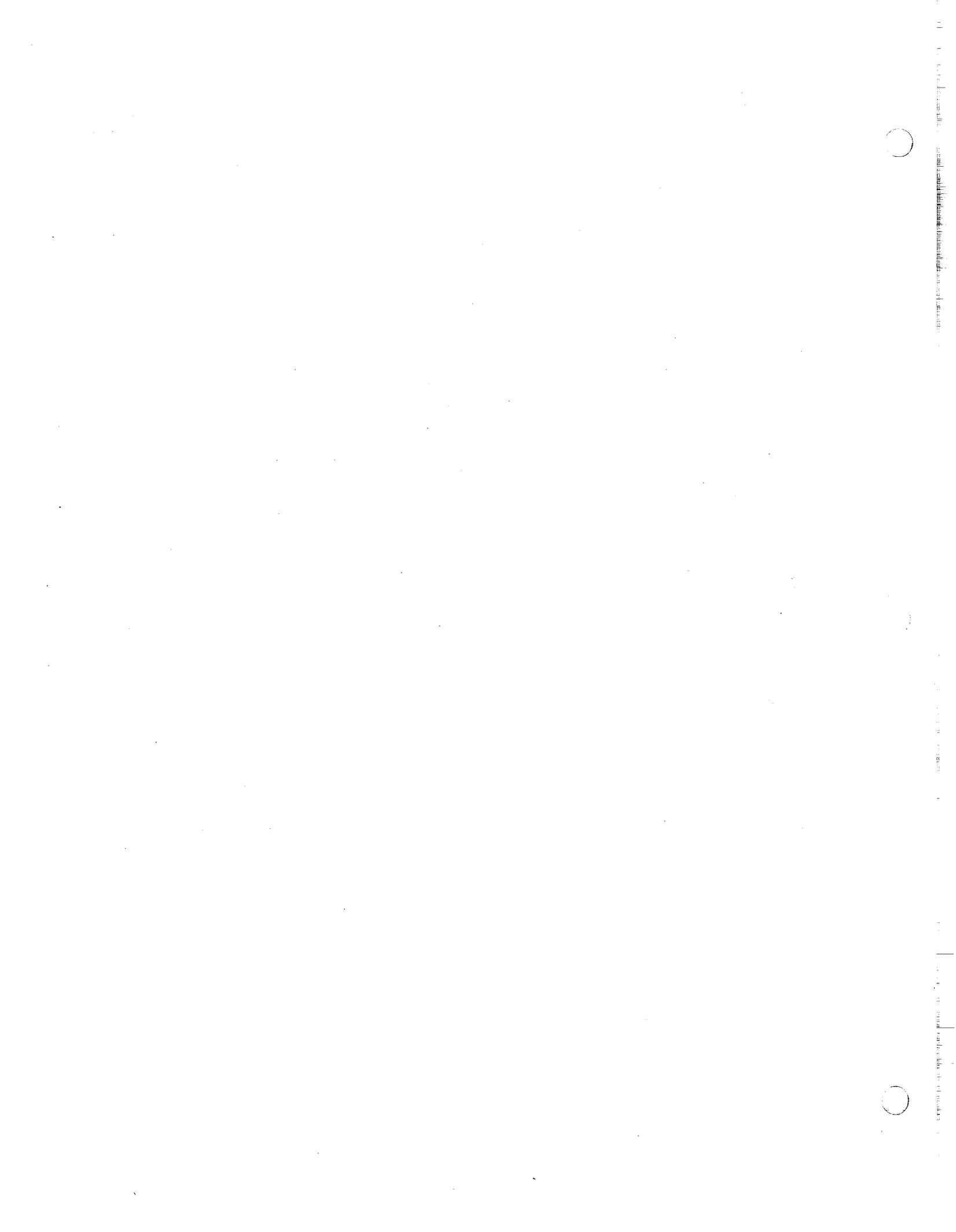
RAD-NUC  
Tritium  
Johnston Labs 2  
July 1973

### Tritium Monitor

Johnston Labs Triton 1055B



Class	Portable
Principle of Operation	Positive displacement pump draws air through submicron filter and electrostatic precipitator, then into flow-through ionization chamber. Signal amplified in electrometer circuit, sealed chambers measure external gamma flux, which is subtracted electronically.
Sensitivity and Range	Tritium Ranges: 50, 500, 5000, 50,000 $\mu$ Ci/M <sup>3</sup> Gamma Ranges: 0.25, 2.5, 25, 250 mr/hr
Performance	Accuracy: $\pm 10\%$ of full scale, reproducibility $\pm 3\%$ Zero Drift: After 15 minute warm-up less than 2% of FS in 24 hours Time Constants: 20 sec Gamma Compensation: Up to 5 mr/hour for uniform external field Air Flow: 2 liters/minute, 0.5 micron filter
Requirements	Power: 115/230 Vac, 50/60 Hz, 25 Watts, or NiCd battery, 6 hours Size: Weight: 14 kg (30 lb)
Features	Alarm for fail-safe operation, variable set point, fail-safe Cleanable chambers, recorder output, remote operation mode Solid-state circuitry, tritium calibrator accessory
References	Manufacturer's specifications
Cost	Model 1055B \$2,295.
Address	Johnston Laboratories, Inc. 3 Industry Lane Cockeysville, MD 21030 (301) 666-9500



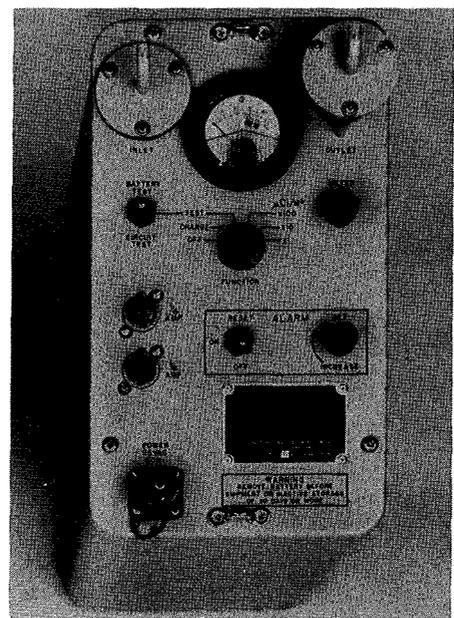
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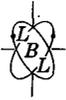
INSTRUMENTATION  
FOR ENVIRONMENTAL  
MONITORING

RAD-NUC  
Tritium  
Johnston Labs 3  
Aug. 1972

Tritium Gas Monitor  
Johnston Labs Triton 1125



Class	Portable
Principle of Operation	A positive displacement pump draws air through a submicron filter and electrostatic precipitator removing interfering particles, ions, humidity and smoke. Air then goes into the flow-through ion chamber. Ionization within the chamber is detected and amplified by an electrometer circuit which is calibrated to read $\mu$ curies/ $M^3$ of tritium. Identical sealed ion chamber is used to subtract the external gamma flux, with subtraction performed electronically.
Sensitivity and Range	Tritium Ranges: 100, 1000, and 10,000 $\mu$ Ci/ $M^3$
Performance	Accuracy: $\pm 15\%$ full scale Zero Drift: After 5 minute warm-up less than 5% FS in 30 minutes; less than 3% FS in 4 hours Time Constant: Approximately 20 seconds Gamma Compensation: up to 5 mR/hr for uniform radiation field Decontamination: Reads less than 5 $\mu$ Ci/ $M^3$ in 5 minutes after 15 minute exposure to 1000 $\mu$ Ci/ $M^3$ of tritium gas Airflow: Approximately 2 liters/min. Operating Temperature: +40° F to +150° F
Requirements	Power: 109-121 V ac, 50-65 Hz, 5 W, or battery Size: 7.2 x 12.7 x 14.2 inches (18 x 32 x 36 cm) Weight: 17 lbs (8 kg)
Features	Automatic changeover to battery operation on power failure. Military type construction. Suitable for other beta emitting fission product gases: $C^{14}$ , $A^{41}$ , $R_n^{222}$ , $S^{35}$ , $K_r^{85}$ , $X_e^{133}$ .
References	1) Manufacturer's specifications 2) J.R. Waters, Johnston Labs Internal Report #JLI-506, "Pitfalls and Errors in the Measurement of Tritium in Air"
Cost	\$1995



INSTRUMENTATION  
FOR ENVIRONMENTAL  
MONITORING

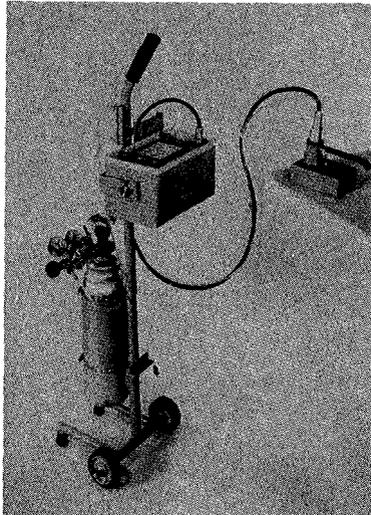
RAD-NUC  
Tritium  
Johnston Labs 3  
Page 2 Aug. 1972

Remarks                    Other models available. Instrument use high above sea level will require a correction factor due to reduced air density in chamber. "Designed for military and other rugged requirements and does not have adequate sensitivity for health monitoring" (Private communication, L. Gevins). Carrying case and straps included.

Address                    Johnston Laboratories Inc.  
3 Industry Lane  
Cockeysville, MD 21030  
(301) 666-9500



Radionuclide Monitor  
Johnston Labs Model LB 124



Class	Mobile
Principle of Operation	Proportional gas flow-counter probe, counter gas is 75% methane and 25% air, two probes available.
Sensitivity and Range	Sensitivity: Model LB 6280 -- $\approx 50$ pCi/cm <sup>2</sup> for H <sup>3</sup> Model MZ 94/190 -- $<1.0$ pCi/cm <sup>2</sup> for $\alpha$ 's -- $<2.0$ pCi/cm <sup>2</sup> for $\beta$ 's Range: 10 <sup>2</sup> , 10 <sup>3</sup> , 10 <sup>4</sup> , 10 <sup>5</sup> , 10 <sup>6</sup> cpm, semi-log scale Efficiency: Model LB 6280 -- $\approx 40\%$ Model MZ 94/190 -- $\approx 25\%$
Sampling	Window: Model LB 6280 -- 0.16" x 1.42", 0 mg/cm <sup>2</sup> Model MZ 94/190 -- 3.7" x 7.48", 0.4 mg/cm <sup>2</sup>
Performance	Temperature Range: Time Constant: 1, 4, 20 sec, selectable
Requirements	Power: 115 Vac, 50/60 Hz, 14 Watts Batteries -- 2 NiCd, 15 hrs Probe Size: Model LB 6280 -- 9 cm W, 15 cm D, 16 cm H (3.5" x 5.9" x 5.1") Model MZ 94/190 -- 12.5 cm W, 24 cm L, 2.7 cm H (4.9" x 9.5" x 1.1") Probe Weight: Model LB 6280 -- 2 kg (4.4 lb) Model MZ 94/190 -- 2.2 kg (4.7 lb)
Features	Recorder output, audible count rate indicator
References	Manufacturer's specifications
Cost	Model LB 124                      \$2,195.*
Address	Johnston Laboratories, Inc. 3 Industry Lane Cockeysville, MD 21030 (301) 666-9500

\*As of the present price list; changes in price may be made subject to regulations.



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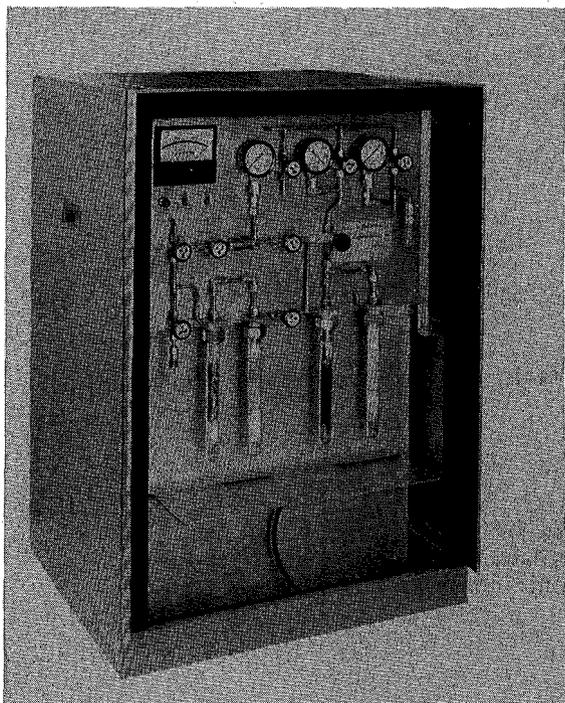


INSTRUMENTATION  
FOR ENVIRONMENTAL  
MONITORING

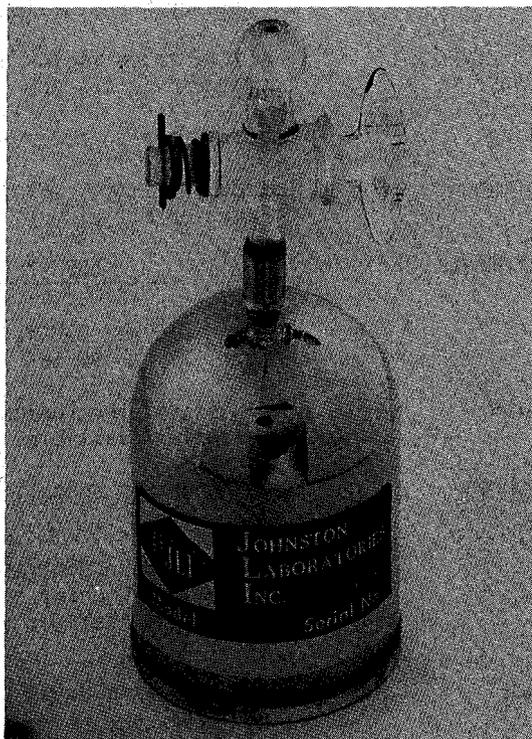
RAD-NUC  
Radon-222 & Daughters  
Johnston Labs 5  
August 1973

### Radon Analysis System

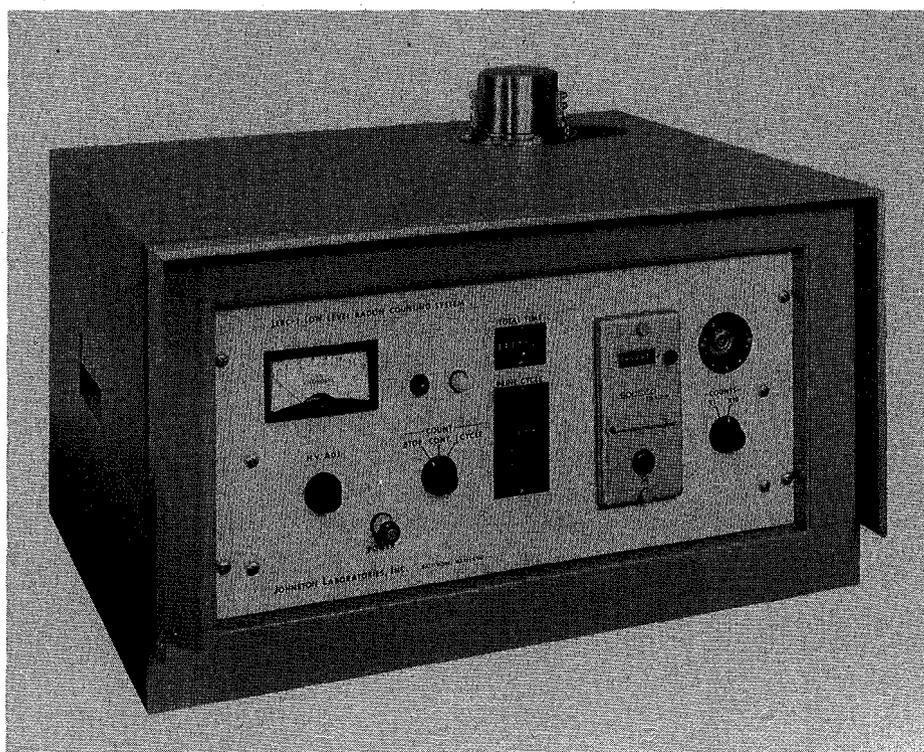
Johnston Models RCTS-2, LAC-2, LLRC-2



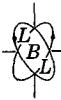
Model RCTS-2



Model LAC-2

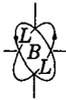


Model LLRC-2



Class	Laboratory																
Principle of Operation	Radon is concentrated and purified in Model RCTS-2, and transferred to Model LAC-2, a Lucas chamber. Scintillations produced in the chamber are counted by a photomultiplier in the LLRC-2 analyzer.																
Sensitivity and Range	1 pCi of radon yields 5 cpm Background is 0.1 cpm																
Sampling	Batch, traps to remove water and CO <sub>2</sub>																
Performance	Reproducibility: ±5% Collection Efficiency: 95%																
Requirements	<table><thead><tr><th></th><th><u>LLRC-2</u></th><th><u>RCTS-2</u></th><th><u>LAC-2</u></th></tr></thead><tbody><tr><td>Power:</td><td></td><td></td><td></td></tr><tr><td>Size:</td><td>16" H, 23" W, 20" D</td><td>40" H, 28" W, 25.5" D</td><td>5" H, 2" dia</td></tr><tr><td>Weight:</td><td>95 lb</td><td>229 lb</td><td></td></tr></tbody></table>		<u>LLRC-2</u>	<u>RCTS-2</u>	<u>LAC-2</u>	Power:				Size:	16" H, 23" W, 20" D	40" H, 28" W, 25.5" D	5" H, 2" dia	Weight:	95 lb	229 lb	
	<u>LLRC-2</u>	<u>RCTS-2</u>	<u>LAC-2</u>														
Power:																	
Size:	16" H, 23" W, 20" D	40" H, 28" W, 25.5" D	5" H, 2" dia														
Weight:	95 lb	229 lb															
Features	Solid state electronics, automatic tape printout/timing/recycling, accessories available.																
References	Manufacturer's specifications																
Cost	<table><tbody><tr><td>RCTS-2 Radon Concentrator (less vacuum pump)</td><td>\$5,995.</td></tr><tr><td>LAC-2 Radon Counter (quartz window)</td><td>295.</td></tr><tr><td>LLRC-2 Radon Analyzer (electronic console)</td><td>4,195.</td></tr></tbody></table>	RCTS-2 Radon Concentrator (less vacuum pump)	\$5,995.	LAC-2 Radon Counter (quartz window)	295.	LLRC-2 Radon Analyzer (electronic console)	4,195.										
RCTS-2 Radon Concentrator (less vacuum pump)	\$5,995.																
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LLRC-2 Radon Analyzer (electronic console)	4,195.																
Address	Johnston Laboratories, Inc. 3 Industry Lane Cockeysville, MD 21030 (301) 666-9500																

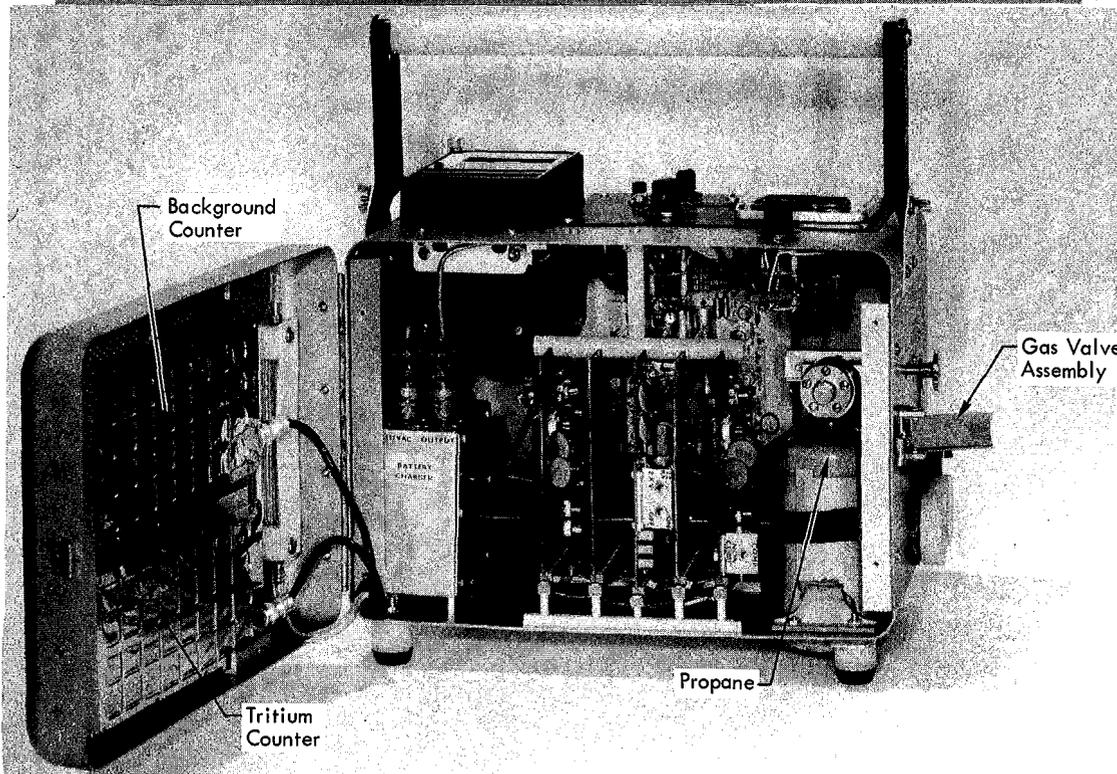
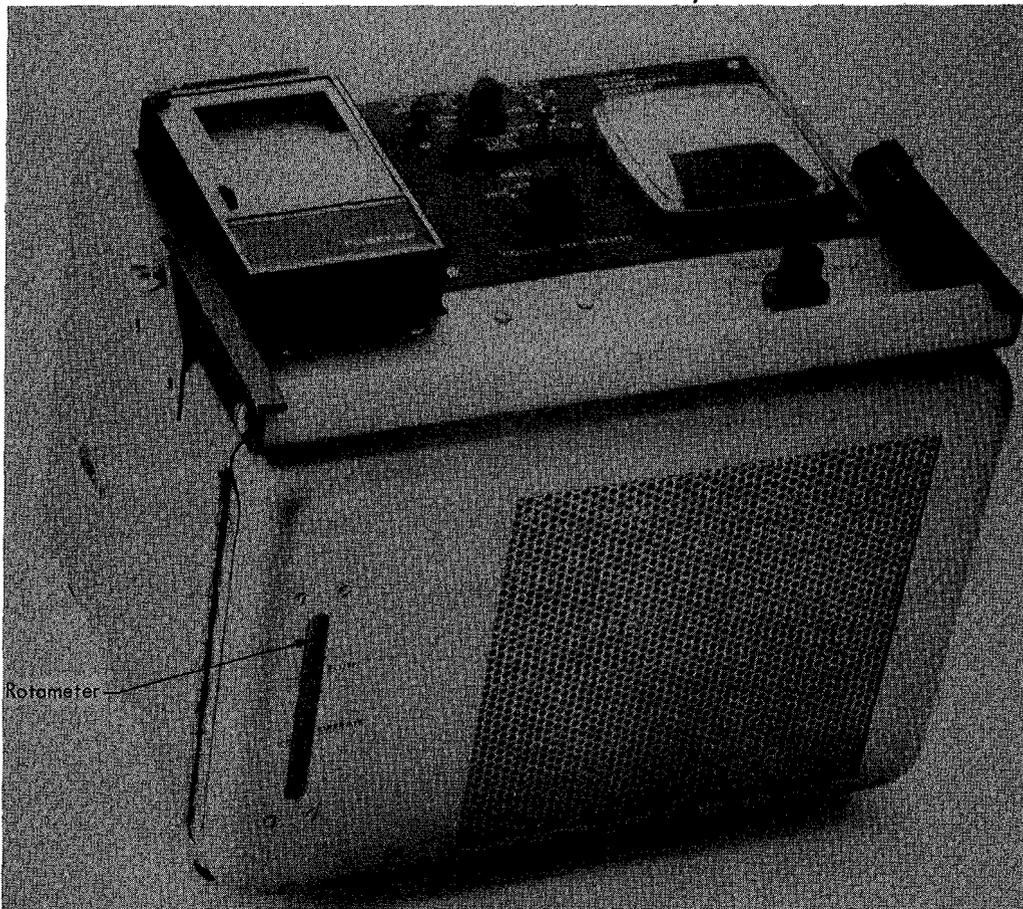
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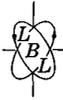


INSTRUMENTATION  
FOR ENVIRONMENTAL  
MONITORING

RAD-NUC  
Tritium in Air  
Lawrence Livermore Laboratory  
Sept. 1972

Tritium in Air Monitor  
Lawrence Livermore Laboratory





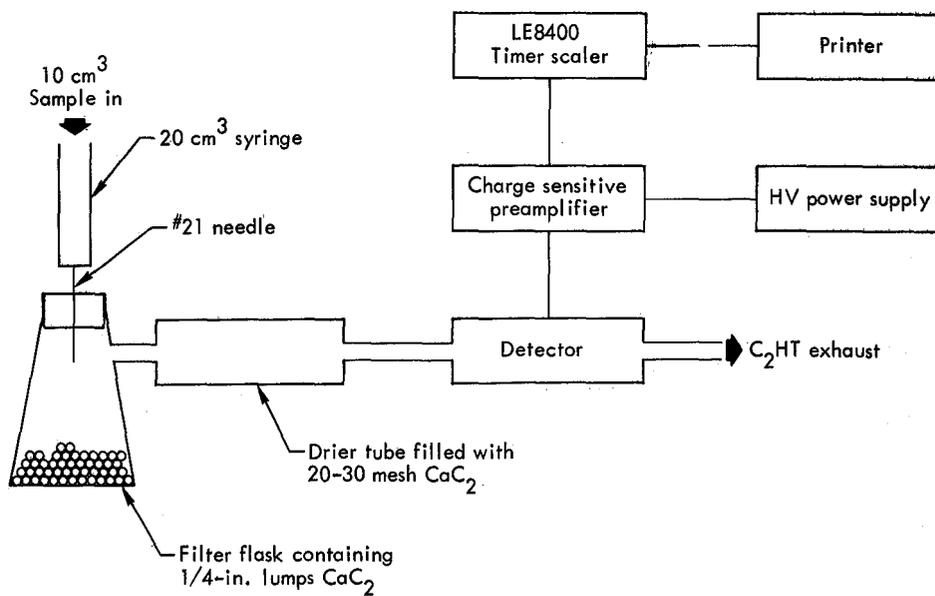
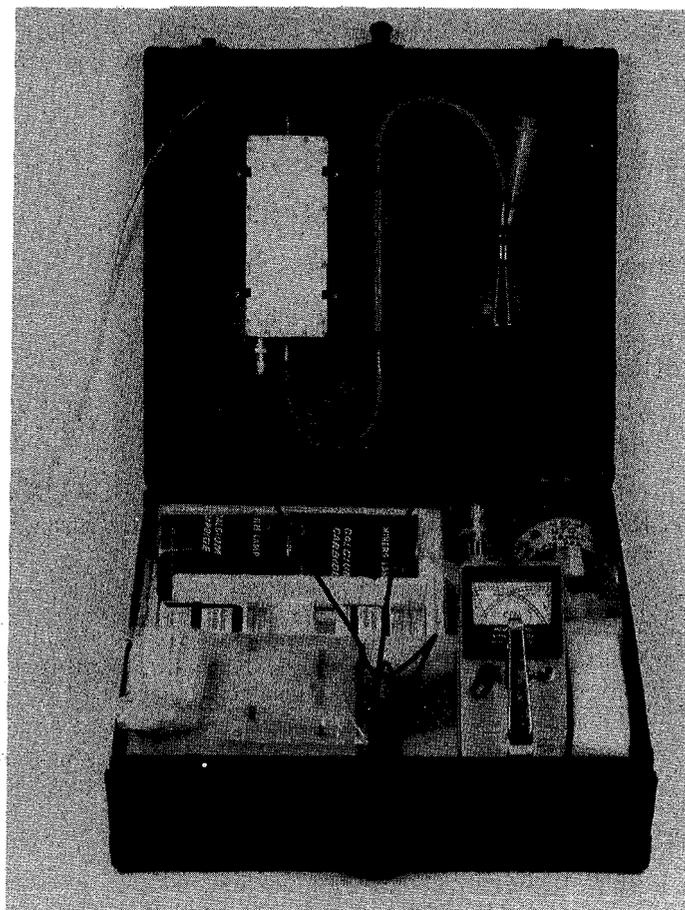
Class	Portable
Stage of Development	Not commercially available
Principle of Operation	Large-area, thin-window gas-flow proportional counter. Two-counter system. First counter has Formvar window thin enough for tritium beta penetration, while second counter has aluminized Mylar window opaque to tritium betas. Window is porous to counting gas (propane); flow is from positive inside pressure. Instrument is more responsive to HTO than HT.
Sensitivity and Range	Sensitivity: About 5 pCi/cm <sup>3</sup> of air in 3 mR/hour gamma flux
Sampling	Continuous
Requirements	Power: NiCd batteries (rechargeable) Size: 9' x 9' x 13" Weight: 20 lb
Features	1) Can count tritium while rejecting higher-energy betas (Ar-41, Kr-85). 2) At flow rate of 55 cm <sup>3</sup> /min, propane bottle lasts 26 hours.
References	S. Block, D. Hodgekins, and O. Barlow, "Recent Techniques in Tritium Monitoring by Proportional Counters," Report UCRL-51131, Lawrence Livermore Laboratory
Cost	Not commercially available
Address	Lawrence Livermore Laboratory Livermore, CA 94550 Attn: S. Block (415) 447-1100



INSTRUMENTATION  
FOR ENVIRONMENTAL  
MONITORING

RAD-NUC  
Tritium in Urine  
Lawrence Livermore Laboratory 2  
Sept. 1972

Tritium Urinalyzer  
Lawrence Livermore Laboratory

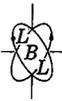




INSTRUMENTATION  
FOR ENVIRONMENTAL  
MONITORING

RAD-NUC  
Tritium in Urine  
Lawrence Livermore Laboratory 2  
Page 2 Sept. 1972

Class	Portable
Stage of Development	Not commercially available
Principle of Operation	Calcium carbide and water react ( $\text{CaC}_2 + 2\text{H}_2\text{O} + \text{C}_2\text{H}_2 + \text{Ca}(\text{OH})_2$ ). Acetylene is then used as the counting gas in a proportional counter. With tritium in urine, $\text{C}_2\text{HT}$ is the counting gas.
Sensitivity and Range	Sensitivity: 1000 pCi/cm <sup>3</sup> of urine
Sampling	Batch basis, 10 cm <sup>3</sup> urine per sample
Performance	Accuracy: ±5% to ±10% deviations per sample
Requirements	Power: NiCd batteries (rechargeable) Size: 6" x 18" x 20" [suitcase] Weight: 20 lb
Features	<ol style="list-style-type: none"><li>1. Reaction vessel is a 150 cm<sup>3</sup> vacuum filter flask sealed at top with a rubber stopper.</li><li>2. Hypodermic needle used to insert 10 cm<sup>3</sup> of urine into vessel.</li><li>3. Water-saturated acetylene is dried by flow through 20-30 mesh calcium carbide in aluminum cylinder.</li><li>4. Proportional counter is identical to that described in another instrument note except it is made gas-tight with aluminum-covered sides.</li></ol>
References	<ol style="list-style-type: none"><li>1. S. Block, D. Hodgekins, and O. Barlow, <u>Recent Techniques in Tritium Monitoring by Proportional Counters</u>, Report UCRL-51131, Lawrence Livermore Laboratory.</li><li>2. S. Block and J.E. Dixon, <u>A Portable Tritium-in-Urine Monitoring Kit</u>, Report UCRL-50007-72-1 (Hazards Control Progress Report No. 42).</li></ol>
Cost	Not commercially available
Address	Lawrence Livermore Laboratory Livermore, CA 94550 Attn: S. Block (415) 447-1100

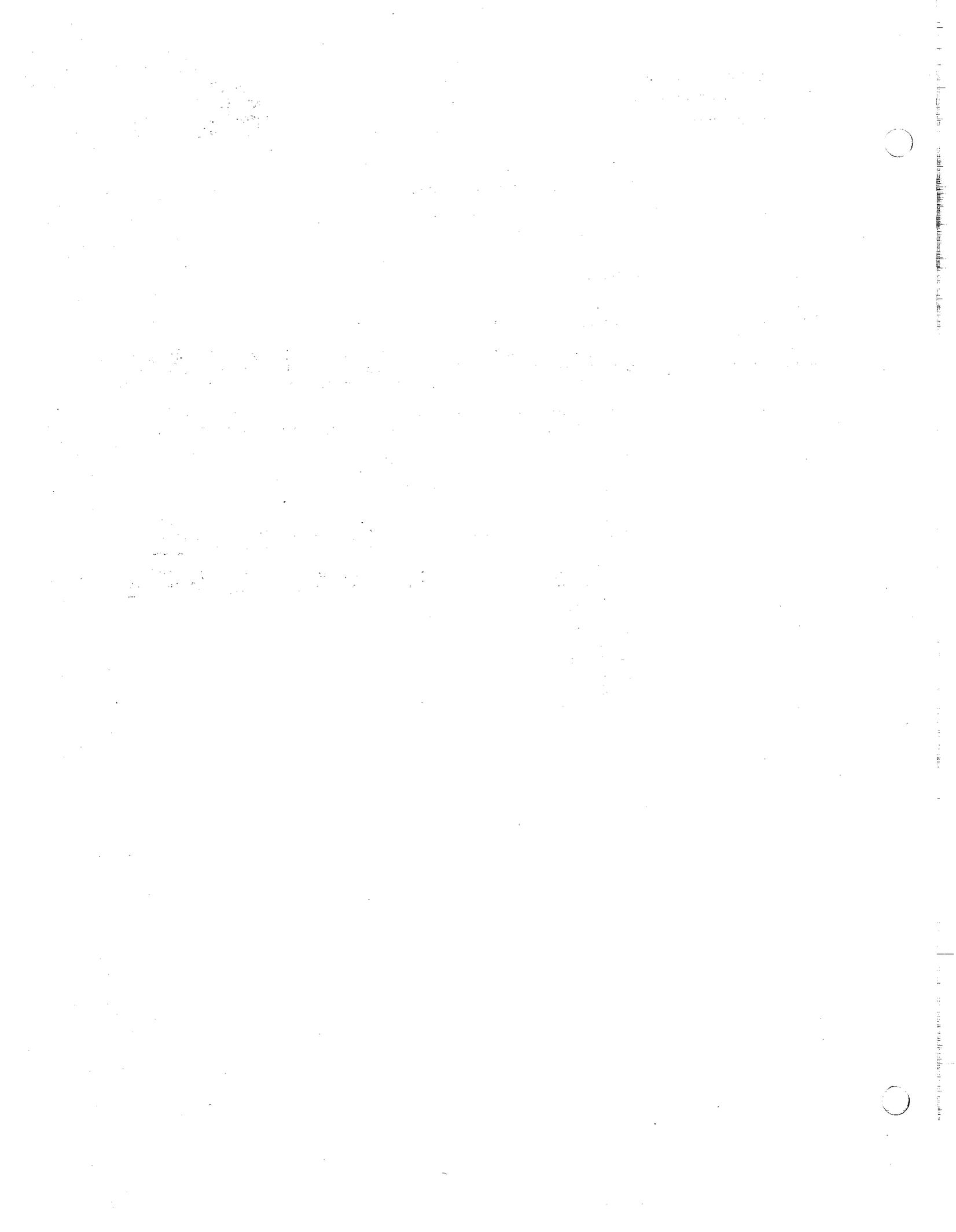


INSTRUMENTATION  
FOR ENVIRONMENTAL  
MONITORING

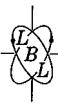
RAD-NUC  
Krypton-85  
Mound Lab  
Jan. 1973

<sup>85</sup>Kr Liquid Scintillation System  
Mound Laboratory

Class	Laboratory
Stage of Development	Not commercially available
Principle of Operation	Sample gas is expanded into evacuated vial, introduced at 25 torr pressure and dissolved into a 25 ml polyethylene vial filled with toluene-based liquid scintillator, and counted by a photomultiplier system.
Sensitivity	Minimum Detectable Concentration: $\approx 15$ pCi/cm <sup>3</sup> of air or any gas Counting Efficiency: 92.5% of <sup>85</sup> Kr betas are counted above threshold
Sampling	Limit of $\leq 25$ ml of krypton can be introduced
Performance	Accuracy: Individual measurements to $\pm 2.2\%$
References	<ol style="list-style-type: none"> <li>1. M.L. Curtis, S.L. Ness, L.L. Bentz, "Simple Technique for Rapid Analysis of Radioactive Gases by Liquid Scintillation Counting," Analytical Chem. <u>38</u>, 636 (1966); this paper describes the method.</li> <li>2. R.E. Shuping, C.R. Phillips, A.A. Moghissi, "Low Level Counting of Environmental <sup>85</sup>Kr by Liquid Scintillation," Analytical Chem. <u>41</u>, 2082 (1969).</li> </ol>
Address	M.L. Curtis Monsanto Research Corporation Mound Laboratory P.O. Box 32 Miamisburg, OH 45342 (513) 866-7444



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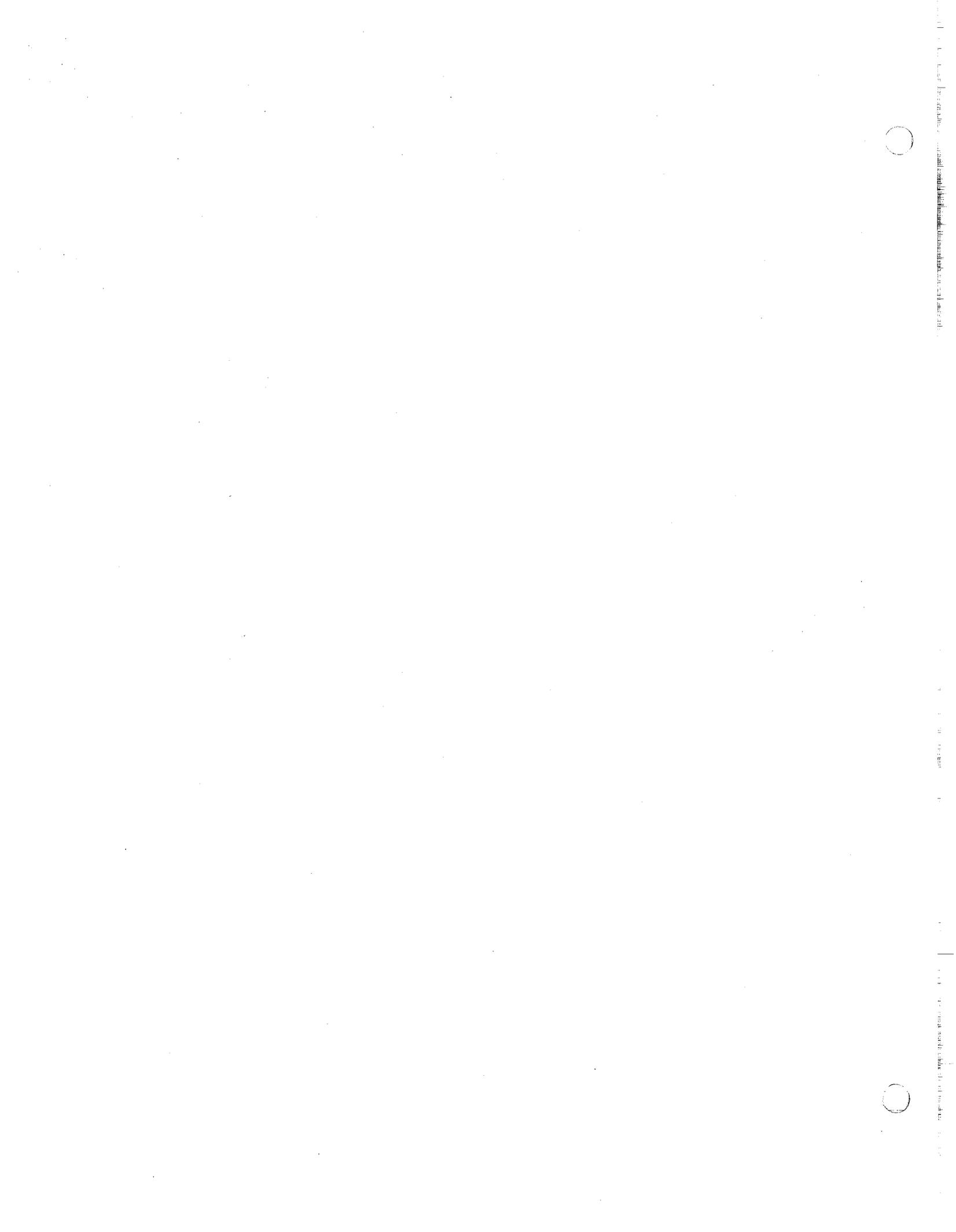


INSTRUMENTATION  
FOR ENVIRONMENTAL  
MONITORING

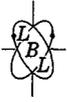
RAD-NUC  
Krypton-85  
New York State  
Aug. 1972

85Kr Plastic Scintillation System  
New York State Department of Public Health

Class	Laboratory
Stage of Development	Not commercially available
Principle of Operation	Krypton gas is concentrated using vacuum-cryogenic techniques, introduced into a 4-cc glass vial containing plastic scintillator shavings, and counted by a photomultiplier tube.
Sensitivity	Minimum detectable concentration = 1 pCi/m <sup>3</sup> of air. Counting Efficiency: 94% of <sup>85</sup> Kr betas are counted above threshold.
Sampling	Sample of about 0.5 m <sup>3</sup> of air, concentrated to 0.5 ml of krypton, can be counted.
Performance	Accuracy: Individual measurements of background <sup>85</sup> Kr (in the 10 pCi/m <sup>3</sup> range) are determined to ±10%. Interferences: Methane must be removed by burning with added O <sub>2</sub> . N <sub>2</sub> , O <sub>2</sub> , and CH <sub>4</sub> did not affect counting efficiency.
References	N.I. Sax, J.D. Denny, R.R. Reeves, "Modified Scintillation Counting Technique for Determination of Low Level <sup>85</sup> Kr," Analytical Chem. <u>40</u> , 1915 (1968).
Address	N.I. Sax Radiological Sciences Laboratory New York State Dept. of Health Albany, NY 12201  R.R. Reeves Rensselaer Polytechnic Institute Troy, NY



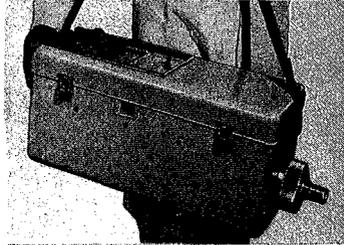
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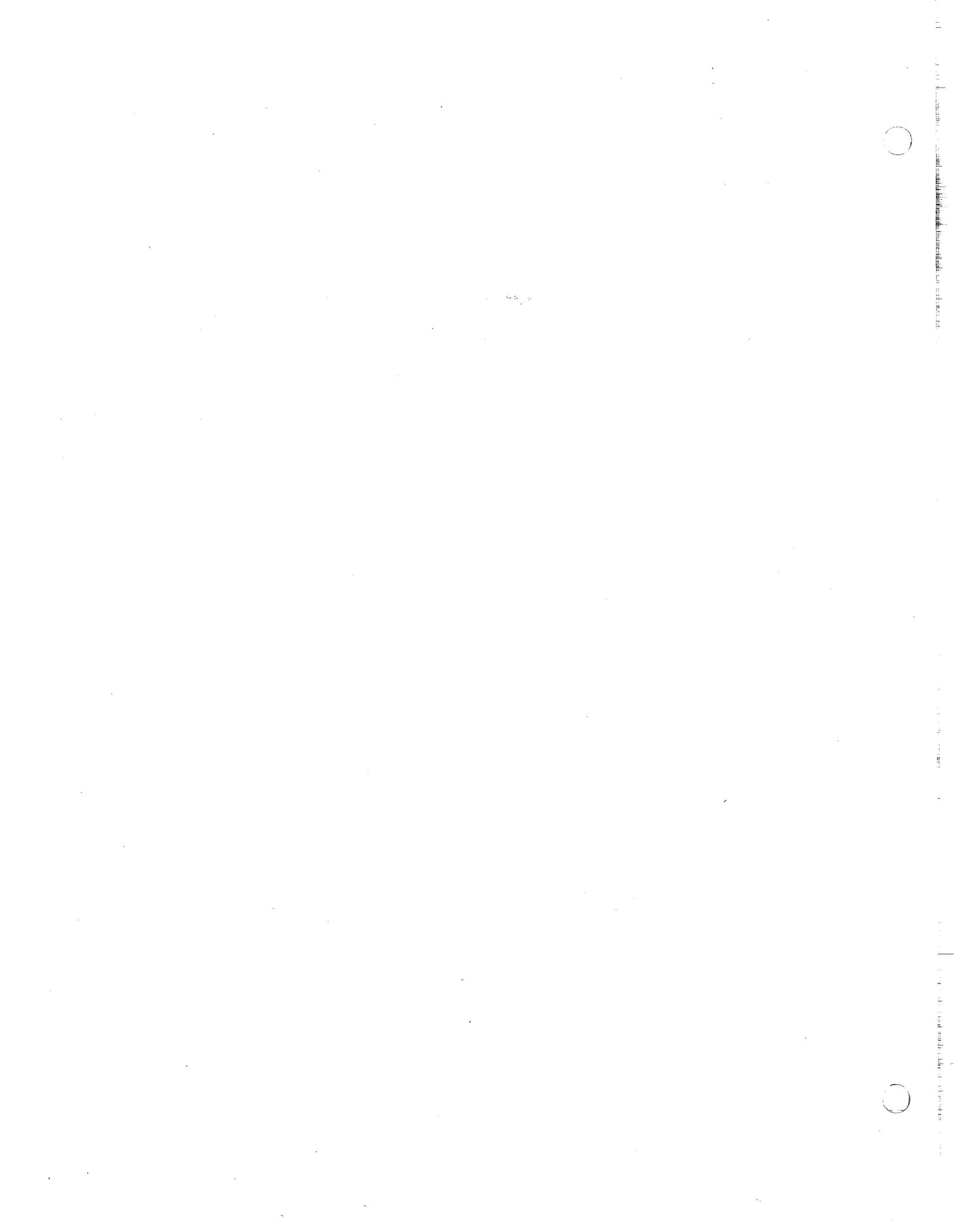
INSTRUMENTATION  
FOR ENVIRONMENTAL  
MONITORING

RAD-NUC  
Tritium  
Nuclear Enterprises  
Dec. 1971

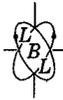
Tritium in Air Monitor  
Nuclear Enterprise Model 1762



Class	Portable, hand-held
Principle of Operation	Ion chamber, amplifier, meter read-out
Sensitivity and Range	Energy sensitivity: Range 10 to 15,000 pCi/cm <sup>3</sup> Interferences:
Sampling	Continuous
Performance	Meter accuracy: Drift: 2% Temperature range:
Requirements	Power: Size: Weight:
Features	
Cost	
Address	Nuclear Enterprises Inc. 935 Terminal Way San Carlos, California 94070 (415) 593-1455



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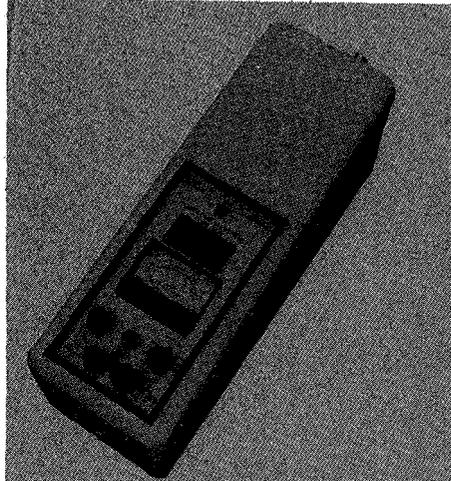


INSTRUMENTATION  
FOR ENVIRONMENTAL  
MONITORING

RAD-NUC  
Tritium, air  
Nuclear Enterprises 2  
Mar. 1972

### Tritium in Air Gas Monitor

#### Nuclear Enterprise Model 1762A



**Class** Portable, hand-held radioactive gas monitor

**Principle of Operation** A dual concentric gamma compensated ionization chamber, which enables measurements of HTO (tritiated water vapor) and other radioactive gases in air. Current from the chamber is measured. A dust filter and an electrostatic precipitator is used for the removal of small ionization particles and gaseous ions to ensure that the current measured is due the actual radioactive disintegrations within the ionization chamber.

**Sensitivity and Range** Range: 10-50, 150, 1500, and 15,000 pCi/cm<sup>3</sup> of HTO in air

Gas	Meter Reading for 1 MPC
HTO	5 pCi/cm <sup>3</sup>
Argon 41	4 pCi/cm <sup>3</sup>
Krypton 85	10 pCi/cm <sup>3</sup>
Radon	21.5 pCi/cm <sup>3</sup>
Xenon 133	40 pCi/cm <sup>3</sup>

**Performance** Accuracy: ± 20% of reading above 1 MPC (MPC of HTO in air is 5 pCi/cm<sup>3</sup> in England)

Temperature coefficient: ± 1% of full scale per 10°C change

**Requirements** Power: Mallory Batteries - RM-42-R, (500 hrs); TR-134-R, (400 hrs); TR-134-R, 2 ea. (200 hrs)

Everyready Batteries - Type 126, 3 ea., motor, (8-9 hrs)

Size: 29.8 cm H, 20.3 cm W, 64.5 cm L (11.75", 8", 25.25")

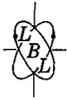
Weight: 12.25 Kg (27 lbs)

**Cost**

**Address** Nuclear Enterprises Ltd.  
935 Terminal Way  
San Carlos, Calif. 94070  
(415) 593-1455



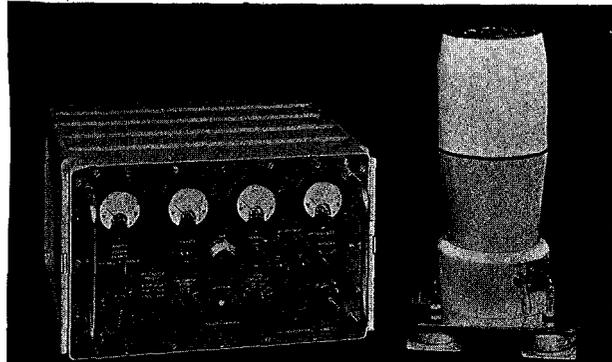
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INSTRUMENTATION  
FOR ENVIRONMENTAL  
MONITORING

RAD-NUC  
Iodine  
Nuclear Enterprise 3  
Dec. 1971

Iodine 131 Monitor  
Nuclear Enterprise NE 8423

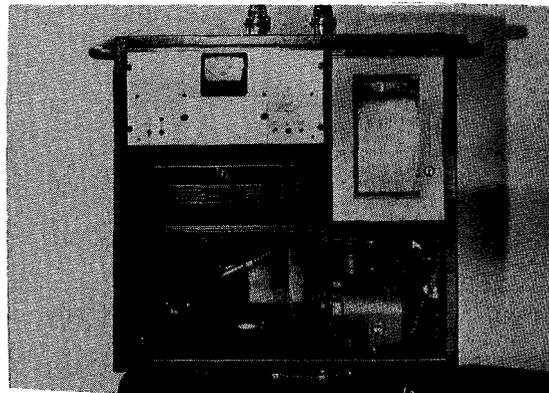
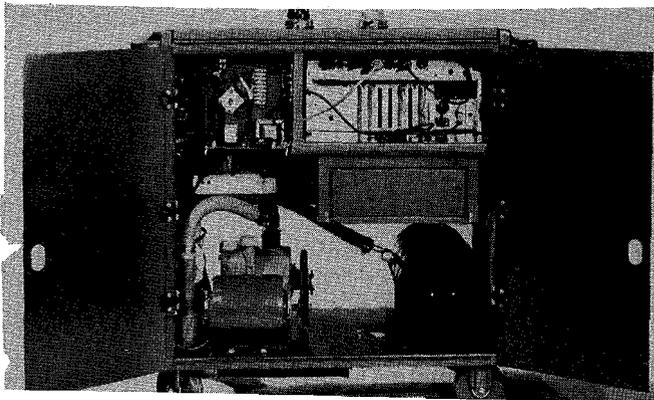


Class	Portable
Principle of Operation	Gamma detector and three discriminators select I-131 energy fission products, meter read-outs.
Sensitivity and Range	Energy Sensitivity: Range: 0.007 $\mu$ Ci to 0.65 $\mu$ Ci Interferences:
Sampling	Continuous, sample size 7 cm dia., 6 cm thick
Performance	Accuracy: Temperature range: Response time:
Requirements	Power: 12 V d.c. car battery Size: Weight:
Features	"Subtraction" electronics displays I-131 in presence of higher and lower energy fission products. Rate meters display channels A,B,C, and B-(A+C)
Cost	
Address	Nuclear Enterprise Inc. 935 Terminal Way San Carlos, California 94070 (415) 593-1455





Air Radiation Monitor  
Nuclear Measurements AM-21

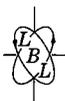


Class	Field
Principle of Operation	The monitoring system consists of two parts, the pump system and the detecting system. Housed within a lead shield assembly is a gamma scintillation detector used for detecting gamma emitters below 1 MeV and a particulate collector; as the particulate matter builds up, the detector converts the ionizing events into pulses which are counted on ratemeter, also air is entering and leaving the lead housing at a flow rate from 3-10 cfm through the filters for deposition of particulates.
Sensitivity and Range	Count rate will be 318 cpm for Iodine <sup>131</sup> at 10% yield, 5 cfm airflow rate, and 10 <sup>-10</sup> µc/cc constant activity.
Sampling	Airflow through filters and charcoal cartridge
Performance	Ratemeter accuracy: ±10% of actual input cartridge Time constant: 200 seconds at 50 c/m to 1 second at 50,000 c/m Stability: Drift not more than ±2% daily, ±20% annually except in narrow window spectrometry
Requirements	Power: 90-130 V ac, 60 Hz, 5-7 amp Size: 53 cm X 1 M x 1 M (21" x 39" x 39") Weight: 250 Kg (550 lbs)
Features	Graphic recorder provision, alarm controls
Cost	\$4390 with Esterline-Angus recorder
Remarks	Other air monitors available similar to the above are given in the following table.
Address	Nuclear Measurement Corporation 24260 North Arlington Avenue Indianapolis, Indiana 46218 (317) 546-2415



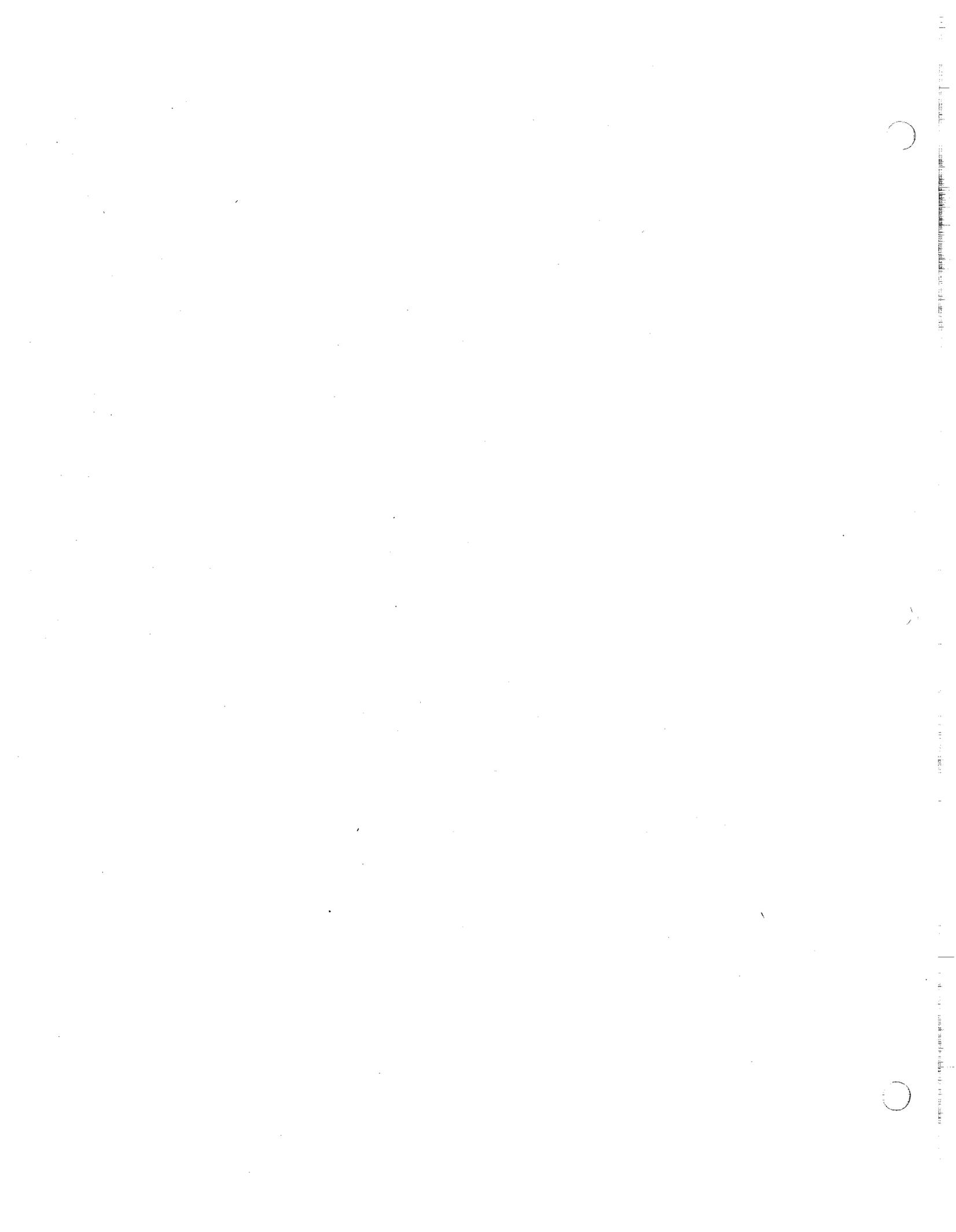
NMC Air Monitors

MODEL	RADIATION MONITORED	PRICE	DETECTORS	RECORDER	FIXED FILTER SA - 20 SERIES
AM - 2A	Alpha	\$2740	SC-2a	1 Single Channel or 1 Rustrak	1 C/M
AM-3A		\$3220	SC-2A	1 Single Channel or 1 Rustrak	
AM-2B	Beta	\$2960	SC-2B	1 Single Channel or 1 Rustrak	21 C/M
AM-3B	Beta	\$3440	SC-2B	1 Single Channel or 1 Rustrak	
AM-2D	Gross Beta-Gamma	\$2860	DGM-2	1 Single Channel or 1 Rustrak	21 C/M
AM-3D	Gross Beta-Gamma	\$3415	DGM-2	1 Single Channel or 1 Rustrak	
AM-2F	Alpha, Beta, Gamma	\$3010	FGM-2	1 Single Channel or 1 Rustrak	30 C/M
AM-3F	Alpha, Beta, Gamma	\$3580	FGM-2	1 Single Channel or 1 Rustrak	
AM-2G	Gamma	\$3780	SC-2-2S	1 Single Channel or 1 Rustrak	Gross 128 C/M SPEC. 10 C/M
AM-3G	Gamma	\$4410	SC-2-2S	1 Single Channel or 1 Rustrak	
AM-2I	Iodine	\$3890	SC-2-1S	1 Single Channel or 1 Rustrak	Gross 97 C/M SPEC 10 C/M
AM-3I	Iodine	\$4530	SC-2-1S	1 Single Channel or 1 Rustrak	
AM-22I	Iodine + Alpha or Beta or Gamma	\$5450	SC-2-1S and DGM-2 or PC-21-FW	1 Dual Channel or 2 Rustrak	Iodine 10 C/M GROSS 97 C/M
AM-33I		\$6015		"	
AM-2P	Alpha or Beta or	\$3310	PC-21-FW	1 Single Channel or 1 Rustrak	a < .1 C/M b < 30 C/M
AM-3P	Alpha + Beta	\$3880	PC-21-FW	"	
AM-22P	Alpha + Beta separately	\$3780	PC-21-FW	1 Dual Channel	a < .1 C/M
AM-33P	but simultaneously	\$4360	PC-21-FW	or 2 Rustrak	b < 30 C/M
PAPM-11	Plutonium	\$4580	SC-2A	"	10 C/M

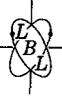


NMC Air Monitors

MODEL	MOVING FILTER SA-30 SERIES	MOVING FILTER SA-30 SERIES	FIXED FILTER SA-3L	MOVING FILTER SA-30 SERIES	MOVING FILTER SA-3L
AM-2A	-	-	20%	-	-
AM-3A	1 C/M	1 C/M	-	20%	11%
AM-2B	-	-	32%	-	-
AM-3B	21 C/M	40 C/M	-	32%	18%
AM-2D	-	-	10%	-	-
AM-3D	21 C/M	30 C/M	-	10%	5.7%
AM-2F	-	-	35%	-	-
AM-3F	30 C/M	30 C/M	-	32%	17%
AM-2G	-	-	19%	-	-
AM-3G	GROSS 128 C/M Spec. 10 C/M	GROSS 120 C/M Spec. 35 C/M	-	19%	8%
AM-21	-	-	17% GROSS 3% IODINE	-	-
AM-31	GROSS 97 C/M Spec 10 C/M	IODINE 30 C/M Gross 115 C/M	-	10% GROSS 15% Iodine	10% GROSS 15% Iodine
AM-221	-	-	17% GROSS 3% IODINE	-	-
AM-331	IODINE 10 C/M Gross 97 C/M	IODINE 30 C/M Gross 115 C/M	-	17% GROSS 3% Iodine	17% GROSS 3% Iodine
AM-2P	-	-	$\alpha$ 37% $\beta$ 37.5%	-	-
AM-3P	$\alpha$ <.1 C/M $\beta$ <30 C/M	$\alpha$ <1 C/M $\beta$ <40 C/M	-	$\alpha$ 37% $\beta$ 37%	$\alpha$ 20% $\beta$ 22%
AM-22P	-	-	$\alpha$ 37% $\beta$ 37%	-	-
AM-33P	$\alpha$ <10/M $\beta$ <30 C/M	$\alpha$ <10/M $\beta$ <40 C/M	-	$\alpha$ 37% $\beta$ 37%	$\alpha$ 35% $\beta$ 32%
PAPM-11	10 C/M	-	-	35%	-



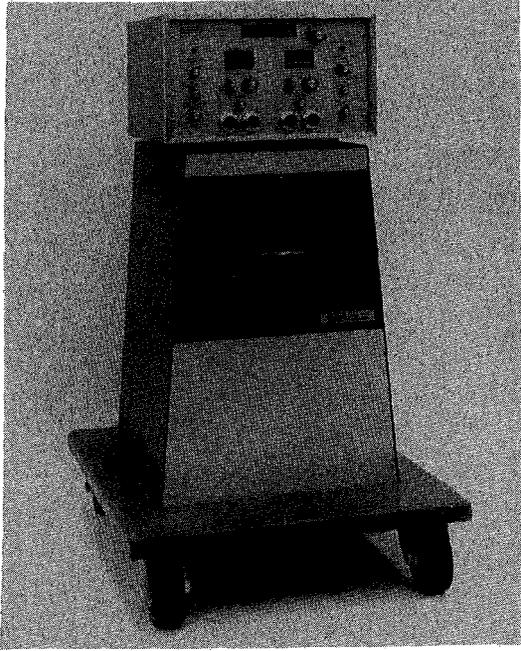
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INSTRUMENTATION  
FOR ENVIRONMENTAL  
MONITORING

RAD-NUC  
Packard  
Nov. 1972

Scintillation Detector  
Packard Model 446



Class	Laboratory
Principle of Operation	Cylindrical sample chamber is surrounded by a 4" annulus of liquid scintillator and six 3" dia. photomultiplier tubes. The tubes are magnetically shielded and have individual controls for focus and gain. A lead shield surrounds the entire detector to reduce background. A handhold is provided to insure reproducible geometry for forearm counting.
Sensitivity and Range	Sensitivity: <sup>60</sup> Co, 48% eff.; <sup>131</sup> I, 46% eff.; <sup>59</sup> Fe, 44% eff.; <sup>137</sup> Cs, 38% eff. Range: 150 keV or higher
Sampling	Aluminum cylinder 4.25" D, 8" L.
Performance	Accuracy: Temperature:
Requirements	Power: Size: 100 cm H, 95 cm W, 79 cm D (39.5", 37.3", 31") Weight: 77 kg (about 1700 lb)
Features	
Cost	Model 446 \$8,900 Less shielding and cabinet \$5,855
Address	Packard Instrument Co., Inc. 2200 Warrenville Road Downers Grove, IL 60515 (312) 696-6000



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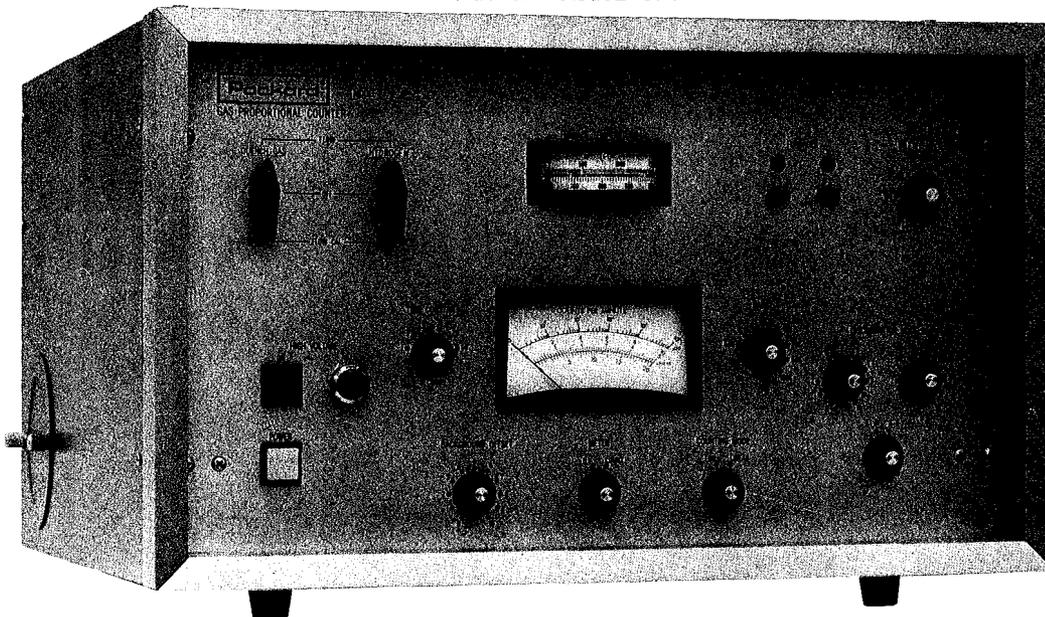


INSTRUMENTATION  
FOR ENVIRONMENTAL  
MONITORING

RAD-NUC  
Tritium, Carbon-14  
Packard 2  
Feb. 1973

### Gas Proportional Counter

Packard Model 894



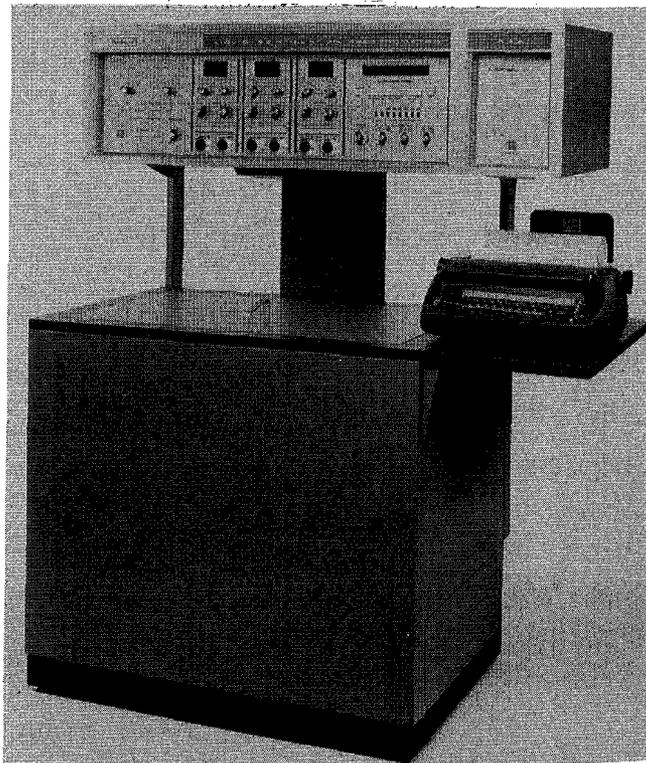
Class	Laboratory
Principle of Operation	Detects tritium or carbon-14 labelled organic compounds as they elute from a gas chromatographic column using a flow-through ion-multiplication detector. A copper oxide furnace oxidizes all organic compounds to carbon dioxide and water to eliminate condensation of high boiling point compounds in the proportional tube.
Sensitivity and Range	Sensitivity: Betas -- 200 dpm/peak carbon-14 labelled compounds Range: 10 linear ranges, 100 to 100,000 cpm full scale Five decade log scale Counting Efficiency: 58% Tritium 80% Carbon-14
Sampling	Continuous
Performance	Accuracy: Temperature: Operates at room temperature Inlet heating to 300°C Furnace heating to 800°C Time Constant: Six ranges, 1 to 50 sec
Requirements	Power: 120-220 Vac (switch), 50 or 60 Hz, 720 W Size: 35.6 cm H, 54.6 cm W, 61 cm D (14", 21.5", 24") Weight: 90 kg (200 lb)
Features	Use with Model 7401 Gas Chromatograph or other gas chromatographs which have been fitted with stream splitters and heated exit ports. Background about 30 cpm (variable offset). Variable upper and lower discriminators. Strip chart recorder and scaler output jacks. Right or left-hand capability to couple with any gas chromatograph
Cost	\$4400
Address	Packard Instrument Company, Inc. 2200 Warrenville Road Downers Grove, IL 60515 (312) 696-6000





Liquid Scintillation Spectrometers

Packard Model 3390 and Others



Class	Laboratory
Principle of Operation	Liquid scintillation spectrometer system. Matched bialkali photomultiplier tubes in coincidence mode detect light; pulse height analysis electronics provides spectrum information. Background subtraction done automatically. Automatic calibration with external source arrangement.
Sensitivity and Range	Sensitivity: Unknown Energy Range: Can detect any beta, alpha, electron capture, X-ray, or weak gamma ray isotopes.
Sampling	300 sample capacity with up to 10 repetitive counts per sample or up to 10 preset number of complete cycles for counting all samples. Also has automatic and group counting mode that allows empty samples with a group.
Performance	Accuracy: Unknown Temperature: Controlled temperature (maintained $\pm 0.5^{\circ}\text{C}$ over range $-5^{\circ}\text{C}$ to $+15^{\circ}\text{C}$ ) or ambient temperature
Requirements	Power: 100-130 or 200-260 V ac, 50-60 Hz, 1000 VA (max) Size: Height 62-1/4 to 67-1/4", adjustable (158 to 171 cm); width 41-1/2 to 57" (depending on readout device); depth 31" (79 cm) Weight: 1000 lb (450 kg)



Features

1. Three separate pulse-height analysis channels.
2. Single-, double-, triple-labeled capability.
3. Independent controls for normalizing each photomultiplier tube.
4. Preset count, background subtraction, low-level reject for each sample channel.
5. Automatic external standardization with radium-226 and americium-241 sources.
6. On-line computation and display of net count rate and standard deviation.
7. Gamma pulse-height capability.
8. Automatic sample oxidizers: Models 300, 305, and 306 (optional).
9. Flow measurement capability using flow-cell and adapter.
10. Model 544 Absolute Activity Analyzer automatically adjusts to optimize counting rate and background conditions and computes quench-corrected dpm for single- or double-labeled samples (optional).
11. Model 3050 extra shielding package for low-level counting (optional).
12. Model 3090 Automatic Channel Selection (optional).

Other Models

Model 3385: Identical to Model 3390 except not available with Model 544 Absolute Activity Analyzer.

Model 3375: Identical to 3380 except not available with 544 Absolute Activity Analyzer.

Model 3330: Similar to 3390 except does not have on-line computation feature, one-source instead of two-source calibration system, some other differences.

Models 2101, 2111, 2211, and 2311: Have fewer features; similar to Model 3320.

Model 3380: Identical to Model 3390 except 200-sample capacity and not available with Model 3090 Automatic Channel Selection.

Model 3320: Identical to Model 3330 except 200-sample capacity.

Models 2425 and 2450: Similar to Model 3390 except Servo-Tray sample changer. Model 2425 has 150sample capacity. Model 2450 has 450-sample capacity.

References

Manufacturer's specifications

Cost	Model 3390	\$17,095.	Model 3330	\$15,595.
	544	4675.	3305	8665.
	513	1435.	2450	14,500.
	545	1670.	2425	10,615.
	542	525.	2311	9565.
	3385	16,095.	2211	7990.
	3380	16,595.	2111	6205.
	3375	15,595.	2101	5200.

Address

Packard Instrument Company Inc.  
2200 Warrenville Road  
Downers Grove, IL 60515  
(312) 969-6000

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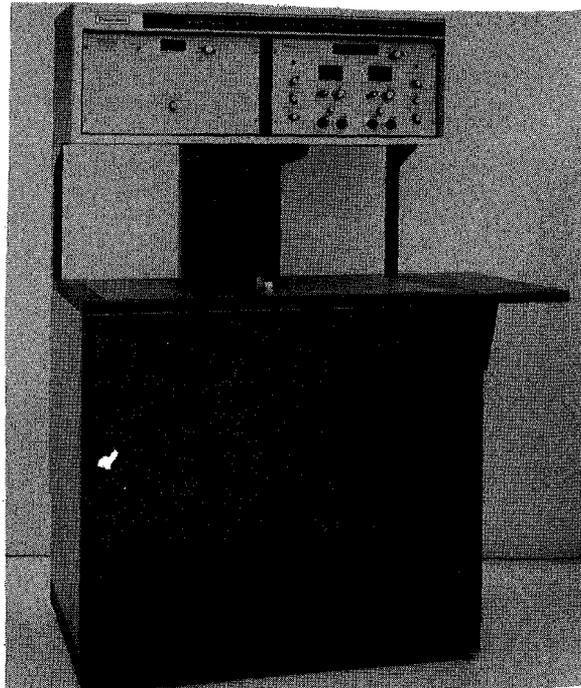
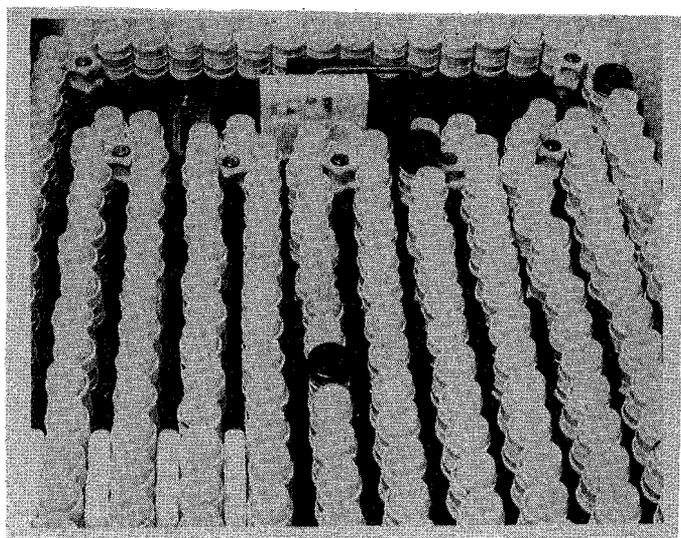


INSTRUMENTATION  
FOR ENVIRONMENTAL  
MONITORING

RAD-NUC  
Spectrometer  
Packard 4  
Jan. 1973

Auto-Gamma Spectrometer

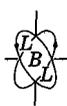
Packard 5000 Series



Class	Laboratory
Principle of Operation	Automatic 300 sample controlled temperature gamma spectrometer. Two diametrically opposed NaI(Tl) scintillation crystals, photomultipliers. Automatic background subtraction, low-level count rejection. Three independent pulse-height analysis channels with three scalers.
Sensitivity and Range	Sensitivity: Energy Range: 10 keV or higher
Sampling	300 sample capacity changer with repeat, group, and automatic counting modes. Solid or liquid samples.
Performance	Accuracy: Temperature: Maintained within $\pm 0.5^\circ\text{C}$ or ambient operation
Requirements	Power: 117 or 234 Vac $\pm 10\%$ , 50 or 60 Hz, 10 A (max) Size: 170 cm H, 105 cm W, 79 cm D (67", 41.5", 31") Weight: 680 kg (1500 lb)
Features	1. Preset time/preset count modes up to nine isotopes. 2. Coarse and continuous calibrated fine gain adjustment. 3. Digital printer output, IBM Selectric or Teletype option.
References	Manufacturer's specifications
Cost	Model 5320 \$13,250 5220 12,735 5120 11,190 5385 16,135
Address	Packard Instrument Company Inc. 2200 Warrenville Road Downers Grove, IL 60515 (312) 696-6000



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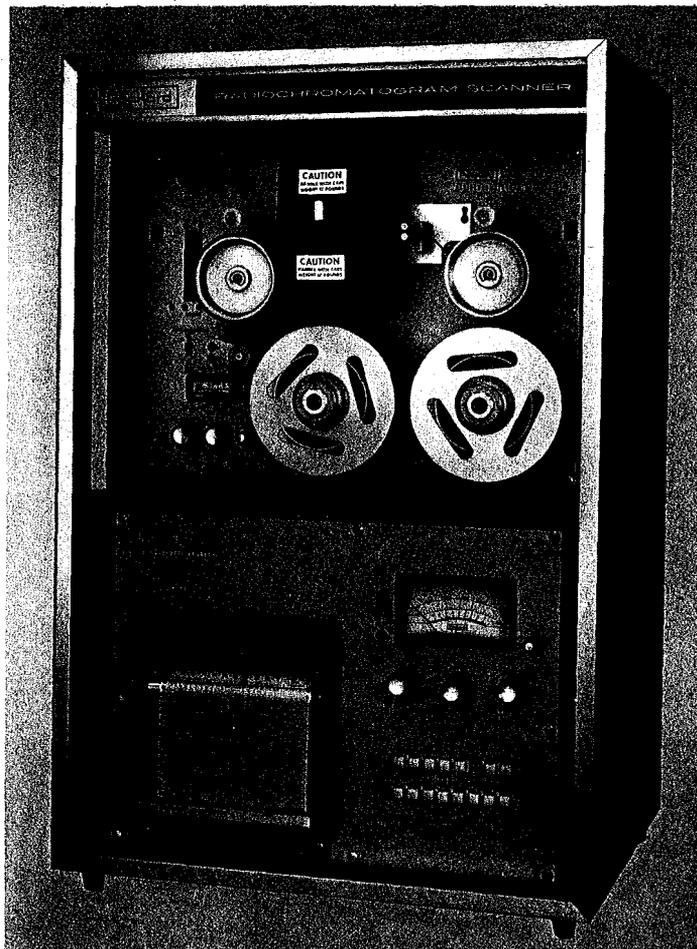


INSTRUMENTATION  
FOR ENVIRONMENTAL  
MONITORING

RAD-NUC  
Beta Scanner  
Packard 5  
Nov. 1972

### Radiochromatogram Scanner

#### Packard Model 7201



Class	Laboratory
Principle of Operation	Radioactivity is measured by two windowless G-M tubes arranged in 4 $\pi$ configuration to scan both sides of the strip. Resolution between adjacent spots is controlled by two adjustable collimators.
Sensitivity and Range	Sensitivity: Beta emitting nuclides Range: Ratemeter has 9 linear count ranges and log scale $10^2$ , $10^3$ , $10^4$ , $10^5$ , $10^6$ , and $3 \times 10^2$ , $10^3$ , $10^4$ , $10^5$ .
Sampling	Accepts chromatograms from 0.5" to 2" wide, lengths up to 150 feet, all types of papers up to Whatman No. 3MM
Performance	Accuracy: See "Time Constants" Temperature: $-9^\circ\text{C}$ to $+50^\circ\text{C}$  Time Constants: From 0.1 to 300 sec (8 settings) Scanning Speed: From 0.02 cm/min to 200 cm/hr (16 ea.)





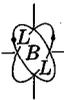
Liquid Scintillation System  
Searle, Nuclear Chicago, Isocap 300

ISOCAP/300 PROGRAM CHART

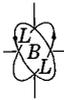
Solvent system sample	CPM	<sup>3</sup> H	<sup>14</sup> C	<sup>32</sup> P	<sup>36</sup> CL	<sup>35</sup> S	<sup>3</sup> H/ <sup>14</sup> C	<sup>3</sup> H/ <sup>32</sup> P	<sup>125</sup> I	<sup>131</sup> I	NOTES
Toluene/oil soluble	H	1B	2B	3B	7B	2B	8	10	7A	7B	8
	L	4B	5B	6B	7B	5B	8	10	7A	7B	
Toluene-NCS/dilute acids	H	1B	2B	3B	7B	2B	8	10	7A	7B	8,12
	L	4B	5B	6B	7B	5B	8	10	7A	7B	
/muscle and light colored tissue	H	1B	2B	3B	7B	2B	8	10	7A	7B	4,7,8,12
	L	4A	5B	6B	7B	5B	8	10	7A	7B	
/liver and dark colored tissue	H	1B	2B	3B	7A	2B	9	10	7A	7A	4,7,9,12
	L	4A	5A	6B	7A	5A	9	10	7A	7A	
/blood	H	1B	5A	3B	7A	5A	9	10	4B	7A	4,7,9,12
	L	4A	5A	6A	7A	5A	9	10	4B	7A	
Toluene or xylene + alcohol or cellosolve /aqueous solutions	H	1B	2B	3B	7B	2B	8	10	7A	7B	8
	L	4A	5B	6B	7B	5B	8	19	7A	7B	
Triton-toluene 1:1/ water	H	1B	2B	3B	3B	2B	8	10	2B	3B	1,2,6,7, 8,11
	L	1B	2B	3B	3B	2B	8	10	2B	3B	
/dilute acids	H	1B	2B	3B	3B	2B	8	19	2B	3B	1,2,6,7, 8,11
	L	1B	2B	3B	3B	2B	8	10	2B	3B	
/aqueous solutions	H	1B	2B	3B	3B	2B	8	10	2B	3B	1,2,6,7 8,11
	L	1B	2B	3B	3B	2B	8	10	2B	3B	
Dioxane (Bray's)/ water	H	1B	2B	3B	7B	2B	9	10	7A	7B	3,9,10
	L	1B	5A	6B	7B	5A	9	10	7A	7B	
/dilute acids	H	1B	2B	3B	7B	2B	9	10	7A	7B	3,9
	L	1B	5A	6B	7B	5A	9	10	7A	7B	
/aqueous solutions	H	1B	2B	3B	7B	2B	9	10	7A	7B	3,9
	L	1B	5A	6B	7B	5A	9	10	7A	7B	
Toluene-Cab-O-Sil/ suspended solids	H	1B	2B	3B	3B	2B	-	-	2B	3B	2,5,7
	L	1B	2B	3B	3B	2B	-	-	2B	3B	
any /sample on paper	H	1B	2B	3B	3B	2B	-	-	2B	3B	2,5,7,13
	L	1B	2B	3B	3B	2B	-	-	2B	3B	
Aqueous/Cerenkov	H	-	-	1B	1B	-	-	-	-	1B	7
	L	-	-	1B	1B	-	-	-	-	1B	

The program analysis windows are designed to meet all routine counting conditions. The sample solvent systems shown here were prepared in Nuclear-Chicago's research department, and included a wide variety of applications which were designed to meet the majority of counting requirements.

Program numbers shown on the chart represent the best match of sample system to analysis window and will give optimum performance. The "A" or "B" following the program number designates the channel of interest. For dual label samples both channels are used.

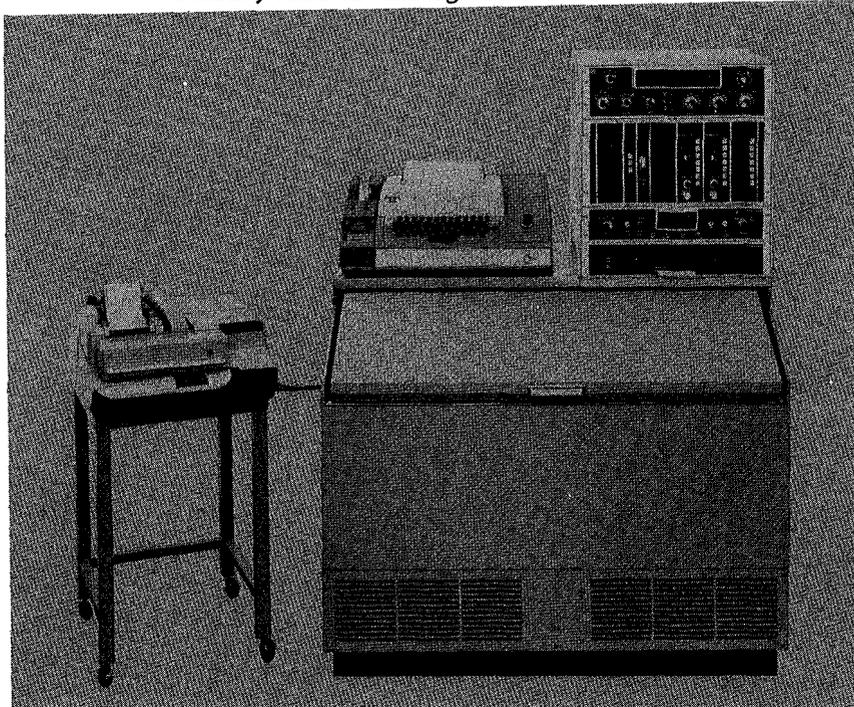


Class	Laboratory
Principle of Operation	Programmable liquid scintillation system, sample changer, two scintillator photomultiplier channels
Sensitivity and Range	Sensitivity: Energy Range:
Sampling	Capacity: 300 samples, bi-directional conveyor Programmable sample windows
Performance	Accuracy: Temperature: Stability: Shift of 14C count rate measured in a dual labelled window (program 8) shall not exceed 0.25% per °C between 15°C and 30°C.
Requirements	Power: 115 V ± 10 Vac, 60 Hz, 230 Vac and/or 50 Hz optional. Size: 75 cm D, 107 cm W, 56 cm H (30" x 42" x 22") Weight: 315 kg (700 lb) shipping.
Features	Optional readouts, stands, calculators
References	Manufacturer's Specifications
Cost	Basic Isocap 300                      \$12,220.
Address	Searle, Nuclear Chicago 2000 Nuclear Drive Des Plaines, IL 60018

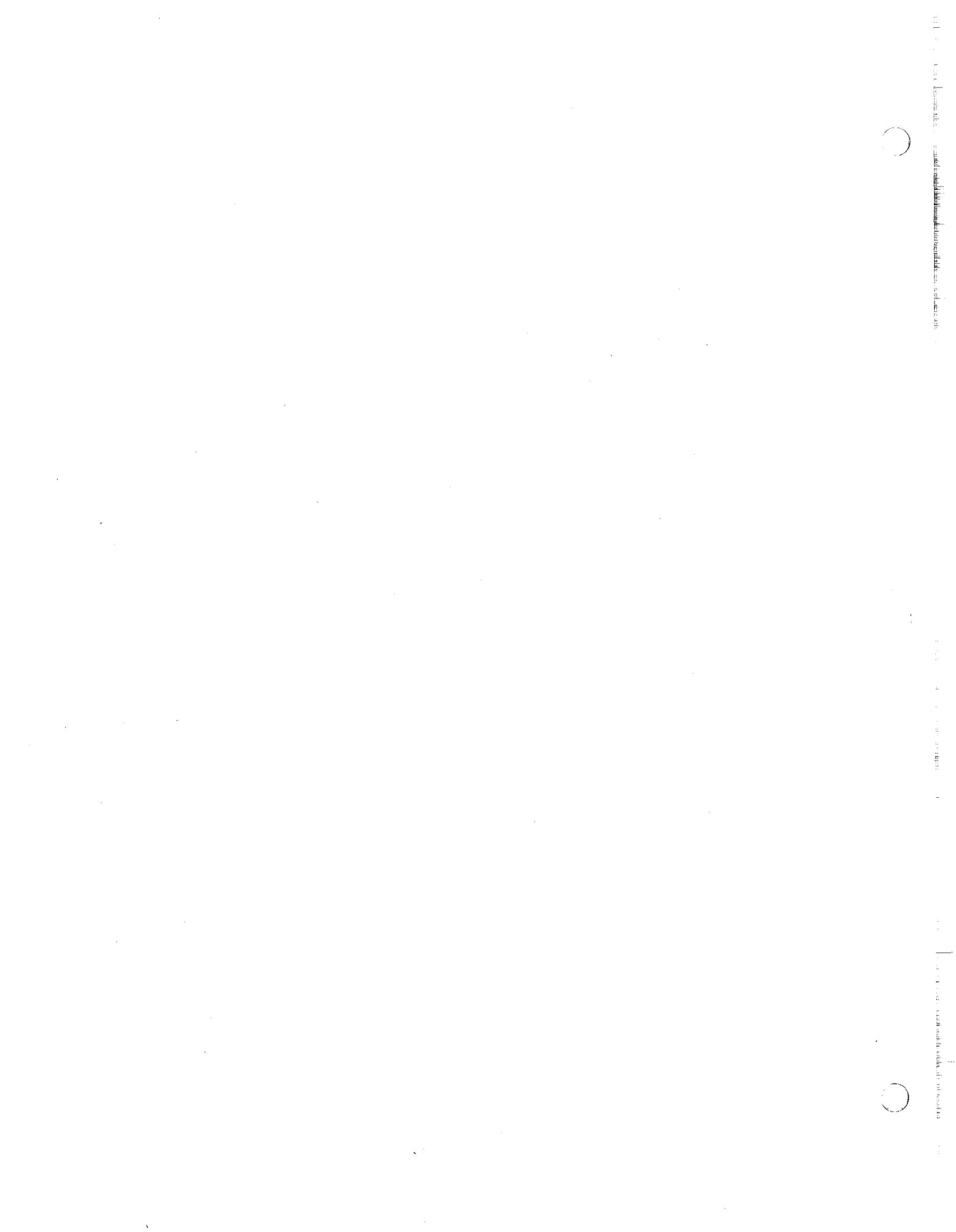


### Liquid Scintillation Systems

#### Searle, Nuclear Chicago Mark II Series

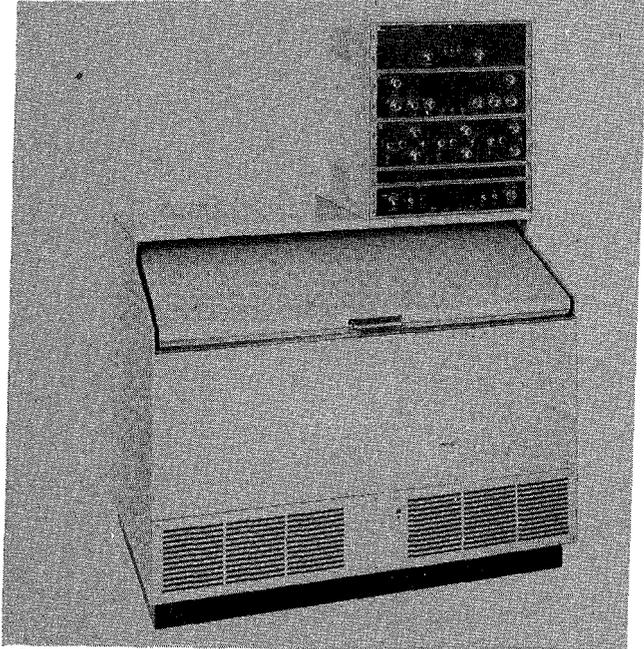


Class	Laboratory	
Principle of Operation	Liquid scintillation system having controlled temperature enclosure for counting single labeled, dual labeled, or intermixed samples, two scintillator-photomultiplier detectors, single channel analyzers, and scaler/timers; external standardization and automatic lister. Single channel analyzers may be pre-set to count $^3\text{H}$ , $^{14}\text{C}$ , $^{32}\text{P}$ , $^3\text{H}$ in the presence of $^{14}\text{C}$ , $^{14}\text{C}$ in the presence of $^3\text{H}$ .	
Sensitivity and Range	Sensitivity: Energy Range:	
Sampling	Capacity: 300 samples Modes: Manual, one cycle, all samples, groups only, group start	
Performance	Accuracy: Temperature: Controlled cooling, automatic defrosting	
Requirements	Power: 115 V $\pm$ 10 Vac, 60 Hz, 230 Vac and/or 50 Hz optional, 1000 watts normal consumption, 2500 start-up. Size: cm H, 79 cm D, 107 cm W ( " H, 31" D, 42" W)	
Features	Optional three channel system, automatic self-calibration, standardization module, auto-programmer module, Teletypewriter <sup>®</sup> , IBM Selectric <sup>®</sup> Typewriter, Olivetti Computer calculator, or card punch.	
References	Manufacturer's specifications	
Cost	Basic Mark II	\$17,305.
Address	Searle, Nuclear Chicago 2000 Nuclear Drive Des Plains, IL 60018	



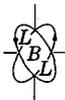


Liquid Scintillation Spectrometer  
Searle, Nuclear Chicago Series 6848

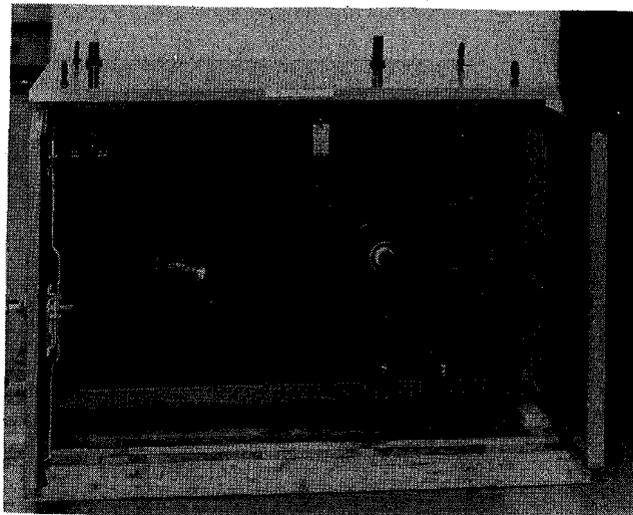


Class	Laboratory
Principle of Operation	Liquid scintillation spectrometer systems having a controlled temperature enclosure for counting single labeled, dual labeled, or intermixed samples of soft beta emitting radioisotopes such C <sup>14</sup> and tritium. Three single channel analyzers, optional readouts.
Sensitivity and Range	Sensitivity: Energy Range:
Sampling	Capacity: 300 samples, samples/min. Modes: Manual, auto stop, all samples, groups only, groups start
Performance	Accuracy: ±0.5% of a Ba <sup>133</sup> external standard. Temperature Control:
Requirements	Power: 115 Vac ±10V, 60 Hz; 230 Vac and/or 50 Hz optional. 1000 W maximum normal power consumption, 2500 W start-up. Size: cm H, 78 cm D, 105 cm W ( " x 31" x 42") Weight: 436 kg (960 lbs)
Features	<ol style="list-style-type: none"> <li>1. Frost free controlled-temperature system</li> <li>2. Low count sample rejection</li> <li>3. Display hold for isotope peaking</li> <li>4. Options: Automatic lister, auto/subtract background subtractor, data calculator</li> </ol>
References	Manufacturer's Specifications
Cost	
Address	Searle, Nuclear Chicago 2000 Nuclear Drive Des Plaines, IL 60018





Continuous Gas Monitoring  
Trapelo Model MGP-1A



Class Laboratory

Principle of Operation Gas sampler, gamma scintillation or G-M tube detector, filter, ratemeter, alarm, and pump

Sensitivity and Range

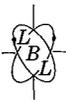
SENSITIVITIES OF TRACERLAB MG-1A AND MG-2A GAS SAMPLERS  
FOR NET SIGNAL EQUAL TO BACKGROUND

Activity	Type and Energy (MeV)		Half-Life	Min. Detectable Conc. µc/cc	
	Gammas	Betas		MG-1A*	MG-2A**
<sup>19</sup> O	1.3 (64%) 0.2 (96%) 0.1 (4%)	4.5 (30%) 2.9 (70%)	30 Seconds	7 x 10 <sup>-7</sup>	3.8 x 10 <sup>-7</sup>
<sup>41</sup> A	1.37 (100%)	1.24 (100%)	110 Minutes	6.25 x 10 <sup>-7</sup>	5.5 x 10 <sup>-7</sup>
<sup>82</sup> Br	γ decay scheme in doubt	0.46 (100%)	36 Hours	-	7.6 x 10 <sup>-7</sup>
<sup>85</sup> Kr <sup>131</sup> I	0.54 (0.35%) 0.36 (87%) 0.64 (9%) 0.72 (3%)	0.69 (100%) 0.61 (87%)	9.4 Years 8.1 Days	3.5 x 10 <sup>-4</sup> 1.7 x 10 <sup>-6</sup>	9.6 x 10 <sup>-7</sup> 1.3 x 10 <sup>-6</sup>
<sup>133</sup> Xe	0.08 (100%)	0.34 (100%)	5.3 Days	4 x 10 <sup>-6</sup>	2 x 10 <sup>-6</sup>

\* With Tracerlab MD-5B Scintillation Detector (250 cpm background, 3" lead shielding).  
\*\* With Tracerlab MD-12C Large Area Beta-Gamma G-M Detector (40 cpm background 2" lead shielding).

Sampling 2 to 10 CFM

Performance Continuous operation

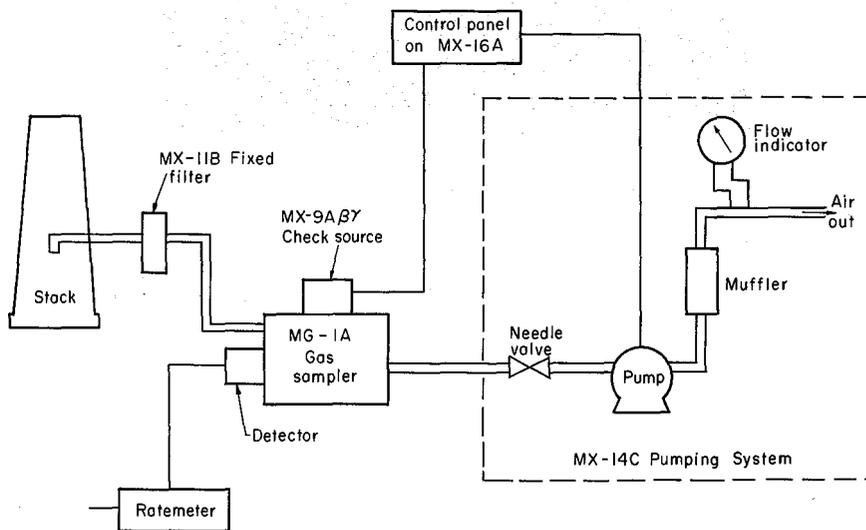


Requirements      Power: 110/220 V ac, 50-60 cps, 1 phase, 900 watts  
                      Size: 82 cm W, 82 cm H, 62 cm D (32.5" W, 32.5" H, 24.5" D)  
                      Weight: About 430 Kg (150 lbs.)

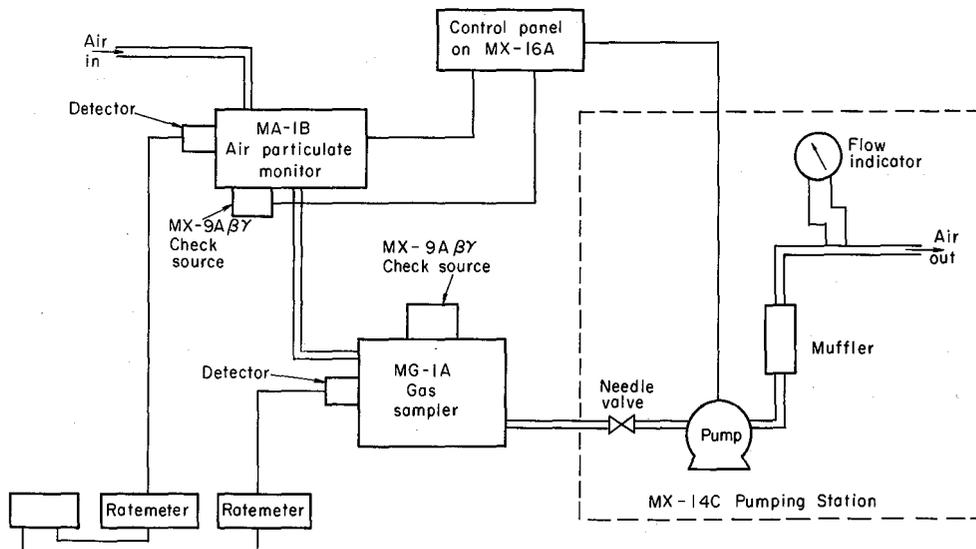
Features            Remote control & Indication. Particulate sampler optional.

Cost                \$5,000

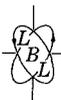
Address            LFE Corporation  
                      Trapelo Division  
                      1601 Trapelo Road  
                      Waltham, Mass. 02154  
                      (617) 890-2000



Schematic Flow Diagram, Stack Monitoring System  
MGP-IA System



Schematic Diagram, MAP-1B/MGP-1A Particulate and  
Gas Monitoring System

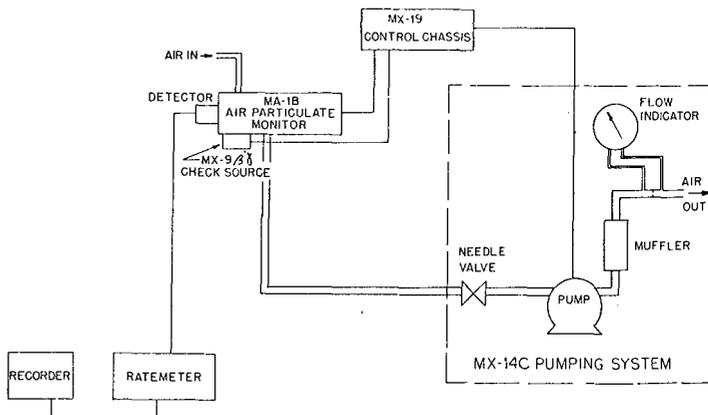
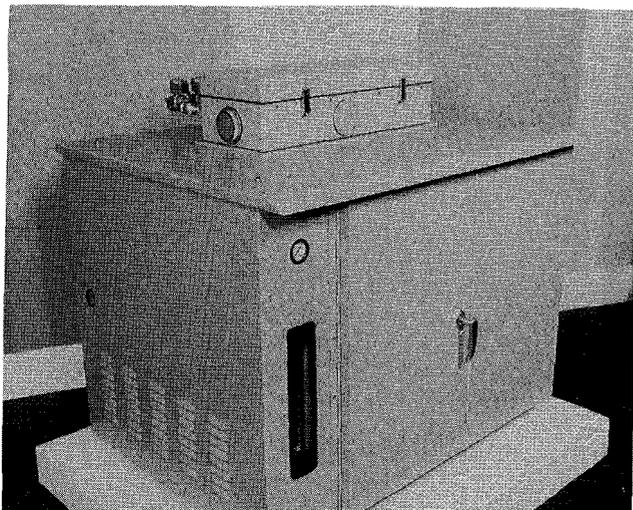


INSTRUMENTATION  
FOR ENVIRONMENTAL  
MONITORING

RAD-NUC  
Particulate  
Trapelo 2  
Mar. 1972

Continuous Particulate Monitor

Trapelo Model MAP-1B



Schematic diagram, MAP-1B.

Class

Laboratory

Principle of Operation

Airborne particulate sampler, moving filter, beta detector, ratemeter, alarms, control panel, enclosure, and pump

Sensitivity and Range

TABLE I  
SENSITIVITIES OF TRACERLAB MA-1A FILTER TAPE TRANSPORT MECHANISM

Detector	Activity	Half-Life	Energy and Type in MeV	Min. Detect. Conc. $\mu\text{c}/\text{cc}$	Back-ground	Net Signal
MD-1B 4 mg/cm <sup>2</sup> Beta-Gamma GM	C <sup>14</sup>	5500y	0.15 $\beta^-$	2.5x10 <sup>-11</sup>	25 cpm	25 cpm
	Ti <sup>304</sup>	3.5y	0.76 $\beta^-$	1.0x10 <sup>-11</sup>		
	Sr <sup>90</sup> -Y <sup>90</sup>	20y	0.54 $\beta^-$ , 2.2 $\beta^-$	4.5x10 <sup>-12</sup>		
	Na <sup>22</sup>	2.6y	0.54 $\beta^+$ , 1.28 $\gamma$	1.0x10 <sup>-11</sup>		
	I <sup>131</sup>	8.1d	0.36 $\gamma$ , 0.61 $\beta^-$	1.0x10 <sup>-11</sup>		
	Co <sup>60</sup>	5.3y	0.31 $\beta^-$ , 1.3 $\gamma$ , 1.1 $\gamma$	8.0x10 <sup>-12</sup>		
	Cs <sup>137</sup>	33y	0.52 $\beta^-$ , 1.1 $\beta^-$	1x10 <sup>-11</sup>		
Pm <sup>147</sup>	2.6y	0.22 $\beta^-$	1.5x10 <sup>-11</sup>			
MD-3B Alpha Scintillation	Po <sup>210</sup>	138d	5.3 $\alpha$	1x10 <sup>-12</sup>	3 cpm	4 cpm
MD-4B Beta Scintillation	C <sup>14</sup>	5500y	0.15 $\beta^-$	1x10 <sup>-9</sup>	20 cpm	20 cpm
	Ti <sup>304</sup>	3.5y	0.76 $\beta^-$	5x10 <sup>-12</sup>		
MD-5B Gamma Scintillation	Co <sup>60</sup>	5.3y	1.1 $\gamma$ , 1.3 $\gamma$	9x10 <sup>-11</sup>	250 cpm	200 cpm
	I <sup>131</sup>	8.1d	0.36 $\gamma$ , 0.61 $\beta^-$	1x10 <sup>-10</sup>		

Measured at air flow rate of 10 scfm, for otherwise uncontaminated filters.

Requirements

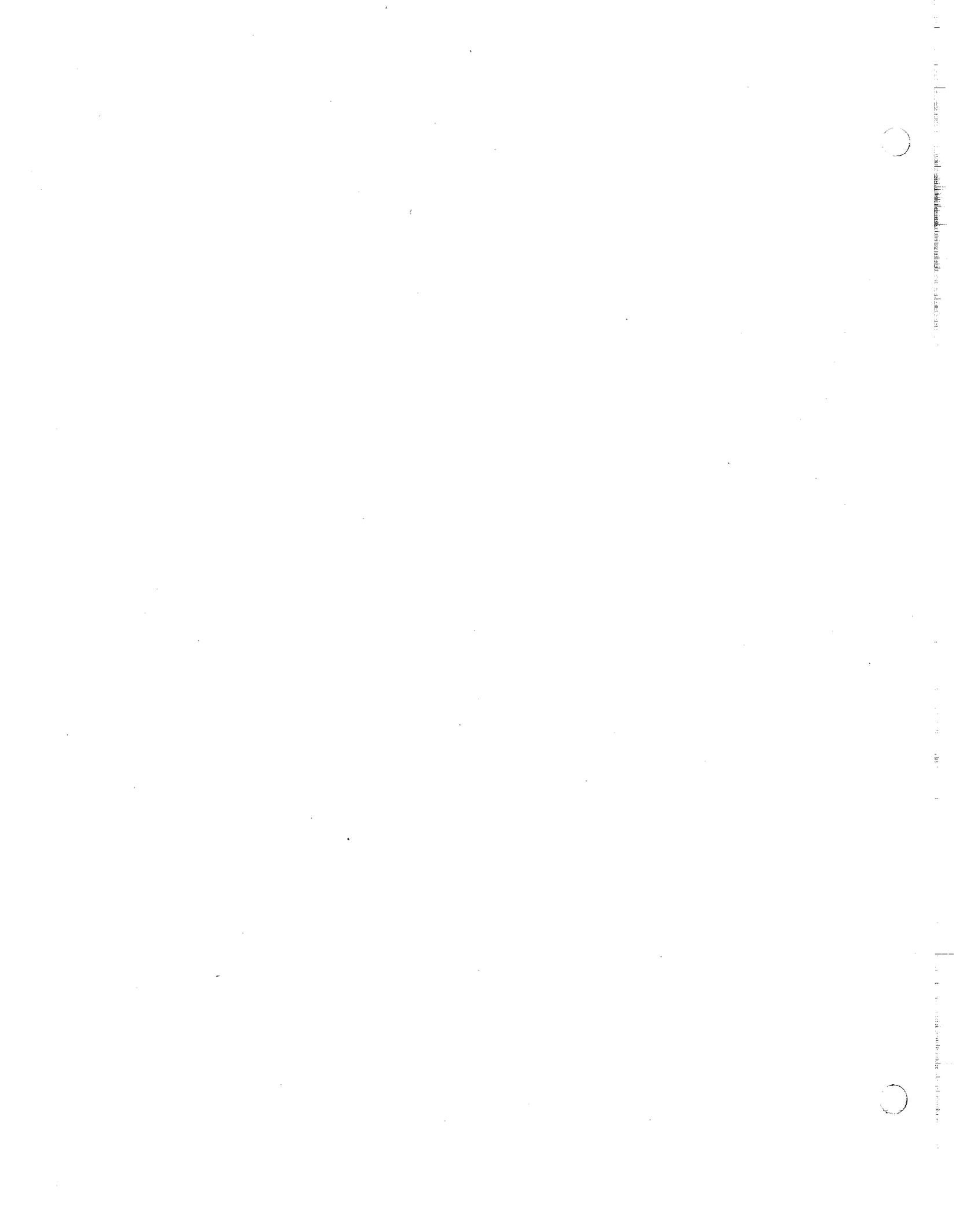
Power: 110/220m 50-60 cps, 1 phase, 1000 watts  
Size: 82 cm W, 101 cm H, 62 cm D (32.5"W, 40" H, 24.5"D)  
Weight: About 270 Kg (600 lbs)

Cost

\$6,000

Address

LFE Corporation  
Trapelo Division  
1601 Trapelo Road  
Waltham, Mass. 02154  
(617) 890-2000



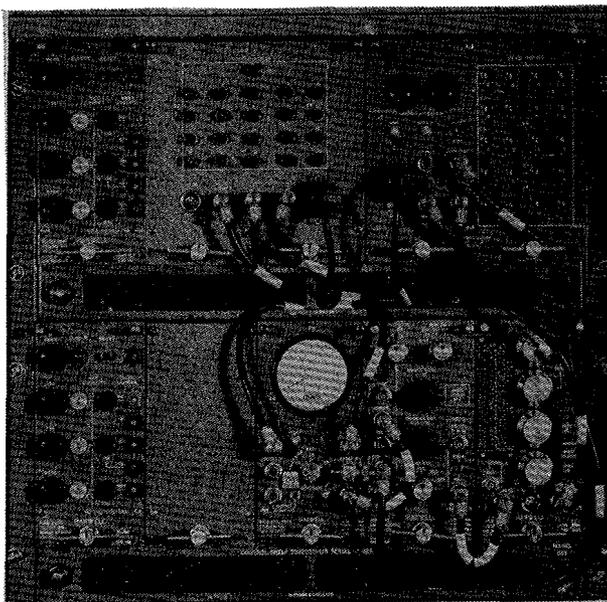
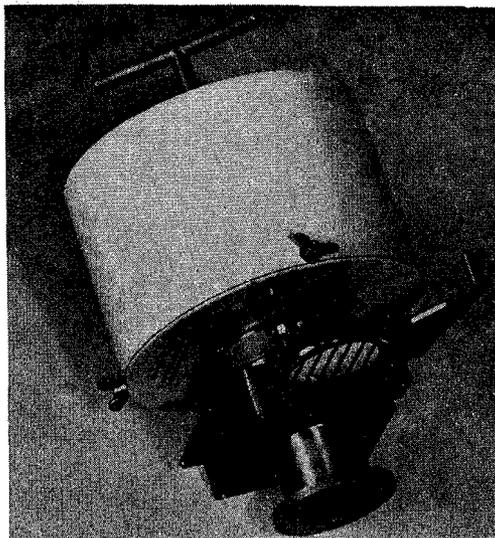


INSTRUMENTATION  
FOR ENVIRONMENTAL  
MONITORING

RAD-NUC  
Plutonium  
UKAEA  
Aug. 1972

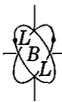
### Plutonium-in-Air Monitor

UKAEA - 3030A



Class	Stationary	
Principle of Operation	Alpha activity deposited on a static filter paper is measured using a silicon detector, amplifier, multiplexer, pulse height selector set to plutonium alphas, ratemeter, array of lamps display, and warning lamp.	
Sensitivity and Range	Energy Range: Activity Range: 1 MPC-hr $\equiv 2 \cdot 10^{-12}$ $\mu\text{Ci}/\text{cm}^3$ for 1 hr Minimum threshold, 0.3 cps $\equiv$ 8 mpc-hr Maximum threshold, 3 cps = 80 mpc-hr	
Sampling	Maximum Number per Installation: 20	
Performance	Accuracy: Temperature: Air Flow: Up to 100 liters/min	
Requirements	Power: Use type 2015 Size: Weight: 3.15 kg (7 lb) for head amplifier sensor kg ( 1b) for electronics	
References	Manufacturer's specifications	
Cost		
Address	Nuclear Enterprises Ltd. Bath Road, Beenham Reading, Berks, ENGLAND Tel.: Woolhampton 2121	Nuclear Enterprises Ltd. 935 Terminal Way San Carlos, CA 94070 (415) 593-1455



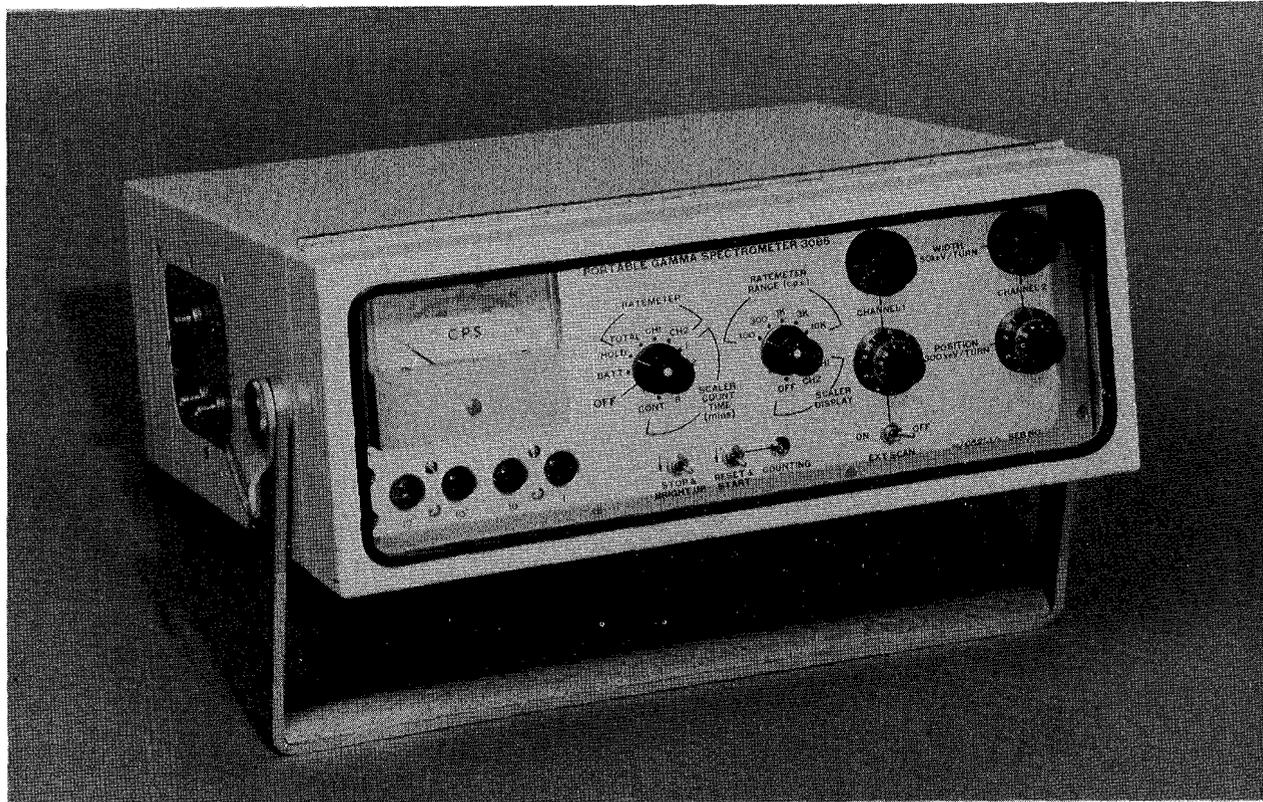


INSTRUMENTATION  
FOR ENVIRONMENTAL  
MONITORING

RAD-NUC  
Uranium  
UKAEA 2  
Oct. 1972

### Uranium Survey Monitor

UKAEA Model 3086



Class	Portable, Field Instrument
Principle of Operation	Two-channel NaI (Tl) scintillation spectrometer, photomultiplier tube, amplifier, 4-decade digital readout and 5-decade meter readout for uranium surveys. Spectrum stabilized using $\text{Am}^{241}$ 59.6 keV reference.
Sensitivity and Range	Energy Range: 0.25 to 3.5 MeV gammas; calibrated controls (300 keV/turn) Channel Width: 0-500 keV (symmetrical expansion); (50 keV/turn) Ranges: 100, 300, 1000, 3000, $10^4$ cps
Performance	Accuracy: Temperature Range: $-10^\circ\text{C}$ to $+60^\circ\text{C}$
Requirements	Power: Eight "D" type cells (R20) with 40 hour life; or external 12V supply Size: 30 cm x 11 cm x 25 cm (12" x 4.5" x 10") Weight: 7 kg
Features	Two separate counting channels; plus internally preset discriminator; waterproof case; 100 mV output for strip-chart recorder. Provision for external control of one channel position, may be used with 1000 meters of standard coaxial cable.
References	Developer's specifications
Cost	On application
Address	Group Commercial Officer Atomic Energy Research Establishment Harwell, Didcot, Berks, ENGLAND Tel.: Abingdon 4141 Telex: 83135 Cable: ATEN, ABINGDON

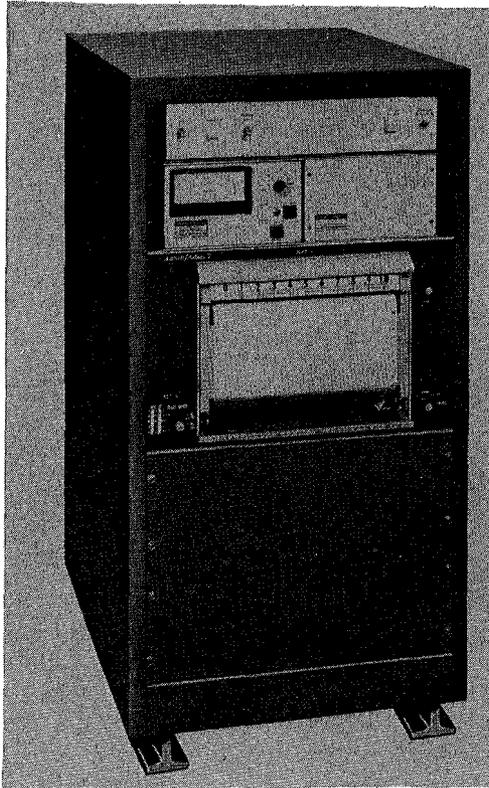




INSTRUMENTATION  
FOR ENVIRONMENTAL  
MONITORING

RAD-NUC  
Iodine  
Victoreen  
Nov. 1971

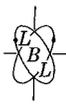
Fixed Filter Air/Iodine  
Monitor-Victoreen Model 840-2



Class	Laboratory and Power Reactor Site Monitoring
Principle of Operation	Air is pumped through a charcoal filter and the iodine particulates are counted.
Sensitivity and Range	From $10^{-12}$ to $10^{-6}$ $\mu\text{c}/\text{cc}$ at 4 cfm Model 843-2 Beta scintillation detector Model 843-3 Gamma scintillation detector Model 843-4 G-M detector
Sampling	Air pump system, 4 cfm air flow, collection eff: 95%
Performance	Meter accuracy: $\pm 2\%$ fullscale Time constant: $10^3$ c/m - 1 minute $10^6$ c/m - 0.2 sec. Temperature range: $0^\circ\text{C}$ - $50^\circ\text{C}$
Requirements	Power: 110 Vac, 10 amps, 60 Hz, single phase (others available on request) Size: 114.3 cm H, 61 cm W, 72.5 cm D (45" H, 24" W, 28.5" D) Weight: 227 Kg (500 lbs)
Features	Single channel analyzer capabilities for iodine monitoring, and airborne particulate monitoring or both simultaneously; alarms for alert, high radiation flow rate, and failure, log and linear scales.
Reference	Manufacturer's specifications



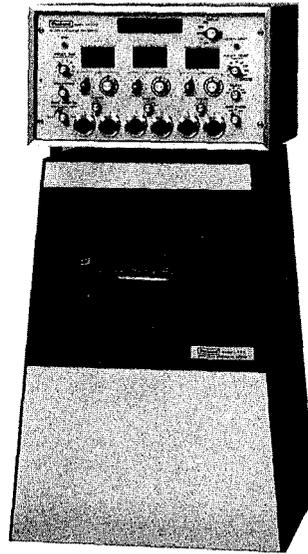
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INSTRUMENTATION  
FOR ENVIRONMENTAL  
MONITORING

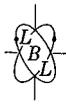
RAD-SYS  
Packard  
Nov. 1972

Scintillation Detector  
Packard Model 446



Class	Laboratory
Principle of Operation	Cylindrical sample chamber is surrounded by a 4" annulus of liquid scintillator and six 3" dia. photomultiplier tubes. The tubes are magnetically shielded and have individual controls for focus and gain. A lead shield surrounds the entire detector to reduce background. A handhold is provided to insure reproducible geometry for forearm counting.
Sensitivity and Range	Sensitivity: Range:
Sampling	Aluminum cylinder 4.25" D, 8" L.
Performance	Accuracy: Temperature:
Requirements	Power: Size: 100 cm H, 95 cm W, 79 cm D (39.5", 37.3", 31") Weight: 771 kg (about 1700 lb)
Features	
Cost	Model 446 \$7,875. Less shielding and cabinet \$4,830.
Address	Packard Instrument Co., Inc. 2200 Warrenville Road Downers Grove, IL 60515 (312) 696-6000

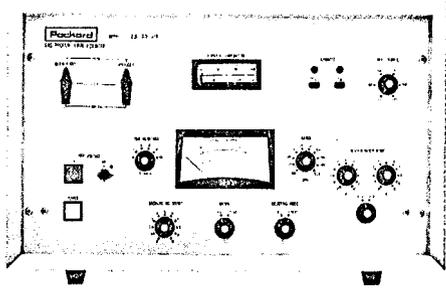




INSTRUMENTATION  
FOR ENVIRONMENTAL  
MONITORING

RAD-SYS  
Tritium, Carbon-14  
Packard  
Nov. 1972

Gas Proportional Counter  
Packard Model 894

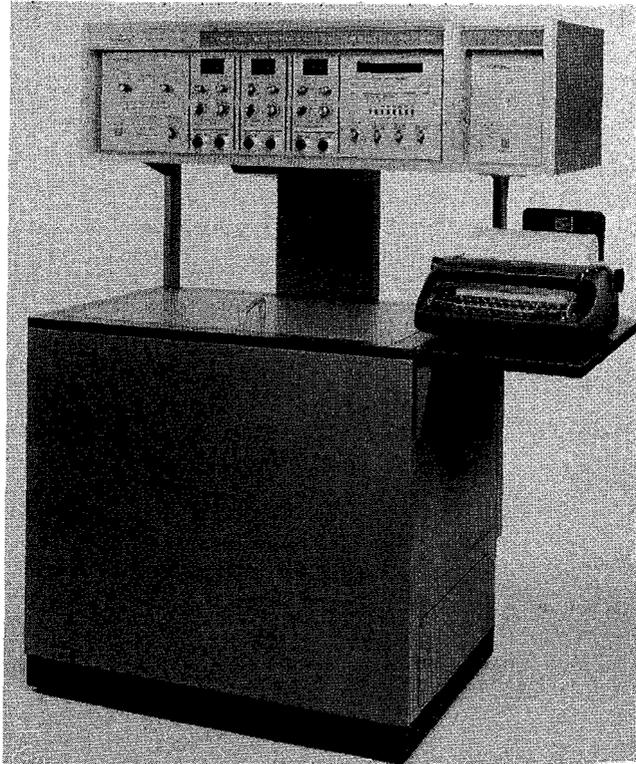


Class	Laboratory
Principle of Operation	Detects tritium or carbon-14 as they elute from a gas chromatographic column using a flow-through ion-multiplication detector. A copper oxide furnace oxidizes all organic compounds to carbon dioxide and water to eliminate sample condensation in the tube.
Sensitivity and Range	Sensitivity: Betas Range: 10 linear ranges, 200 cpm max. Five-decade log scale Counting Efficiency: 58% Tritium 80% Carbon-14
Sampling	Continuous
Performance	Accuracy: Temperature: Operates at room temperature Inlet heating to 300°C Furnace heating to 800°C Time Constant: 2 sec
Requirements	Power: 120-220 V ac (switch) 50 or 60 Hz, 720 watts Size: 35.6 cm H, 54.6 cm W, 61 cm D (14", 21.5", 24") Weight: 90 kg (200 lb)
Features	Use with Model 7401 Gas Chromatograph Background about 30 cpm (variable offset) Variable upper and lower discriminators Simultaneous analog and scaler output Right or left-hand capability to couple with any G.C.
Cost	
Address	Packard Instrument Co., Inc. 2200 Warrenville Road Downers Grove, IL 60515 (312) 696-6000





Liquid Scintillation Spectrometers  
Packard Model 3390 and Others



Class	Laboratory
Principle of Operation	Liquid scintillation spectrometer system. Matched bialkali photomultiplier tubes in coincidence mode detect light; pulse height analysis electronics provides spectrum information. Background subtraction done automatically. Automatic calibration with external source arrangement.
Sensitivity and Range	Sensitivity: Unknown Energy Range: Can detect any beta, alpha, electron capture, X-ray, or weak gamma ray isotopes.
Sampling	300 sample capacity with up to 10 repetitive counts per sample or up to 10 preset number of complete cycles for counting all samples. Also has automatic and group counting mode that allows empty samples with a group.
Performance	Accuracy: Unknown Temperature: Controlled temperature (maintained $\pm 0.5^{\circ}\text{C}$ over range $-5^{\circ}\text{C}$ to $+15^{\circ}\text{C}$ ) or ambient temperature
Requirements	Power: 100-130 or 200-260 V ac, 50-60 Hz, 1000 VA (max) Size: Height 62-1/4 to 67-1/4", adjustable (158 to 171 cm); width 41-1/2 to 57" (depending on readout device); depth 31" (79 cm) Weight: 1000 lb (450 kg)



Features

1. Three separate pulse-height analysis channels.
2. Single-, double-, triple-labeled capability.
3. Independent controls for normalizing each photomultiplier tube.
4. Preset count, background subtraction, low-level reject for each sample channel.
5. Automatic external standardization with radium-226 and americium-241 sources.
6. On-line computation and display of net count rate and standard deviation.
7. Gamma pulse-height capability.
8. Automatic sample oxidizers: Models 300, 305, and 306 (optional).
9. Flow measurement capability using flow-cell and adapter.
10. Model 544 Absolute Activity Analyzer automatically adjusts to optimize counting rate and background conditions and computes quench-corrected dpm for single- or double-labeled samples (optional).
11. Model 3050 extra shielding package for low-level counting (optional).
12. Model 3090 Automatic Channel Selection (optional).

Other Models

- Model 3385: Identical to Model 3390 except not available with Model 544 Absolute Activity Analyzer.
- Model 3375: Identical to 3380 except not available with 544 Absolute Activity Analyzer.
- Model 3330: Similar to 3390 except does not have on-line computation feature, one-source instead of two-source calibration system, some other differences.
- Models 2101, 2111, 2211, and 2311: Have fewer features; similar to Model 3320.
- Model 3380: Identical to Model 3390 except 200-sample capacity and not available with Model 3090 Automatic Channel Selection.
- Model 3320: Identical to Model 3330 except 200-sample capacity.
- Models 2425 and 2450: Similar to Model 3390 except Servo-Tray sample changer. Model 2425 has 150sample capacity. Model 2450 has 450-sample capacity.

References

Manufacturer's specifications

Cost	Model 3390	\$17,095.	Model 3330	\$15,595.
	544	4675.	3305	8665.
	513	1435.	2450	14,500.
	545	1670.	2425	10,615.
	542	525.	2311	9565.
	3385	16,095.	2211	7990.
	3380	16,595.	2111	6205.
	3375	15,595.	2101	5200.

Address

Packard Instrument Company Inc.  
2200 Warrenville Road  
Downers Grove, IL 60515  
(312) 969-6000

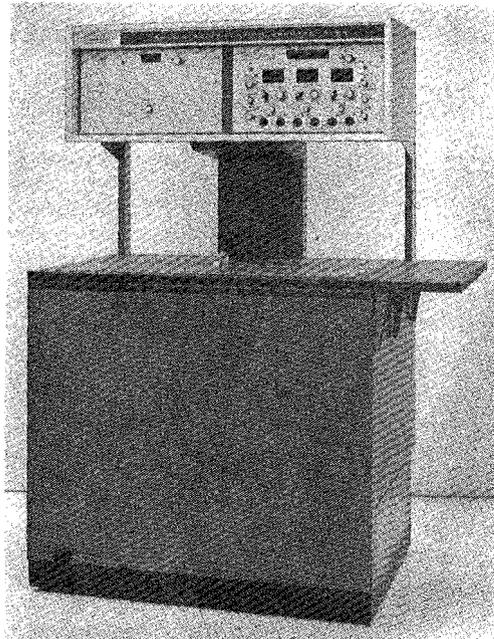
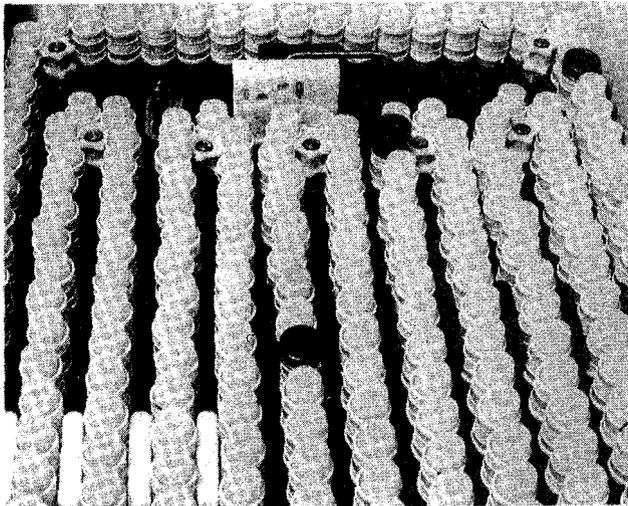


INSTRUMENTATION  
FOR ENVIRONMENTAL  
MONITORING

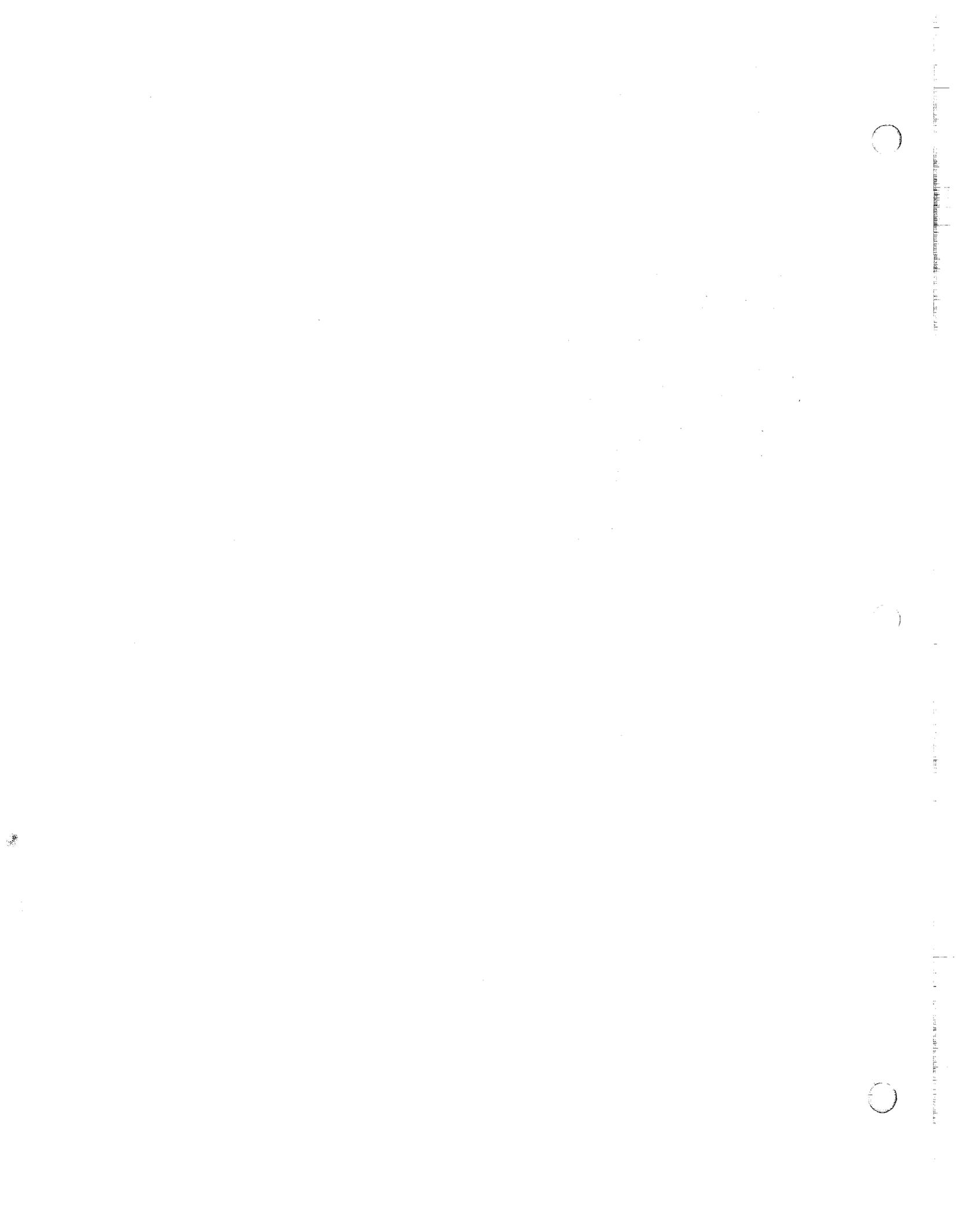
RAD-SYS  
Spectrometer  
Packard  
Jan. 1973

Auto-Gamma Spectrometer

Packard 5000 Series



Class	Laboratory			
Principle of Operation	Automatic 300 sample controlled temperature gamma spectrometer. Two diametrically opposed NaI(Tl) scintillation crystals, photomultipliers. Optional tandem liquid scintillation system for beta counting; Ge(Li) for higher resolution $\gamma$ spectroscopy; and Si (Li) for higher resolution X-rays. Automatic background subtraction, low-level count rejection. Three independent pulse-height analysis channels with three scalers.			
Sensitivity and Range	Sensitivity: Unknown Energy Range: Unknown			
Sampling	300 sample capacity changer with repeat, group, and automatic counting modes. Solid or liquid samples.			
Performance	Accuracy: Unknown Temperature: Maintained within $\pm 0.5^\circ\text{C}$ or ambient operation			
Requirements	Power: 110 or 220 V ac, 50 or 60 Hz, 10 A (max) Size: 170 cm H, 105 cm W, 79 cm D (67", 41.5", 31") Weight: 680 kg (1500 lb)			
Features	<ol style="list-style-type: none"> <li>1. Preset time/preset count modes up to nine isotopes.</li> <li>2. Coarse and continuous calibrated fine gain adjustment.</li> <li>3. Digital printer output, IBM Selectric or Teletype option.</li> </ol>			
References	Manufacturer's specifications			
Cost	Model 5320	\$12,865	Beta Model 5120-83	\$17,900
	5220	12,365	5220-83	18,900
	5120	10,865	5320-83	19,400
	5385	15,665		
Address	Packard Instrument Company Inc. 2200 Warrenville Road Downers Grove, IL 60515 (312) 696-6000			



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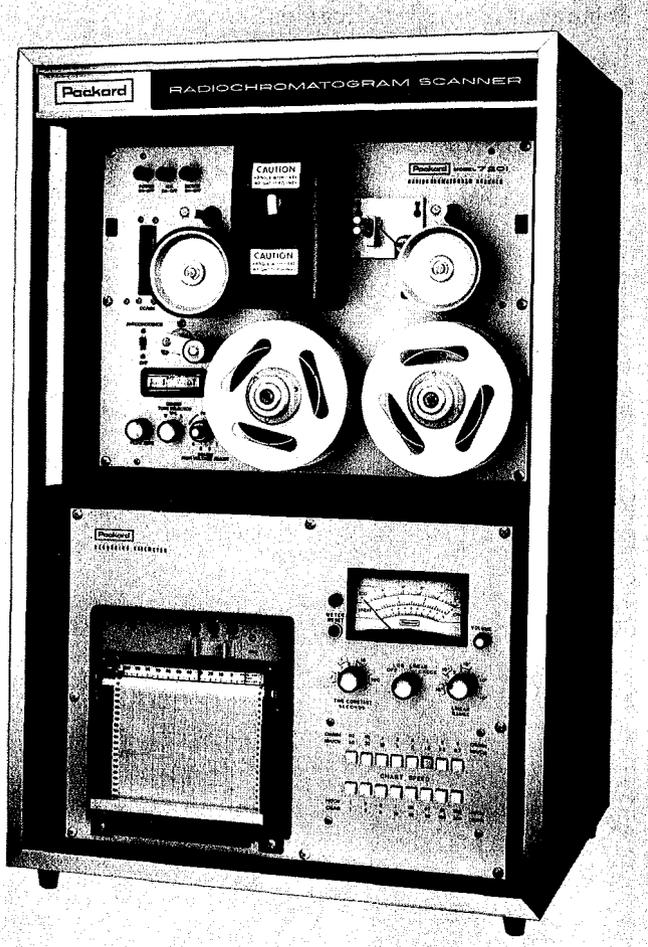


INSTRUMENTATION  
FOR ENVIRONMENTAL  
MONITORING

RAD-SYS  
Beta Scanner  
Packard  
Nov. 1972

### Radiochromatogram Scanner

Packard Model 7201



Class

Laboratory

Principle  
of Operation

Radioactivity is measured by two windowless G-M tubes arranged in 4  $\pi$  configuration to scan both sides of the strip. Resolution between adjacent spots is controlled by two adjustable collimators.

Sensitivity  
and Range

Sensitivity: Beta emitting nuclides  
Range: Ratemeter has 9 linear count ranges and log scale  $10^2$ ,  $10^3$ ,  $10^4$ ,  $10^5$ ,  $10^6$ , and  $3 \times 10^2$ ,  $10^3$ ,  $10^4$ ,  $10^5$ .

Sampling

Accepts chromatograms from 0.5" to 2" wide, lengths up to 150 feet, all types of papers up to Whatman No. 3MM

Performance

Accuracy:  
Temperature:

Time Constants: From 0.1 to 300 sec (8 settings)  
Scanning Speed: From 0.02 cm/min to 200 cm/hr (16 ea.)



INSTRUMENTATION  
FOR ENVIRONMENTAL  
MONITORING

RAD-SYS  
Beta Scanner  
Packard  
Page 2 Nov. 1972

Requirements

Power:  
Size: 88 cm H, 55 cm W, 48 cm D (35", 22.5", 19")  
Weight: 98 kg (215 lb)

References

Manufacturer's specifications

Cost

Model 7201 \$4,115  
With disc integrator \$4,950

Address

Packard Instrument Co., Inc.  
2200 Warrenville Road  
Downers Grove, IL 60515  
(312) 696-6000