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FOR PERIOD ENDING JANUARY 31, 1954

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HEALTH PHYSICS DIVISION
SEMIANNUAL PROGRESS REPORT

THEORETICAL PHYSICS

J. Neufeld
R. H. Ritchie W. S. Snyder

DISORDERING OF SOLIDS BY HEAVY CORPUSCULAR RADIATION

The number of vacant lattice sites in a monatomic substance exposed to heavy corpuscular radiation has been calculated for the range $E > \gamma$, where E is the energy of a recoil atom produced by the radiation and $\gamma = 25A \times 10^3$ ev (A is the mass number of the atoms of the solid).

SPECTRAL DEGRADATION OF NEUTRONS

The change in the energy spectrum of neutrons that have penetrated a slowing-down medium has been calculated. The calculated spectrum has been compared with experimental data and there is fair agreement. This work has been reported.[1]

EFFECT OF RECOMBINATION UPON RADIATION DAMAGE BY X AND GAMMA IRRADIATION

The greater part of damage to living tissue subjected to x or gamma irradiation is thought to be due to indirect chemical effects of ionization products, as opposed to direct hits of the primary ionizing particles upon a vulnerable portion of the cell.

Volume recombination of H and OH radicals which are formed in the decomposition of water by the ionizing particles and which are believed to be responsible for chemical effects[2] may reduce, to some extent, the biological damage done at high dose rates.

To estimate the magnitude of this effect, the equation describing n (the density of radicals of either sign) is written as follows:

$$\alpha n^2 + \beta n - q = 0 \quad .$$

Here α is the number of radicals disappearing through volume recombination per unit time for unit radical density, β is the number disappearing through chemical action per unit time for unit density, and q is the number of radicals generated per unit volume per second by the ionizing radiation. The variation of n with time is neglected, since chemical combination times are of the order of 10^{-9} sec, which is short compared with any realistic irradiation time. By solving this equation for n and determining the source density q at which the recombination effect accounts for the disappearance of half the radicals, a generated radical density of $q = 2\beta^2/\alpha$ is found. Lea[3] estimates that β may be as high as 10^9 sec^{-1} in the cell fluid and assumes a value for α of 4×10^{-10} cm^3/sec. If it is assumed that about one radical of each kind is produced per ion pair and that about 2×10^{12} ion pairs/cm^3 are equivalent to 1 r, then the dose rate would be 2.5×10^{15} r/sec. The extremely high value of the dose rate at which the chemical effects are reduced by one-half through volume recombination shows that this effect will be negligible in any realistic irradiation of living cells.

[1] G. S. Hurst, J. A. Harter, P. N. Hensley, W. A. Mills, and R. H. Ritchie, ORNL-1671 (classified), to be published.

[2] D. E. Lea, *Actions of Radiations on Living Cells*, McMillan, 1947.

[3] *Ibid.*, pp. 52, 60.

PHYSICS OF NUCLEAR RADIATION

G. S. Hurst, W. A. Mills, C. E. Melton[1]
T. E. Bortner, M. Slater, L. W. Cochran[2]
W. G. Stone, J. W. Cure, III[1], R. H. Ritchie[3]

NEUTRON-SCATTERING EXPERIMENT

The purpose of the neutron-scattering experiment in the determination of the neutron dose at a point in air was threefold: (1) to determine the variation in dose at different distances (D) between the source and point, and to check the reliability of the inverse-square law for estimating the correct dose, (2) to determine the variations in dose with changes in height (H) of the source and detector (point) above a large block of concrete, 3 to 4 in. thick, and the effect of neutrons scattered from the concrete on the dose, and (3) to determine whether or not the detector can be shielded by a paraffin cone so that the dose at the point is due only to scattered neutrons.

The experimental arrangement is shown in Fig. 1. The neutron source is Po^{210}-B, and the counter is a proportional counter, ethylene-polyethylene type,[4] with a pulse integrator[5] for measuring the dose.

The differences in the neutron dose as a function of height above concrete at various distances between the source and the detector are shown graphically in Figs. 2a through 2f. These differences are important since they enable an experimenter to determine how much neutron scattering can be expected from concrete walls, floors, etc. For distances up to 1 meter between the counter and source, the dose due to scattered neutrons is less than 2% of the dose due to the primary beam of neutrons if the counter and source are at least 1.5 meters away from the concrete wall or floor. Also, it should be noted that if the distance between the source and counter is greater than 1.5 meters, there is a height above the concrete at which the dose rate is at a maximum. This is attributable to the scattering of the neutrons in the concrete at some distance below the surface. The neutrons are effectively soaked up by the concrete, because, when they are scattered at some distance below the surface, they must go through a considerable thickness of concrete to reach the detector. Figure 3 is a graph of the dose rates as read from the flat portions of the curves of Figs. 2a through 2f (i.e., for heights above the concrete where neutron scattering from the concrete is negligible and for the corresponding distances between source and detector). It is noted that the inverse-square law gives the correct dose rate up to a distance of about 1.5 meters between the counter and source. At distances greater than 1.5 meters the dose rate is appreciably greater than that predicted by the inverse-square law. It is believed that this increase is due to air scattering. It was determined that a 44-cm-long paraffin frustum, having diameters of approximately 3 and 10 cm, placed between the source and counter cut off the primary beam of neutrons from the detector, thereby reducing the dose due to the primary beam of neutrons by a factor greater than 1000. Comparison of the shapes of the curves with and without the paraffin (Figs. 2c and 2d compared with Figs. 4a and 4b) indicates that the dose at the detector with the paraffin in front of the source is due only to scattered neutrons.

Complete details of this experiment will be reported by J. W. Cure, III in a thesis for the degree of Master of Science in physics at Vanderbilt University.

ELECTRON-DRIFT VELOCITY

In the experiment for measuring the attachment coefficient of electrons, the drift velocity of the electrons under specific conditions has been determined. The apparatus described by Stevenson[6]

[1] AEC Fellow in Radiological Physics (Vanderbilt University, cooperating university).

[2] Consultant from Department of Physics, University of Kentucky.

[3] Theoretical Physics Group, Health Physics Division.

[4] G. S. Hurst and R. H. Ritchie, *Radiology* 60, 864 (1953).

[5] F. M. Glass and G. S. Hurst, *Rev. Sci. Instr.* 23, 67 (1952).

[6] A. Stevenson, *Rev. Sci. Instr.* 23, 93 (1952).

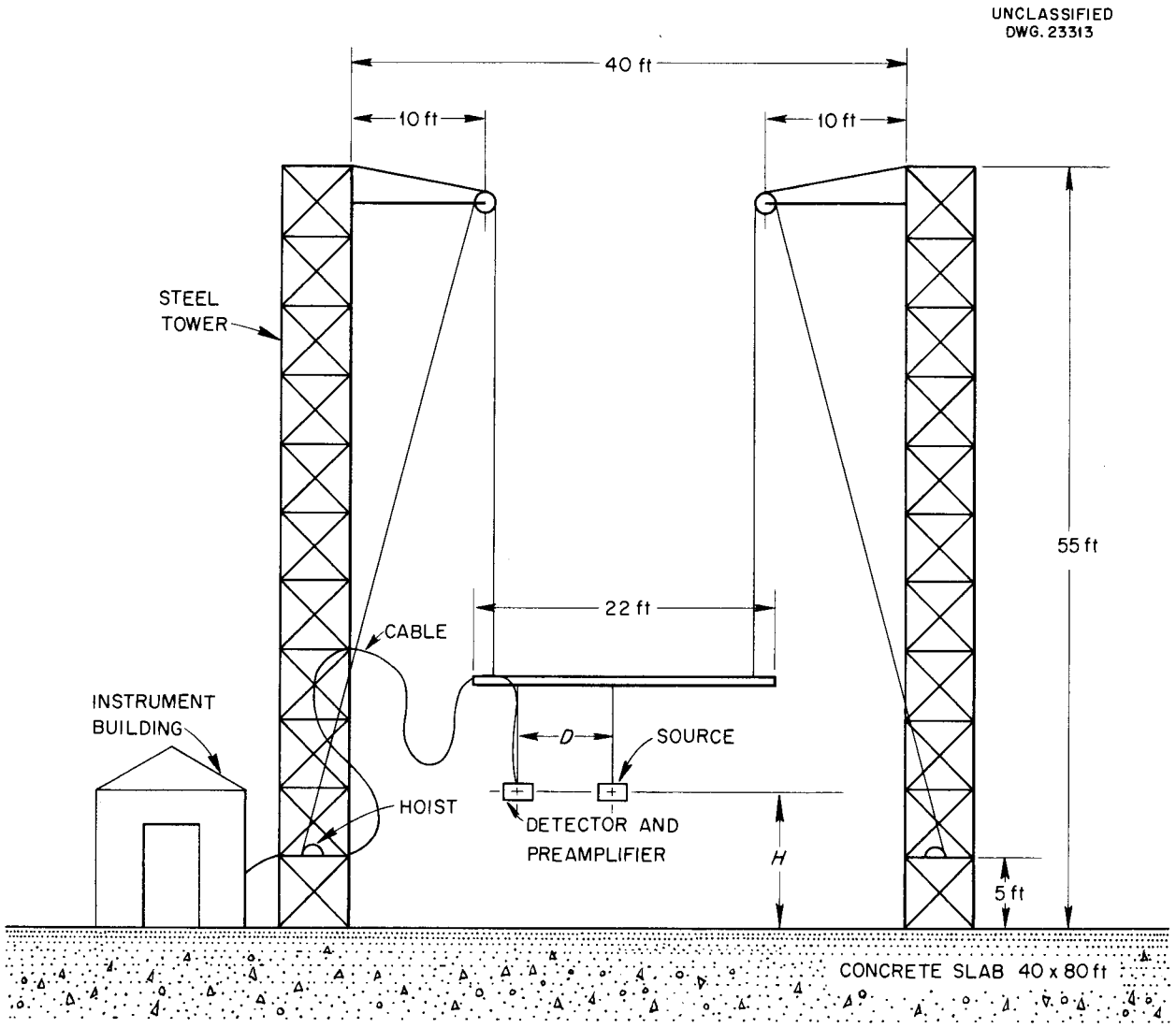

Fig. 1. Experimental Arrangement for Scattering of Fast Neutrons.

was used, and the data obtained (Fig. 5) for pure N_2 are in good agreement with those reported by Klema and Allen.[7] A mixture of 90% argon and 10% methane showed a rapid rise to a maximum drift velocity of 6.6 cm/μsec at a field strength of 0.1 volt/cm-mm of Hg, followed by a gradual decrease to 3.1 cm/μsec at a field strength of 0.8 volt/cm-mm of Hg, after which the value remained constant to the maximum measured field strength of 2.0 volts/cm-mm of Hg. These results for the argon-methane mixture have an important practical implication, since such mixtures are commonly used in proportional counters and ionization chambers. The data indicate that an optimum voltage can be chosen so as to give a minimum electron-collection time. Further, the results are of interest since the maximum drift velocity in the mixture is approximately four times that in pure argon.

[7] E. D. Klema and J. S. Allen, *Phys. Rev.* 77, 661 (1950).

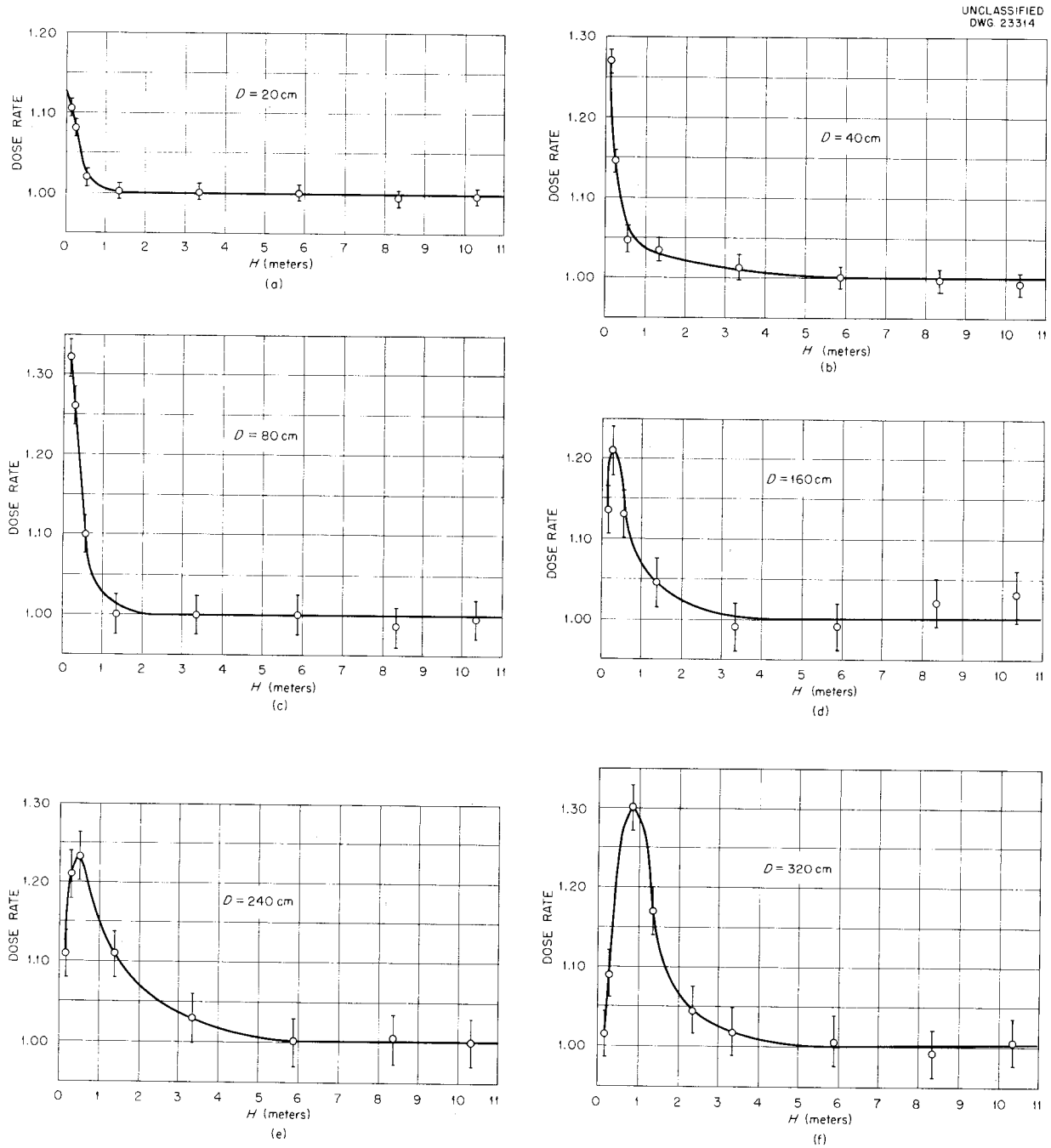

Fig. 2. Neutron Dose as a Function of Height Above Concrete for Various Distances Between Source and Detector.

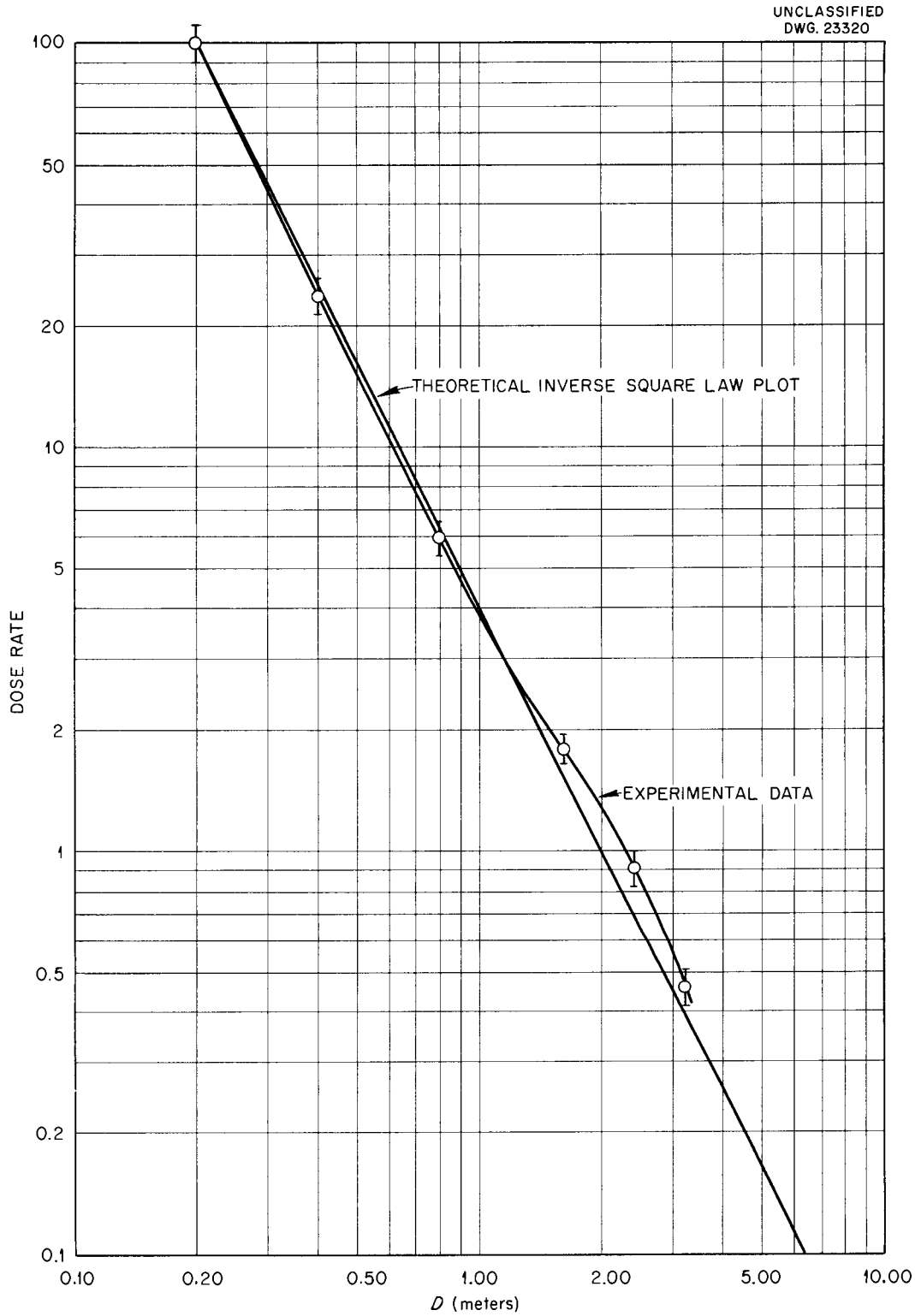

Fig. 3. Comparison of the Measured Dose and the Dose as Calculated from the Inverse-Square Law.

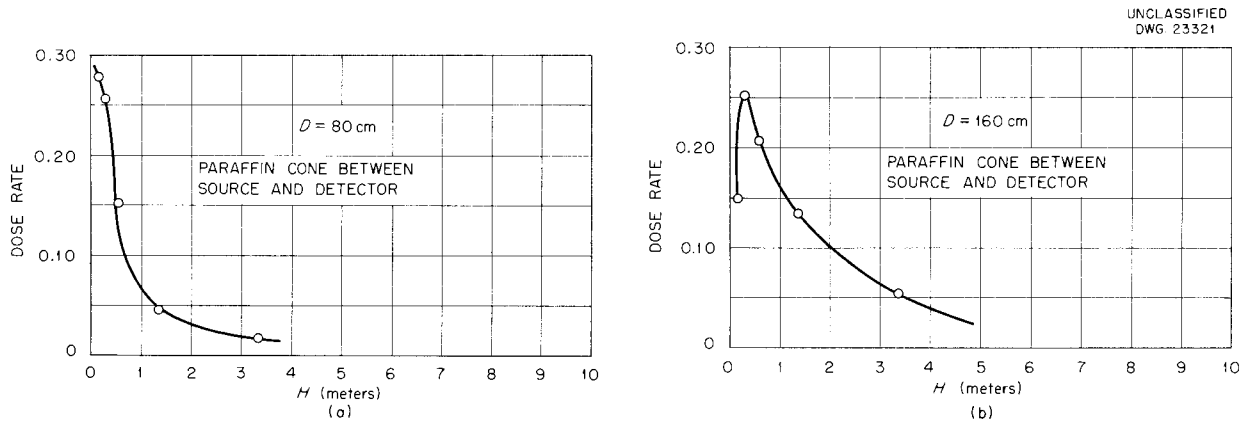

Fig. 4. Dose of Scattered Neutrons as a Function of Height Above Concrete for Two Distances Between Source and Detector.

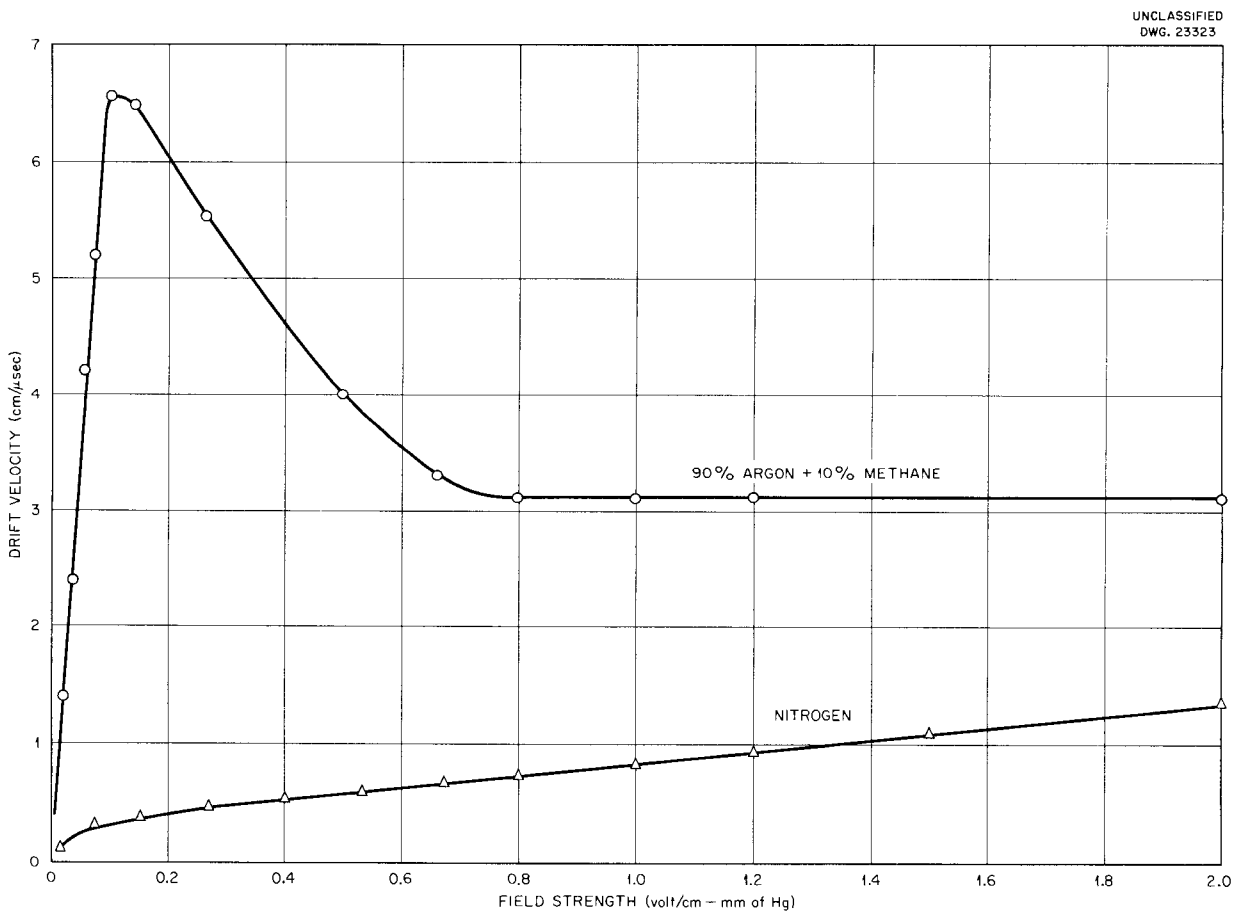

Fig. 5. Electron Drift vs. Field Strength.

IONIZATION OF MIXTURES OF GASES BY PLUTONIUM ALPHA PARTICLES

Measurements of the ionization of gas mixtures by alpha particles from Pu^{239} has continued, and two types of mixtures are of interest. The first class of gas mixtures is one for which the W (average electron volts dissipated by the alpha particle per ion pair produced) changes gradually as the proportions of the two gas components are changed. It was found that for this case the W for a gas mixture (W_m) may be written in terms of the values of W of the components W_1 and W_2 having pressures P_1 and P_2, as follows:

$$\frac{1}{W_m} = \left(\frac{1}{W_1} - \frac{1}{W_2}\right) P + \frac{1}{W_2}$$

where $P = P_1/(P_1 + aP_2)$ and a is constant for a given two components. For the following mixtures, a has the values shown in Table 1.

A complete paper on this work has been accepted for publication in the *Physical Review* and is scheduled to appear in the April 15, 1954 issue.

For the second class of gas mixtures, the W value shows a minimum when a small percentage of the second gas is added to argon. This is true of gases having an ionization potential less than 14.5 volts. The minimum value of W for mixtures of argon and these gases is given in Fig. 6, where W is plotted as a function of the ionization potential of the gases. The percentage of the gas which gives the minimum W value is indicated in parentheses below the point. No amount of H_2 or N_2 (the ionization potentials are above 14.5 volts) added to argon would depress the W value below that of pure argon, while gases having ionization potentials below 14.5 show a minimum in the value of W below that of pure argon for the mixtures. An attempt is being made to explain the results by considering the effect of a radiation transition state in argon (between 14.5 and 15.5 volts) which has a long lifetime; hence, when the impurity is added, there is a good chance of many collisions between the excited argon atoms and the impurity. Some of these collisions would result in a transfer of the energy of the excited state to ionization of the impurity atoms; hence the increased ionization would be explained.

TABLE 1. VALUES OF CONSTANT a FOR VARIOUS MIXTURES

MIXTURE	a
N_2-H_2	0.28
N_2-A	0.53
N_2-O_2	1.06
He-A	0.75
He-H_2	3.55
He-N_2	8.47
He-CH_4	0.68
A-H_2	0.56
H_2-CH_4	4.03

APPLIED DOSIMETRY

Measurements of the fast-neutron dose were made at two biological exposure facilities. One facility is at the CP3′ reactor, Argonne National Laboratory, and is used by H. H. Vogel, Jr. and associates for the irradiation of mice and other small animals. The other facility is the 86-in. cyclotron located at Y-12 and used by the Biology Division, Oak Ridge National Laboratory.

At the ANL facility, the neutron dose was measured with a proportional counter which measures the dose directly. Details of the counter have been reported by Hurst and Ritchie.[8] The source of fast neutrons at the facility is a 1-in.-thick natural-uranium plate irradiated by thermal neutrons from the reactor, giving a fission spectrum which is modified by 3 in. of lead. The dose was measured at different locations in the animal exposure cage for various operating levels of the reactor. The results agreed within 14% with the data previously obtained by Vogel using a Victoreen condenser chamber calibrated with the Rossi tissue-equivalent chamber. The close agreement may be explained by the facts that (1) the gamma contamination in the facility is low and (2) the Victoreen was used in the location where it was calibrated with the tissue chamber.

To measure the dose at the Y-12 facility, two methods were used – a proportional counter and fission threshold detectors. Not only was the dose measurement at the cyclotron of interest to the

[8] G. S. Hurst and R. H. Ritchie, *Radiology* 60, 864-868 (1953).

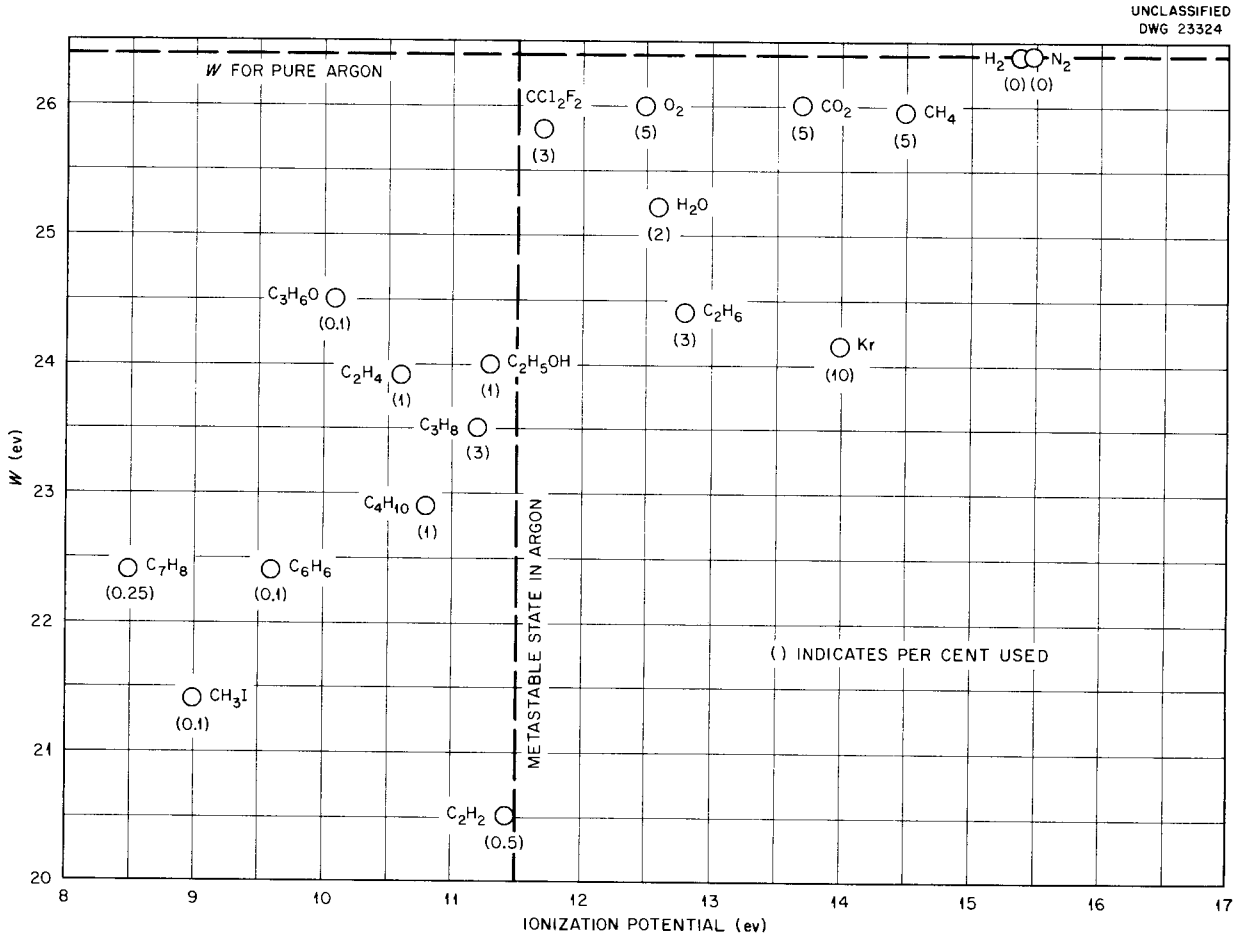

Fig. 6. *W* Values vs. Ionization Potentials of Gases Used as Impurities in Argon.

Biology Division but also the cyclotron afforded an opportunity to compare the two methods of dosimetry. Since the proportional counter is essentially a detector for low neutron intensities and the threshold detectors are usable only for high intensities, a comparison could be made by varying the beam current of the cyclotron; however, even at the very lowest beam current, the neutron intensity was so great that a new and very insensitive proportional counter had to be designed. Dose rate measurements with the two methods agreed within 20%. The difference may be due to the fact that in using the threshold detectors the dose is calculated by means of first-collision tissue-dose curves, whereas the proportional counter measures dose directly when calibrated with a neutron source, Po-B, the strength of which is known. Therefore the proportional counter determination is believed to be the more accurate of the two methods. Use of the fission threshold detectors does give valuable information on the fast-neutron spectrum, and this method of dosimetry is very satisfactory for field use at atomic tests.

IONIZATION PRODUCED BY RECOIL ATOMS IN METHANE

The value of W, the average energy required to produce an ion pair, has been measured in methane for the recoil atoms resulting from the alpha decay of thorium C and thorium C′.

The number of ion pairs produced by a recoil particle is of the order of 10^3. To measure such a small number with reasonable accuracy, a proportional counter which gives considerable gas

amplification within the counter itself is used. To calibrate the gain of such a counter, a source which will produce a determinable number of ions is placed within the counter.

The experimental arrangement similar to that used by Madsen[9] is shown in Fig. 7. A long proportional counter, with an active length sharply defined by field tubes, has at one end a plutonium source used for calibration; and it is so well collimated that the alphas traverse the entire length. Each alpha produces a pulse of measurable size. By reference to the tables of Hirshfelder and Magee,[10] the number of ion pairs corresponding to the size of the pulse can be calculated if the energy of the alphas from plutonium, the measured pressure of the methane in the counter, and the value of W for alphas in methane are known. The dimensions of the counter and the pressures used are such that the calibrating pulses are from 10 to 70 times as large as those produced by the recoil atoms to be studied.

The source of recoil atoms is Pb^{212} (ThB) from natural thorium nitrate. The ThB is deposited on a mica window (thickness 1 mg/cm^2) which is sealed into the wall of the long counter at the midpoint. The surface of the window facing the long counter is coated with a very thin electrically conducting layer of silver, over which is deposited the ThB. The other face is covered with an opaque coating of graphite, over which is placed a brass plate $\frac{3}{8}$ in. thick with a collimating hole $\frac{1}{4}$ in. in diameter. This side of the window forms one end of an alpha counter. All recoil atoms from the ThB are stopped in the window, but the alphas penetrate easily. The alphas which enter the alpha counter come from disintegrations so oriented that the associated recoil atoms expend all their energy within the volume of the long proportional counter.

The long proportional counter is connected to a preamplifier and A-1 amplifier whose low output terminal is connected to a single-channel analyzer (Fig. 8). The output of this single-channel analyzer is fed to one input of a coincidence circuit. The alpha counter is connected to a preamplifier and an A-1 amplifier. The discriminator output of the A-1 is lengthened to a pulse about 4 μsec long so as to overlap the signal from the long counter

[9] B. S. Madsen, *Kgl. Danske Videnskab. Selskab, Mat. fys. Medd.* 23, No. 8 (1945).

[10] J. O. Hirshfelder and J. L. Magee, *Phys. Rev.* 73, 207 (1948).

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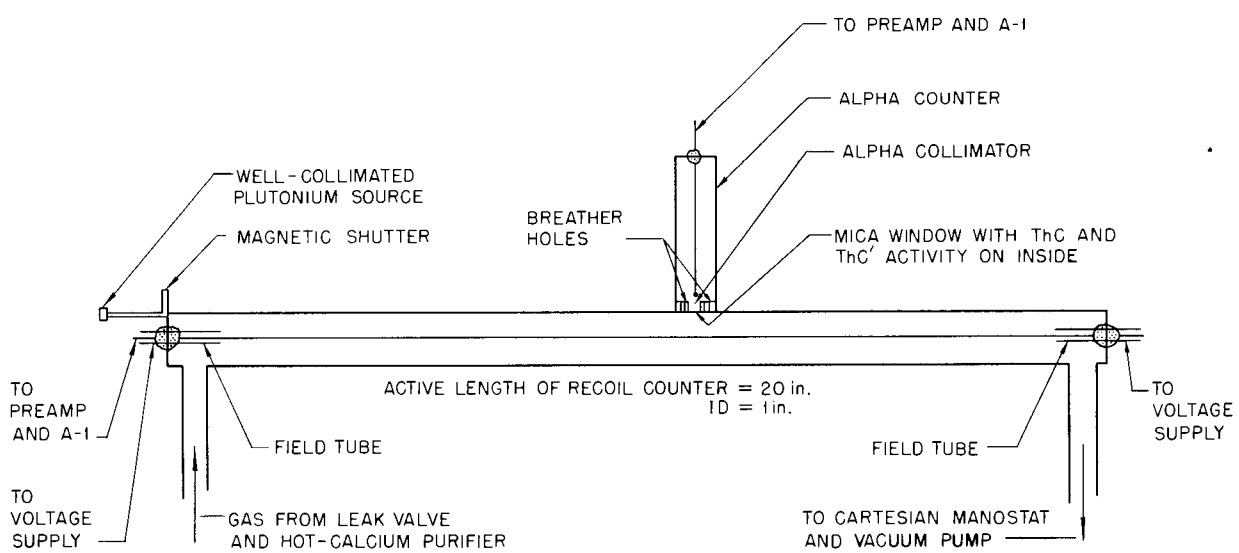

Fig. 7. Recoil-Measurement Proportional Counter.

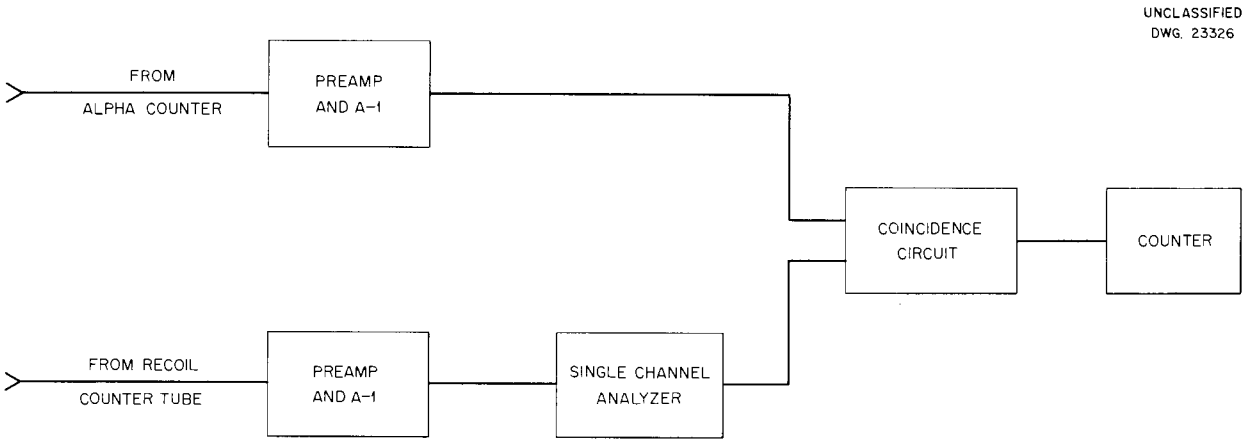

Fig. 8. Schematic Diagram of Recoil Ionization Measurement Circuits.

which is delayed by the single-channel analyzer. The output from the A-1 amplifier of the alpha counter is connected to the other input of the coincidence circuit.

A plot is made of the number of coincidences per unit time for various base lines of the single-channel analyzer, which gives two well-defined peaks corresponding to the pulse heights of ThC and ThC′. The same type of plot is made without the alpha counter and coincidence circuit to get the pulse height of the calibrating alphas.

The slow counting rate and the short half-life of the Pb^{212} (10.6 h) limited the counting statistics to the order of 10% probable error. The location of the peaks on the plots seems to be somewhat better than this. Table 2 gives the final data.

To compare the value of W of the recoil atoms in methane obtained in this experiment with the value obtained by Madsen for the W in 95% argon and 5% air, it is necessary to calculate the differential W, that is, the value of W calculated from the difference in energy of the ThC′ recoil atom (166,000 ev) and that of the ThC atom (115,000 ev) divided by the difference in ion pairs produced by the two recoil atoms. This calculation yielded a value of 107.8 ev per ion pair for methane. Madsen's value for the argon-air mixture was 67 ev per ion pair.

TABLE 2. MEASUREMENTS OF W USING SINGLE-CHANNEL ANALYZER*

EXPERIMENT NO.	PRESSURE OF METHANE (cm Hg)	W = ELECTRON VOLTS PER ION PAIR	
		ThC′ Recoil Atom	ThC Recoil Atom
1	1.078	125.0	141.0
2	1.056	130.7	131.5
3	1.056	126.9	137.4
4	1.824	126.4	149.6
5	1.464	117.9	121.3
6	1.822	128.2	132.7
7	1.056	130.1	145.7
Average		126.5	137.0

*W (differential) = 107.8 ev per ion pair for the ThC′ recoil atom in the energy region from 166 to 115 kev.

RADIATION MEASUREMENTS

F. J. Davis P. W. Reinhardt
J. A. Harter

URANIUM PROSPECTING[1]

The radiation-survey instrumentation of the second DC-3 aircraft has been completed. A series of flights over four sources (Ra, Co^{60}, Cs^{137}, and Au^{198}) placed on islands in Cherokee Reservoir were made at 10 altitudes from 200 to 2000 ft above the water surface. Plotting the measured intensity of radiation (counts per second) times height vs. height on semilog paper (Fig. 9) shows that the data can be represented as a straight line for altitudes above 400 feet.

Using 0.00118 g/cc as the density for air, the values of half-thickness correspond to absorption coefficients given in col. 3, Table 3. The corresponding theoretical values calculated from the Klein-Nishina formula are given in the last column. The value 1.07 Mev shown for radium is the energy corresponding to the 0.062 cm^2/g experimental value.

A straight line indicates that the buildup factor B (the increase in intensity due to the scattered radiation) is proportional to the height; so the intensity at a height h can be expressed as

$$I = k_1 \frac{e^{-\mu h} B}{h^2} = k_2 \frac{e^{-\mu h}}{h} ,$$

where μ is the linear absorption coefficient in air.

For altitudes below 400 ft, the buildup factor may be represented by $B = 1 + a(\mu h)$, since as h approaches zero the buildup factor should approach unity. For aerial surveying of radiation the first relation is sufficiently accurate. If this relation is used and if a plane source such as fall-out from atomic tests is integrated, the variation in intensity with altitude is a simple exponential of μh. Also in measuring a radioactive ore outcrop, if it is assumed that the buildup factors in the ground and in the air are proportional to the respective absorption coefficients of ground and air, a simple exponential is obtained. Thus in both cases the relation $I = ke^{-\mu h}$ holds.

The experimental data shown in Fig. 10 were taken at various locations but are normalized to an altitude of 5000 ft above sea level to correct for change of μ with altitude. If the half-thickness of air for radium gammas shown in Fig. 9 as 310 ft is corrected to 5000 ft above sea level, it becomes 360 ft, which compares very well with the half-thickness of air above ground shown as 370 ft in Fig. 10 for natural activity.

Tests were made with various forms of lead shielding for detector collimation of different discriminator settings of the amplifier, thus restricting the energies of radiation detected. The results showed that the lead shielding did not sufficiently improve the differentiation between a wide area of low-intensity radiation and a small area of high intensity, and hence it was omitted to save weight. The best signal-to-noise ratio was obtained with the discriminator set to accept all energies above approximately 50 kev.

A 6- or 12-volt battery-operated scintillation detector, weighing approximately 31 lb without the

[1] In cooperation with the U.S. Geological Survey.

TABLE 3. EXPERIMENTAL AND CALCULATED ABSORPTION COEFFICIENTS

SOURCE	E (Mev)	μ_{exp} cm^2/g	μ_{th} cm^2/g
Co^{60}	1.25	0.055	0.056
Ra	(1.07)	0.062	
Cs^{137}	0.66	0.081	0.077
Au^{198}	0.41	0.101	0.095

recorder, has been designed and built for use in a car or light aircraft. The unit contains a built-in calibration system consisting of a simple Q pulser (ORNL-Q-1066). The d-c voltage for the Q pulser is supplied from an RM12 mercury cell, and the output pulse is used to standardize the amplifier gain and to calibrate the rate meter. By switching input to the Q pulser from the RM12 battery to a voltage divider across the negative high-voltage supply, the stability of the high voltage can be checked directly with the amplifier discriminator.

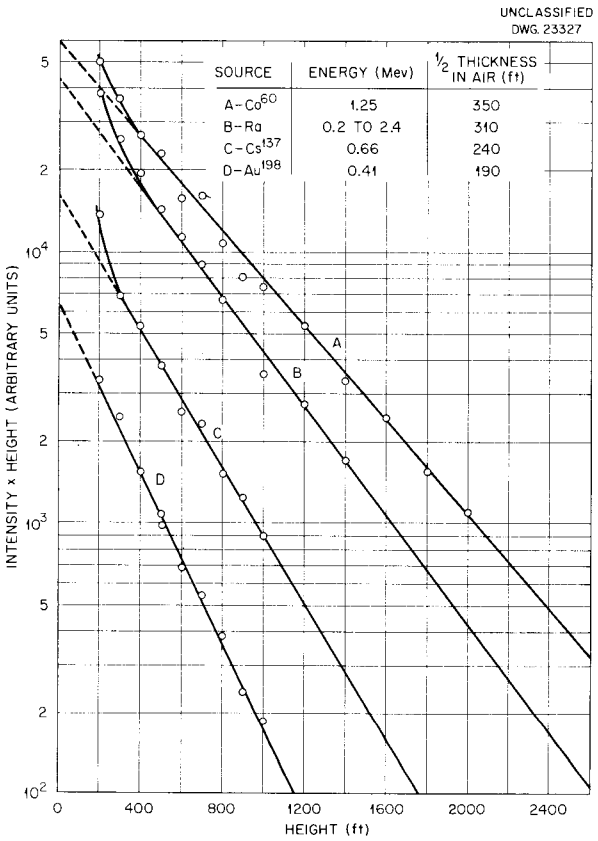

Fig. 9. Point-Source Measurements.

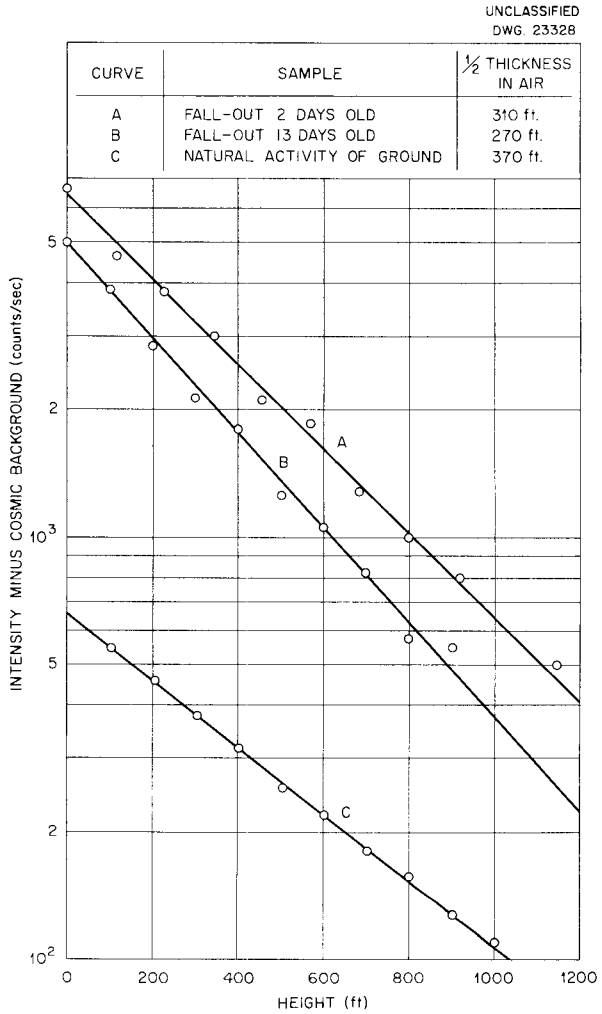

Fig. 10. Broad-Source Measurements.

RADIATION DOSE

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SPECTROGRAPHIC ANALYSIS OF HUMAN TISSUE[1]

The entire procedure of the spectrographic laboratory for the analysis of trace elements in human tissue is being changed. Instead of dry ashing samples of tissue and analyzing the ash by visual comparison of the spectrum of the tissue ash with standard spectra, a method of wet ashing, precipitating the elements to be analyzed with organic precipitants, and analyzing by means of an internal standard is now being employed.

The tissue is homogenized, dried under vacuum, and wet ashed with a mixture of nitric, perchloric, and sulfuric acids. The elements for which the sample is to be analyzed are precipitated with 8-hydroxyquinoline and thionalide (β-aminonaphthalide thioglycolic acid), using indium as a carrier and later as an internal standard. At the proper pH, sodium and potassium are not precipitated and thus are not present in the sample to decrease the spectrographic sensitivity for certain elements as in the old method. The precipitate is ashed at 450°C, and the ash is ground with graphite to make an arcing mixture. The entire treatment is carried out in fused-silica vessels.

The spectrographic procedures are much the same as have been reported previously. The d-c arc-cathode layer technique is used, the image of the cathode layer being focused on the collimator of the spectrograph. Quantitative analyses involving indium as an internal standard have replaced the visual comparison method previously reported. Blanks are carried through the entire procedure, and corrections are made for impurities in the reagents used in the new method.

The sensitivity for most elements has been increased from two- to tenfold by the new method, and the accuracy is many times greater than before. This method can be used in the analysis for approximately 30 elements.

[1]University of Tennessee research and development contract.

RADIOCHEMICAL ANALYSIS

A new technique for the separation of tracer amounts of radioactive strontium from bulky quantities of calcium has been achieved. The separation is possible because Versene forms stable complexes with calcium, whereas complexes with strontium do not exhibit such stability, provided the pH is controlled within a limited range. The separation is carried out on a cation-exchange resin such as Dowex-50. By this system it is possible to recover more than 99% of Sr^{89} tracer from as much as 600 mg of calcium with a separation factor of strontium from calcium of 10^3.

The solution containing alkaline earths is chelated with disodium dihydrogen versenate at a pH of 7.5. A 20% additional amount of chelating agent is added. The pH of the solution is then adjusted to 5.5, and the solution is passed over a column containing Dowex-50 (sodium cycle). Extraneous sodium and calcium can be eluted from the resin with 0.8 *N* HCl. The Sr^{89} can be eluted with stronger acid.

A procedure for separating radioactive strontium from urine, using the general technique outlined previously, is being worked out. Since urine contains appreciable quantities of magnesium, and since the stability constants of versenate chelates of magnesium and strontium are almost alike, an additional separation of magnesium from strontium can be made by eluting the magnesium with ammonium citrate. A recovery of more than 97% of tracer Sr^{89} from a 1500-ml urine sample can be achieved. Approximately 2 to 3 mg of alkaline earths are present in the eluent, and this small amount can be easily counted without further separation.

The foregoing procedure could be used for recovery of minute amounts of strontium from bulky water samples, from bone, and from urine.

EXTERNAL DOSE

Measurement of Ionizing Radiation by High-Frequency Variation

Experiments on the use of a ferroelectric material in the frequency-controlling circuit of a radio-frequency oscillator were continued. As previously described,[2] ionizing radiation varied the electrostatic charge on a capacitance made with a ferroelectric material, thus changing its dielectric constant and hence the frequency. Improvements on this technique extended the sensitivity range down to about 15 mr/hr. This work is described by E. Denman in a master's thesis to be submitted to Vanderbilt University.

Further improvements were made in the method by using capacitances of the order of 10 to 20 $\mu\mu$f. These were constructed with the cooperation of the ORNL Research Shop under the direction of R. J. Fox. These capacitances consisted of barium titanite 0.05 to 0.15 cm in thickness and less than 1 mm^2 in surface area. The capacitances were placed between phosphor-bronze springs that served as electrodes. The leakage resistance of these capacitances was less than that of larger pieces used previously, and their application for detection of ionizing radiation by frequency variation increased the sensitivity.

A detailed report by R. J. Fox on the construction of the capacitances will be included in the final report of the study.

In order to determine whether gamma radiation would produce measurable changes in the natural frequency of a cut quartz crystal, two crystals were exposed to 3×10^7 r of Co^{60} gamma radiation. One crystal retained its original frequency (990 kc), while the other, having initially the same frequency, had its resonant frequency lowered by 79 cps.

Measurement of Thermal Neutrons by High-Frequency Variation

The aforementioned method was applied to the detection of thermal neutrons by using a crystal of the piezoelectric material potassium pentaborate ($KB_5O_8 \cdot H_2O$) as the frequency-controlling element in a radio-frequency circuit. The material was supplied by H. Jaffe of the Brush Development Company. As may be seen in Fig. 11, the crystal was operated at series resonance. Its thickness was about 1.5 mm and the frequency about 1.03 Mc. The thickness was in the approximate direction of the X axis, and the vibration is a thickness shear mode.

The curve of frequency as a function of temperature for this cut of crystal has an extreme value at 25°C, and its frequency is therefore relatively stable against slight temperature variations at room temperature. When the crystal is exposed to thermal neutrons, the boron (n,α) reaction will occur, where $B^{10} + n \rightarrow Li^7 + \alpha + 2.79$ Mev. The amount of normal boron was about 48 mg/cm^2. Using the absorption cross section of 718 barns for this reaction, the macroscopic cross section is about 2, and therefore practically all thermal neutrons are absorbed. The following effects per captured neutron can be expected:

1. One boron atom will be removed, thus producing a vacancy in the lattice.

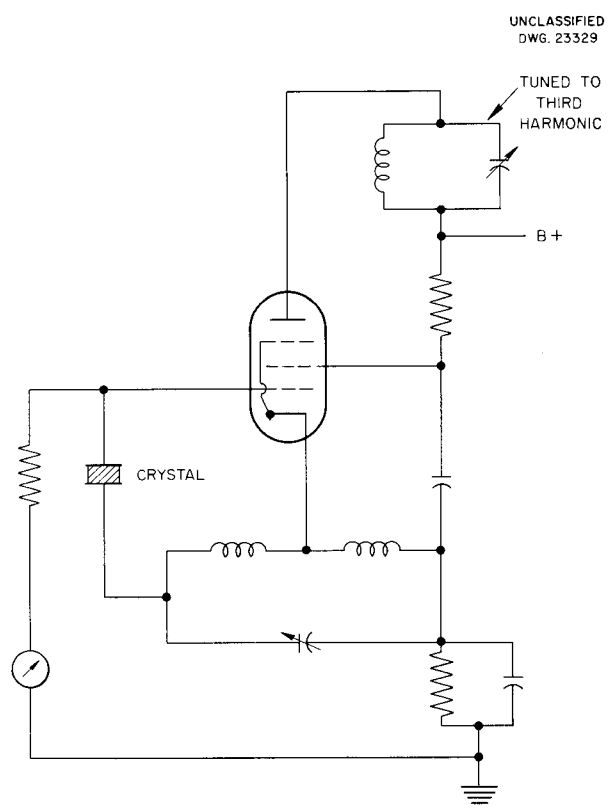

Fig. 11. Circuit for Studying Irradiated Piezoelectric Crystals.

[2] H. H. Hubbell *et al.*, *H-P Semiann. Prog. Rep. July 31, 1953*, ORNL-1596, p. 11.

2. The kinetic energy of the He^4 and the Li^7 will displace other atoms inside the lattice.

3. Recoils produced by the helium and lithium will produce further interstitials and vacancies in the lattice.

4. After loss of their kinetic energy, the helium and lithium will become impurity atoms.

The frequency of the crystal is given by

$$F = \frac{1}{4\pi x}\sqrt{\frac{E}{\rho}} = \frac{1}{2}\sqrt{\frac{EYZ}{mx}}, \tag{1}$$

where

Y,Z = length and width,
m = mass of the crystal,
x = thickness,
ρ = density,
E = elastic modulus,
F = frequency of vibration.

Any change in E or ρ will vary the frequency. The frequency variation can be represented by

$$\frac{\partial F}{F} = \frac{1}{2}\frac{\partial E}{E} + \frac{1}{2}\frac{\partial Y}{Y} + \frac{1}{2}\frac{\partial Z}{Z} - \frac{1}{2}\frac{\partial x}{x}. \tag{2}$$

To determine whether temporary exposure to different temperatures would affect the frequency permanently, the crystal was exposed to temperature variations from about 0 to 50°C. After 20 to 30 min in a room with an ambient temperature between 24 to 25°C, the frequency returned to its original value within ±25 cps. Since the thermal-neutron channel in the ORNL Graphite Reactor contained some gamma radiation, the crystal was exposed to an equivalent x-ray dose. No measurable effect was observed. Some uncertainty in all measurements was introduced by the variation of about 25 cps, due to the variation of the room temperature between 24 and 25°C. Standard equipment with quartz-crystal control and interpolation oscillator permitted observation of every cycle per second of frequency variation on the oscilloscope. The crystal was placed in the slow-neutron channel, where the temperature was kept below 50°C. No frequency variation was measured at exposures of the order of 10^{12} thermal neutrons/cm^2. At magnitudes exceeding 10^{14} neutrons/cm^2 effects were observed.

Setting n, the total number of neutrons per square centimeter, and ΔF, the frequency change after irradiation (plus sign is increase, minus sign is decrease of frequency), the results given in Table 4 were obtained.

Film Dosimetry for Fast Neutrons

Work had previously been done on the problem of converting the NTA film badge to a true dosimeter

TABLE 4. VARIATION IN FREQUENCY

n (neutrons/cm^2)	ΔF (cps)	APPROXIMATE TIME FROM END OF EXPOSURE TO FREQUENCY MEASUREMENT
2.25×10^{14}	−391	½ hr
	−444	2 hr
1.73×10^{14}	−218	½ hr
	−156	2 hr
2.23×10^{14}	−272	½ hr
	−150	1 day
	−126	2 days
3.2×10^{14}	−601	½ hr
	−175	3 days
	− 58	6 days

by means of a compensating proton radiator and absorber consisting of laminations of paper and aluminum to adjust the over-all response, in tracks per neutron, to the calculated dose per neutron as a function of energy. It had been decided that the response should be fitted to the first-collision dose curve, as has already been reported.[3] The structure of the packet and the agreement of dose-energy response with calculated values still remained to be developed and evaluated.

Although the best fit to the first-collision dose curve required a thinner film base as the first proton radiator, J. Spence and W. F. Swann of Eastman Kodak recommended that the original base be retained to avoid both manufacturing and processing difficulties due to a tendency of thin film to curl. The component laminations were recalculated for the original 0.008-in. film base. The maximum deviation of the response of the packet from the dose curve was +10% at 5 Mev. This is considered acceptable since (1) under the most favorable conditions, the reading errors are of this magnitude and (2) the monitoring is usually of a broad neutron spectrum and thus would tend to produce a smaller average deviation than 10%.

The final packet devised is symmetrical to the plane of the emulsion in that it contains a blank film base facing the emulsion and one sheet of aluminum foil on each side. The blank film base makes the dosimeter independent of effect of reversal of the badge, though not completely nondirectional. The aluminum foil allows the manufacturer to use present packaging machinery with only minor adjustments, merely substituting the aluminum foil for the usual black paper. The resulting packet is energy independent within 10% up to 10 Mev. A measurement with a Po-Be source showed agreement between observed and calculated track counts within the probable error of the count. If higher-energy neutrons are to be monitored, additional shields and radiators may be incorporated in the badge.

A report on this project has been written and submitted to *Nucleonics* for publication.

Energy Losses of Electrons in Foils

Beta Spectrometer. The large beta-ray spectrometer of the solenoidal type mentioned in earlier progress reports is practically complete and has undergone preliminary tests. Correcting coils have been added, and the magnetic field has been measured and found to be uniform within 0.2%. The controls on the console have been completed and are operating dependably. These include a current regulator which maintains the current constant within 0.01% at any specified value and a ripple corrector which eliminates generator brush noise by feeding an equal but out-of-phase signal to a geometrically similar coil which is interwound with the main coil. An automatic sequencer has been installed which directs the spectrometer to scan any desired energy range and which prints the time required to accumulate a predetermined number of counts both on the spectrum under investigation and on the background. In addition, interlocks have been installed which shut down the instrument in case of vacuum, power, or water failure. The motor-generator has been enclosed in a soundproof housing, and a fan has been installed to circulate outside air through this housing.

The slits have been constructed, installed, and aligned with the magnetic field by noting the performance of the instrument on *K* conversion lines appearing in the spectra of Sc^{46} and Ag^{110} (629 and 886 kev, respectively).

Accelerator. During this period the electrostatic analyzer was installed and is now operating. It was found necessary to add collimating slits in front of the analyzer to achieve the required resolution. The source of electrons originally was a tungsten filament, but his has now been discarded in favor of a commercial unit taken from a 5-in. cathode-ray tube. A method of breaking the tube envelope and introducing the gun into the accelerator without destroying the activation of the cathode has been found. The new electron gun delivers high currents in a well-focused beam with narrow energy spread.

A technique of making thin Zapon films has been worked out. Films of about 10 $\mu g/cm^2$ are placed on an electromesh backing and have proved sufficiently sturdy to permit vacuum deposition of metallic layers of about the same thickness. A film holder and vacuum-lock arrangement allow easy introduction of the foils into the beam. The various meters in use in the instrument have been calibrated, and an energy calibration of the decelerating and deflecting system has been made.

[3] *Ibid.*, p. 12.

Aluminum foils of about 10 $\mu g/cm^2$ have been placed in the beam of 45-kev electrons, and a peak in the energy-loss curve at 15.65 ± 0.05 ev has been found. This agrees with the plasma oscillation theory of Pines and Bohm,[4] if the effective number of free electrons per atom in aluminum is taken as 2.93.

Film Dosimetry

Studies have been continued on the ORNL personnel monitoring and security badge in the hope of developing a technique for routine interpretation of the film blackening in terms of gamma-ray energy and exposure and beta-ray exposure. Sample badges were exposed to several more x- and gamma-ray energies in addition to those previously reported[5] and to combinations of various doses of uranium beta rays and different x- and gamma-ray doses and energies. Considerable analysis of the data has produced a system which promises to permit satisfactory interpretation of a film badge exposed to a single moderately narrow range of x- or gamma-ray energies, with or without additional uranium beta-ray exposure. Exposures of the film to radiations of more than one energy are much more difficult to interpret correctly, and the problem is still under study.

A new type of film, Du Pont 555, was investigated. It has a much wider dose range than any one now in use, and, if it proves to be sufficiently stable, it might replace the three films currently used. Somewhat lower exposures can be read on this film, and the variation of its sensitivity with energy is about the same as that of other films studied.

[4] D. Pines and D. Bohm, *Phys. Rev.* 82, 625 (1951); 85, 338 (1952); 92, 609 (1953); 92, 626 (1953).

[5] H. H. Hubbell *et al.*, *op. cit.*, p. 13.

AIRBORNE RADIOPARTICULATE CONTAMINATION

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W. D. Cottrell
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B. G. Saunders
J. W. Thomas

RADIOACTIVE-DUST STUDIES AND PROTECTIVE EQUIPMENT EVALUATION PROGRAM (PEEP)

Joint Chemical Corps–ORNL Program

This period marks the end of the cooperative support of the Protective Equipment Evaluation Program by the U.S. Army Chemical Corps. The program was initiated as a joint effort of the Chemical Corps and ORNL and was authorized on a one-year basis July 1, 1949. The objective of the program was to study the performance of protective equipment such as filtering devices, masks, clothes, etc., when used in the presence of radioactive gases and radioactive submicron particulates. Sponsorship of the program was withdrawn October 1, 1953 by the Chemical Corps, because of a sharp cut in budget. A substantial amount of money was turned over to the group for completion of current activities.

Work prior to January 1, 1952 has been summarized in Quarterly Progress Reports, Protective Equipment Evaluation Program, Y-531, Y-532, Y-605, Y-638, Y-687, Y-724, Y-768, ORNL-1258, and ORNL-1352. Subsequent work has been summarized in Health Physics Division progress reports.

Tests have been made on the performance of gas-mask canisters with radioactive SO_2 and of filter papers with radioactive methylene blue. This research resulted in three reports: Y-700, Y-685, and ORNL-1378. The major effort of this program, however, has been directed toward the development of instruments, since the instrumentation in this field is not adequate. The following nine reports of this endeavor were issued: Y-791, Y-793, ORNL-1413, ORNL-1648, ORNL-1655, ORNL-1656, ORNL-1657, ORNL-1660, ORNL-1666.

Study of Radioparticulate Contamination at X-10

During this period, the Health Physics Division, in cooperation with the Oak Ridge Office, AEC–U.S. Weather Bureau, continued a study of radioparticulate air contamination at X-10.

A histogram (Fig. 12) shows the average number of particles per 1000 ft^3 collected weekly during the period from March 1949 through December 1953 by 10 continuous air monitors stationed throughout the Laboratory area.[1]

As a start in isolating a single source of radioparticulate contamination, a special isotope-recovery process was selected for study. A complete cycle of operation was monitored during the last reporting period, and another cycle was monitored during this reporting period.

Evaluation of the data showed that the major portion of the activity discharged through the stack appeared to be gaseous radioiodine. This gaseous activity masked the particulate activity and made it difficult, by radiochemical analysis or autoradiographic techniques, to identify the nature of the collected particulates. Only those samples containing particulates showed elements other than iodine.

Gaseous and particulate samples were collected simultaneously in the recovery process building and in the stack for a period of 30 hours. The concentration of particulates per unit volume of air inside the building was 3.2 times higher than the concentration of particulates per unit volume of stack effluent.

An underground tank that serves to store wastes from the special isotope-recovery process was monitored during the second cycle studied. This tank proved to be a profuse source of airborne contamination. Peaks of airborne activity occurred whenever wastes were jetted from the process building to the tank. Air was sampled 18 in. away from the vent in the top of the tank. The level of activity two days prior to the first jetting amounted to 10^{-9} $\mu c/cc$. The average activity for the period that covered the entire cycle of operation averaged 2×10^{-6} $\mu c/cc$. The highest activity observed in a 24-hr period was 2×10^{-5} $\mu c/cc$.

Peaks of activity at the tank vent were always accompanied by peaks of activity and particle count at the continuous air monitors located about the Laboratory area. The composition of the activity coming from the tank vent varied with the different stages of the process cycle. Radiochemical analyses showed considerable quantities of various isotopes. Because of a diffuse background, individual particulates could not be resolved for measurement of size when autoradiograms were made from the samples.

[1] The arrows on the histogram mark the periods when large chemical separation processes were known to be in operation in the Laboratory, and in most cases these periods were accompanied by peaks of radioparticulate activity in the area.

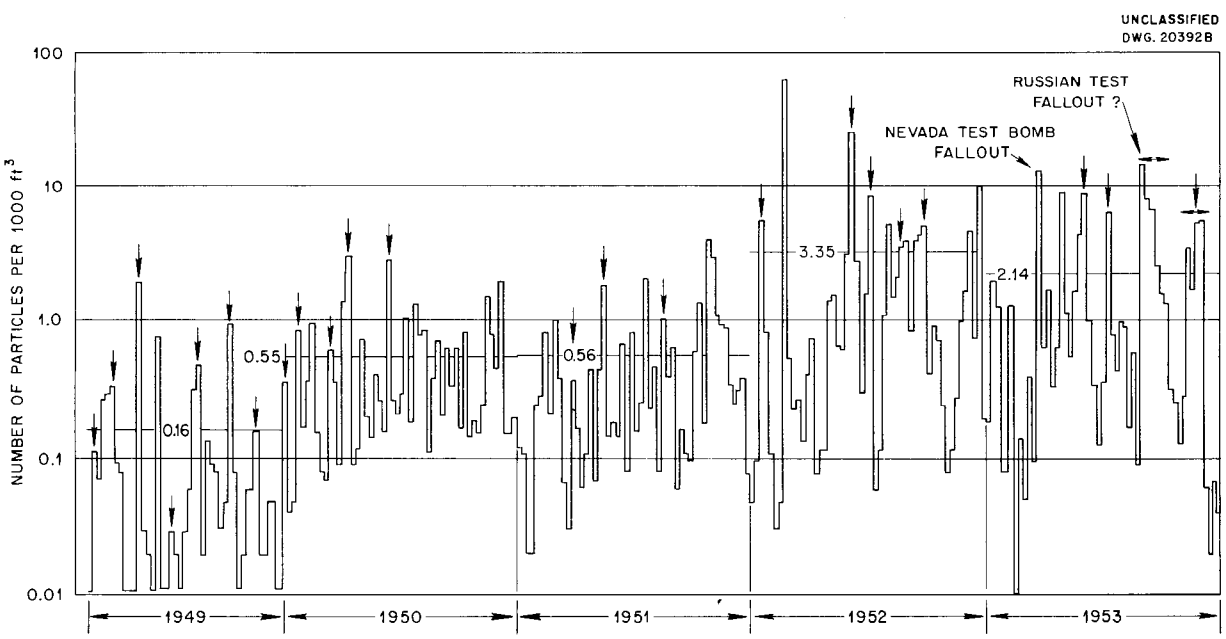

Fig. 12. Trend of Particle Concentration at ORNL from 1949 to 1953.

The concentration of particulates in the process building ranged from 0 to 2600 particles in each 1000 ft^3 of air, the average for one entire cycle amounting to 177 particles in 1000 ft^3. Decay and absorption studies of the samples collected in the process building indicated that a mixture of isotopes was present in the air.

RADIOACTIVE WASTE DISPOSAL RESEARCH

R. J. Morton

WATER AND LIQUID-WASTE DECONTAMINATION PROCESSES

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H. L. Krieger
W. J. Lacy
D. A. Pecsok
O. R. Placak
M. S. Seal
C. P. Straub

Water Decontamination Studies

At the October meeting, the Technical Advisory Board for Research on the Decontamination of Radioactive Waters discussed topics pertinent to radioactive-waste problems and water decontamination and reviewed the "Interim Progress Report on the Decontamination of Radioactive H_2O" compiled by the Radioactive Waste Disposal Research Section of ORNL. The following statements appear to represent the consensus of the Advisory Board as reported at the meeting:

1. The water-decontamination studies at ORNL have progressed in a satisfactory and integrated manner.
2. The military-equipment testing program has progressed to the point where future planning can be limited to the dissemination of operational information.
3. The decontamination research should continue on a selective basis to obtain data on those radioisotopes which are most hazardous and which will probably reach water courses in the most significant quantities.
4. The interim report, as revised and edited, should be prepared for broader distribution as an unclassified document through normal informational channels.
5. The proposed AEC waste-disposal policy, although applying specifically to AEC contractors, has wide implications. The utilization of the NCRP standards as a base is sound. However, coordination with public-health and stream-pollution agencies will be necessary to prevent inadvertent establishment of double standards for individual streams.
6. From a health-protection standpoint, the development of continuous water-monitoring devices can be given a low priority (in the ORNL joint program).
7. The work has progressed to a stage where consideration should be given to the dissolution of the Board as an active administrative group.
8. The creation of an appropriate program development and review group for research on disposal of waste from nuclear reactors warrants consideration.

Process-Waste-Treatment Experimental Plant

Three pilot-scale experimental waste-treatment units have been installed near the settling basin. Experimental studies will be made with these units in an effort to devise a method of treatment applicable to a full-scale installation for reducing the radioactivity in the discharge from the settling basin. A schematic flow diagram for the three units is shown in Fig. 13. Two of the units are approximately 7 ft in diameter and 5 ft in depth, and the third is 5 ft in diameter and approximately 3 ft in depth. They are comprised of a solids contact chamber, an up-flow type of sedimentation chamber, and a sludge-concentrator compartment, as typified by the conventional Accelator unit shown in Fig.

14. The SR (sludge recirculation) Accelator is a new development that is better adapted to water-softening processes. The Erdlator, developed by the Corps of Engineers Research and Development Laboratories, will also be evaluated for treatment of the settling-basin waste. It is anticipated that operation of the experimental plant will be started during the week of February 1, 1954.

The general objectives of these experimental studies will be (1) to determine the efficiency of each of the three types of units in the removal of radioactive constituents from the wastes by water-treatment methods using various coagulating chemicals and various adsorbents, (2) to determine the volume of sludge produced and suitable methods for handling and disposing of this sludge, and

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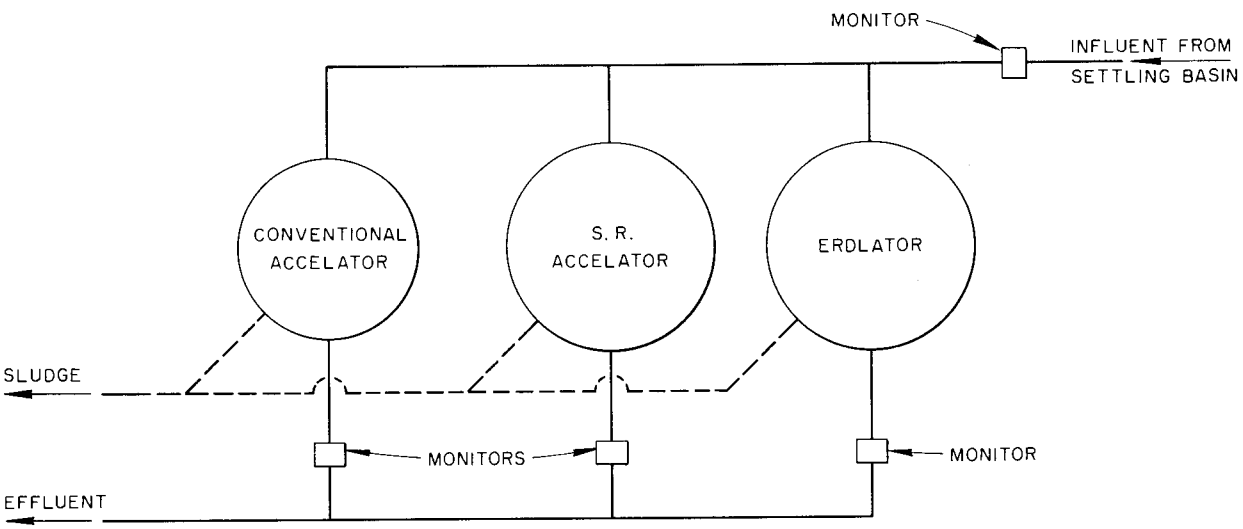

Fig. 13. Flow Diagram of Waste-Treatment Pilot Plant.

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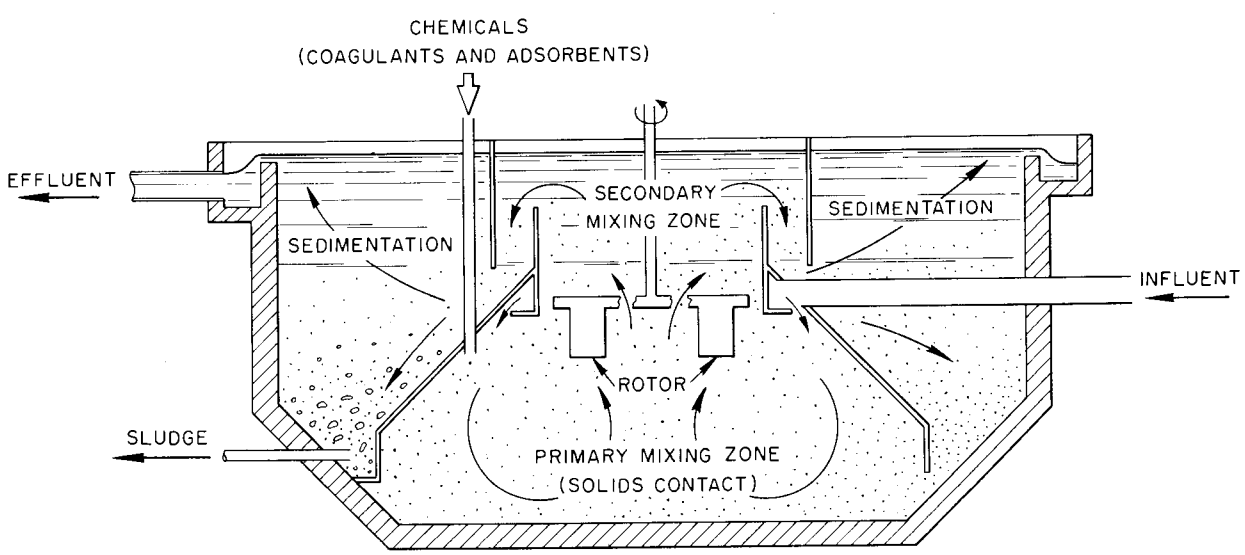

Fig. 14. Diagrammatic Cross Section of Accelator.

(3) to determine whether radiological hazards are likely to develop due to external radiation in the vicinity of the sludge or the equipment.

A continuous monitoring system for the measurement of the levels of radioactivity in the influent and effluent of the treatment units has been designed and partially constructed. The system includes four detectors with their associated preamplifiers (one on the influent and one on each of the three effluent pipes), an automatic switching unit, and recording count-rate meters. Each of the detectors, which were developed especially for these experiments, consists of a Lucite cylinder approximately 3½ in. in diameter and 18 in. in length through which the influent and effluent liquids flow continuously. In each cylinder a thin-walled glass G-M counter tube is suspended concentrically and supported by a watertight compression seal at the top of the cylinder. The monitoring system will be calibrated by periodic sampling of the influent and effluent liquids.

Surface Adsorption – A Mechanism in the Removal of Radioactive Materials from Aqueous Solutions by Clay Slurry

It is well known that suspensions of clay materials are effective in removing many radioisotopes from solutions. If the process of removing radioactive materials from water by clay slurry is similar to true surface adsorption and follows a Freundlich-type reaction, then the log-log plot of the amount of adsorption per unit weight of adsorbent (clay) vs. the nonadsorbed portion (radioactivity) should be a straight line.

To investigate this phenomenon, a jar-test procedure was followed in which "spiked" distilled water and local montmorillonite clay were used. Samples were taken for counting, and a log-log plot of the results obtained are shown in Fig. 15. From these results it can be concluded that the process of surface adsorption appears to be the major mechanism involved in the removal of certain fission products from water by clay slurry.

If this same procedure is used, an approximate idea of the capacity of a unit quantity of clay at 0% salt concentration (distilled water) can be obtained. Extreme dilutions of clay material were necessary (10 to 20 ppm) in order to approach saturation of the clay surface. This limits the accuracy of the results. It was calculated from the test results that about 10 mc of mixed radioactive fission products can be adsorbed on 1 g of montmorillonite clay from a distilled-water solution under the conditions of these tests.

SURVEYS AND EVALUATIONS – FIELD AND LABORATORY

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K. E. Cowser	C. P. Straub
J. M. Garner	H. J. Wyrick
B. Kahn	R. M. Richardson[1]

Continued Study of Waste-Storage Pit No. 2

The construction and observation of waste-storage pit No. 2 have been summarized previously.[2] Up to January 1, 1954, a total of 278,000

[1] U.S. Geological Survey, part time.

[2] T. W. Brockett *et al.*, *H-P Quar. Prog. Rep. Oct. 20, 1952*, ORNL-1420, p. 1.

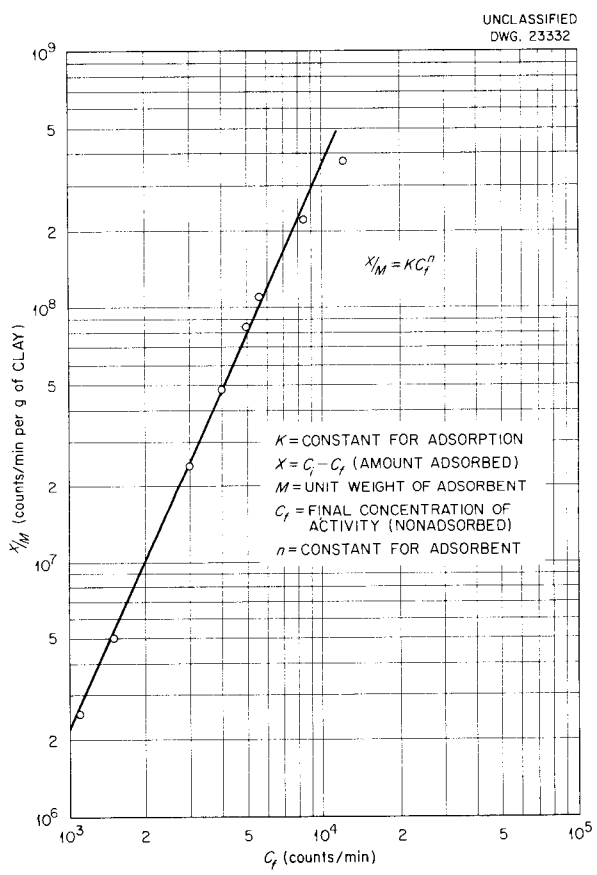

Fig. 15. Adsorption Isotherms for Montmorillonite Clay with 90 min Contact Time in a Mixed-Fission-Product Solution.

gal of liquid radioactive waste containing 8700 curies of activity had been transferred to the pit. Grab samples taken from the three wells adjacent to the pit and counted at approximately 10% geometry have shown a maximum activity of 72,700 counts/min/ml in well No. 52 and 15,400 counts/min/ml in well No. 54. Insofar as can be determined, well No. 53 has remained uncontaminated. Radiochemical analyses continue to indicate that the contamination reaching the wells from the pit is entirely Ru^{106} and its daughter product Rh^{106}. Liquid levels in the pit have a definite and moderately rapid influence on water levels in the two contaminated wells.

To obtain information relative to the amount, location, and entrance point of radioactive contamination with respect to these and other test wells, radiologging equipment was developed. Essentially, it consists of a thin-walled G-M tube and preamplifier encased in a suitable watertight housing, connected to a count-rate meter and Esterline-Angus recorder. The probe is lowered in the well at a rate of ½ ft/min by means of a motor-driven cable reel which may be set to cut off at any desired depth. Figure 16 shows the continuous type of radiolog that is obtained. Calibration of the probe by means of analyzed water samples collected from specific depths will allow use of the radiolog charts to determine the total radioactivity in the well, if it can be assumed that most of the radioactive contaminant is contained in the water rather than the walls of the well at the points of measurement.

Definition of the underground pollution stream originating at the pit will necessitate additional test wells. To facilitate the installation of these wells, special equipment has been made available by the U.S. Geological Survey.

Additional work proposed in connection with the pit study includes the following:

1. electric logging of wells Nos. 52, 53, and 54,
2. radiologging of existing and future wells,
3. controlled sampling of the wells,
4. drilling of additional wells in areas adjacent to the pit,
5. permeability studies of prospective pit sites,
6. a controlled study of a pit of pilot-plant dimensions.

[3] J. M. Garner *et al.*, *H-P Semiann. Prog. Rep. July 31, 1953*, ORNL-1596, p. 28.

Radioactivity in River-Bottom Sediments

A limited resurvey of radioactivity in bottom sediments in the Clinch River from White Wing Bridge to the mouth of the River and in the Tennessee River downstream to Chattanooga was made during the summer of 1953. The detector (Flounder) used in the surveys[3] of 1951 and 1952 was used in this resurvey. The weighted averages of the readings at various cross sections of the river are plotted in Fig. 17, in order to compare the results of the three surveys (1951, 1952, and 1953).

As shown in Fig. 17, the levels of activity observed in sediments in the Clinch River were lower than those found in 1952 but higher than in 1951. In the Tennessee River the 1953 levels were essentially the same as in 1952. Definitive data relative to the conditions of river flow and to the quantities and half lives of radioactive materials discharged during 1953 are insufficient to afford an explanation of the observed decrease in the levels of activity found in the Clinch River sediments for this one year.

Five mud samples from bottom deposits, two from Clinch River, two from Watts Bar Reservoir, and one from Chickamauga Reservoir, were collected to supplement the data from mud samples taken during the 1952 survey. The purpose of the work with mud samples was to develop methods of analysis, to identify and determine the amounts of fission products present, and to obtain information concerning the radioactivity in bottom sediments that originated from natural sources. Measurable amounts of fission products were found in the mud from Clinch River and in Watts Bar Reservoir below the mouth of the Clinch River. The techniques and results of mud analyses are summarized in the following section.

INSTRUMENTATION AND TECHNIQUES

J. M. Garner B. Kahn

Determination of Radioactive Nuclides in River-Bottom Sediments

A method is being developed to determine the amount of radioactive nuclides adsorbed by muds of the Clinch and Tennessee Rivers. Such techniques of analysis are also needed in studies of the movement of radioisotopes through soils, for example, in the vicinity of waste-storage pits. The radioactive nuclides present in the river muds are believed to be some of the major long-lived

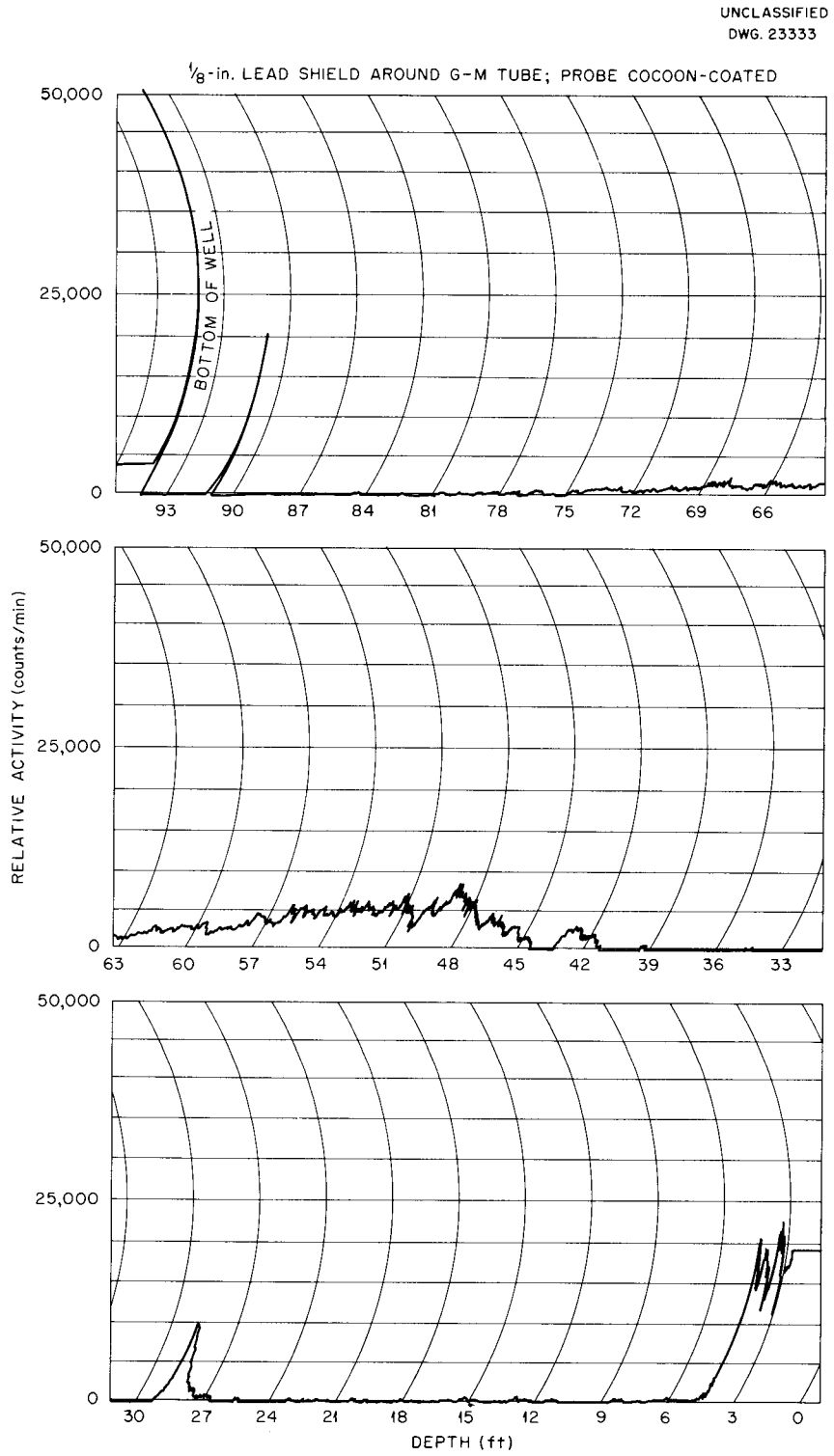

Fig. 16. Radiolog Obtained in Well No. 52 on January 13, 1953.

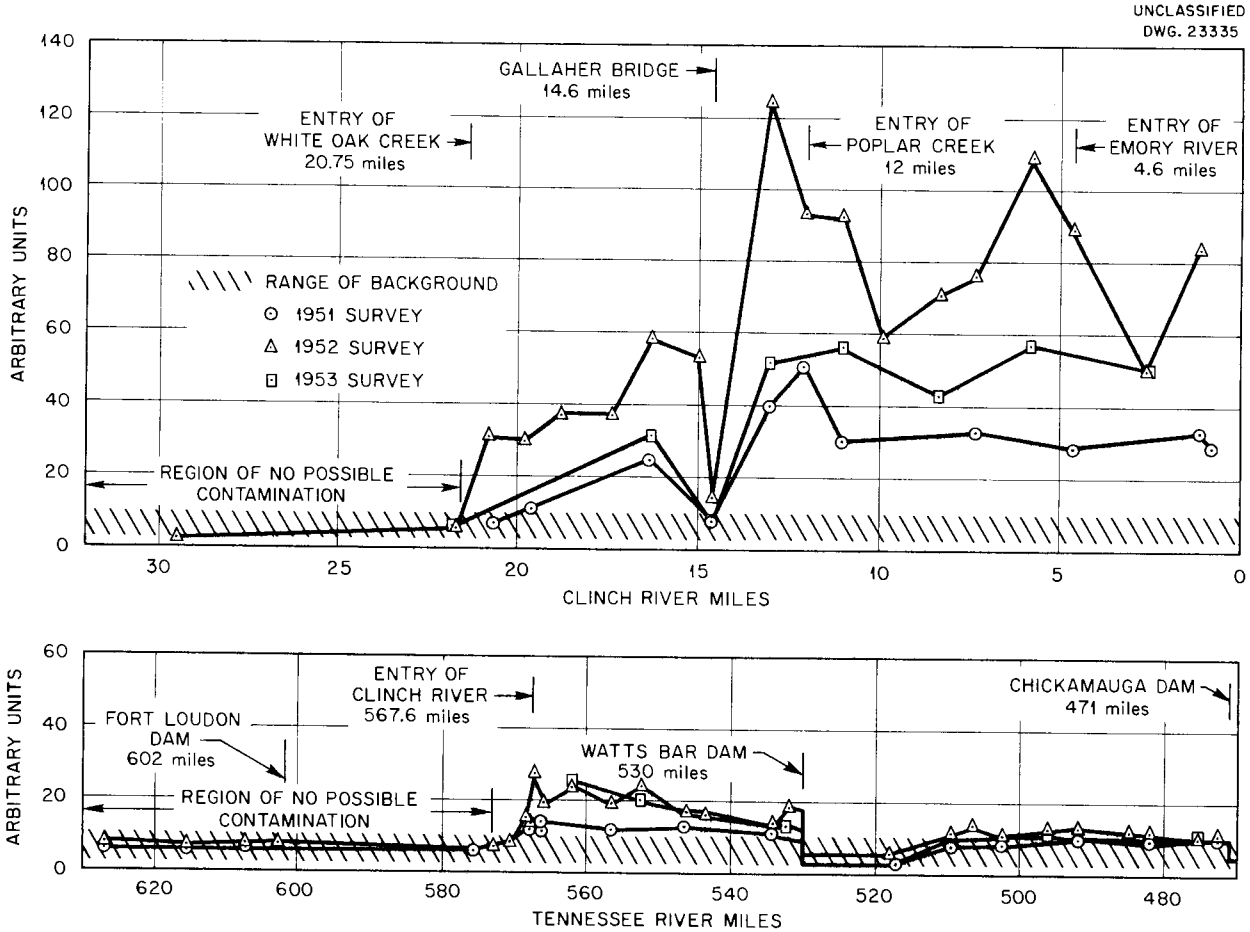

Fig. 17. Weighted Average Concentration of Radioactivity in River Sediments.

fission products – Sr^{89}, Sr-Y^{90}, Y^{91}, Zr-Nb^{95}, Cs-Ba^{137}, Ce-Pr^{144}, and Pm^{147}. The procedure consists in leaching the mud, followed by chemical separation of the radionuclides.

The radionuclides, together with large amounts of solids, are leached from the mud by refluxing with a total of 3 ml of aqua regia per gram of moist mud for three consecutive 12-hr periods. Tracer studies of the efficacy of leaching were made by adsorbing about 1.0 μc of each of the radionuclides listed in Table 5 on 15-g samples of Tennessee River mud (containing almost no radioactivity) in a slightly basic solution, washing the mud with tap water, and then leaching as just described. The efficacy of removal of radionuclides from mud is shown in Table 5.

In order to determine the amount of natural radioactivity which may be ascribed to radium at various points in the Tennessee River, samples from Duck River and from Wilson Dam were analyzed for radon (Rn) by A. Gabrysh of the Experimental Radiation Measurements Section, Health Physics Division. The results, listed in Table 6, indicate no unusual radon activity in the analyzed samples.

Compilation of Radiochemical Methods

Abstracts of unclassified methods for the chemical separation of radionuclides have been compiled.[4] The collection consists of approximately 275 methods for the separation of 85 nuclides. Each abstract contains a summary of the procedure and the associated time, yield, and decontamination factor (when these are reported), as well

[4] B. Kahn, *Separation of Radionuclides—Abstracts of Unclassified Methods*, unpublished.

as references to the original papers. The compilation has been distributed to persons who are developing and using radiochemical methods, in order to obtain their criticisms, corrections, and suggestions for changes and additions. It is planned to revise and bring the abstracts up to date according to the suggestions received.

TABLE 5. LEACHING OF RADIONUCLIDES FROM TENNESSEE RIVER MUD

RADIONUCLIDE	PER CENT REMOVED BY LEACHING
Sr^{90}	98
Y^{90}	98
Zr^{95}	76
Nb^{95}	65
Ru^{106}	96
Cs^{137}	98

TABLE 6. SAMPLES ANALYZED FOR RADON ACTIVITY

SAMPLE	ACTIVITY
Duck River (water)	0.56×10^{-12} curie/liter
Duck River (solids in 1 liter of water)	2.0×10^{-12} curie/g
Wilson Dam (water)	1.12×10^{-12} curie/liter
Wilson Dam (solids in 1 liter of water)	5.4×10^{-12} curie/g

Portable Well-Logging Instrument for Beta Measurements

A portable instrument has been designed and constructed for the detection and measurement of radioactivity, particularly beta activity, in the earth walls above water level and in the water of test wells. This instrument was devised for convenient use in the radiologging of multiple test wells to be drilled in order to determine the underground travel of radioactive contaminants from the two existing ORNL chemical waste pits and from future experimental waste-pit installations. The detector is a commercially available 30- to 35-mg/cm^2 stainless-steel-walled G-M tube. The G-M tube is attached by means of a watertight plug-in connector to the preamplifier housing to which, in turn, is attached 53 ft of five-conductor, shielded rubber-covered cable. This cable leads to a battery power supply and to a modified portable survey meter.

A portable well-logging instrument is essential for use in relatively inaccessible locations. Advantages of the present design are its beta sensitivity, the ease of replacing the G-M tube, the ease of changing types of detectors so as to cover a wide range of activity levels in the test wells, and the ability to protect the detector against contamination by covering it with a watertight disposable thin rubber sheath. One limitation of the present design is that the detector will not withstand water pressures caused by submergence to depths greater than 17 feet.

ECOLOGICAL STUDY[1]

L. A. Krumholz
W. T. Helm
W. T. Miller
E. R. Eastwood

Virtually the entire period has been spent in the preparation of the final report on the findings of the survey, and that report is nearly complete.

The extent to which migratory water fowl are using White Oak Lake as a resting and feeding area is still under study; thus the program for banding and releasing migratory waterfowl at the lake has been continued. Banding was begun in September 1953 in order to capture the resident and early migrant wood ducks. From that time through January 31, 1954, a total of 4881 waterfowl were caught in the eight traps designed for that purpose. Of those, 1108 new individuals were banded; the remainder had been previously banded, either by this group or by others, and were recaptures.

The numbers of birds of each species that were banded or recaptured are listed in Table 7.

Of the waterfowl banded at White Oak Lake during the 1952-1953 season, a total of 30 individuals, which represents 5% of those banded, were recaptured at the same location during the current season, as follows: 16 mallards, 8 black ducks, 5 wood ducks, and 1 coot. Also, three birds banded at other stations, as yet unknown to us, were caught at White Oak Lake.

[1] In cooperation with the Tennessee Valley Authority.

TABLE 7. BIRDS BANDED OR RECAPTURED FROM SEPTEMBER 1953 TO JANUARY 31, 1954

	MALLARD	BLACK DUCK	WOOD DUCK	BALD-PATE	PIN-TAIL	GREEN-WINGED TEAL	GAD-WALL	COOT	TOTAL
Banded	800	118	142	10	4	8	1	25	1108
Recaptured	2787	637	255	28	28	8	0	30	3773
Total	3587	755	397	38	32	16	1	55	4881

EDUCATION, TRAINING, AND CONSULTATION

E. E. Anderson M. R. Ford
M. F. Fair T. H. J. Burnett
K. Z. Morgan

AEC FELLOWSHIP PROGRAM

The 1953-1954 group of 22 AEC Fellows in Radiological Physics, which reported to Vanderbilt University on September 28, completed the first quarter of the academic work with very creditable grade records. The 1952-1953 group completed their training in applied health physics at the Laboratory on August 29. Of this group, three fellows were granted extensions of their fellowships to complete work for the master's degree and chose research problems in the Health Physics Division. They have completed the experimental work and have accepted employment.

OTHER TRAINING PROGRAMS

Two members of the Education, Training, and Consultation Section gave a three-week course in health physics in the Reactor School. In January, a 2 hr per week course in nuclear physics was started for members of the Applied Health Physics Section who have come to the Laboratory recently. The course will continue for approximately six months.

Lt. Elwood Lloyd, Army Chemical Center, Fort McClellan, Alabama, spent the month of January in training in the applied Health Physics Division.

LECTURES

Lectures on radiation hazards and current practices in radiation protection were given at Virginia Polytechnic Institute and the University of Mississippi in conjunction with the ORINS-ORNL Traveling Lecture Program, and two lectures on radiation protection were given at the Medical Military Symposium of the U.S. Navy Medical School. Seminars given during the reported period consisted of one on health physics for R. T. Overman's radioisotopes course at ORINS, one on radiological safety to the ORINS advanced medical course, one on nuclear physics and one on radiation protection of the Fifth Annual Nuclear Science Seminar of the Naval Research Reserve Group.

CONSULTATION

The relative toxicity of U^{233} was investigated to determine the extent to which the hazards associated with handling U^{233} would influence the costs of manufacturing fuel elements. Chiefly because of its greater specific activity and also because of differences in energies, the hazard in the processing of U^{233} would be 133 times that of U^{235} for the same loss in mass. However, preliminary tests indicate that the additional precautions required for U^{233} will lead to only a 20% increase in costs. In the course of this study[1] the effective energy (that is, the sum of the energies of the radiations of the parent and the daughters weighted for their relative biological effectiveness, normalized to the RBE of alphas) of radium, used as a reference for biological damage value, was recalculated and found to be 16.18 effective Mev.

[1] T. H. J. Burnett, ORNL CF-53-9-77 (Sept. 16, 1953).

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J. W. Thomas, *The Diffusion Battery Method for Aerosol Particle Size Determination*, ORNL-1648 (Dec. 14, 1953).

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T. E. Bortner, *Range of P^{32} Betas in Different Atomic Number Absorbers*, Tennessee Academy of Science, Nov. 27, 1953, Oak Ridge, Tenn.

G. S. Hurst, *Fast Neutron Dosimeters, Care and Breeding Of*, Shielding Meeting, Argonne National Laboratory, Nov. 12, 1953.

W. S. Snyder, *On the Solution of Integral Equations*, American Mathematical Society, Aug. 31 to Sept. 4, 1953, Kingston, Ontario, Canada.

I. H. Tipton, *Results of Spectrographic Analysis of Tissues of 40 Individuals*, Tennessee Academy of Science, Nov. 27, 1953, Oak Ridge, Tenn.

B. Kahn and W. S. Lyon, *Gamma Ray Spectroscopy in Analytical Radiochemistry*, Southwest Region Meeting of American Chemical Society, Dec. 10 to 12, 1953.

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E. E. Anderson, *Radiation Protection*, Sigma Delta Epsilon Honorary Women's Science Research Fraternity, Madison, Wis., Sept. 8, 1953.

E. E. Anderson, *Nuclear Physics Review*, 5th Annual Nuclear Sciences Seminar, Naval Research Reserve, Oak Ridge, Tenn., Dec. 2, 1953.

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B. G. Saunders and W. D. Cottrell, *The Spinning Top*, 8th Tripartite Meeting, Army Chemical Center, Edgewood, Md., Sept. 28, 1953.

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