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# HEALTH PHYSICS DIVISION

## SEMIANNUAL PROGRESS REPORT

FOR PERIOD ENDING JULY 31, 1953

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# HEALTH PHYSICS DIVISION SEMIANNUAL PROGRESS REPORT

for

Period Ending July 31, 1953

K. Z. Morgan, Director

DATE ISSUED

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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

A general method has been developed for determining the number of vacant lattice sites or interstitial atoms in a monoatomic substance exposed to heavy corpuscular radiation. The colliding atoms are assumed to be within the energy range for which the orbital picture can be applied. By following the treatment of Bohr, the energy range is divided into regions of excessive screening, moderate screening, Rutherford scattering, and electronic collisions. The first three regions cover the range in which the velocity of the atom is less than the velocity of the orbital electron of the hydrogen atom. In this range, which extends over $0 \leq E \leq \gamma = 25\ A \times 10^3$ ev (where $A$ is the mass number of the atoms in the solid), only elastic collisions are considered. The theory predicts that the number of vacant lattice sites is approximately $(E + a)/2a$, where $a$ is the energy required to displace an atom from its lattice site. Thus the energy used in producing vacancies is approximately half the total energy absorbed by the solid.

Work on the problem for energies $E > \gamma$, where electronic collisions are important, is in progress.

### ENERGY LOSSES OF MOVING CHARGED PARTICLES

The stopping power of a particle having charge $Ze$ and moving with velocity $v$ has been determined by Bohr for $K = 2Ze^2/\hbar v \gg 1$ and $\eta_s = 2v/u_s \gg 2$, where $u_s$ is the orbital electron velocity that is effective in stopping the particle. Bohr's results are formulated differently for $K/\eta_s < 1$ and for $K/\eta_s > 1$ and are based entirely on an orbital picture of the processes involved. In an alternative method that has been developed, the space surrounding the particle track is subdivided into three regions: (1) the region of validity of the Rutherford formula (adjacent to the track), (2) the intermediate region involving large perturbations for which no adequate theory exists at present, and (3) the region of validity of the quantum perturbation theory (the region most remote from the track). By extrapolating the formulas valid in regions 1 and 3 into region 2, the contribution to the stopping power that is due to the region 2 has been evaluated; and by adding the contribution resulting from regions 1 and 3 to region 2, an expression has been derived for the total stopping power that is applicable for any $K \gg 1$ and $\eta_s \gg 2$. The physical picture used for deriving this expression is somewhat similar to that used by Bohr for $K/\eta_s < 1$, but it is different from the one used by Bohr for $K/\eta_s > 1$. As an alternative of Bohr's formula for the stopping number for $K/\eta_s > 1$, namely

$$\sigma_s = \frac{2\pi Z^2 e^4}{mv^2} \ln \frac{\eta_s^3}{K} ,$$

the following expression is proposed:

$$\sigma_s = \frac{2\pi Z^2 e^4}{mv^2} \left[ \ln (\lambda^2 \eta_s^2 + 1) + 2 \left(\lambda \frac{K}{\eta_s}\right) K_0 \left(\lambda \frac{K}{\eta_s}\right) K_1 \left(\lambda \frac{K}{\eta_s}\right) \right] ,$$

where $\lambda$ is a known function of $K$ and $\eta_s$.

### COOPERATIVE PROGRAM WITH SHIELDING GROUP

During the 1953 Summer Shielding Session at ORNL, it was necessary to solve a neutron transport problem for design purposes. An approximate analytical solution which avoids the numerical work involved in solving the problem by the multigroup approach has been worked out. The details have been published elsewhere.[1]

---

[1] E. P. Blizzard and H. G. Goldstein (ed.), ORNL-1575 (classified).

# PHYSICS OF NUCLEAR RADIATION

G. S. Hurst
T. E. Bortner
W. G. Stone
W. A. Mills
T. A. Barr, Jr.
L. W. Cochran

## NEUTRON DOSIMETRY

In the neutron dosimetry studies, tissue-depth dose curves have been obtained with a phantom (dimensions: 6 by 6 in. and 12 in. high) and Po-B and Po-Be neutron sources. The ratio of the dose at surface to the dose in air without the phantom is 1.1, and the depth-dose curve drops by $1/e$ in approximately 10 cm. Complete results will be reported by W. A. Mills, who will offer the work as a thesis for the Master of Science degree from Vanderbilt University.

A project involving the measurement of a fast neutron spectrum has been completed. Details will appear in a classified report.

## IONIZATION BY ALPHA PARTICLES

The average number of electron volts used in producing an ion pair in various gases has been measured for alpha particles from plutonium. Values for pure gases and mixtures are given in Table 1.

TABLE 1. VALUES OF $w$ FOR PURE GASES AND MIXTURES

| GAS | $w$ (ev per ion pair) |
|---|---|
| Nitrogen | 36.3 |
| Argon | 26.4 |
| Ethylene | 28.0 |
| Hydrogen | 37.0 |
| Freon | 29.2 |
| Sulfur hexafluoride | 34.8 |
| Boron trifluoride | 36.2 |
| Tissue-equivalent gas* | 30.8 |
| Air (both dry and 50% RH at 24°C) | 35.0 ± 1% |
| Freon 50%, $SF_6$ 50% | 32.2** |

*Mixture used by Rossi-Failla to obtain soft-tissue-equivalent gas: $CH_4$, 64.4%; $CO_2$, 32.4%; $N_2$, 3.2%.

**Calculations based on energy division according to stopping powers, partial pressure, and $w$ values for pure gases gave 32.8 ev per ion pair.

The value for He, 46.0 ev per ion pair, is approximately 4 ev higher than the value measured by other investigators. If helium is contaminated with less than 1% of argon, nitrogen, hydrogen, or methane, the value of $w$ drops in each case to about 30 ev per ion pair, which is the value usually found in the literature. A complete report on the value of $w$ for many gas mixtures is now in preparation.

## IONIZATION BY RECOIL ATOMS

The ionization produced by the recoil uranium atoms from the decay of $Pu^{239}$ ($Pu^{239} \longrightarrow He^4 + U^{235}$) is being studied with the use of a proportional counter whose length is large compared with its diameter and which has a collimated well in the wall for the plutonium source. The counter is filled with various gases at such a low pressure that the uranium atoms lose all their energy in the counter, whereas the alpha particles travel across the counter and dissipate their energy largely in the wall and thus produce a very small pulse in the counter. Another alpha emitter which produces alpha particles that travel the long length of the counter serves as a calibration source.

## ATTACHMENT COEFFICIENT FOR ELECTRONS

The equipment for measuring the electron attachment coefficient (that is, the probability that in moving 1 cm in a gas, an electron will be attached to an atom or a molecule) is now being assembled and tested.

## APPLIED DOSIMETRY

The irradiation of primates at the Bulk Shielding Facility, the first phase of the joint project of the USAF and the Health Physics Division, has been completed and a classified report is being prepared.

The participation of the Health Physics Division in the phase of the program at the University of Texas has been mainly that of assisting in the design and construction of the irradiation facility and of the fast neutron dosimeters of the proportional-counter type.

# RADIATION MEASUREMENTS

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

## URANIUM PROSPECTING[1]

The instrumentation for airborne prospecting has been redesigned, and duplicate instruments to equip the second aircraft have been constructed. In the redesign, particular attention has been given to eliminating temperature effects. A speaker system has been added so that an audio signal is given when the radiation reaches a prescribed level; the pitch of the signal increases as the radiation increases. This makes it possible to locate the position on the ground from which the peak radiation is received without having to refer to the recorder chart. An additional chart-marking system is included in the revised instrument: one set of edge markers gives edge marks at 12-sec intervals, and another set on the opposite side of the record automatically marks every tenth edge mark; every hundredth edge mark is indicated by a wide mark.

When the equipment is installed in the second aircraft, a series of flight tests over sources of various energies will be made for determining the optimum shielding of the scintillation detectors. When the aircraft is operated at 500 ft above the ground and large areas of low activity are scanned, small areas of high grade ore are difficult to locate. In order to improve the detection of small areas, various methods of collimating the detectors with lead shielding have been tried. Experiments performed on the ground indicate that little advantage can be gained unless the detectors are limited to the harder radiation, which will thus reduce the scattered radiation component. This means reducing the counting rate from the source by about one-half; however, since the error that results from statistics is still small compared with variations in background, the use of lead shielding appears to be advantageous. Further test flights over a simulated ore outcrop of 60 tons of carnotite ore at Grand Junction, Colorado, are scheduled.

Flights were made over regions of fall-out from the spring tests in Nevada. Altitude curves showed an exponential variation of radiation intensity with height which compare favorably with curves reported previously.[2] The half thickness of air in the earlier report was 260 ft for fall-out that was 13 days old; the recent curve gives the half thickness as 310 ft for fall-out that was two days old. These data are as would be expected, since the fission products with shorter half lives emit harder radiations.

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

[2] F. T. Davis and P. W. Reinhardt, *H-P Quar. Prog. Rep. Jan. 20, 1952 to July 20, 1952*, ORNL-1353, p. 14.

# RADIATION DOSE

H. H. Hubbell

A. G. Barkow
R. D. Birkhoff
A. W. Blackstock
J. S. Cheka
M. J. Cook
M. R. Ford
A. F. Gabrysh
E. D. Gupton
P. N. Hensley
K. Z. Morgan
F. H. W. Noll
H. K. Richards
M. Slater
A. W. Smith
I. Tipton

## INTERNAL DOSE

### Maximum Permissible Concentrations

The previously published values[1] of maximum permissible concentrations of radioisotopes were submitted to the Subcommittee on Permissible Internal Dose of the National Committee on Radiation Protection. The conclusions of this subcommittee were published on March 20, 1953, by the National Bureau of Standards.[2] In the light of the recent data, some of the values are being reconsidered and may be revised. Further developments will be published soon.[3]

The Committee on Permissible Dose for Internal Radiation of the International Commission on Radiological Protection held a meeting in Copenhagen, Denmark, July 13 to 17, 1953. A report entitled *Maximum Permissible Internal Dose* is being prepared and will be submitted for publication by January 1954.

### Maximum Permissible Concentration Experiments on Mice

An experiment is in progress to determine which is the critical organ for $Co^{60}$, to determine the ratio of $Co^{60}$ in the critical organ to that in the whole body, and to check the conclusion that a continuous dose of $2 \times 10^{-2}$ $\mu c$ of $Co^{60}$ per cubic centimeter in drinking water consumed for a lifetime will not build up to give a weekly dose to the critical organ that is greater than the maximum permissible dose. It is also hoped that it will be possible to determine the percentage of cobalt that passes from the gastrointestinal tract to the blood. With the cooperation of the Biology Division, a number of C-57 strain mice are being given individual doses of 32 $\mu c$ of $Co^{60}$ directly into the gastrointestinal tract, while another group is being given drinking water containing $2 \times 10^{-2}$ $\mu c$ of $Co^{60}$ per cubic centimeter. The subsequent concentrations of the element in the various organs are to be determined and the dose rate to the organ will be calculated.

### Spectrographic Analysis of Human Tissue[4]

The Spectrographic Laboratory at the University of Tennessee has completed a preliminary survey of trace elements in human tissue. The results of a semiquantitative estimation of trace elements in the normal tissues of 41 autopsies of human beings are shown in Figs. 1 through 11. The numbers in parentheses, which appear under the chemical symbols of the elements, are the limits of detection in parts per million in ash. The other numbers indicate the number of samples in which the element appeared. When an element appeared in over half the samples, a geometric mean value for the concentration was calculated, and it appears as the shaded portion of the bar. In every case, the dotted line at the top of the bar indicates the highest value that appeared; the dotted line in the shaded portion indicates the lowest value, this being determined only in those instances where the element appeared in every sample analyzed.

---

(1) K. Z. Morgan, M. J. Cook, and M. R. Ford, *H-P Prog. Rep. Jan. 20, 1952 to July 20, 1952*, ORNL-1353, p. 15.

(2) *Maximum Permissible Amounts of Radioisotopes in the Human Body and Maximum Permissible Concentrations in Air and Water*, Handbook 52, National Bureau of Standards (March 20, 1953).

(3) K. Z. Morgan and M. R. Ford, "Developments in Internal Dose Determinations," *Nucleonics* (to be published).

(4) University of Tennessee research and development contract.

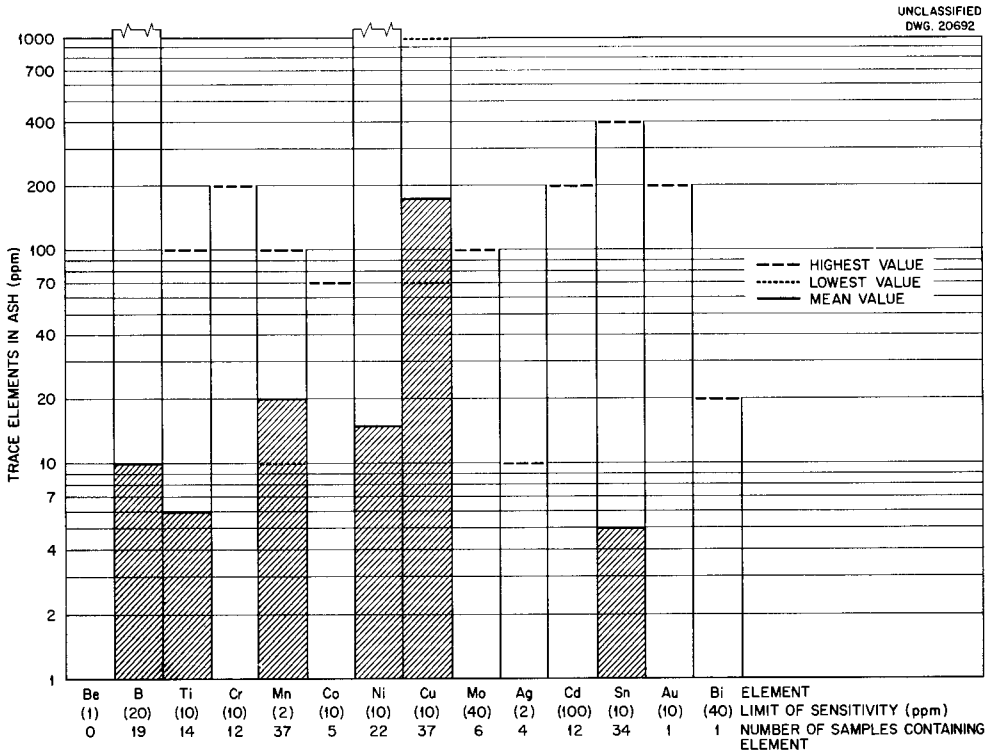


**Fig. 1. Spectrographic Analyses of 37 Samples of Human Muscle for Trace Elements.** Average Ash, 1.1%.

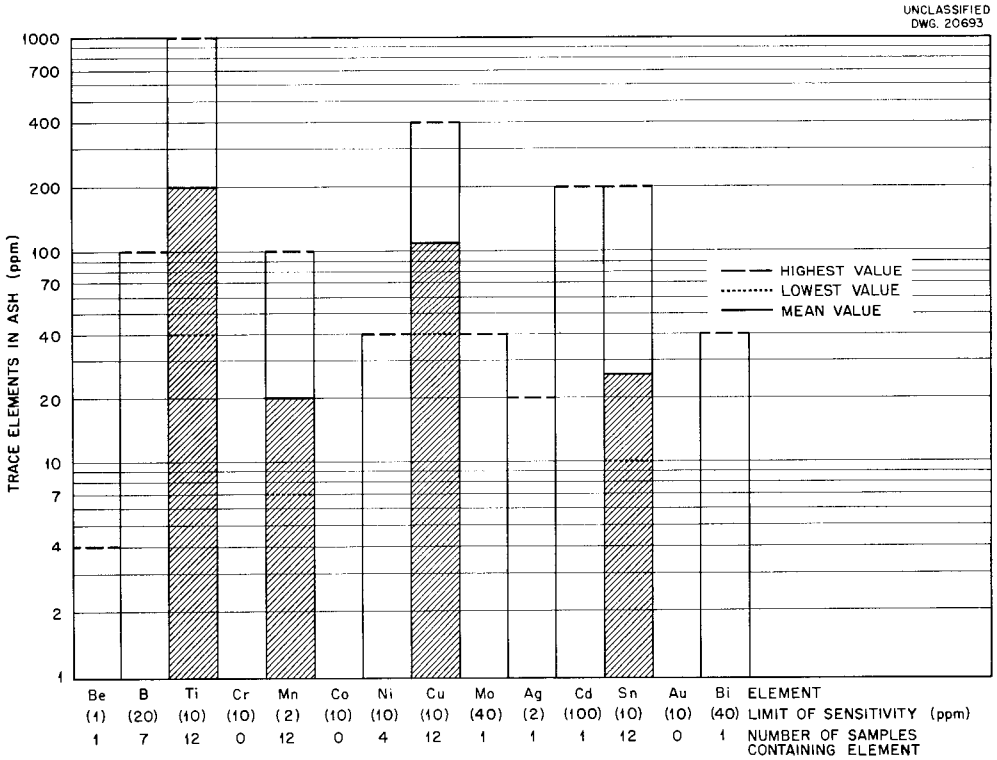


**Fig. 2. Spectrographic Analyses of 12 Samples of Human Lung for Trace Elements.** Average Ash, 1.1%.

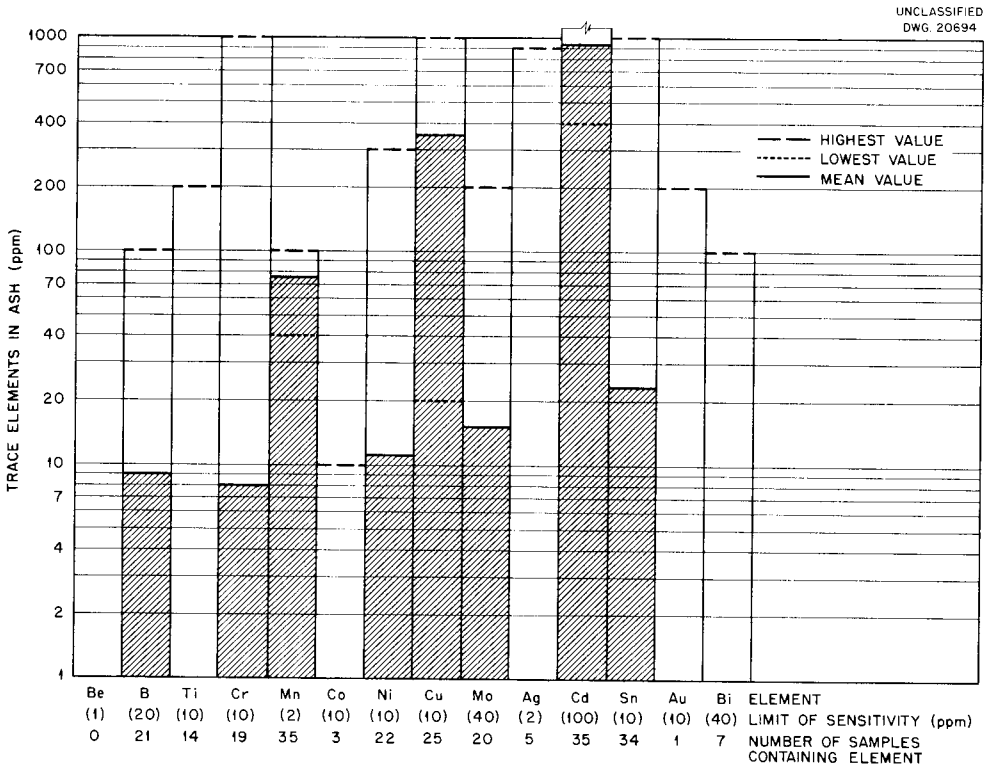


**Fig. 3. Spectrographic Analyses of 35 Samples of Human Kidney for Trace Elements.** Average Ash, 1.1%.

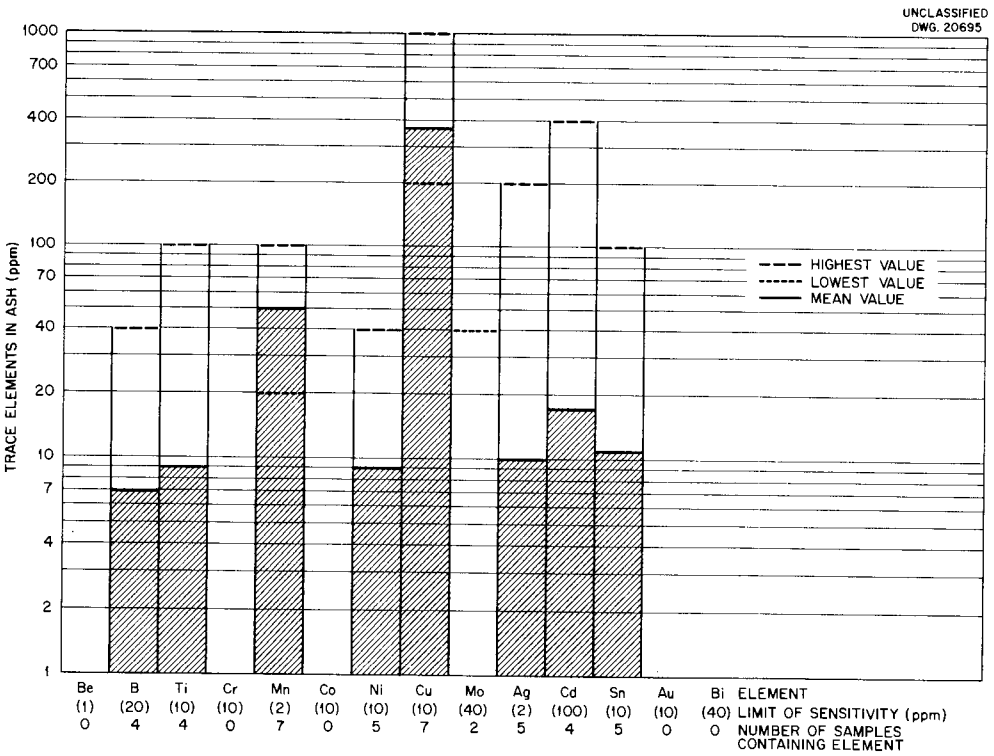


**Fig. 4. Spectrographic Analyses of 7 Samples of Human Intestine for Trace Elements.** Average Ash, 0.7%.

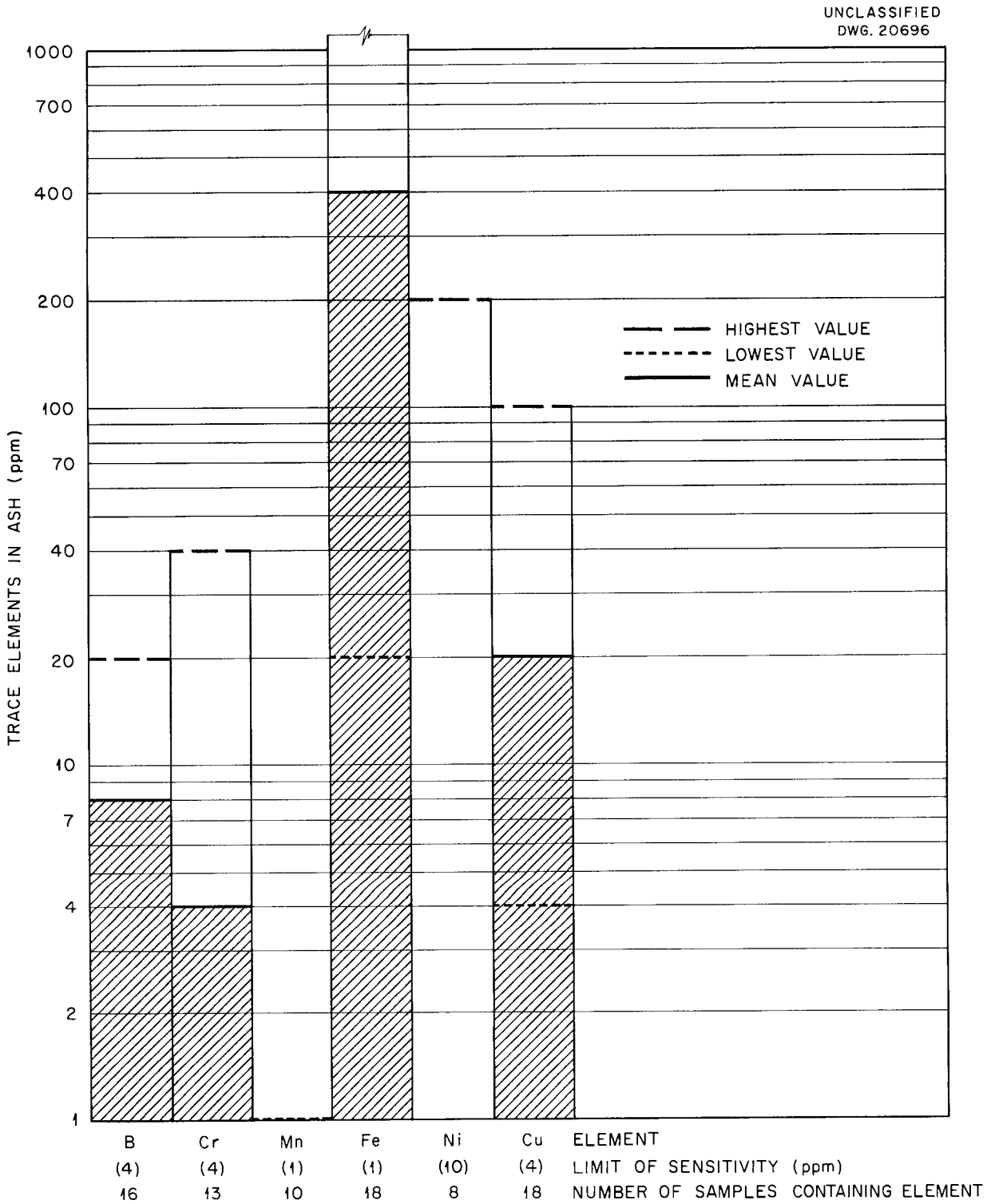


**Fig. 5. Spectrographic Analyses of 18 Samples of Human Cartilage for Trace Elements.** Average Ash, 2.4%.

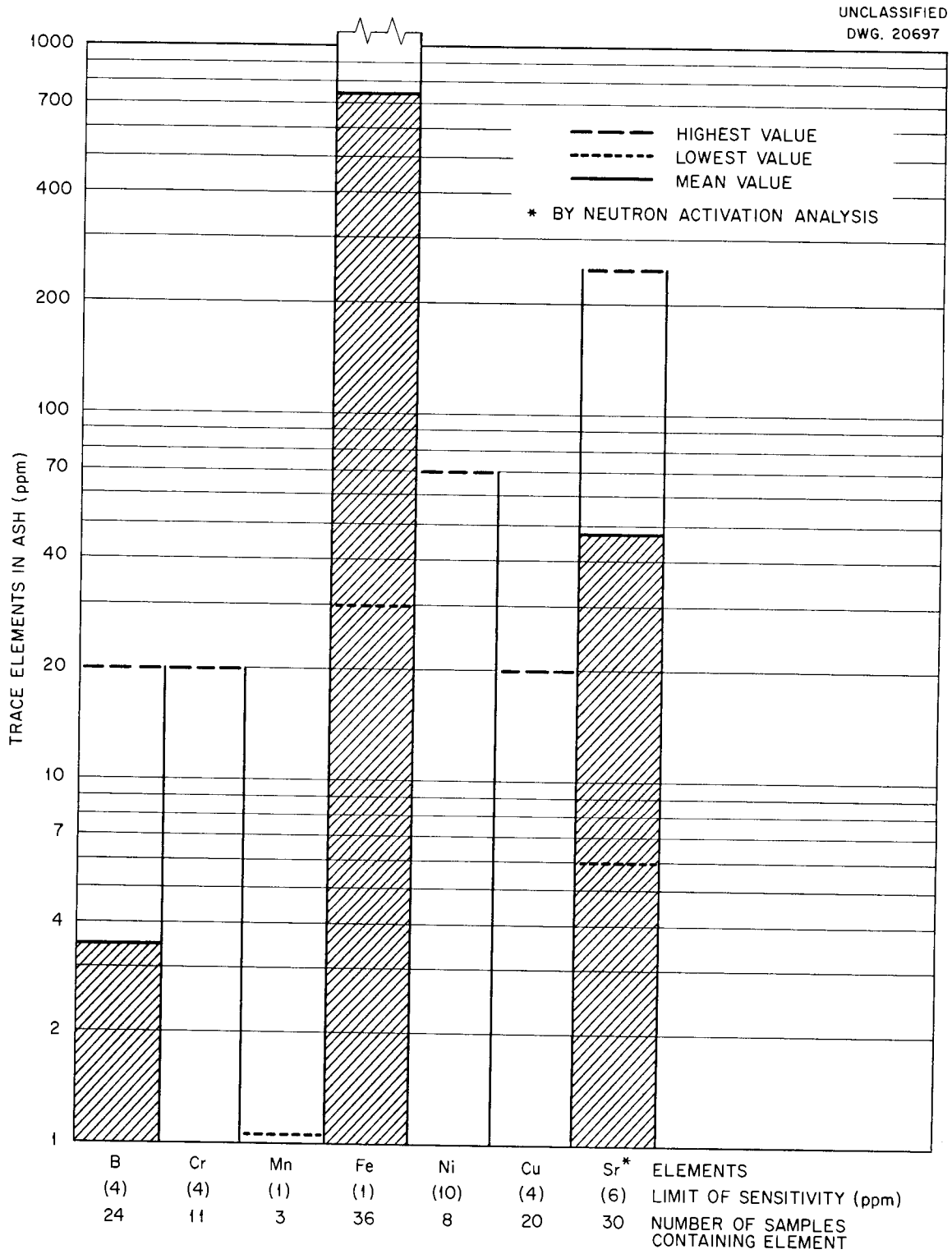


**Fig. 6. Spectrographic Analyses of 36 Samples of Humon Bone for Trace Elements.** Average Ash, 15.7%.

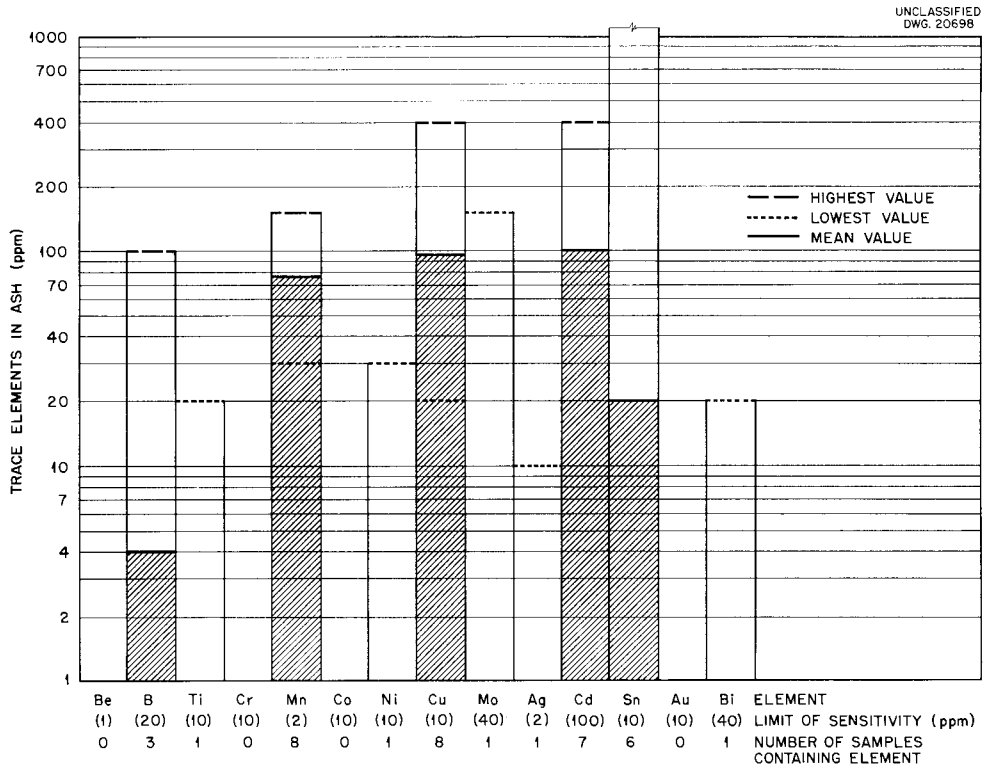


**Fig. 7. Spectrographic Analyses of 8 Samples of Human Pancreas for Trace Elements.** Average Ash, 1.1%.

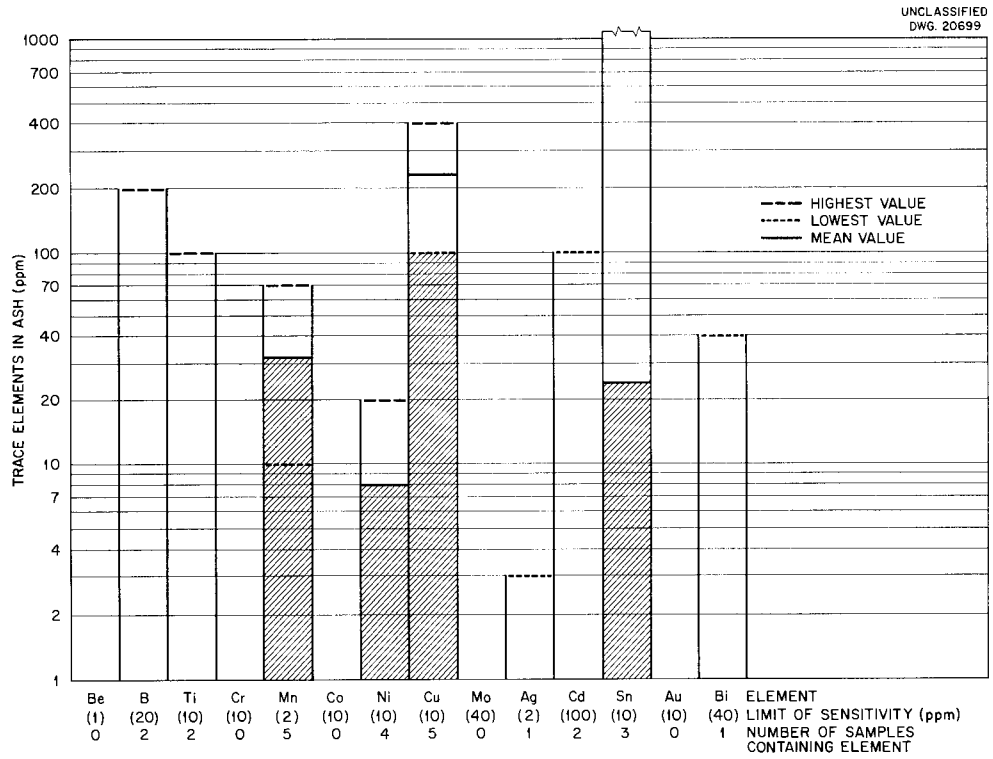


**Fig. 8. Spectrographic Analyses of 5 Samples of Human Stomach for Trace Elements.** Average Ash, 0.8%.

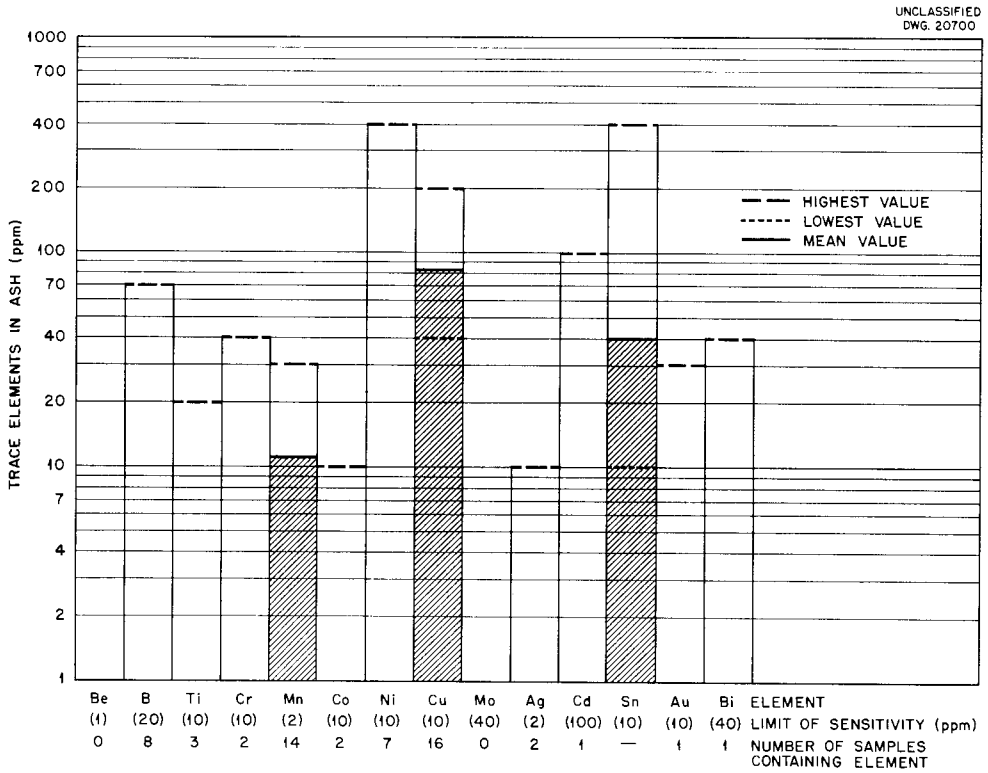


**Fig. 9. Spectrographic Analyses of 16 Samples of Human Spleen for Trace Elements.** Average Ash, 1.4%.

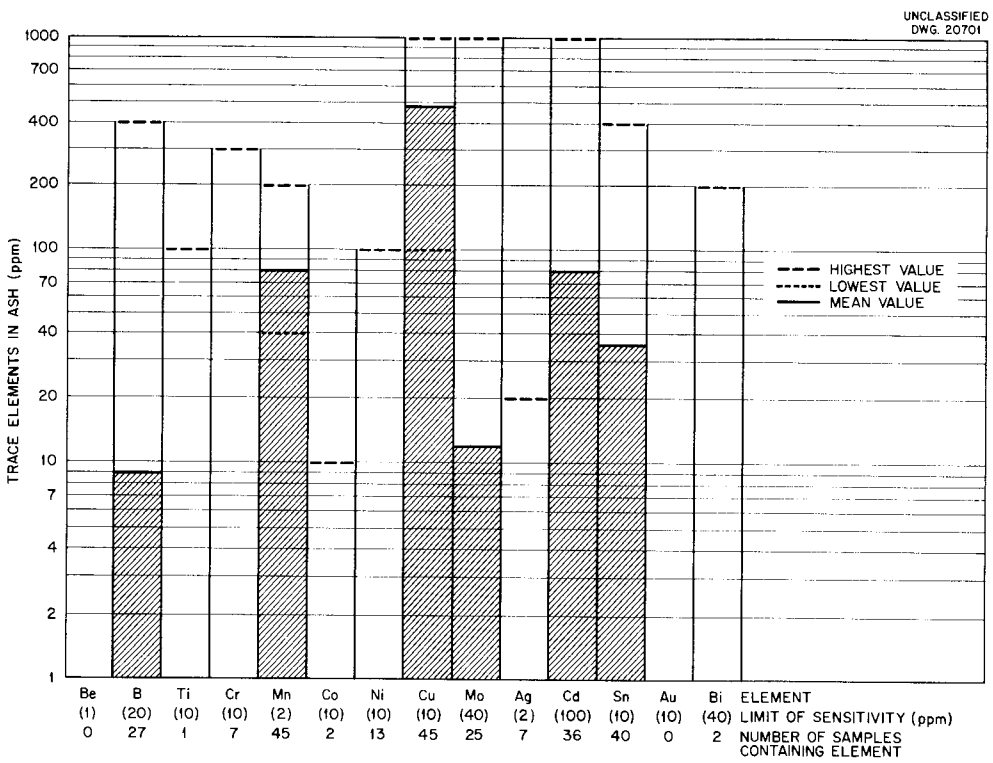


**Fig. 10. Spectrographic Analyses of 45 Samples of Human Liver for Trace Elements.** Average Ash, 1.4%.

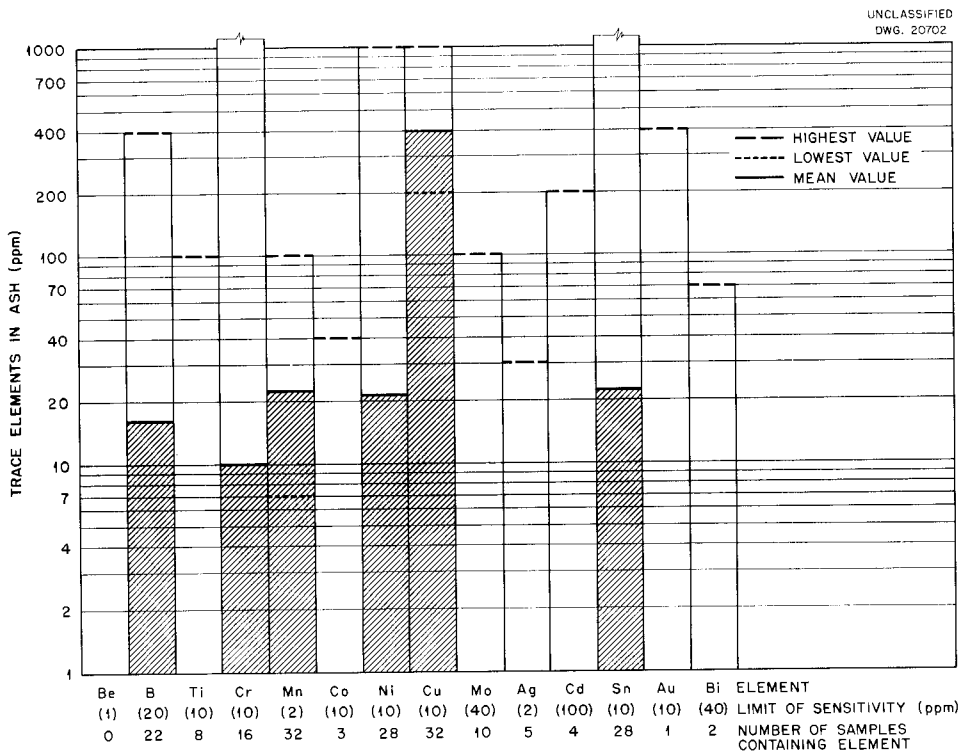


**Fig. 11. Spectrographic Analyses of 32 Samples of Human Heart for Trace Elements.** Average Ash, 1.0%.

Because of the high concentration of potassium and sodium in the ash, the sensitivity for many elements was greatly reduced. Zinc, for instance, could not be measured but was present in nearly every sample.

No vanadium was detected, but it could have been lost in the ashing process. A method for trapping the volatile products of the ashing is being investigated, and, if successful, a complete analysis that includes volatile elements can be made. A detailed report on procedures and the results of the preliminary survey has been written.[5]

### Radiochemical Analysis

Work has continued on the development of an analytical procedure that requires a shorter time for the determination of radioactive strontium in urine than does the present method. Preliminary investigations have been made on the use of Versene (ethylenediamine tetra-acetic acid) for selectively chelating calcium. This would provide a means of separating radioactive strontium from the calcium in urine.

The order of chelation of the alkaline earths by Versene is Ca > Mg > Sr. By controlling the pH of a urine specimen at 6.0 to 6.5 – with an excess of magnesium present – and by allowing the solution to flow over the column of a cation-exchange resin (Dowex-50 in sodium cycle), essentially all the radioactive strontium is taken up on the resin. Some magnesium is also retained on the column, but practically all the calcium will pass through as the chelate.

Subsequent separation of the radioactive strontium by using the procedure already established is made easier by the decreased amount of calcium present. The time required for an analysis may be reduced by a factor of 2 or more.

## EXTERNAL DOSE

### Measurement of Ionizing Radiation by High-Frequency Variation[6]

Experiments were continued on the use of an electrometer as the frequency-controlling element

[5] I. H. Tipton, W. D. Foland, F. C. Bobb, and W. C. McCorkle, *Spectrographic Determination of Trace Elements in Human Tissue*, CF-53-8-4 (Aug. 3, 1953).

[6] H. K. Richards, *Phys. Rev.* 89, 897A (1953).

in a high-frequency circuit. As previously described, the electrometer leaf rotates on the quartz-fiber suspension as the electrostatic charge is changed by ionizing radiation. This turning varies the capacity that is seen by the radio-frequency circuit and thereby changes the frequency. A sensitivity of 0.5 mr/hr or better was observed when an ion-collecting volume of about 100 $cm^3$ was used. As little as 2000 beta particles from $Sr^{90}$ and $Y^{90}$ have been measured.

To utilize the ferroelectric effect to eliminate moving parts, the electrometer was replaced by a capacitance that had a dielectric of barium titanate. The barium titanate was in the form of a cylinder about 6 mm in length and 1 $mm^2$ in cross section that had silver-plated electrodes on the ends. The capacitance was about 25 $\mu\mu f$. This capacitance was connected in series with an ordinary, highly insulated capacitance, and the combination formed a section of a frequency-determining network of a 5-megacycle per second, quartz-crystal-controlled oscillator. The center frequency could be varied by 5 to 10 kilocycles per second, since a coil in series with the crystal broadened the resonance. The connection between the barium titanate and the ordinary capacitance was joined to the inner electrode of an ionization chamber of about 100 $cm^3$ volume and charged to between 300 and 700 volts dc. This polarizing voltage reduces the dielectric constant of the barium titanate and therefore the capacitance; thus the oscillator frequency is increased by several thousand cycles per second. When the ionizing radiation in the chamber discharges the electrode, the polarizing voltage is lowered, the capacitance of the barium titanate is increased, and the frequency is thereby reduced. Reproducible measurements of dose rates from 35 to 500 mr/hr have been obtained. At present, the sensitivity is limited by the finite conductivity of the ferroelectric material and by temperature effects.

### Backscattering of Beta Particles

Calculations of backscattering for 0.6-Mev monoenergetic beta particles showed that for scatterers with low atomic numbers ($Z$) normally incident particles are largely reflected by single scattering up to depths of 100 to 200 $mg/cm^2$. If the loss of particle energy that is due to absorption and the number of particles lost are taken into account, most of the reflected particles, even from elements of medium $Z$, have suffered single scattering. Elements with high $Z$ produce principally multiple scattering.

A plot of the calculated total number of electrons scattered between 90 and 180 deg vs. $Z$ has the general shape which is observed experimentally. As yet, numerical agreement has not been obtained, and multiple scattering will have to be considered.

### Film Monitoring for Fast Neutrons[7]

Work has continued toward making the NTA film badge into a true neutron dosimeter by means of a compensated proton radiator. This radiator consists of laminations of aluminum foil and paper which adjust the film response to neutrons to agree with the dose vs. energy curve. The radiator was constructed to match the film response to W. S. Snyder's multiple-collision dose calculation.[8]

A question remained as to whether the multiple-collision dose calculation was the correct dose curve to use. If the dose was measured in air, this curve would give the dose to a body that is in the same position as the film. A series of exposures was made with and without a 20-cm-dia paraffin phantom that backed the film badges. A film placed 1 cm from the phantom (the separation produced by the badge clip) showed a track population 30% higher than that of a film without the phantom. Since the neutrons reflected from a phantom (or a body) also record on the film, film response matched to a first collision dose curve[9] gives a better approximation to the true body dose than one based on Snyder's curve – the latter gives too high a reading. Consequently, the first collision dose curve was calculated by use of the latest available cross-section data. The lamination thicknesses of the radiator were recalculated to match this curve.

One of the dimensions requiring change is the thickness of the film base. This problem was discussed with representatives of Eastman Kodak Company and, the technical staff at Eastman Kodak

(7) J. S. Cheka, *H-P Quar. Prog. Rep. Oct. 20, 1952*, ORNL-1420, p. 7; *H-P Quar. Prog. Rep. Jan. 20, 1953*, ORNL-1488, p. 3; *Phys. Rev.* 90, 353A (1953).

(8) W. S. Snyder and J. Neufeld, *Maximum Permissible Neutron Flux for Fast and Thermal Neutrons*, AECU-2328 (1953).

(9) This curve was recalculated by using the latest available data. Earlier data were given by K. Z. Morgan in *Health Control and Nuclear Research* (to be published). Morgan's curve was published by G. S. Hurst, R. H. Ritchie, and H. N. Wilson, *Rev. Sci. Inst.* 22, 981 (1951).

is checking on the feasibility of using a thinner plastic base. If a thinner base cannot be produced without curl, a different plastic will be tried.

The microscopy techniques recently reported[6] for counting proton tracks in film have been modified. It was found that with partly degenerate tracks, in the presence of fog, track counts at 440X magnification did not agree with those at approximately 900X. However, if 10X wide-field eyepieces and a 90X oil-immersion objective are used, and if the traverse method of reading is retained, there is still a great advantage because the total area scanned is greater than that obtained in reading discrete fields. The field diameter of 0.019 cm with a traverse length of 1.32 cm gives an area of 0.025 $cm^2$, as compared with the 0.0024 $cm^2$ that is covered by 12 discrete fields in the present method. Statistical error for the permissible exposure of one week is based on about 60 tracks rather than on the 6 tracks used at present.

Since the traverse method of reading film takes about twice as long as the scanning of 12 fields, splitting the group of neutron films and reading half the films on alternate weeks would reduce the monitoring period to two weeks per film. However, under the present circumstances, such a program is not feasible because the proton latent images fade with time. A survey of the literature showed that exposure to 100% relative humidity at 35°C for 6 hr eradicates all latent images, even those of alpha tracks. A test was run to determine whether desiccation would prevent the rapid fading. A group of films was uniformly exposed to neutrons from a polonium-boron source and then divided into two sets. Both sets were stored at 24°C, one at a relative humidity of about 50% and the other in a desiccator over Drierite. Specimens of each set were developed at intervals from day zero to about two months. The set stored at 50% relative humidity showed 88% track retention after 11 days and 50% retention after 33 days. The desiccated set showed 95% retention after 21 days, 88% retention after 30 days, and 50% retention after 64 days. Previous tests have shown that under normal conditions at Oak Ridge, track retention is about 88% in seven days. Allowance is made for this in the present method of calibration.

In order to take advantage of low relative humidity to inhibit fading, a new sealed badge that contains a desiccant should be designed for NTA film or the films should be desiccated and sealed individually in aluminum foil during manufacture. The latter may be more feasible, since the proposed dosimeter packet requires layers of aluminum.

### X-Ray Shielding Room

Measurements were made on the attenuation of the beam from the 250-kv x-ray machine in samples of ordinary concrete block and of barytes-loaded concrete block. On the basis of these measurements, it was calculated that the dose rate in the direct beam could be reduced to about 7 mr/hr at a distance of 7 ft from the target by about 16 in. of ordinary concrete, 4.5 in. of barytes concrete, or $\frac{3}{8}$ in. of lead. Barytes concrete was used in the construction of a shielding room for the x-ray machine because it provided, at the least cost, walls that are not unreasonably thick. The x-ray tube and stand were moved inside, and the controls remain outside. In order to allow a safety factor, the walls were made 6 in. thick and the ceiling was covered with $\frac{3}{16}$ in. of lead to absorb scattered radiation. The door and window were lead-sheathed, and baffles were placed in the vents. Preliminary measurements indicate that the beam is attenuated to 10 mr/hr, even when it is aimed directly at the wall at a distance of 5 feet. The room considerably increases the safety and usefulness of the x-ray machine.

### Film Dosimetry

The new, personnel monitoring and security badge adopted by ORNL was extensively tested for the Applied Health Physics Section. As has been reported,[10] the badge contains filters of 0.040-in. copper, 0.030-in. cadmium, and 0.020-in. lead, and an open window region. The variation in transmission of x rays and gamma rays through filters of different atomic numbers permits the blackening on the film to be interpreted in terms of both the dose and the energy of the radiation. Heavily filtered beams from the x-ray machine and a radium source permitted study of this effect with energy variations from about 20 to 700 kev, and over a dose range from 10 mr to 3000 r. Results for Du Pont type 502 film are shown in Fig. 12.

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[10] D. M. Davis and J. C. Hart, *A Review of Film Dosimetry*, paper presented at the Meeting on Industrial Health sponsored by US–AEC, Los Angeles, California, April 24, 1953.

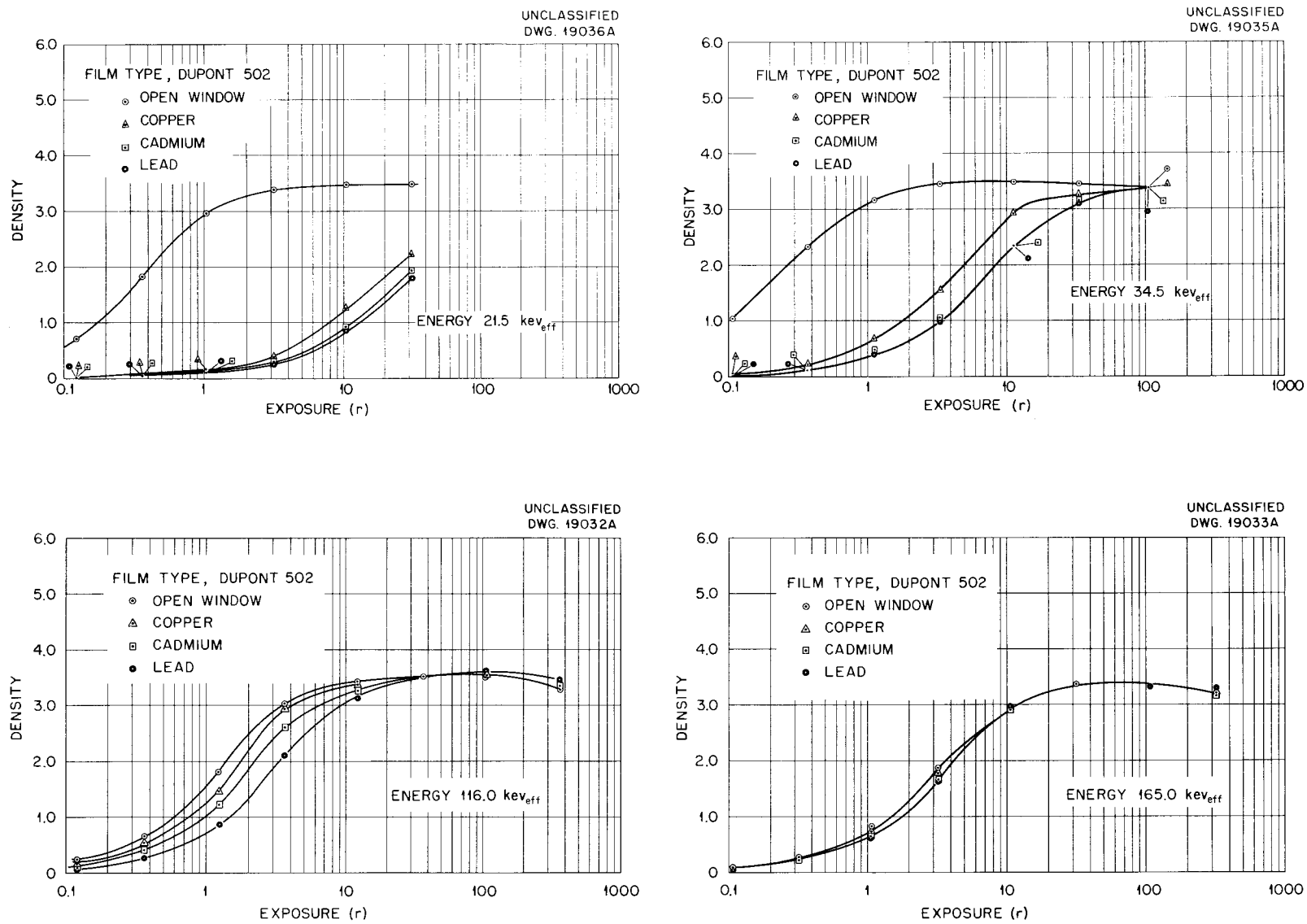


**Fig. 12. Response of Du Pont Type 502 (Sensitive) Film as a Function of Exposure for Different Effective Radiation Energies behind the Four Filtered Areas in the Film Badge.**

### Energy Losses of Electrons in Foils

**Beta Particle Spectrometer.** A paper to be published soon[11] describes experiments on the determination of energy losses of electrons in passing through thin foils of materials of various atomic numbers. The 624-kev $K$ conversion electrons from $Cs^{137}$ were allowed to pass through thin absorbers of beryllium, aluminum, copper, tin, tantalum, and lead. The energy spectra of the transmitted electrons were analyzed with the small, high-resolution, solenoidal, beta spectrometer. Agreement with the theory of Landau was found for the displacement of the peaks of the energy-loss distribution, but the widths of the distributions were about 1.8 times the Landau values. An extension of the simple theory by Blunck and Leisegang treated the resonance transfers of energy to the atomic electrons more completely and predicted wider distributions for the heavy elements. This prediction agreed with the data obtained, but disagreement was still found for the light elements.

The larger beta spectrometer mentioned in earlier progress reports[12] is nearing completion. The fabrication of the vacuum chamber, 10 ft in length and 3 ft in diameter, was completed at K-25, and the instrument has now been installed in Building 2001. Plumbing and electrical work have been substantially completed, and the unit has been vacuum tested. Work is progressing on the magnet-current control and the ripple-correcting circuit. Tests are in progress to determine the geometry of the compensating coils needed to correct the magnetic field to homogeneity within 0.1% throughout the useful volume of the instrument. The slit and vacuum lock systems have been completed, and plans have been drawn up for the detector assembly. A console has been constructed and several control and monitoring units have been installed in it. The unit is designed to analyze electron beams having energies from about 100 kev to 2 Mev.

**Accelerator.** The high-voltage accelerator mentioned in previous progress reports[13] was designed for several purposes. One of the purposes was the study of energy losses of electrons in foils at incident-electron energies from a few kev up to 125 kev; thus the range of the beta spectrometer was extended and overlapped. By first accelerating the electrons, then passing them through the foil, then slowing them down to nearly zero energy before collection, it appeared possible to obtain an energy resolution of a few parts per million.[14] However, when very slow electrons approached the collector, the beam defocussed so badly that the high resolution was lost. Numerous attempts to prevent the loss of resolution by the addition of grids and magnetic focussing coils were unsuccessful. Therefore an electrostatic analyzer has been built to resolve the beam after it has been decelerated to 100 volts, but is still well focussed.

Recent work on the accelerator has included completion of a high-voltage transfer switch, a precision high-voltage voltmeter, and controls to permit the use of the high-voltage supply for the x-ray machine to operate either the x-ray tube or the accelerator.

### Special Tests

At the request of the Radiation Instruments Branch of the AEC in Washington, a classified project involving the calculation of an expected induced gamma-ray dose and its experimental verification was attempted during the recent Nevada tests with the cooperation of the ORNL Physics of Nuclear Radiations Section of the Health Physics Division.

### Standardization of Radiation Monitoring Film Sensitometry

At the request of the Photographic Sensitometry Committee of the American Standards Association, ORNL has been represented for the past two years on the Subcommittee on X-ray Sensitometry. This subcommittee has prepared standards for determining the speed and contrast of medical x-ray film and of film for industrial radiography. A standard for determining the useful range of film for radiation monitoring has been the subject of repeated discussions and some experimental work, particularly at ORNL and at the National Bureau of Standards. The draft of the standard has undergone several revisions, and it is hoped that a final form acceptable to the representatives of the photographic industry and other interested parties may be completed soon.

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(11) R. D. Birkhoff and F. Kalil, *Phys. Rev.* 91 (to be published).

(12) R. D. Birkhoff and A. W. Smith, *H-P Prog. Rep. Jan 20, 1952 to July 20, 1952*, ORNL-1353, p. 9.

(13) R. D. Birkhoff, A. W. Smith, and J. Bergstein, *H-P Quar. Prog. Rep. Oct. 20, 1952*, ORNL-1420, p. 7; R. D. Birkhoff, J. Bergstein, M. Slater, and H. H. Hubbell, *H-P Quar. Prog. Rep. Jan. 20, 1953*, p. 4.

(14) J. Bergstein and R. D. Birkhoff, *Phys. Rev.* 91, 223 (1953).

# AIRBORNE RADIOPARTICULATE CONTAMINATION

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## RADIOACTIVE-DUST STUDIES AND PEEP

### 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, resumed a study of radioparticulate air contamination at X-10. The immediate objective was to evaluate the degree of airborne contamination produced by all the research and production operations in the Laboratory area.

The data collected over a period of four years by ten continuous air monitors stationed throughout the Laboratory area were reduced to histograms. Each histogram shows the number of particles per 1000 $ft^3$ of air collected weekly by a continuous air monitor (CAM) during the period of March 1949 to December 1952. The histogram (Fig. 13) for the average of the same ten monitors shows a general upward trend in the number of particles per 1000 $ft^3$ during the four-year period as follows:

| | 1949 | 1950 | 1951 | 1952 |
|---|---|---|---|---|
| Particles per 1000 $ft^3$ | 0.16 | 0.55 | 0.56 | 3.35 |

The increase in the number of particles per 1000 $ft^3$ collected each year is ascribed to the greater amounts of radioactive materials handled and to the new processes placed in operation.

A log was compiled of all ORNL process and reactor operations. A preliminary survey indicates that a significant correlation exists between the operation of certain of these processes and the peaks of the histograms.

As a start in isolating a single source of radioparticulate contamination, one particular operation was selected for study during a complete cycle of operation. Four CAM stations were chosen on the

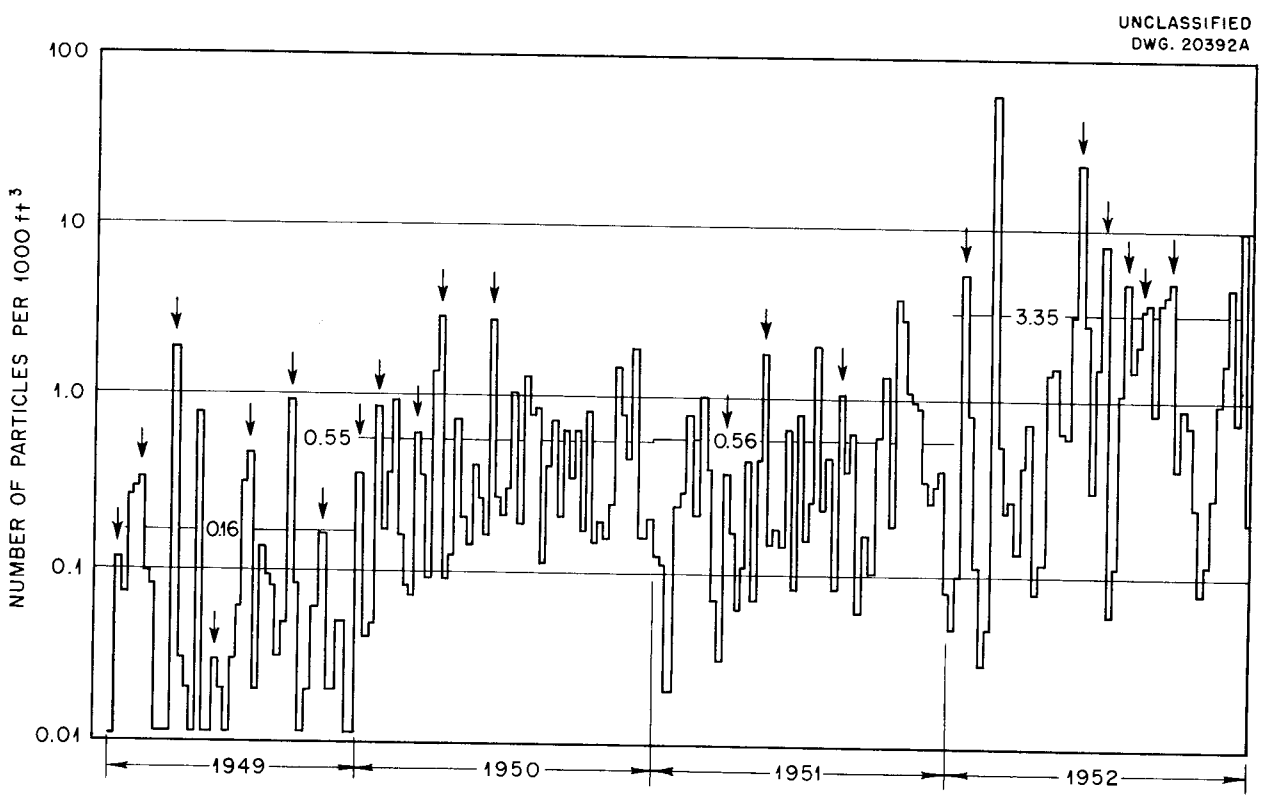


**Fig. 13. Trend of Particle Concentration at ORNL from 1949 to 1952.**

basis of wind direction, and collections were made on Hollingsworth and Vost Co. No. 70 filter papers. Millipore (molecular) filters were also set up at the four stations, as well as in the process area, and collections were made continuously throughout the process run. Off-gas and ventilation air from the process were sampled by cascade impactors and by millipore filters before they were discharged into the atmosphere. Fall-out trays were located about the X-10 area, and the roof of Building 2001 was scrubbed with water and the water was collected in a tank. Radiochemical analyses have been made of representative samples. CAM and millipore filters have been autoradiographed, and the number of particles on each filter has been counted. Work is continuing on the identification of the collected particles and all data, including weather conditions, are being correlated with the operational stages of the process.

### Activity Escaping From the Graphite Reactor Filter House

The primary objective of the work was to determine whether the filter house is effectively cleaning the effluent from the ORNL graphite reactor. A secondary objective was to investigate the value of activated carbon in removing radioactive xenon and krypton from the effluent gases.

A flow-type ionization chamber that was equipped with a vibrating-reed electrometer and a recorder[1] was used to sample the effluent gases. Chemical Corps No. 6 filter paper was used to sample the particulate matter. It was found that if alpha contamination is present on the filter papers, it corresponds to an activity of less than $5 \times 10^{-12}$ $\mu c/cm^3$[2] of the effluent gases.

When the filter house effluent is passed directly into the flow ion chamber, the ion current produced in the 1-liter ionization chamber is about $1.0 \times 10^{-12}$ ampere. For a 0.17-Mev $\beta^-$ emitter, this is equivalent[1] to $1.5 \times 10^{-4}$ $\mu c/cm^3$ of effluent. However, most of this activity is due to $A^{41}$ (1.2-Mev $\beta^-$), and a smaller amount is due to krypton and xenon and the decay product of these gases. The ion chamber has not been calibrated for this higher $\beta^-$ energy. An order-of-magnitude estimate is that $10^{-3}$ to $10^{-4}$ $\mu c/cm^3$ of $\beta^-$ and $\gamma$ activity is present in the effluent less than 1 min after it has left the filter house.

Several experiments were made to estimate the effectiveness of activated carbon in removing argon, krypton, and xenon from the effluent. The flow rate was 10 liters/min through a 300-ml carbon bed. In the first of two techniques used, the filter-house effluent was passed directly into the ionization chamber, a sample was isolated in the chamber, and the decay-rate was observed. In the second technique, the experiment was repeated except that the effluent was passed through an activated-carbon layer at room temperature and then passed into the ionization chamber for observation of the decay rate. The second procedure was varied in that the carbon layer was cooled to −170°C. The results are shown in Fig. 14.

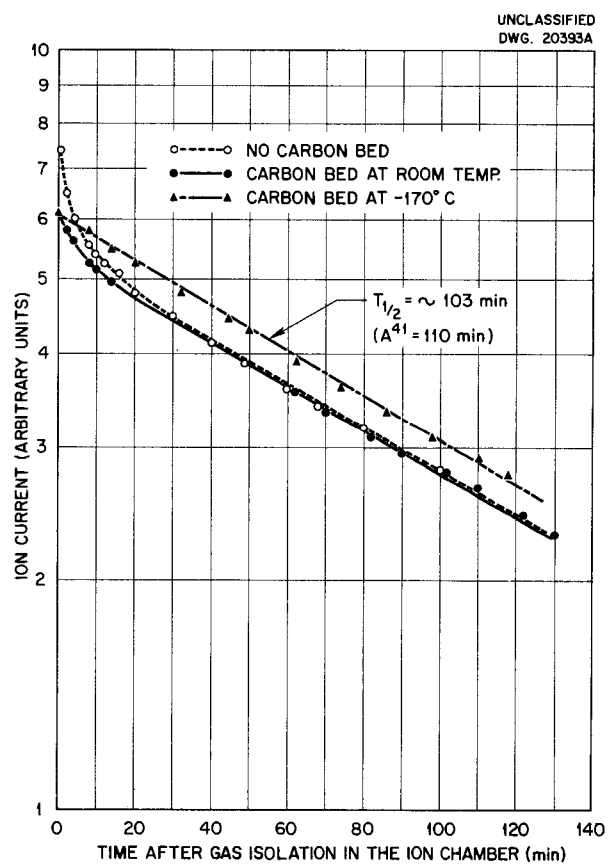


**Fig. 14. Activity Decay of Effluent from Filter House.**

[1] J. W. Thomas and R. L. Quinn, *Calibration of a Flow-Type Ionization Chamber with Radioactive Sulfur Dioxide*, Y-793 (Aug. 20, 1951).

[2] *Maximum Permissible Amounts of Radioisotopes in the Human Body and Maximum Permissible Concentrations in Air and Water*, Handbook 52, National Bureau of Standards (March 20, 1953), gives $5 \times 10^{-12}$ $\mu c/cm^3$ as the permissible concentration of assay alpha-emitting radioisotope in the air beyond the area of control for short periods of exposure. The value given for the beta emitter $A^{41}$ is $5 \times 10^{-7}$ $\mu c/cm^3$.

The second technique was similar to that used by Coughlen[3] and Spiewak.[4] The filter-house effluent, after passing through the charcoal bed, was passed through four No. 6 filter papers in series. The first two papers were placed adjacent to each other, as were the last two papers. There was a distance of 66 cm between the two sets of papers, which corresponded to a flight time of 2 sec. This arrangement detects the radioactive daughters of krypton and xenon. Krypton and xenon penetrate the first two papers as gases, decay to particulate daughters enroute to the second set of papers, and are collected by the second set of papers. Three experiments were made in which there was (1) no carbon bed in the line ahead of the filter papers, (2) a carbon bed at room temperature ahead of the papers, (3) a carbon bed at −78°C ahead of the papers. The results indicated that even at room temperature the carbon bed was removing a large fraction (~50%) of the krypton and/or the xenon, and that at −78°C essentially all the krypton and the xenon were removed from the filter house effluent. (Argon is known to condense at −186°C.)

The results are confirmed in a general way by the decay curves of Fig. 14. Although no experiment was made at −78°C in which the decay-rate technique was used, it is apparent from the curves that the carbon bed removes a large fraction of the very short-lived krypton and/or xenon isotopes at room temperature, and that with the carbon bed at −170°C, there appears to be very little activity present, except $A^{41}$.

Because of the limited scope of this investigation, the results are not conclusive, but the following two points seem to be fairly well established:

1. The filter house is at present operating efficiently, and little or no alpha contamination passes through. The activity that does escape from the filter house (~$10^{-3}$ to $10^{-4}$ $\mu c/cm^3$) has a short half life and is largely due to $A^{41}$, xenon, and krypton.
2. A carbon bed at −78°C will efficiently remove krypton and xenon from the filter house effluent.

[3] C. P. Coughlen, *Determination of Potential Sources of Area Atmospheric Radio-Active Contamination*, ORNL-677 (June 8, 1950).

[4] I. Spiewak, *Notes on Pile Filter House Efficiency*, M-4393 (Aug. 17, 1949).

**Cloud Chamber for Measuring Aerosol Concentration**

One method of measuring the concentration of otherwise invisible particles in a gas is to use a cloud chamber of special design. The chamber creates a supersaturated state of the vapor. The particles act as nucleation centers for liquid droplets and thus the apparent diameters are enlarged a thousandfold or more so that they can be seen and counted.

In the operation of the cloud chamber of H. L. Green, which was loaned to this section by the Ministry of Supply of Great Britain, a beam of light from a zirconium arc passes through the chamber. The cross section of the beam is defined by a slit that is imaged at the center of the chamber by a microscope objective. Thus only droplets occupying a known volume are illuminated. The droplets are photographed at 90 deg to the beam path. A count of the droplet images on the film gives the particle concentration of the aerosol in the expanded chamber.

For a concentration of $10^5$ particles per $cm^3$, about 40 droplets are photographed at a time. In order to obtain a statistically valid count, many photographs of the aerosol sample must be taken; therefore a repeating or so-called "continuous-action" cloud chamber is useful. The Green cloud chamber is of this type. A fresh sample of aerosol is drawn into the chamber every 2 sec, and a cam-operated piston creates an expansion during the cycle. With this apparatus, the shutter must be cocked and released and the film changed by hand for each cycle.

Inasmuch as the Green apparatus is being returned, it was decided to build a similar apparatus for use in future studies. The apparatus has been designed and is partly built. A 16-mm motion picture camera will be used to obtain photographs of the droplets. One of the most objectionable features of the Green apparatus is that the droplets are in motion and can be stopped only by introduction of a current of air into the system; however, tests show that a G-E flash tube (type FT-230) is sufficiently intense (200 watt-sec) and fast enough ($10^{-4}$ sec) to "stop" the droplet motion and yet give adequate exposure. The new machine will use this type of illumination, synchronized with the camera shutter.

Provision has been made so that either of two types of cloud chambers can be used in the apparatus. In the first, a *volume-defined* cloud chamber,

a rotating valve (1) admits aerosol into the chamber, (2) closes the chamber, (3) expands the chamber by allowing vacuum to operate a sylphon bellows, and (4) removes the aerosol. In the second, a *pressure-defined* cloud chamber, a cam actuates three solenoid valves through the same cycle, except that in step 3 the aerosol expands into an evacuated chamber.

### Atomic Weapons Test Fall-Out at ORNL

Five radioactive particles collected at ORNL on March 19, 1953 during the Nevada atomic weapons test fall-out were examined as to appearance, size, and activity.

All the particles examined were spherical and had the appearance of having been formed from molten material. All appeared to be black and opaque, with the exception of one which was glassy and transparent. Their diameters ranged from 18 to 24 microns. There was no correlation between the activities of the particles and their sizes. The beta activities ranged from approximately $1.1 \times 10^3$ to $1.7 \times 10^5$ dpm. Half-life and absorption measurements showed that the particles did not contain the same relative amounts of radioactive components.

A detailed account will be found in a memorandum to be issued by this section under the title *Atomic Weapons Test Fall-Out at ORNL on March 19, 1953.*

## RADIOACTIVE WASTE DISPOSAL RESEARCH

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### WATER AND LIQUID WASTE DECONTAMINATION PROCESS

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### Water Decontamination Studies

A re-examination of the data obtained in an earlier $3^4$ factorial study in which the effects of four parameters (three levels of each) were investigated simultaneously indicated that it would be possible, by setting up a $\frac{1}{3}$ replication of the $3^4$ factorial experiment, to reduce the number of samples evaluated from 243 to 54 and still observe the effects of the variables. By utilizing this new procedure, coagulation studies were made in which the effects of the day-to-day variations, activity levels (approximately 200, 2000, and 20,000 counts/min/ml), coagulants used ($FeSO_4$, $FeCl_3$, and alum), and coagulant dose (1, 2, and 6 gr/gal) on the removal of specific radioisotopes was noted. The radioisotopes investigated included $P^{32}$, $Sc^{46}$, $Cr^{51}$, $Sr^{89}$, $Y^{91}$, $Zr^{95}$-$Nb^{95}$, $Mo^{99}$, $Ru^{103}$, $I^{131}$, $Cs^{137}$-$Ba^{137}$, $Ba^{140}$-$La^{140}$, $Pr^{142}$, $Ce^{144}$-$Pr^{144}$, $Pm^{147}$, $Sm^{153}$, $W^{185}$, $Re^{186}$, and a fission product mixture; all were obtained from the Operations Division of ORNL. The results are summarized in Table 2.

In a series of experiments with the use of an electrolytic ion-exchange cell in which permselective ion-exchange membranes were employed to separate the central chamber of the cell from the anode and cathode compartments, it was found that more than 99% of mixed-fission-product activity could be removed from distilled water at flow rates of 5 ml/min. In other studies with $Cs^{137}$ and $I^{131}$ (singly and in combination), removals of 99.5% were obtained with $Cs^{137}$, and removals of about 95 to 96% were obtained with equal mixtures of $Cs^{137}$ and $I^{131}$ activities.

A mixed-fission-product waste solution was passed through a Greensand (Zeo-Dur) ion-exchange column to the point of breakthrough of hardness and radioactivity; then the column was regenerated with 20% NaCl. Six such cycles have been studied. Prior to the seventh pass, the bed was exhausted with $CaCl_2$; then the mixed radioactive waste solution was passed through the bed. The four results obtained thus far show: (1) that an average of about 90% of the activity is removed to the hardness breakthrough, (2) that 20% NaCl may be used for regeneration, (3) that about 50% of the adsorbed activity is removed by regeneration, and (4) that with a calcium-exhausted bed, removals of about 70 to 80% of the activity may still be obtained.

TABLE 2. COAGULATION RESULTS – $3^4$ FACTORIAL EXPERIMENTS

| RADIOISOTOPE | COAGULANT DOSE[a] (gr/gal) | REMOVAL (%)[b,c] | | |
|---|---|---|---|---|
| | | Low | Intermediate | High |
| | | Coagulant: $FeSO_4$ | | |
| $P^{32}$ | 1 | 90-95 | 95-99 | 95-99 |
| | 2 | 95-99 | 95-99 | 95-99 |
| | 6 | 95-99 | 95-99 | $99^+$ |
| $Sc^{46}$ | 1 | 75-90 | 95-99 | 95-99 |
| | 2 | 75-90 | 95-99 | 95-99 |
| | 6 | 75-90 | 95-99 | 95-99 |
| $Cr^{51}$[d] | 1 | 0-25 | 0-25 | 25-75 |
| | 2 | 25-75 | 0-25 | 0-25 |
| | 6 | 0-25 | 0-25 | 25-75 |
| $Sr^{89}$ | 1 | 0-25 | 0-25 | 0-25 |
| | 2 | 0-25 | 0-25 | 0-25 |
| | 6 | 0-25 | 0-25 | 0-25 |
| $Y^{91}$ | 1 | 90-95 | 90-95 | 90-95 |
| | 2 | 95-99 | 95-99 | 90-95 |
| | 6 | 95-99 | 95-99 | 90-95 |
| $Zr^{95}$-$Nb^{95}$ | 1 | 25-75 | 95-99 | 95-99 |
| | 2 | 95-99 | 75-90 | 95-99 |
| | 6 | 90-95 | 95-99 | 95-99 |
| $Mo^{99}$ | 1 | 0-25 | 0-25 | 0-25 |
| | 2 | 0-25 | 0-25 | 0-25 |
| | 6 | 0-25 | 0-25 | 0-25 |
| $Ru^{103}$ | 1 | 75-90 | 75-90 | 90-95 |
| | 2 | 75-90 | 75-90 | 90-95 |
| | 6 | 75-90 | 95-99 | 90-95 |
| $I^{131}$[e] | 1 | 0-25 | 0-25 | 25-75 |
| | 2 | 25-75 | 0-25 | 0-25 |
| | 6 | 0-25 | 25-75 | 0-25 |
| $Cs^{137}$-$Ba^{137}$ | 1 | 0-25 | 0-25 | 0-25 |
| | 2 | 0-25 | 0-25 | 0-25 |
| | 6 | 0-25 | 0-25 | 0-25 |
| $Ba^{140}$-$La^{140}$ | 1 | 25-75 | 25-75 | 25-75 |
| | 2 | 25-75 | 25-75 | 25-75 |
| | 6 | 25-75 | 25-75 | 25-75 |
| $Pr^{142}$ | 1 | 90-95 | 90-95 | 75-90 |
| | 2 | 95-99 | $99^+$ | 95-99 |
| | 6 | 90-95 | $99^+$ | $99^+$ |
| $Ce^{144}$-$Pr^{144}$ | 1 | 75-90 | | 95-99 |
| | 2 | 75-90 | 95-99 | $99^+$ |
| | 6 | 90-95 | 95-99 | |

TABLE 2 (continued)

| RADIOISOTOPE | COAGULANT DOSE[a] (gr/gal) | REMOVAL (%)[b,c] | | |
|---|---|---|---|---|
| | | Low | Intermediate | High |
| **Coagulant: $FeSO_4$** | | | | |
| $Pm^{147}$ | 1 | 90-95 | 95-99 | |
| | 2 | 75-90 | | 75-90 |
| | 6 | | $99^+$ | $99^+$ |
| $Sm^{153}$ | 1 | 75-90 | 75-90 | 90-95 |
| | 2 | 90-95 | 95-99 | 95-99 |
| | 6 | 95-99 | 90-95 | $99^+$ |
| $W^{185}$ | 1 | 25-75 | 25-75 | 25-75 |
| | 2 | 75-90 | 0-25 | 25-75 |
| | 6 | 75-90 | 95-99 | 25-75 |
| $Re^{186}$ | 1 | 0-25 | 0-25 | 0-25 |
| | 2 | 0-25 | 0-25 | 0-25 |
| | 6 | 25-75 | 0-25 | 0-25 |
| Fission-product mixture | 1 | 75-90 | 25-75 | 75-90 |
| | 2 | 75-90 | 75-90 | |
| | 6 | 75-90 | | 75-90 |
| **Coagulant: $FeCl_3$** | | | | |
| $P^{32}$ | 1 | 25-75 | 95-99 | 95-99 |
| | 2 | 75-90 | 95-99 | 95-99 |
| | 6 | 95-99 | 95-99 | 95-99 |
| $Sc^{46}$ | 1 | 25-75 | 95-99 | 90-95 |
| | 2 | 25-75 | 95-99 | 95-99 |
| | 6 | 75-90 | 90-95 | 95-99 |
| $Cr^{51}$[d] | 1 | 0-25 | 0-25 | 0-25 |
| | 2 | 0-25 | 0-25 | 25-75 |
| | 6 | 0-25 | 0-25 | 0-25 |
| $Sr^{89}$ | 1 | 0-25 | 0-25 | 0-25 |
| | 2 | 0-25 | 0-25 | 0-25 |
| | 6 | 0-25 | 0-25 | 0-25 |
| $Y^{91}$ | 1 | 25-75 | 25-75 | 90-95 |
| | 2 | 25-75 | 75-90 | 95-99 |
| | 6 | 95-99 | 90-95 | 95-99 |
| $Zr^{95}$-$Nb^{95}$ | 1 | 0-25 | 25-75 | 25-75 |
| | 2 | 25-75 | 25-75 | 25-75 |
| | 6 | 0-25 | 75-90 | 95-99 |
| $Mo^{99}$ | 1 | 0-25 | 0-25 | 0-25 |
| | 2 | 25-75 | 0-25 | 0-25 |
| | 6 | 25-75 | 25-75 | 0-25 |

TABLE 2 (continued)

| RADIOISOTOPE | COAGULANT DOSE[a] (gr/gal) | REMOVAL (%)[b,c] Low | Intermediate | High |
|---|---|---|---|---|
| | | Coagulant: $FeCl_3$ | | |
| $Ru^{103}$ | 1 | 25-75 | 25-75 | 25-75 |
| | 2 | 25-75 | 75-90 | 75-90 |
| | 6 | 75-90 | 75-90 | 95-99 |
| $I^{131}$ [e] | 1 | 25-75 | 25-75 | 0-25 |
| | 2 | 0-25 | 25-75 | 25-75 |
| | 6 | 25-75 | 0-25 | 25-75 |
| $Cs^{137}$-$Ba^{137}$ | 1 | 0-25 | 0-25 | 0-25 |
| | 2 | 0-25 | 0-25 | 0-25 |
| | 6 | 0-25 | 0-25 | 0-25 |
| $Ba^{140}$-$La^{140}$ | 1 | 0-25 | 25-75 | 25-75 |
| | 2 | 25-75 | 25-75 | 25-75 |
| | 6 | 25-75 | 25-75 | 25-75 |
| $Pr^{142}$ | 1 | 90-95 | 90-95 | 95-99 |
| | 2 | 90-95 | 95-99 | 95-99 |
| | 6 | 95-99 | 95-99 | $99^+$ |
| $Ce^{144}$-$Pr^{144}$ | 1 | 25-75 | 90-95 | |
| | 2 | 90-95 | | 75-90 |
| | 6 | 90-95 | 75-90 | 95-99 |
| $Pm^{147}$ | 1 | | 25-75 | 75-90 |
| | 2 | 0-25 | 90-95 | |
| | 6 | 75-90 | | 90-95 |
| $Sm^{153}$ | 1 | 25-75 | 90-95 | 90-95 |
| | 2 | 90-95 | 90-95 | 90-95 |
| | 6 | 95-99 | 95-99 | $99^+$ |
| $W^{185}$ | 1 | 25-75 | 25-75 | 0-25 |
| | 2 | 25-75 | 25-75 | 25-75 |
| | 6 | 25-75 | 25-75 | 75-90 |
| $Re^{186}$ | 1 | 0-25 | 0-25 | 0-25 |
| | 2 | 0-25 | 0-25 | 0-25 |
| | 6 | 0-25 | 0-25 | 0-25 |
| Fission-product mixture | 1 | 25-75 | | 25-75 |
| | 2 | 75-90 | 75-90 | 25-75 |
| | 6 | 25-75 | 25-75 | |
| | | Coagulant: Alum | | |
| $P^{32}$ | 1 | 95-99 | 95-99 | 95-99 |
| | 2 | 95-99 | 95-99 | 95-99 |
| | 6 | $99^+$ | 95-99 | $99^+$ |

TABLE 2 (continued)

| RADIOISOTOPE | COAGULANT DOSE[a] (gr/gal) | REMOVAL (%)[b,c] | | |
|---|---|---|---|---|
| | | Low | Intermediate | High |
| | | Coagulant: Alum | | |
| $Sc^{46}$ | 1 | 75-90 | 95-99 | 95-99 |
| | 2 | 75-90 | 95-99 | 95-99 |
| | 6 | 75-90 | 95-99 | $99^{+}$ |
| $Cr^{51}$[d] | 1 | 0-25 | 0-25 | 0-25 |
| | 2 | 0-25 | 0-25 | 0-25 |
| | 6 | 25-75 | 0-25 | 25-75 |
| $Sr^{89}$ | 1 | 0-25 | 0-25 | 0-25 |
| | 2 | 0-25 | 0-25 | 0-25 |
| | 6 | 0-25 | 0-25 | 0-25 |
| $Y^{91}$ | 1 | 90-95 | 90-95 | 90-95 |
| | 2 | 95-99 | 95-99 | 90-95 |
| | 6 | 95-99 | 95-99 | 95-99 |
| $Zr^{95}$-$Nb^{95}$ | 1 | 95-99 | 95-99 | 95-99 |
| | 2 | 75-90 | 95-99 | 96-99 |
| | 6 | 95-99 | 95-99 | 95-99 |
| $Mo^{99}$ | 1 | 0-25 | 0-25 | 0-25 |
| | 2 | 0-25 | 0-25 | 0-25 |
| | 6 | 25-75 | 0-25 | 0-25 |
| $Ru^{103}$ | 1 | 25-75 | 25-75 | 25-75 |
| | 2 | 25-75 | 25-75 | 90-95 |
| | 6 | 25-75 | 75-90 | 90-95 |
| $I^{131}$[e] | 1 | 0-25 | 0-25 | 0-25 |
| | 2 | 0-25 | 0-25 | 25-75 |
| | 6 | 25-75 | 0-25 | 0-25 |
| $Cs^{137}$-$Ba^{137}$ | 1 | 0-25 | 0-25 | 0-25 |
| | 2 | 0-25 | 0-25 | 0-25 |
| | 6 | 0-25 | 0-25 | 0-25 |
| $Ba^{140}$-$La^{140}$ | 1 | 25-75 | 25-75 | 25-75 |
| | 2 | 25-75 | 25-75 | 25-75 |
| | 6 | 25-75 | 25-75 | 25-75 |
| $Pr^{142}$ | 1 | 90-95 | 90-95 | 95-99 |
| | 2 | 75-90 | 90-95 | 95-99 |
| | 6 | 95-99 | 95-99 | 95-99 |
| $Ce^{144}$-$Pr^{144}$ | 1 | 95-99 | 95-99 | $99^{+}$ |
| | 2 | 95-99 | 95-99 | |
| | 6 | 90-95 | | 99+ |
| $Pm^{147}$ | 1 | 75-90 | | 95-99 |
| | 2 | | 95-99 | 95-99 |
| | 6 | 90-95 | 95-99 | |

TABLE 2 (continued)

| RADIOISOTOPE | COAGULANT DOSE[a] (gr/gal) | REMOVAL (%)[b,c] | | |
|---|---|---|---|---|
| | | Low | Intermediate | High |
| | | Coagulant: Alum | | |
| $Sm^{153}$ | 1 | 90-95 | 95-99 | 95-99 |
| | 2 | 25-75 | 95-99 | 95-99 |
| | 6 | 95-99 | 95-99 | 95-99 |
| $W^{185}$ | 1 | 0-25 | 0-25 | 0-25 |
| | 2 | 25-75 | 0-25 | 0-25 |
| | 6 | 25-75 | 25-75 | 25-75 |
| $Re^{186}$ | 1 | 0-25 | 0-25 | 0-25 |
| | 2 | 0-25 | 0-25 | 0-25 |
| | 6 | 0-25 | 0-25 | 0-25 |
| Fission-product mixture | 1 | 25-75 | 75-90 | |
| | 2 | 75-90 | | 25-75 |
| | 6 | 75-90 | 75-90 | 25-75 |

[a]Value of 1 gr/gal represents 1 gr/gal of primary coagulant ($FeSO_4$, $FeCl_3$, or alum), 0.4 gr/gal of sodium silicate, and 1 gr/gal of lime. Similar ratios hold for doses of 2 and 6 gr/gal.

[b]Removals are indicated in the following class intervals: 0-25, 25-75, 75-90, 90-95, 95-99, and $99^+$. These values supplement those reported in ORNL-1488.

[c]The low, intermediate, and high activity levels are ~200, 2000, and 20,000 counts/min/ml.

[d]The low, intermediate, and high activity levels in this instance were 10, 100, and 200 counts/min/ml.

[e]Values somewhat higher than those previously reported.

Several commercial units which are available for producing deionized water have been evaluated for their efficacy in removing mixed fission products. One unit, designed primarily for producing deionized water for steam irons, storage batteries, etc., contained a small quantity of mixed-bed resin and a chemical compound to indicate exhaustion of the bed by a change of color. With a total flow of 1 to 2 liters, the maximum removals did not exceed 90%. Another unit that was studied contained 0.07 $ft^3$ of mixed bed resin and produced good results. Breakthrough of both activity and hardness occurred at about 60 gal of flow (2 gpm/$ft^2$ flow rate), and the average removal of activity amounted to 99.8%, with a range of removal of 99.2 to 99.97%.

The effect of such factors as pH, buffer solution, substrates, carrier concentration, clay concentration, kind of clay, exchange capacity, etc. on the removal of $Co^{60}Cl_2$, $Zn^{65}Cl_2$, $Se^{75}Cl_4$, $K_8Ta^{182}O_{19}$, $Na_2Ir^{192}Cl_6$, and $Tl^{204}(NO_3)_3$ from water by clays has been investigated. A jar-test laboratory procedure was employed. With an initial concentration of 10,000 counts/min/ml, clay concentrations of 2500 to 5000 ppm will remove about 90% of the radioelements listed. With the clays and the radioisotopes employed, there appeared to be no relation between the base-exchange capacity and the removal of activity. It was found that pH, added carrier, added salts, and substrate had a decided effect on removals by the clays investigated.[1]

A study was made to compare the removal of radioactive contaminants from water by four ion-exchange resins and three types of clay. Fines

[1]C. P. Straub and H. L. Krieger, *Removal of Radioisotopes from Waste Solutions – Soil Suspension Studies*, 8th Industrial Waste Conference, Purdue University, May 5 to 6, Lafayette, Indiana.

(less than 60 mesh) of Permutit Q, Zeo-Dur, Amberlite XE-69, and Amberlite MB-3, were compared with a local clay from the X-10 Area, montmorillonite, and a montmorillonite-bentonite clay. The three contaminants used were (1) W-6, a concentrate from the ORNL waste evaporator, (2) MFP-3, a three-year old dissolver solution containing approximately 20% trivalent rare earths, 20% strontium, 20% cerium, 16% cesium, 3% ruthenium, and 21% undetermined, and (3) the three radioisotopes shown in Table 3. A jar-test procedure using Oak Ridge tap water with five different dosages of each of the adsorbent materials was employed. The results obtained, with the lowest and the highest dosage in each test as shown in Table 3, indicated that, with the test methods used, $Zr^{95}$-$Nb^{95}$ was easy to remove, up to 99% of the $Ce^{141-144}$-$Pr^{144}$ could be removed by MB-3 (2700 ppm) or local clay (4000 ppm), and that 2700 ppm of either Permutit Q, Amberlite XE-69 or montmorillonite clay would remove at least 92% of the $Sr^{90}$-$Y^{90}$.

### Process Waste Studies

The statistical evaluation of all data pertaining to the volume of flow to and from the settling basin, the activity of both influent and effluent, and the total curies discharged is being continued to provide specific information for the design of full-scale facilities for the treatment of the radioactive process waste waters. The design of the full-scale plant will be predicated upon the results obtained in the operation of the pilot-scale experimental units now being installed.

## SURVEYS AND EVALUATIONS – FIELD AND LABORATORY

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### Estimates of Downstream Effects in the Event of Emergency Contamination in the Clinch River

Studies of surface water conditions in the Clinch and Tennessee River systems in relation to hazards that might result from an emergency release or escape of radioactive materials into the Clinch River have been in progress for about two years. This is a cooperative effort of ORNL and the Health and Safety Division of the Tennessee Valley Authority. The purpose is to make available basic data and methods of analysis in a form which can be used promptly for estimating the extent and

TABLE 3. COMPARISON OF REMOVAL OF RADIOACTIVE MATERIALS FROM WATER BY ION-EXCHANGE RESINS AND CLAYS

| CONTAMINANT | ACTIVITY* (counts/min/ml) | DOSAGE OF ADSORBENT MATERIAL (ppm) | REMOVAL (%) | | | | | | |
|---|---|---|---|---|---|---|---|---|---|
| | | | Ion-Exchange Resin Fines | | | | Clays | | |
| | | | Permutit | | Amberlite | | Montmorillonite–bentonite | Montmorillonite | Local X-10 Area** |
| | | | Q | Zeo-Dur | XE-69 | MB-3 | | | |
| W-6 solution | 5,000 | 450 | 61.3 | 46.3 | 54.0 | 38.4 | 74.8 | 59.5 | 58.8 |
| | | 2700 | 76.3 | 62.7 | 64.9 | 86.9 | 81.2 | 67.7 | 63.4 |
| MFP-3 as nitrates | 48,000 | 450 | 50.7 | 52.5 | 57.4 | | 42.6 | 49.8 | |
| | | 2700 | 83.9 | 66.2 | 89.5 | | 59.8 | 69.5 | |
| $Sr^{90}$-$Y^{90}$ as chlorides | 4,100 | 450 | 77.7 | 68.0 | 92.5 | | 73.9 | 75.5 | |
| | | 2700 | 92.7 | 77.0 | 93.0 | | 88.9 | 92.7 | |
| $Zr^{95}$-$Nb^{95}$ as oxalate | 2,000 | 450 | 92.3 | 96.3 | 93.7 | 83.1 | 97.9 | 99.3 | 93.5 |
| | | 2700 | 93.7 | 97.7 | 95.1 | 97.9 | 98.6 | 0.9 | 98.5 |
| $Ce^{141,144}$-$Pr^{144}$ as chloride | 8,000 | 450 | 53.1 | | | 95.4 | | | 98.7 |
| | | 2700 | 53.8 | | | 99.0 | | | 99.3 |

*Count rates are at approximately 10% geometry.

**Dosages are an exception and were 1000 and 4000 ppm.

duration of potential hazards and for planning the necessary countermeasures.

The TVA has completed tabulations of hydraulic data, and two reports containing charts, tables, and explanatory text are now available for use as needed.[2,3] The reports are designed particularly to aid in the determination of river flows, time of water travel, dilution factors, and dispersion of masses of contaminated water in the rivers from Norris Dam to as far downstream as Chattanooga. A manual with illustrative examples and instructions for use of the basic charts and tables is being prepared by sanitary engineers of the TVA.

At the request of the Physics Division of the Laboratory, an analysis of one specific problem was made; the results were used in the preparation of a Reactor Safeguard Report on the Tower Shielding Facility. In the analysis of this problem, the data of the TVA report[2] were used, and TVA specialists were consulted. It was assumed that a large quantity of mixed fission products of various ages might be accidentally discharged into the Clinch River near ORNL during a 6-hr period. From the assumed conditions, the total quantity, the average half life, and the approximate composition of the radioactive mixture that would be dispersed were calculated with the use of the tables and charts prepared by Hunter and Ballou;[4] and the peak concentration in microcuries per cubic centimeter at the point of entry could be estimated for any given volume of stream flow in the Clinch River.

The data provided by TVA made it possible to estimate (1) the combination of conditions in the rivers that would cause the highest peaks of radioactivity in the water at various downstream locations, (2) the time of water travel from the point of entry of the contaminants to each of four downstream locations of particular interest and the reduction of activity by radioactive decay during these time intervals, and (3) the percentage of reduction of the maximum peak concentration in the Clinch River by dispersion and by dilution before the activity reached each of the downstream locations. Some of the pertinent values obtained by analysis of the problem are shown in Table 4.

The last two lines in Table 4 show maximum permissible concentration (MPC) values of radioactive contamination in drinking water for emergency use for periods of one day and of 30 days. With the use of the methods developed by Morgan and Straub,[5] these values were estimated for the particular mixture of radioisotopes that would reach each of the four locations. In the Clinch River at K-25 and at the mouth of the river, the estimated peak concentrations would be nearly equal to the one-day emergency MPC values. In the Tennessee river, the peak concentrations would be far lower than the 30-day emergency MPC values.

## INSTRUMENTATION AND TECHNIQUES

H. G. Conner, B. Kahn, J. M. Garner, F. Kalil, R. L. Nichols

### Counting Response of Crystal Spectrometer[6]

The determination of the counting response of a thallium-activated sodium iodide scintillation spectrometer on which progress was reported previously[7] has been completed. Over-all counting efficiencies were determined experimentally for sources placed 2.45 cm from a cylindrical crystal $1\frac{1}{2}$ in. in diameter and 1 in. in length. The sources were radioisotopes emitting x and gamma rays with energies between 31 kev and 2.76 Mev. Their disintegration rates were found by coincidence counting, $4\pi$ counting, and by the use of a calibrated high-pressure ion chamber. The variation of counting efficiency with energy of gamma radiation is given in Fig. 15. The counting efficiency has been corrected for absorption of gamma rays in

(2) *Effect of Accidental Spillage of Radioactive Waste in Clinch River: Report No. 1, Time of Travel, Probable Flows, and Dispersion*, Nov. 10, 1952 (unpublished report prepared for the Division of Health and Safety by the Hydraulic Branch, Division of Water Control Planning, Tennessee Valley Authority).

(3) *Effect of Accidental Spillage of Radioactive Waste in Clinch River: Report No. 2, Water Control Possibilities and Their Results*, April 20, 1952 (unpublished report prepared for the Division of Health and Safety by the Hydraulic Branch, Division of Water Control Planning, Tennessee Valley Authority).

(4) H. F. Hunter and N. E. Ballou, *Simultaneous Slow Neutron Fission of $U^{235}$ Atoms. I. Individual and Total Rates of Decay of the Fission Products*, ADC-65 (Feb. 24, 1949); H. F. Hunter and N. E. Ballou, *Nucleonics* 9:5, C-2 (Nov. 1951).

(5) K. Z. Morgan and C. P. Straub, *Maximum Permissible Exposure Levels of Ionizing Radiation for Peacetime Operations and for Emergency Conditions* (unpublished).

(6) In cooperation with W. S. Lynn, Analytical Chemistry Division.

(7) T. W. Brockett *et al.*, *H-P Quar. Prog. Rep. Jan. 20, 1953*, ORNL-1488, p. 10.

TABLE 4. ESTIMATED FACTORS AND EFFECTS FROM ASSUMED DISPERSION OF RADIOACTIVE CONTAMINANTS INTO THE CLINCH RIVER UNDER ADVERSE* RIVER CONDITIONS

| | LOCATION OF MAXIMUM PEAK CONCENTRATION AFTER DISPERSION | K-25 WATER INTAKE | MOUTH OF CLINCH RIVER | WATTS BAR DAM | CHATTANOOGA WATER INTAKE |
|---|---|---|---|---|---|
| Miles above mouth of Clinch River | 20.8 | 13.2 | 0.0 | | |
| Miles above mouth of Tennessee River | | | 567.7 | 529.9 | 465.3 |
| Stream flow, cfs | 1,000 | 1,000 | 1,000 | 18,900 | 25,500 |
| Time of water travel, days (cumulative from point of entry) | 0.25 | 3.50 | 9.75 | 17.25 | 25.75 |
| Total radioactivity, curies (approximate, after reduction by decay) | 100,000 | 60,000 | 50,000 | 42,000 | 37,000 |
| Reduction of maximum peak concentration: | | | | | |
| By dispersion, % | 0 | 38 | 76 | 97 | 99 |
| By dilution, % | 0 | 0 | 0 | 95 | 96 |
| Peak concentration, $\mu c/cm^3$ | $1.7 \times 10^{-1}$ | $62 \times 10^{-3}$ | $20 \times 10^{-3}$ | $11 \times 10^{-5}$ | $2.4 \times 10^{-5}$ |
| Emergency maximum permissible concentration values, $\mu c/cm^3$: | | | | | |
| For 1-day use | | $58 \times 10^{-3}$ | $55 \times 10^{-3}$ | | |
| For 30-day use | | $1.7 \times 10^{-3}$ | $1.5 \times 10^{-3}$ | $1.4 \times 10^{-3}$ | $1.2 \times 10^{-3}$ |

*Adverse river conditions are those that would result in the highest concentrations of contaminants at downstream locations, that is:

Pool elevations,

Watts Bar Reservoir – 740 ft

Chickamauga Reservoir – 682 ft

Stream flows as in table

Thermolly stratified conditions in Clinch River Embayment and Watts Bar Reservoir.

the MgO reflector and in the aluminum container which enclose the crystal, and for the escape of x rays from the crystal.

The calibrated detector has been used to determine the intensities of gamma rays in some radioisotopes with decay schemes that are not well known, such as $K^{42}$ and $Rh^{106}$. It has also been used to identify gamma-emitting radioisotopes in dust and water samples when the activity was sufficient (0.01 $\mu c$ or greater) to warrant use of the spectrometer.

### Analytical Methods of Radioisotope Determination

Methods of radiochemical analysis are being compiled to assist those interested in the determination of radioisotopes in water supplies. The chemical procedures selected are those on which information is readily accessible in the unclassified

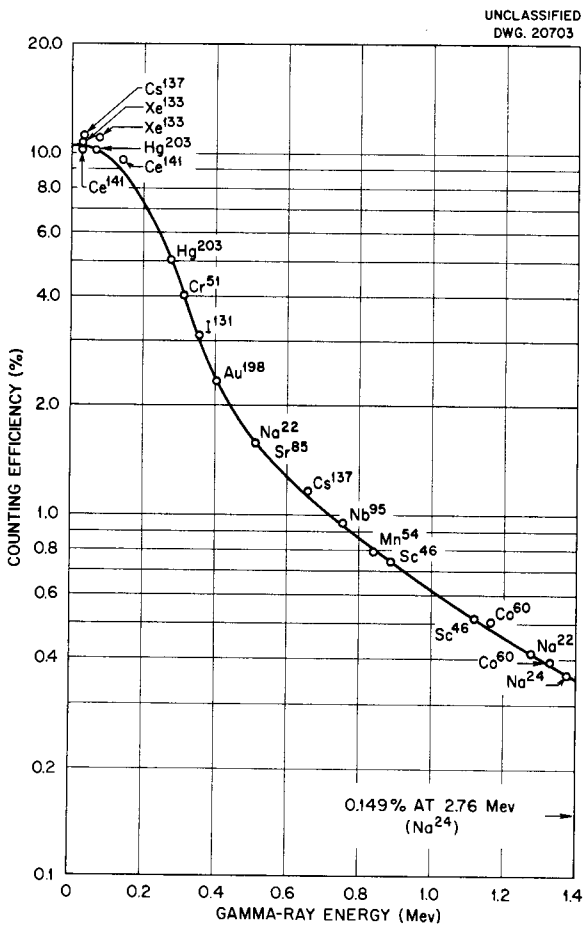


**Fig. 15. Counting Efficiency of Thallium-Activated Sodium Iodide Scintillation Spectrometer.**

literature; pertinent references will be listed. The procedures for the separation of radionuclides have been abstracted and indexed and in some cases have been modified so as to be of specific use for the above purpose. It is intended that this compilation will serve as a source book for use at ORNL and at other laboratories.

### Radioactivity in River Bottom Sediments

The previously reported[8] survey of radioactivity in bottom sediments in the Clinch and Tennessee Rivers was repeated during the summer of 1952. The same detector "flounder" was used for both surveys. The weighted average of the readings for a particular cross section of the river was obtained by weighting the individual observations by the total area represented by each individual reading. The weighted average reading of each cross section is plotted as a point in Fig. 16 and in which the results of the two surveys (1951 and 1952) are compared. The more recent survey shows a considerable increase in the radioactivity in river bottom sediments in the Clinch River below the mouth of White Oak Creek. Increases in radioactivity are also noticeable in the Tennessee River for a distance of approximately 200 miles downstream. Although it is too soon to draw conclusions, these data indicate the possibility of a trend toward increasing accumulations of radioactive materials in the rivers with time.

[8] E. R. Eastwood *et al.*, *H-P Quar. Prog. Rep. Jan. 20, 1952*, ORNL-1277, p. 1.

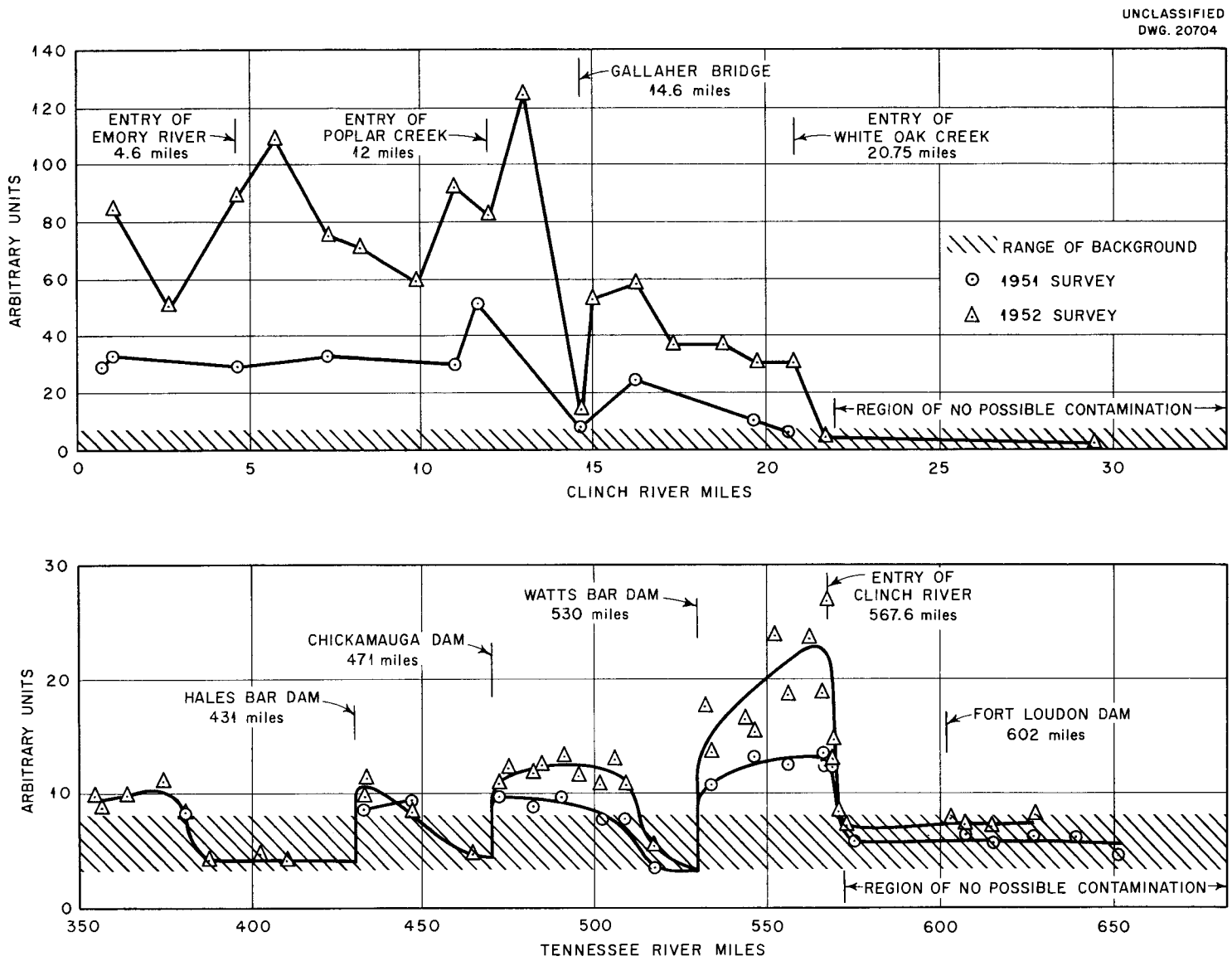


**Fig. 16. Weighted Average Concentration of Radioactivity in River Sediments.**

## ECOLOGICAL STUDY[1]

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W. T. Helm
W. T. Miller
E. R. Eastwood

The reports of the fifth and sixth semiannual estimates of the fish population in White Oak Lake have been completed. These reports will not be distributed as previously, but will be incorporated in the final over-all report which is being written. In April, the entire fish population of White Oak Lake was killed with rotenone and removed from the lake. Preliminary figures indicate that more than 250,000 fish with a total weight of over 25,000 lb were recovered during the study.

During the past winter season a total of 1203 migratory waterfowl was captured in traps on White Oak Lake. Of that total, 647 birds were banded and 556 banded birds were recaptured. The eight species of waterfowl banded were mallard, black duck, wood duck, pintail, gadwall, baldpate, green-winged teal, and coot. Reports of 20 ducks banded at White Oak Lake have been returned by hunters who shot the birds in Tennessee, Alabama, Kentucky, Louisiana, and Texas. The carcass of one bird was found in Ontario, Canada.

In March, a muskrat bearing approximately 1 $\mu c$ of radiostrontrium per gram of bone tissue was captured in the X-10 area. This concentration of radiostrontrium is believed to have induced an osteogenic sarcoma in the right tibiofibula with consequent metastases into the lungs and kidneys. Estimates by photometric methods indicate that the bone of the animal had been receiving a dose of approximately 1.5 rep/hr.

All field work on the ecological survey program has been discontinued, and the entire efforts of the group are being expended toward the preparation of a final report. It is estimated that the report will be completed by the end of the present fiscal year.

[1] In cooperation with the Tennessee Valley Authority.

## EDUCATION, TRAINING, AND CONSULTATION

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

### AEC FELLOWSHIP PROGRAM

The 1952-53 group of AEC Fellows in Radiological Physics completed the year of graduate study at Vanderbilt University with very creditable records, and reported to the Laboratory June 9 for 12 weeks of field training in the Health Physics Division. Three members of the Division taught a six-credit graduate course in Radiological Physics at Vanderbilt University during the last two quarters of the academic year. While designed primarily for the AEC Fellows, this course is open to other students in the University.

### LECTURES

Lectures on current practices in radiation protection were given at two universities in conjunction with the ORINS-ORNL Traveling Lecture Program. Seminars given during the reported period consisted of two on nuclear and health physics for R. T. Overman's groups at ORINS; one for officers in training at the Navy Medical School at Bethesda, Maryland, one at Harwell, England, and one at Stockholm, Sweden. A paper on the developments in internal dose determinations was given at the American Industrial Hygiene Conference in Los Angeles.

### CONSULTATION

A detailed justification of the fencing based on direct radiation estimates for the Tower Shielding Reactor was prepared.[1] Calculations were made of the airborne hazards associated with the following types of catastrophe postulated for this reactor: a power excursion sufficient only to melt down the

[1] T. H. J. Burnett, ORNL-1550, Appendix A (classified.

fuel plates that would result in a total release of halogens and rare-gas fission products as a thermally cold cloud; and an Al-$H_2O$ reaction that would result in the total release of all fission products in a hot cloud of considerable thermal content. The data from the calculations, in terms of permissible concentrations, were supplied to the Oak Ridge Office, AEC–U.S. Weather Bureau for inclusion in the feasibility report to be submitted to the Reactor Safeguards Committee. Table 5 gives the maximum concentrations which are permissible for a single dose of 10 rep to the specified body organ during the exposure times given by the meteorologists. For the values given in Table 5, the isotope mixture is taken to be that present at time 0 + $10^3$ sec.

Values for maximum permissible contamination of Laboratory personnel and surfaces in relation to the maximum permissible concentrations in air and water have been developed. The factors determining the dispersal of contamination were studied and the limits were developed on the basis of experimental data. Considerable progress was made toward agreement and acceptance of these values by X-10, Y-12, and K-25; final acceptance will give a uniform basis for contamination control at the three Oak Ridge plants.

**TABLE 5. SPECIFIC CONCENTRATIONS OF FISSION PRODUCTS FOR A SINGLE DOSE OF 10 rep TO THE SPECIFIED BODY ORGAN***

| TYPE OF CATASTROPHE | EXPOSURE TIME (sec) | PRODUCT CONCENTRATIONS REQUIRED TO DELIVER A DOSE OF 10 rep (μc/ml) | | |
|---|---|---|---|---|
| | | Halogen | Total Maximum Fission Products | Submersion-volatile Fission Products** |
| Thermally cold cloud | | | | |
| Night | 30 | $2.8 \times 10^{-3}$*** | Not present | 1.14 |
| Day | 64 | $1.3 \times 10^{-3}$*** | Not present | 0.54 |
| Thermally hot cloud | | | | |
| Night | 228 | $3.7 \times 10^{-4}$ | $3.2 \times 10^{-3}$*** | $1.5 \times 10^{-1}$ |
| Day | 624 | $1.4 \times 10^{-4}$ | $1.2 \times 10^{-3}$*** | $5.5 \times 10^{-2}$ |

*Critical organ
for halogens: thyroid gland
for fission products: lungs or gastrointestinal tract
for submersion-volatile fission products: total body

**The value would be $1.9 \times 10^{-2}$ if the dosage rate were limited to 20 rep/hr.

***These values are closest to those actually reached under the assumed conditions of the accident.

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