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ANNUAL REPORT FOR 1962

OAK RIDGE NATIONAL LABORATORY

operated by

UNION CARBIDE CORPORATION

for the

U.S. ATOMIC ENERGY COMMISSION

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HEALTH PHYSICS DIVISION

APPLIED HEALTH PHYSICS ANNUAL REPORT FOR 1962

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OAK RIDGE NATIONAL LABORATORY
Oak Ridge, Tennessee
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UNION CARBIDE CORPORATION
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1.0 SUMMARY

The environmental monitoring data processed during 1962 indicated that Laboratory operations did not contribute appreciably to the contamination of the environment with radioactive materials. However, the effects of airborne radioactive materials generated from nuclear weapons tests conducted at various locations throughout the world continued to be reflected in certain parts of the environmental monitoring data. This effect is illustrated in the tabulations below where certain 1962 averages are compared with the 1961 averages:

1) The concentration of radioactive materials in air (as determined by air filtration techniques) increased by a factor of about 2.5.

2) The concentration of radioactive materials in rain water increased by a factor of 3 to 5.

3) The Sr-90 concentration in raw milk increased by a factor of about 3.

4) The I-131 concentration in raw milk increased by a factor of about 12.

5) The general radiation background increased by a factor of 2 in the East Tennessee area.

6) Gamma measurements taken over river bottom sediment increased by a factor of about 2 in Fort Loudoun Reservoir—a body of water that is fed by the Tennessee River and not affected by liquid waste releases originating from Oak Ridge operations.

The calculated average concentration of radioactive materials in the Clinch River resulting from waste releases originating from ORNL operations decreased from 18 per cent of the $(MPC)_w$ in 1961 to 7.4 per cent of the $(MPC)_w$ in 1962. The measured concentration at a point about 16 miles below the entry of ORNL waste releases averaged 4.9 per cent of the $(MPC)_w$ during 1962.

There were no personnel exposures recorded during 1962 which exceeded maximum permissible levels. The highest total body dose recorded was 4.6 rem which is 38 per cent of the maximum permissible annual dose. There were no cases involved where the internal deposition of radioactive material in the body was estimated to have exceeded one-half of a maximum permissible body burden averaged over a 12-month period. As of December 31, 1962, only one individual had accumulated a total body dose from external radiation which exceeded the age proration formula 5(N-18).

The Laboratory experienced 55 unusual occurrences during 1962—all of which were classified as minor events. Both the frequency rate and severity index of unusual occurrences continued to drop during 1962.

2.0 ENVIRONMENTAL MONITORING

The Health Physics Division monitors for air-borne radioactivity in the East Tennessee area by the use of three separate monitoring networks. The local air monitoring (LAM) network consists of ten stations which are located within the immediate ORNL area (Fig. 1); the perimeter air monitoring (PAM) network consists of seven stations which are located on the perimeter of the AEC controlled area (Fig. 2); and the remote air monitoring (RAM) network consists of seven stations which are located outside the AEC controlled area at distances of from 12 to 75 miles from ORNL (Fig. 3). The monitoring networks provide for the collection of (1) air-borne radioactivity by air filtration techniques, (2) radioparticulate fall-out material by impingement on gummed paper trays, and (3) rain water for measurement of radioactivity as rainout. The filter data are representative of radioparticulate matter which might be considered respirable; the gummed paper data are representative of radioparticulate fall-out; and the rain water data provide information on the soluble and insoluble fractions of the radioactive content of fall-out material.[1]

Low level radioactive liquid wastes originating from Oak Ridge National Laboratory (ORNL) operations are discharged, after preliminary treatment, to White Oak Creek which is a small tributary of the Clinch River. Liquid waste releases are controlled so that resulting average radioactive concentrations in the Clinch River comply with maximum permissible concentrations established for populations in the neighborhood of an atomic energy installation as recommended by the National Committee on Radiation Protection (NCRP) and the Federal Radiation Council (FRC).

The radioactive content of White Oak Creek water is determined at a number of points (Fig. 4) along its course between the point where wastes are first discharged and the entry of the stream into Clinch River waters. Water samples are also collected at a number of locations along the Clinch River beginning at a point above the entry of wastes into the river via White Oak Creek and ending at Center's Ferry (near Kingston, Tennessee) about 16 miles downstream from the entry of White Oak Creek. Water samples of the above type are analyzed for gross radioactivity and for certain specified long-lived radionuclides. Using the maximum permissible concentration values for drinking water, $(MPC)_w$, for each isotope as recommended by NCRP, a weighted average $(MPC)_w$ for the mixture of radionuclides is calculated on the basis of the isotopic distribution in the water. The average concentrations of gross activity are used for control purposes.

Raw milk and potable water samples are collected at six sampling stations within a radius of 25 miles from the Laboratory. The milk samples

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[1] A detailed discussion concerning techniques used in processing air and water samples for environmental monitoring purposes is given in ORNL-2601.

are analyzed for Sr-90 and I-131 whereas the potable water samples are analyzed only for the Sr-90 content. The purpose of this sampling program is twofold: first, samples collected in the immediate vicinity of the Laboratory provide data by which one may evaluate the possible effect of waste releases originating from Laboratory operations; second, samples collected remote to the immediate vicinity of the Laboratory area provide background data which is essential in establishing a proper index from which the intentional or accidental release of radioactive materials originating from Oak Ridge operations may be evaluated.

Aerial background surveys are made at least once each calendar quarter over the ORNL area (Fig. 5) and for several miles from ORNL in the general direction of the prevailing winds. Using light aircraft and flying at speeds of approximately 120 miles per hour, experiments have shown that, at an altitude of approximately 300 feet, it is possible to detect I-131 contamination upon grasslands with reasonable accuracy by scintillation detectors down to levels of about 0.5 $\mu c/m^2$. Thus, light aircraft, equipped with portable scintillation detectors and used in the manner described above, provide a practical means of detecting significant quantities of I-131 deposited on ground surfaces.

Background gamma radiation measurements are made monthly at a number of locations within the Oak Ridge geographical area and less frequently at locations throughout other portions of the East Tennessee area. These measurements are taken with calibrated GM and scintillation type detectors at a distance of three feet above the surface of the ground.

Annual surveys have been made of the Clinch and Tennessee Rivers since 1951 in order to evaluate the radioactive content of bottom sediments and to provide data relative to the dispersion of liquid wastes released from the Oak Ridge area.

2.1 Atmospheric Monitoring

The average concentrations of radioactive materials in the atmosphere, as measured by filtration methods provided by the LAM, PAM, and RAM network system during 1962, were fairly consistent between networks as follows:

Network	Concentration ($\mu c/cc$)
LAM	3.7×10^{-12}
PAM	3.6×10^{-12}
RAM	4.3×10^{-12}

The LAM network value of 3.7×10^{-12} μc/cc is about 0.37 per cent of the $(MPCU)_a$[2] based on occupational exposure. When evaluated in terms of the maximum permissible concentration for persons residing in the neighborhood of an atomic energy installation, the PAM and RAM network values represent 3.6 and 4.3 per cent of the $(MPCU)_a$ respectively. The averages recorded by the three networks during 1962 increased over the 1961 averages by a factor of about 2.5. Table 1 gives a tabulation of data derived from filtration techniques for each station within each network. The weekly averages for each network are graphically illustrated in Fig. 6.

Radioparticulate fall-out as measured by the gummed paper technique[3] was highest during the first and last quarters of 1962. The peak value for each network was measured during Week No. 45 (Fig. 7). The abundance of radioactive particulates collected on the air monitor filters followed the same trend as that for the gummed paper collectors but to a more pronounced degree (Fig. 8). The radioparticulate count per 1000 cubic feet of air sampled by filtration techniques at each air monitoring station within each network is given in Table 1. The average number of particles per square foot collected each week by gummed paper fall-out collector techniques is shown for each monitoring station within each network in Table 2.

2.2 Water Analyses

Rain Water - There was an increase in the radioactive content of rain water at all stations during 1962 as follows:

	Concentrations (μc/cc)	
Network	1961	1962
LAM	1.6×10^{-7}	10×10^{-7}
PAM	2.5×10^{-7}	11×10^{-7}
RAM	4.1×10^{-7}	13×10^{-7}

The lack of significant variations between network averages during 1961 and 1962 indicates that the radioactivity in rain water collected at these stations is not of local origin. As in 1961, average concentrations were

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[2]The $(MPCU)_a$ is defined as the maximum permissible concentration for an unknown mixture of radioisotopes in air. NBS Handbook 69, Table 4, p. 94 gives exposure values applicable to various mixtures of radionuclides and establishes guide lines for deriving the $(MPCU)_a$.

[3]The gummed paper collector presents a collection surface of 1 square foot. Radioparticulates per square foot are determined by autoradiography.

slightly higher within the RAM network—stations located several miles distant from ORNL. Averaged values as derived for each station are shown in Table 3.

Clinch River Water - A total of 1436 beta curies of radioactivity was released via White Oak Creek to the Clinch River during 1962 (Table 4). Radiochemical analyses of the effluent passing through White Oak Dam indicated that about 94 per cent of the radioactivity consisted of Ru-106 which represented an increase in Ru-106 content of about five per cent over the value observed during 1961. The percentage of Sr-90 in the effluent leaving White Oak Dam was about 1.3 per cent which was essentially the same as the level recorded during 1961.

The calculated average concentration of radioactive materials in the Clinch River at Clinch River Mile (CRM) 20.8 (the point of entry of White Oak Creek into the river) was 3.4×10^{-7} μc/ml which represents 7.4 per cent of the weighted average $(MPC)_w$ recommended for persons who reside in the neighborhood of an atomic energy installation (Table 5). The calculated value is based on concentrations released from White Oak Dam and the dilution afforded by the river; it does not include amounts of radioactive materials (e.g., fall-out) that may have entered the river upstream from CRM 20.8. The average concentration of radioactive materials in the Clinch River did not exceed 25 per cent of the $(MPC)_w$ during any given week during 1962 (Fig. 9).

The measured average concentration of radioactivity in the water taken from the Clinch River at CRM 41.5 (above the entry of White Oak Creek) was 1.7 per cent of the weighted average $(MPC)_w$ (Table 5). It is of interest to note that the concentration of Sr-90 in the river above the entry of White Oak Creek is essentially the same as the calculated value for White Oak Creek effluent at CRM 20.8 assuming uniform dilution of the two streams. The radioactive materials in the river upstream presumably are the result of fall-out from world-wide nuclear weapons testing and natural causes.

The measured average concentration of radioactive materials in the Clinch River at CRM 4.5 (near Kingston, Tennessee) was 1.7×10^{-7} μc/ml. This value represents about 4.9 per cent of the $(MPC)_w$ as applied to persons residing in the neighborhood of an atomic energy installation.

Potable Water - Potable water samples collected within a radius of 25 miles of ORNL indicated an average concentration of Sr-90 of about 0.5 μμc/liter. Concentrations ranged from a minimum of 0.1 μμc/liter to a maximum of 2.1 μμc/liter. These values are well below the upper limit of Range I for Sr-90 as specified by the Federal Radiation Council.[4]

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[4] Background Material for the Development of Radiation Protection Standards, Report No. 2, Staff Report of the FRC, p. 18, September, 1961.

2.3 Milk Analyses

The average concentration of Sr-90 in raw milk samples collected from within a 25-mile radius of the Laboratory was 33 μμc/liter. The average concentration of I-131 was 96 μμc/liter. These data are compared with six other locations within the United States[5] as shown in Fig. 10. Both the average Sr-90 and I-131 values derived from samples processed at ORNL fall within FRC Range II limits if one assumes the average daily intake per individual to be one liter per day. (See Appendix 9.1 for a discussion concerning I-131 analytical techniques.)

Sr-90 and I-131 concentrations in raw milk samples during 1962 were higher than the 1961 values by factors of 3 and 12 respectively. Most of this increase may be attributed to fall-out resulting from world-wide nuclear weapons testing. However, local operations had some effect on the I-131 concentrations. Approximately 102 curies of gaseous wastes were released from plant operations between May and October of 1962. A large percentage of the release consisted of I-131 and arose from defects which developed in the off-gas cleaning system at a processing facility.[6] The defect was corrected during October.

2.4 Background Measurements

Background measurements were taken at a number of locations (established in 1961) in the East Tennessee area during routine servicing visits to the remote air monitoring stations. Measurements were made at each location on a frequency of once each five weeks. Average background readings and the location of each station are presented in Fig. 11. The average background level during 1962 as measured at these stations was on the order of 19 μR/hr. The 1962 value was 27 per cent higher than the average value recorded for the last quarter of 1961.

Background measurements made on the ORNL site during 1962 were determined by film monitoring techniques. The system utilizes moisture-proof film packets which are located on a grid that covers the ORNL area. Films are processed each quarter. The average background level for the Laboratory as determined by the film technique was 0.11 mR/hr (Fig. 12). The 1962 average background was about 10 per cent higher than the 1961 level.

Background measurements made monthly with a calibrated GM tube monitor at five selected stations adjacent to the Laboratory area yielded an average background reading of approximately 0.03 mR/hr during 1962 (Fig. 12). The 1962 value is about 2.5 times the average background measured in the Oak Ridge area in 1943 prior to the start-up of the Oak Ridge Graphite

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[5]Report by the U. S. Public Health Service, Vol. III, No. 1, January, 1962.

[6]J. F. Manneschmidt, Laboratory Facilities - Waste Disposal, Report for December, 1962, ORNL CF-63-1-73.

Reactor. The 1962 value does not differ significantly from averages observed throughout the eastern section of the United States.

Aerial surveys with light aircraft were performed at least once each quarter during 1962 in order to maintain current recordings of background levels essential to the evaluation of a ground contamination problem following a major release of air-borne radioactivity. Typical chart recordings taken on flights over the Laboratory area for the years 1959 through 1962 are shown in Fig. 13.

2.5 Annual Survey of the Clinch and Tennessee Rivers

The annual survey of the Clinch and Tennessee Rivers was carried out by the Applied Health Physics Section during the summer of 1962. The survey of the Tennessee River extended from Fort Loudoun Reservoir downstream through Guntersville Reservoir. (The techniques and procedures used are described in ORNL-2847.)

Figures 14 and 15 show the gamma count rate at the surface of Clinch and Tennessee River bottom silt at certain river mile markers for the years 1961 and 1962. An examination of Fig. 14 shows the longitudinal dispersion of radioactivity in Clinch River bottom silt in 1962 to be essentially the same as that of 1961 but of smaller magnitude. This decrease was to be expected due to a decrease in waste releases from Oak Ridge operations during 1962. (A total of 2187 curies was discharged to the Clinch River during the 12-month period which ended in July of 1961 just prior to the 1961 survey; only 1700 curies were released during the corresponding period in 1962.)

The increase in the gamma count rate in Tennessee River bottom silt in 1962 (Fig. 15) appears to be due primarily to fall-out from world-wide nuclear weapons testing. This conclusion is supported by the fact that bottom silt background readings taken in Fort Loudoun Reservoir increased from 8.9 cts/sec in 1961 to 16 cts/sec in 1962.

Table 6 shows the average concentrations of the major radionuclides found in Clinch River water upstream from the outfall of White Oak Creek. The data in Table 6 also support the conclusion that the 1962 river survey data were influenced by fall-out. Increased average concentrations of Sr-90, $Ru^{103-106}$, $Zr\text{-}Nb^{95}$, and Ce-144 were detected in the Clinch River at CRM 41.5 (20.7 miles upstream from the mouth of White Oak Creek). Increases such as this are attributed to nuclear weapons testing fall-out. (Increased concentrations of fall-out material in river water would have very little effect upon the relatively high gamma count rate in the Clinch River but would significantly influence the relatively low count rate of the Tennessee River silt.)

The average gamma count rate at the surface of river bottom silt in the Clinch and Tennessee Rivers for the years 1951-1962 is presented in Fig. 16.

In comparison to the 1961 data: a decrease was observed in the average gamma count rate in the Clinch River; an increase was observed in the Tennessee River.

Results of the radiochemical analysis of the Clinch River silt collected during the 1961 and 1962 surveys are given in Table 7.

2.6 Effect of Waste Releases in the Clinch River under Static-Flow Conditions

The Tennessee Valley Authority drastically curtailed water releases from Norris Dam during a 10-day period which began on August 4th in order to provide for certain activities associated with the construction of Melton Hill Dam. During this 10-day period it was estimated that there would be essentially a no-flow condition at the point where White Oak Creek discharges into the stream. Thus, conditions prevailed during August which permitted a limited study concerning the effect of radioactive releases to the Clinch River during static-flow conditions.

It was decided to allow discharges from White Oak Lake to proceed in a normal manner and evaluate concentrations of radioactivity in the river on a daily basis during and following the 10-day interval. Three additional sampling stations were established in the river at CRM 18.0, CRM 14.5, and CRM 11.0. Samples were collected daily at each of these locations during the period August 8th through August 17th and analyzed for gross beta activity. The average concentrations observed at each of the three locations during the period under study was 1.1×10^{-7} $\mu c/ml$, 0.7×10^{-7} $\mu c/ml$, and 1.7×10^{-7} $\mu c/ml$ respectively. The maximum concentration of radioactivity detected was 8.9×10^{-7} $\mu c/ml$ and was detected in the sample collected on August 15th at CRM 11.0.

Samples were collected in the river approximately one mile above and below the outfall of White Oak Creek on August 11th to permit an evaluation relative to the movement of activity upon entry into the river. These data indicated that the material tended to pond near the mouth of White Oak Creek and on occasion flowed upstream a short distance. In fact, the maximum activity measured during this phase of the study was 1.4×10^{-5} $\mu c/ml$ and was found in a sample collected approximately 100 feet upstream from the mouth of White Oak Creek. Subsequent measurements made following the resumption of normal flow in the river indicated that ponded material had a tendency to break up rather quickly and the normal dilution process was effective in reducing concentrations to those usually observed further downstream under regular stream flow conditions.

2.7 Improvements in LAM Network Equipment

Extensive remodeling of LAM network equipment was almost concluded during 1962. The fixed filter was replaced by a step-type moving tape which passes through a filter head under a beta-gamma sensing device The equipment at the station is housed in a metal building (Fig. 17). A new central panel board for recording telemetered information from the air monitors was installed at environmental monitoring headquarters

(Fig. 18). The central panel board contains multipoint recorders for recording information from both the local (LAM) and perimeter air monitoring (PAM) networks. It also contains an alarm device which will be activated by a maximum permissible concentration of radioactivity at any of the air monitors, a range changing switch for each air monitor which changes the recorder range for information being recorded, a tape break alert for each air monitor, and individual switches for remotely advancing the tape at each air monitor in the local network. The new air monitors provide considerably more versatility to the local air monitoring system which lends itself to a better and more rapid evaluation of airborne radioactivity in an emergency situation. Work is currently in progress for the addition of 12 more of the new type air monitors to the local air monitoring network in 1963.

The perimeter air monitoring station (station # 37) located on Hickory Creek Bend on the north side of the Clinch River was relocated in November of 1962. The station was moved to the south side of the Clinch River near the intersection of Hickory Creek and Buttermilk Roads. The relocation was necessary due to the fact that the access road to the old location will be inundated when Melton Hill Reservoir on the Clinch River is filled in 1963.

3.0 PERSONNEL MONITORING

It is the policy of Oak Ridge National Laboratory to monitor the radiation exposure of each individual who enters the Laboratory premises to ensure (1) that personnel exposure is kept to the lowest practical level within permissible limits and (2) to provide a record of any radiation exposure sustained by individuals resulting from Laboratory operations. Personnel monitoring is accomplished by means of personnel meters, personnel surveys for radioactive contaminants, analysis of body fluids, and in vivo gamma counting techniques.

The principal personnel meter for monitoring external radiation exposure is the ORNL Badge-Meter Model II (Fig. 19) which is both a combination radiation exposure meter and security identification badge. The Model II badge-meter is issued to all employees and other individuals who are assigned to work at the Laboratory for an extended period of time. Individuals who enter the Laboratory premises for shorter periods of time are issued a temporary security pass (Fig. 20) which contains a monitoring film packet partially shielded by approximately .015" of indium foil.

In order to control day-to-day exposures and to provide for work assignment scheduling, pocket-type ionization chambers (pocket meters, Fig. 21) are provided for the use of individuals who work in areas where the dose rate and working time is such as to result in an accumulated exposure of 20 mrem or more in a single work day. In addition, other types of personnel metering devices including the pocket screamer (PRM, Fig. 22) are issued when special job assignments and/or exposure conditions are such that the film meter and/or pocket meter alone will not provide the degree of exposure control required.

Administrative procedures, zoning techniques, and special work equipment are used routinely in the work areas to effect control over the radioactive contamination of personnel or where there are indications of past exposure which would warrant further investigation. In addition, internal dose measurements are computed periodically from body fluids analysis techniques. The frequency of these determinations depends upon the radioactive contaminant exposure potential in the work area where personnel participate in work assignments. Ordinarily, the sampling frequency ranges from about once each month for those individuals who work routinely with radioisotopes to once every three years for persons whose potential for exposure to internal emitters is highly limited or even unlikely. Whole body counting techniques are employed when applicable.

3.1 External Exposures

During 1962, no employee received an external radiation dose which exceeded the maximum permissible levels recommended by the Federal Radiation Council (FRC). The highest total body radiation dose received by an employee was about 4.6 rem or 38 per cent of the maximum permissible annual dose. From the above (4.6 rem dose) one sees that no employee received a total body external exposure in excess of the maximum permissible yearly average of 5 rem as derived from the age proration formula 5(N-18).

The record shows that 4,956 persons who were monitored were classified as "employees" during 1962. Of this number, 99.86 per cent (see Table 8) received total body exposures of less than one-third of the maximum permissible annual dose. In all, only 17 persons exceeded an exposure of 3 rem.

As of December 31, 1962, the highest cumulative dose of total body radiation received by an employee was approximately 80 rem. This dose was accrued over an employment period of about 18 years and represented an average annual exposure of about 4.4 rem. The ten highest cumulative doses in this exposure category (see Table 9) ranged downward from the high of 80 rem to approximately 54 rem. Two individuals (each of whom had accumulated only ten years of employment service) had recorded cumulative exposures which resulted in an average annual exposure of 6.8 and 5.8 rem respectively. The other eight employees had recorded cumulative exposures that averaged less than 5.0 rem per year.

At the close of the year, only one employee had a cumulative total body dose which exceeded the age proration formula 5(N-18). Practically all of the dose recorded for this employee (67.6 rem) resulted from an accident which occurred during 1957 and at the end of 1962 represented about 135 per cent of the dose permitted by the age proration formula (see Table 10).

3.2 Internal Exposure

During 1962 there were no cases of internal exposure where the deposition of radioactive materials within the body was estimated to have averaged greater than half of a maximum permissible body burden.[7] One case had developed involving a build-up in the body of transuranic alpha emitters which appears to be approaching 40 per cent of the maximum permissible body burden. At this particular time the elimination rate is not sufficiently known so as to permit a precise estimate. However, it appears that the situation is such as to predict that the final estimate will run between 35 and 45 per cent of the maximum permissible body burden. Two other employees had accumulated body burdens of Pu-239 which were about 35 per cent of the maximum permissible value.[8] Health Physics procedures require that individuals who exceed 30 per cent of a maximum permissible body burden be placed on a work assignment where the potential for internal exposure is reduced.

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[7] AEC Manual Chapter 0502 requires an evaluation of the radiation exposure status of an employee when monitoring techniques indicate that a body burden equals or exceeds 50 per cent of a maximum permissible limit.

[8] Handbook 69 values are the basis for these determinations.

3.3 Monitoring Resume

Pocket Meters - The number of pocket meters processed during 1962 (Table 11) totaled 358,525 which represented an increase of about five per cent over the number issued during 1961. This increase was due in part to the fact that supervision in the work areas has come to rely more and more on the daily record generated from pocket meter data as a means of achieving day-to-day radiation control.[9] From Table 11 it will be seen that out of the 358,525 meters issued only 93 meters were returned for processing in a non-readable condition. In all, only 1220 meters were recorded as off-scale. The off-scale readings resulted from all of the usual causes (leakers, mishandling, tampering, etc.) and included those meters that were subjected to radiation exposure above 0.2 rad—the upper readable limit for this model meter. The number of paired off-scale readings (above 0.2 rad)[10] totaled 87 for the year while there was a statistical probability of about 9×10^{-4} that two "leakers" would be paired randomly prior to issuance. Thus, from a statistical point of view, only two pairs of pocket meters were issued during 1962 which should have recorded unreliable off-scale readings.

Film Meters - The total number of monitoring films processed during 1962 was 89,370 (Table 11). Of this number about 22 per cent were NTA films used for neutron monitoring. Only about 17 per cent of the NTA films were checked for neutron exposure—the processing criteria being based upon whether or not the monitored individuals had a work history involving potential neutron exposure. All other films (beta-gamma sensitive) were processed according to standard procedures and the monitoring results recorded for the record.

During the last quarter of 1962, the NTA films were desiccated and sealed in moisture-proof "pouch" paper. Cheka[11] has shown that fading of the latent image is reduced by an appreciable factor when the NTA type emulsion is packaged in a "moisture-proof" container. Thornton, Davis,

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[9] In the immediate years following the advent of the atomic energy industry, considerable difficulty was experienced with pocket meters in that there were many defects inherent in the manufacturing process. These defects led to a high frequency in the leakage rate with the result that some individuals justifiably lost confidence in this method of monitoring. Since about 1950 ORNL has had excellent results with pocket meters and their performance during 1962 is not much different from that experienced over the past 12 years.

[10] When paired off-scale readings occur, the film badge-meter is processed to establish the rem dose.

[11] "A Neutron Film Dosimeter", J S. Cheka, Proceedings of the Health Physics Society, First Annual Meeting, Ann Arbor, Michigan, June 25-27, 1956. Also, see "A Neutron Film Dosimeter", J. S Cheka, Nucleonics, Vol. 12, No. 6, p. 6, 1954.

and Gupton[12] experimented with moisture-proof pouch paper and found that the latent image fading could be controlled while the film remained in the badge-meter (see Fig. 23). With this added sensitivity it is no longer necessary to process the NTA film on a monthly cycle with the result that it is now possible to achieve complete film monitoring for beta, gamma, and neutron exposure on a regular quarterly exchange cycle.[13]

Body Fluids Analysis - The urine sampling program was expanded during 1962 with provisions being made for the sampling of all Laboratory employees on a routine three-year cycle. (Employees who work regularly with radio-isotopic mixtures would be sampled more frequently in accordance with past procedures.) Consequently, there was almost a twofold increase in the number of analyses (6,718) performed during 1962 over the number (4,150) recorded during 1961. About 85 per cent of the analyses (Table 12A) included Sr-90 and gross alpha determinations.

Whole Body Counter[14] - During the calendar year 1962 the routine counting program included 395 human counts. Most of these were 20-minute counts in the chair position using an 8" x 4" NaI (Tl) crystal located in a fixed geometry relative to the chair.[15] As a result of instrumentation and program improvements completed at the end of 1962, it is anticipated that the scope of the routine counting program can be expanded (a) to include a greater number of routine counts of potentially exposed persons, and (b) to allow for a greater number of baseline counts on individuals prior to new work assignments involving radioactive materials handling.

Measurable amounts of internal radioactive contamination were found in 44 persons (see Table 12B). The highest indicated internal exposure measured during 1962 involved a person who was examined 15 days after inhalation exposure to I-131 vapor. By comparison with a similar inhalation case where the individual was examined within a few hours after exposure, it was estimated that the initial intake of I-131 was approximately 1.5 μc by inhalation (one-third of the maximum permissible quarterly intake).

Laundry Monitoring - Approximately 460,000 articles of wearing apparel passed through the laundry monitoring unit during 1962. About three per cent of the items checked were found, following laundering, to be above maximum permissible contamination limits for contamination zone clothing.

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[12] "The ORNL Badge Dosimeter and Its Personnel Monitoring Applications", W. T. Thornton, D. M. Davis, E. D. Gupton, ORNL-3126, December 5, 1961.

[13] Prior to the use of the moisture-proof wrapper, it was necessary to process NTA films on a two to four week cycle for persons who worked regularly in areas where neutron exposure was likely.

[14] Data provided by B. R. Fish, et al., Health Physics Technology Section.

[15] Health Physics Division Annual Progress Report (for the period ending July 31, 1961), ORNL-3189, pp. 222-224, discusses in vivo counting techniques.

4.0 LABORATORY OPERATIONS MONITORING

The Applied Health Physics Annual Report for 1961 (ORNL-3284) discusses the evolution of the concept of the term UNUSUAL OCCURRENCE and lays down certain ground rules for classifying radiation accidents, or near accidents, in accordance with the unusual occurrence concept. In general, an unusual occurrence is considered to have taken place when one or more of the following occurs:

1. A violation of a Health Physics regulatory policy.

2. An event which might have resulted in significant personnel exposure or facility contamination under less fortunate circumstances.

3. An event which might have had public relations significance under less fortunate circumstances.

4. An event where the radiation dose exceeds 1/3 of a maximum permissible quarterly dose.[14]

5. A radiation or contamination incident of a magnitude sufficient to result in a significant curtailment of operations.

An unusual occurrence is considered to be a major event when (1) an individual receives a radiation dose in excess of maximum permissible limits as recommended by the FRC, (2) Laboratory operations result in the contamination of the environment surrounding the Laboratory area in excess of levels recommended by the FRC, or (3) the cost of reclaiming a laboratory facility following a radiation incident exceeds $5,000. All other unusual occurrences are considered to be minor events.

It is obvious from the above that a major event constitutes an incident of significant proportions if for no other reason than that the recommended maximum permissible limits have been breached (no matter how slight) and/or normal operating costs have been increased appreciably. In the case of the minor event, it is not immediately obvious that the situation has taken on significant proportions as the incident may merely call attention to the fact that operations have become marginal. Where a Health Physics regulatory policy has been violated—the event is significant only in an executory sense; where an event "might have resulted in significant personnel exposure or facility contamination" under less fortunate circumstances—the event takes on the color of a "near miss" and does not necessarily develop into a significant event; again, as in the case where "operating limits" (limits set below the recommended maximum permissible limits) are involved—the event is only of executory interest.

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[14] The maximum permissible quarterly dose is based on FRC values.

Thus, it seems appropriate to define an unusual occurrence as being a significant occurrence when the event is such as to (1) exceed a recommended maximum permissible limit and/or (2) require a work stoppage in an operation while clean-up measures are instituted following a radioactive contaminant release. Obviously, the above definition distinguishes between the major event and the minor event only in the matter of costs involved in the restoration of an operating facility following an unusual occurrence and in those instances involving "near misses" or executory matters. An event where the recommended maximum permissible limits are exceeded is always categorized as a significant occurrence and will be reported as a major event regardless of the degree of overage.

4.1 Unusual Occurrences Summarized, 1960-62

The Laboratory experienced 55 unusual occurrences during 1962, and for the first time since this method of reporting was originated in 1960, no major incidents were recorded. (See Table 13.) The 55 events which occurred during 1962 represented a reduction in unusual occurrences of about 25 per cent over 1961. The 1961 total was about 14 per cent less than the 1960 total. Thus, the number of events recorded in 1962 was only about 61 per cent of the 1960 total.

4.2 Significant Occurrences, 1960-62

Slightly less than 55 per cent of the unusual occurrences recorded during 1962 and 1961 were categorized as significant.[15] However, about 69 per cent of all unusual occurrences recorded in 1960 were in the significant category. Thus, the frequency rate for significant occurrences dropped in 1962 and 1961 to about 86 per cent of the 1960 frequency rate.

4.3 Personnel Exposures

No personnel exposures occurred during 1962 which were classifiable as significant (Table 13, Part II). In fact, there were only seven instances where the planned operational exposure limit[16] was exceeded. Although the 1962 total slightly exceeded the 1961 total, the 1962 experience was by far the best recorded during the three-year period which ended with 1962. (One major event was recorded in 1960; two major events were recorded in 1961.)

4.4 Contamination Incidents

About 30 of the 55 unusual occurrences which occurred during 1962 involved radioactive contaminant releases (Table 13, Part III). In fact, all 30 of these events were categorized as significant and some work re-

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[15] The significant occurrence is discussed in the introductory comments.

[16] Planned exposures are calculated so as not to exceed 1/3 of a recommended maximum permissible quarterly dose.

strictions and/or special clean-up measures were required. Even so, program losses experienced during 1962 were of a minor nature as it was necessary to utilize interdepartmental assistance to effect clean-up measures only on two occasions (Table 13, Part III, Item 6).

It is evident that the contamination incident governs the unusual occurrence frequency rate and is usually involved (Fig. 24) where the incident takes on significant proportions.[17] During the three years which ended with 1962, there were 131 unusual occurrences that could be classified as significant. Of the 131 events, the contamination of the premises was involved 129 times. However, the importance of the containment program (begun in 1960) and an accelerated health physics program is evidenced by the fact that the contamination incident frequency rate dropped to about two-thirds of the 1960 rate in 1961 and was further reduced in 1962 to about half of the 1960 rate.

4.5 Unusual Occurrence Frequency Rate

As a general rule the frequency rate will be somewhat related to (1) the quantity of radioactivity handled, (2) the number of radiation workers assigned to the work unit, (3) the type of operating facilities utilized, or (4) the radiation hazard potential associated with a particular operation. In the discussion which follows, no attempt has been made to evaluate the frequency rate in terms of the above factors. Consequently, the data do not necessarily reflect the degree of adequacy of performance within a particular work unit.

Frequency Rate Among the Laboratory Divisions - During 1962 there were 55 unusual occurrences recorded among 11 Laboratory Divisions (Table 14). Four of the 11 Divisions recorded unusual occurrences at or above the mean (5 events) and about 76 per cent of all events were attributed to these four operating groups as follows:

1. Isotopes (18 events)
2. Chemical Technology (13 events)
3. Operations (6 events)
4. Analytical Chemistry (5 events)

There were 217 unusual occurrences (Table 14) recorded among 19 Laboratory Divisions during the three-year period which ended with 1962. Using the mean value (~ 12 events) for comparison, about 80 per cent of all the unusual occurrences were recorded among six of the 19 operating groups as follows:

1. Chemical Technology (49 events)
2. Isotopes (41 events)
3. Operations (32 events)
4. Reactor (18 events)
5. Analytical Chemistry (12 events)
6. Engineering and Mechanical (12 events)

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[17] Cf. Item 2 of Part I in Table 13 with the totals shown in Part II of Table 13.

From Table 14 it will be observed that the Reactor Division recorded no events during 1962. (Table 14 shows the unusual occurrence frequency rate for all Divisions of the Laboratory for the three-year period which ended with 1962.)

Frequency Rate Among Operating Facilities - Unusual occurrences took place in 22 operating facilities (Table 15) during 1962. (Four events occurred out-of-doors in areas designated as "miscellaneous".) Two or more events occurred in 14 of the 22 facilities; three or more events occurred in only five of the 22 facilities; and only two of the 22 operating facilities experienced more than three events. Nine events were experienced in Bldg. 3019 and eight events took place in operations conducted at Bldg. 3517. Thus, about one-third of the unusual occurrences experienced during 1962 took place in two of the 22 operating facilities that recorded unusual occurrences. Even though operations in Bldg. 3019 were responsible for about 16 per cent of the 1962 total, there was some improvement over the 1961 experience as the number dropped from 16 events in 1961 to nine events in 1962. The reverse was true in operations being conducted in Bldg. 3517 where the total number of events recorded in 1962 increased from three events recorded in 1961 to eight events recorded in 1962.

Over the three-year period which ended with 1962 (Table 15) there were 197 unusual occurrences recorded in 38 operating facilities. (Twenty events, shown under the heading "miscellaneous" in Table 15, occurred out-of-doors in areas that could not be classified generally as operating facilities.) About 45 per cent of the 197 events occurred in only five of the 39 operating facilities as follows:

1.	Bldg. 3019	(36 events)
2.	Bldg. 3517	(17 events)
3.	Bldg. 7500	(12 events)
4.	Bldg. 9201-2	(12 events)
5.	Bldg. 3042	(11 events)

Five facilities recorded between six and eight events which represented about 18 per cent of the three-year total as follows:

6.	Bldg. 4501	(8 events)
7.	Bldg. 3001	(7 events)
8.	Bldg. 3025	(7 events)
9.	Bldg. 9204-1	(7 events)
10.	Bldg. 3028	(6 events)

The remaining events (about 37 per cent) were spread over 29 of the 39 operating facilities with 12 facilities recording only one event each during the three-year period.

It should be noted that only two of the 10 operating facilities which recorded six or more events during the three-year period which ended in 1962 involved the operation of a nuclear reactor. The HRE (located in Bldg. 7500) and the Graphite Reactor (located in Bldg. 3001) recorded 19 events which represents about 15 per cent of the 123 events recorded in these 10 facilities. Thus, chemical operations continue to be the principal source of the unusual occurrence at ORNL.

4.6 Unusual Occurrence Distribution (Day of the Week)

The unusual occurrence frequency rate continued to be slightly higher on Friday—the last full workday of the 5-day regular work week—during 1962. Although these statistics (see Table 16) are not immediately conclusive, it is interesting to note that during 1960 and 1961 about one-third fewer events occurred on Thursday as compared with the other four regular workdays beginning with Monday and ending with Friday. However, this trend changed during 1962 where Tuesday replaced Thursday as the low frequency day and the number of events occurring on Tuesday was about half of the number occurring on other regular workdays. An analysis of work schedules may possibly lead to an understanding of the relationship between the unusual occurrence frequency rate and causative factors which may be eliminated.

5.0 LABORATORY SUPPORT FACILITIES

Staff technicians who perform laboratory analysis for the various monitoring units may be responsible for a variety of tasks ranging from chemical analysis to instrumentation applications. In general, each technician has a specific group of assignments for which he is especially responsible; however, most of the technicians who are assigned to laboratory duties are proficient in all phases of the analytical program and from time-to-time are assigned to each of the various laboratory support facilities.

5.1 Radiography

There were 93,469 pieces of film processed by the radiography units during 1962 (Table 17). About 95 per cent of the 1962 output involved films used in meters for personnel monitoring purposes that are processed on an assembly line at a relatively high rate. The remaining five per cent consisted largely of autoradiograms fabricated from 14 x 17 inch films which are used in radioparticulate studies conducted by the Environmental Monitoring units.[18]

Autoradiographic techniques developed by the Los Alamos Laboratories for determining plutonium concentrations in urine[19] were modified and adapted for use during 1962. The process utilizes a chemical separation technique following which the plutonium in various chemical forms is plated out on a 1/2-inch stainless steel planchet. The plated planchet is then placed in contact with a nuclear track emulsion for a 168-hour exposure period. After development, the number of alpha tracks in the emulsion is determined by microscopy. From microscopy data the deposition of plutonium in the body is estimated. The method is about ten times more sensitive than methods which use conventional counting techniques. However, at ORNL, the conventional counting techniques are used most of the time as a high degree of sensitivity is not required.

A film reader which converts density readings directly into dose units was developed and put into service during 1962. The principal components consist of (1) an Ansco model 12 densitometer, (2) a four-place digital voltmeter, (3) an amplifier for the conversion of the densitometer meter current, (4) a power supply, and (5) panel meters which may be used in lieu of the digital voltmeter. All components are mounted on a console (Fig. 25) at which the operator sits during the reading of films. The digital voltmeter and amplifier convert densities to mR equivalent readings. The normal function of the densitometer is not altered and it

[18] Methods described in ORNL-2601, "Radioactive Waste Management at Oak Ridge National Laboratory".

[19] "A New Procedure for Plutonium Analysis", Health Physics, Vol. 6, No. 3-4, October, 1961.

may be used in a conventional manner to determine densities up to a density of 4.0. The useful range of the device runs from approximately 0 to 1500 mR equivalent when used with duPont emulsion 555 developed for three minutes under standard darkroom conditions. As monitoring films seldom are exposed to readings in excess of 1500 mR equivalent, it is only very rarely required that density-dose graphs need to be prepared. Thus, the conventional three-step operation which requires a density determination, the preparation of a graph, and the picking off of dose readings from densities has been replaced by a single operation. The principal advantages gained by the use of the film reader described above lies in the reduction of clerical errors coupled with minor savings in time.

5.2 Counting Facilities

There was a drop of about ten per cent in the number of samples processed by the counting facilities during 1962 as compared with the previous year. A breakdown showing the number and type of samples processed is found in Table 18. The reduction in the number of samples handled was due to a sizable reduction in operations involving the rehabilitation of contaminated operating facilities. Counting facilities were expanded during 1962 in order to provide for greater capability in threshold detector measurements. Two complete threshold detector foil counting units were made available and a 4 x 5 inch thalium activated sodium iodide crystal was obtained for a single channel gamma analyzer to enable blood-sodium activation analysis.

5.3 Radiochemical Analysis

Radiochemical analysis is required extensively in conjunction with internal dose determinations where body fluids are analyzed and in the Environmental Monitoring program. During 1962 these laboratory units processed 7,382 body fluid specimens (Table 19), and examined 13,753 environmental monitoring samples (Table 20). The methods used by the various analytical groups have been described elsewhere.[20]

5.4 Dose Analyses

A significant addition to the dose analyses program during 1962 involved a method for estimating the plutonium body burden from urinalysis data by the use of high speed digital computers.[21] In this system computer codes are used to estimate the body burden by the power function equation, by intake minus total excretion, or by the accumulated intake. There are three modes of operation specifying the date of an estimation:

[20] ORNL Master Analytical Manual.

[21] The Estimation of a Body Burden of Pu from Urinalysis Data, W. S. Snyder, Proceedings of the Seventh Annual Bio-Assay and Analytical Chemistry Meeting, Argonne National Laboratory, October 12-13, 1961.

one mode gives estimates on each data-point day, another on specified dates, and the third on integral multiples of some specified number of days. Programming was done with the IBM 7090 located at the Oak Ridge Central Data Processing Facility. Numerous advantages accrue in the use of this method. The principal advantage lies in the speed with which body burden estimates can be made from accumulated urinalysis data.

5.5 Health Physics Instrumentation

The Instrumentation and Controls Division is responsible for the final development of health physics instruments, the recommending of specific models for purchase, and for operational maintenance on health physics instruments used in the field. The Applied Health Physics Section collaborates in the development and design of health physics instruments specifying criterial requirements.

During 1962 the following instruments were designed and/or put into use:

1. A fall-out monitor (ORNL Q-2256) for measuring the beta, gamma, and alpha components of fall-out material was designed, tested, and approved for fabrication. Approximately 24 of these devices are scheduled to be installed as part of the ORNL Emergency Radiation Monitoring system.

2. A neutron hazard monitor (ORNL Q-2562) which activates an alarm at a pre-set flux level and is sensitive to neutron "bursts" was designed, tested, and approved for fabrication. This device will be used in the ORNL Emergency Radiation Monitoring system.

3. A light transmission spectrometer (B & L Spectromic 20) was modified to provide a means of evaluating chemical dosimeters of the type currently being tested for use in the ORNL Model II Badge-Meter.

4. An improved environmental air monitoring station network which feeds monitoring data to a central control panel located at monitoring headquarters has been installed. The new system enables rapid dissemination of monitoring data in the event of an accidental release of air-borne radioactivity over the Laboratory premises.

5. A scintillation, alpha sample counter (ORNL Q-2287) for use with the ORNL Model Q-2188 Scaler was designed, tested, and approved for fabrication. A few of these counters have been placed in service.

6. Improvements have been made in the design of the Continuous Alpha Air Monitor (Model Q-2340) which simplifies the interpretation of data and eliminates certain operational

difficulties. These modifications are being incorporated in all instruments now in use and the design changes have been included in specifications governing instruments to be fabricated in the future.

7. A Semi-Automatic Film Reader for use with monitoring films used in the ORNL Model II Badge-Meter has been designed and partially fabricated. The device feeds dose data directly to an IBM punch card machine and, when perfected, will reduce handling requirements, simplify data processing techniques, and reduce common transcription errors inherent in manually operated systems.

5.6 Calibration Facility

Some 6585 portable instruments and 6082 films were calibrated for various source materials during 1962 (Table 21). There was an increase of approximately 15 per cent in the number of battery-powered portable instruments calibrated in 1962 compared with the number calibrated in 1961. The totals for other calibrations performed differed only slightly from the totals for 1961.

The inventory of battery-powered portable instruments (Table 22) at the close of 1962 increased by about 10 per cent over the 1961 inventory. The general inventory has increased from 511 battery-powered instruments in 1957 to 901 such instruments in 1962. Improvements in the calibration techniques permitted this almost twofold increase in the calibration program and allowed a corresponding annual decrease in personnel requirements from 3.5 man years to 2.9 man years.

6.0 PUBLICATIONS AND REPORTS

6.1 Publications

E. L. Sharp and W. P. Ellis, "Smear Techniques for Surface Monitoring for Radioactive Materials", ORNL Central Files Number 62-10-1, October 1, 1962.

W. W. Ogg, "Report of Health Physics Advisor for Reactor Start-up at Atomic Energy Research Institute, Republic of Korea, Seoul, Korea", ORNL Central Files Number 62-5-52, May 15, 1962

W. W. Ogg, "Health Physics Report to the Government of Korea", IAEA, TA Report Number 59.

W. W. Ogg, "Part II of Hazards Evaluation for ROKAERI Triga Mark II Reactor".

F. F. Haywood, et al., "Technical Concept—Operation BREN", CEX 62.01.

F. F. Haywood, et al., "Operation Plan and Hazards Report—Operation BREN".

6.2 Interdepartmental Reports

Applied Health Physics Quartery Report - January, February, and March of 1962, CF 62-5-65.

Applied Health Physics Quarterly Report - April, May, and June of 1962, CF 62-8-84.

Applied Health Physics Quarterly Report - July, August, and September of 1962, CF 62-11-74.

Applied Health Physics Quarterly Report - October, November, and December of 1962, CF 63-3-51.

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TABLE 1 CONCENTRATION OF RADIOACTIVE MATERIALS IN AIR - 1962
(Filter Paper Data—Weekly Average)

Station Number	Location	Long-Lived Activity 10^{-13} μc/cc	No. of Particles by Activity Ranges[a] $< 10^5$ d/24 hr	10^5-10^6 d/24 hr	10^6-10^7 d/24 hr	$> 10^7$ d/24 hr	Total	Particles Per 1000 ft^3
			Laboratory Area					
HP-1	S 3587	38	128	1.6	0.00	0.00	129	3.1
HP-2	NE 3025	43	122	1.9	0.04	0.00	124	3.5
HP-3	SW 1000	37	129	2.1	0.10	0.02	131	2.1
HP-4	W Settling Basin	21	91	1.2	0.04	0.00	93	1.6
HP-5	E 2506	51	115	1.2	0.04	0.04	117	3.9
HP-6	SW 3027	33	136	1.5	0.02	0.02	137	2.4
HP-7	W 7001	40	115	1.8	0.00	0.00	117	2.3
HP-8	Rock Quarry	39	132	1.5	0.00	0.02	133	2.5
HP-9	N Bethel Valley Rd.	31	145	1.6	0.00	0.00	146	2.3
HP-10	W 2075	38	126	1.3	0.00	0.00	128	3.1
Average		37	124	1.6	0.02	0.01	125	2.7
			Perimeter Area					
HP-31	Kerr Hollow Gate	34	135	1.6	0.04	0.04	137	2.7
HP-32	Midway Gate	37	132	2.1	0.02	0.00	134	2.6
HP-33	Gallaher Gate	32	113	1.4	0.00	0.02	114	2.2
HP-34	White Wing Gate	34	153	1.5	0.00	0.00	155	3.0
HP-35	Blair Gate	39	168	1.6	0.00	0.02	169	3.3
HP-36	Turnpike Gate	39	158	2.2	0.02	0.04	161	3.2
HP-37	Hickory Creek Bend	34	114	1.6	0.02	0.00	115	2.3
Average		36	139	1.7	0.01	0.02	141	2.8
			Remote Area					
HP-51	Norris Dam	43	139	2.3	0.04	0.00	141	2.6
HP-52	Loudoun Dam	42	130	2.8	0.10	0.00	133	2.4
HP-53	Douglas Dam	44	150	2.6	0.02	0.00	153	2.8
HP-54	Cherokee Dam	40	164	2.4	0.04	0.02	167	3.0
HP-55	Watts Bar Dam	45	157	2.0	0.04	0.00	159	2.9
HP-56	Great Falls Dam	46	166	2.3	0.00	0.00	168	3.1
HP-57	Dale Hollow Dam	38	171	1.6	0.00	0.04	172	2.9
Average		43	154	2.3	0.03	0.01	157	2.8

[a]Determined by filtration techniques.

TABLE 2 RADIOPARTICULATE FALL-OUT - 1962

(Gummed Paper Data—Weekly Average)

Station Number	Location	Long-Lived Activity 10^{-13} μc/cc	No. of Particles by Activity Ranges					Total Particles Per Sq. ft.
			< 10^5 d/24 hr	10^5-10^6 d/24 hr	10^6-10^7 d/24 hr	> 10^7 d/24 hr	Total	
			Laboratory Area					
HP-1	S 3587	15	79	2.1	0.12	0.06	81	42
HP-2	NE 3025	17	88	2.3	0.04	0.06	91	49
HP-3	SW 1000	15	83	2.0	0.15	0.06	86	42
HP-4	W Settling Basin	14	73	2.3	0.08	0.04	75	48
HP-5	E 2506	14	86	2.0	0.08	0.04	91	50
HP-6	SW 3027	16	101	2.9	0.02	0.02	104	61
HP-7	W 7001	15	89	2.4	0.02	0.06	92	49
HP-8	Rock Quarry	17	89	2.6	0.00	0.08	91	46
HP-9	N Bethel Valley Rd.	16	88	2.9	0.06	0.12	91	41
HP-10	W 2075	15	100	2.3	0.04	0.00	103	55
Average		15	88	2.4	0.06	0.05	91	48
			Perimeter Area					
HP-31	Kerr Hollow Gate	17	103	2.13	0.13	0.10	105	47
HP-32	Midway Gate	16	99	2.6	0.10	0.06	102	46
HP-33	Gallaher Gate	14	82	2.4	0.10	0.00	85	42
HP-34	White Wing Gate	18	104	2.2	0.19	0.08	106	47
HP-35	Blair Gate	15	124	2.0	0.06	0.04	126	50
HP-36	Turnpike Gate	16	109	3.5	0.08	0.02	112	57
HP-37	Hickory Creek Bend	16	85	2.3	0.04	0.08	87	47
Average		16	101	2.5	0.10	0.05	103	48
			Remote Area					
HP-51	Norris Dam	14	86	2.2	0.12	0.04	89	36
HP-52	Loudoun Dam	13	70	2.7	0.06	0.06	73	29
HP-53	Douglas Dam	13	77	2.7	0.06	0.08	80	35
HP-54	Cherokee Dam	14	81	2.9	0.13	0.06	84	35
HP-55	Watts Bar Dam	16	81	2.2	0.14	0.08	83	37
HP-56	Great Falls Dam	14	98	2.2	0.06	0.02	100	39
HP-57	Dale Hollow	14	96	2.0	0.08	0.06	98	33
Average		14	84	2.4	0.09	0.06	87	35

TABLE 3 CONCENTRATION OF RADIOACTIVE MATERIALS IN RAIN WATER - 1962
(Weekly Average by Stations)

Station Number	Location	Activity in Collected Rain Water, μc/cc
	Laboratory Area	
HP-7	West 7001	10.3×10^{-7}
	Perimeter Area	
HP-31	Kerr Hollow Gate	11×10^{-7}
HP-32	Midway Gate	12
HP-33	Gallaher Gate	10
HP-34	White Wing Gate	11
HP-35	Blair Gate	11
HP-36	Turnpike Gate	10
HP-37	Hickory Creek Bend	11
Average		11×10^{-7}
	Remote Area	
HP-51	Norris Dam	14×10^{-7}
HP-52	Loudoun Dam	11
HP-53	Douglas Dam	13
HP-54	Cherokee Dam	11
HP-55	Watts Bar Dam	14
HP-56	Great Falls Dam	16
HP-57	Dale Hollow Dam	11
Average		13×10^{-7}

TABLE 4 LIQUID WASTES DISCHARGED FROM WHITE OAK CREEK, 1962

	Curies		% Deviation from
	Total for Year	Weekly Average	1961 Weekly Average
Beta Activity	1436	27.6	- 34
Transuranic Alpha Emitters	.063	.0012	- 8

NOTE: The weekly average concentration of transuranic alpha emitters in the Clinch River was 3.6×10^{-9} μc/ml which was 28% less than the 1961 value.

TABLE 5 RADIOACTIVITY IN CLINCH RIVER - 1962

Location	Concentration of Nuclides of Primary Concern in Units of 10^{-8} μc/cc						Average Concentration of Total Radioactivity 10^{-8} μc/cc	$(MPC)_w$ [a] 10^{-6} μc/cc	% of $(MPC)_w$
	Sr^{90}	Ce^{144}	Cs^{137}	$Ru^{103-106}$	Co^{60}	Zr^{95}-Nb^{95}			
CRM 41.5[b]	0.16	0.14	0.02	0.78	*	0.42	1.5	0.90	1.7
CRM 20.8[c]	0.15	0.02	0.09	21	0.18	0.09	34	4.6	7.4
CRM 4.5[b]	0.34	0.20	0.07	16	0.32	0.54	17	3.5	4.9

[a] Weighted average $(MPC)_w$ calculated for the mixture, using $(MPC)_w$ values for specific radionuclides recommended in NBS Handbook 69.

[b] Measured values.

[c] Calculated values based on the levels of waste released and the dilution afforded by the river.

* None detected.

TABLE 6 AVERAGE CONCENTRATION OF MAJOR RADIOACTIVE CONSTITUENTS IN THE CLINCH RIVER AT MILE 41.5[a]

Period	Units of 10^{-8} μc/ml					
	Sr^{90}	Ce^{144}	Cs^{137}	$Ru^{103-106}$	Co^{60}	$Zr\text{-}Nb^{95}$
Third Qtr., 1961	0.10	0.05	*	0.45	*	*
Fourth Qtr., 1961	0.08	0.04	0.05	0.59	0.06	0.32
1st Half, 1962	0.20	0.17	0.01	0.90	*	0.68

[a]Sampling station moved from Clinch River Mile 33.2 to Mile 41.5 about January 1, 1962.

*None detected.

TABLE 7 RADIONUCLIDES IN RIVER SILT - 1961-1962
(Units of 10^{-6} μc/g of Dried Mud)

Location	Cs^{137}		Ce^{144}		Sr^{90}		Co^{60}		$Ru^{103-106}$		Zr^{95} + Nb^{95}		TRE* + Y^{90} (as Y^{90})	
	1961	1962	1961	1962	1961	1862	1961	1962	1961	1962	1961	1962	1961	1962
CRM 21.5	1.3	3.2	0.44	11	0.26	0.36	0.32	--	2.7	11	0.50	16	0.7	10
19.1	41	5.2	2.7	3.8	1.0	0.41	5.9	0.7	95	6.1	1.4	6.2	7.8	3.5
16.3	71	58	5.6	5.2	2.0	0.72	11	8.1	159	50	1.7	3.9	16	14
15.2	64	55	4.4	5.2	0.77	0.90	10	7.3	148	46	1.8	4.2	16	13
14.0	127	237	8.2	6.2	1.1	1.8	14	20	153	43	1.8	3.6	31	31
11.0	98	63	9.4	6.9	1.0	1.0	14	8.6	144	68	4.6	5.4	29	16
8.0	81	59	9.4	8.5	1.4	1.0	11	8.6	152	70	2.3	5.4	26	18
5.8	115	94	9.9	8.4	1.0	1.6	14	12	157	68	2.7	6.5	28	22
4.7	112	86	9.9	9.5	1.3	1.2	15	14	148	86	2.7	6.0	20	22
2.6	82	73	7.0	7.7	0.90	0.72	11	10	103	77	1.1	5.6	23	16
1.1	100	56	8.6	13	0.41	0.72	12	9.0	141	76	1.8	11	35	16
Average	81	72	6.9	7.7	0.85	0.95	11	9.8	128	55	2.0	6.7	21	17

*TRE - total rare earths minus cerium.

TABLE 8 DOSE DATA SUMMARY FOR LABORATORY POPULATION INVOLVING EXPOSURE TO TOTAL BODY RADIATION - 1962

Dose Range in Rem Units	Number of Persons Monitored	Percentage of Population
0 - 1	4737	95.59
1 - 2	158	3.19
2 - 3	44	.88
3 - 4	10	.20
4 - 5	7	.14
Above 5	0	0.00
TOTALS	4956	100.00

TABLE 9 PERTINENT DATA REGARDING THE TEN LABORATORY EMPLOYEES WHO HAVE RECEIVED THE HIGHEST CUMULATIVE DOSE OF TOTAL BODY RADIATION AS OF DECEMBER 31, 1962

Employee	Dept./Div. Presently Assigned	Employee's Age on Dec. 31, 1962	Tenure of Employment in Years	Dose in Rem Units	
				Av. Dose Each Yr. of Employ.	Total Cum. Dose
1	Isotopes	43	18	4.4	80.5
2	Isotopes	44	15	4.9	73.8
3	Isotopes	56	18	3.9	70.4
4	Isotopes	55	17	4.0	68.3
5	E and M*	28	10	6.8	67.6
6	Isotopes	38	19	3.3	52.8
7	Isotopes	34	10	5.8	57.6
8	Operations	44	19	2.9	55.8
9	Isotopes	31	11	4.9	54.2
10	Isotopes	43	11	4.9	53.6

*Major portion of exposure received while assigned to the Chemical Technology Division.

TABLE 10 PERTINENT DATA REGARDING EMPLOYEES WHOSE CUMULATIVE TOTAL BODY EXPOSURE EXCEEDS 50 PER CENT OF THE AGE PRORATION FORMULA 5(N-18) AS OF DECEMBER 31, 1962

Employee	Dept./Div. Presently Assigned	Employee's Age on Dec. 31, 1962	Tenure of Employment in Years	Cumulative Dose in Rem Units	Percentage of the Quantity 5(N-18)
A (5)	E and M*	28	10	67.6	135
B (9)	Isotopes	31	11	54.2	83
C (7)	Isotopes	34	10	57.6	72
D	Isotopes	33	13	50.1	67
E (1)	Isotopes	43	18	80.5	64
F (6)	Isotopes	38	19	52.8	60
G	Isotopes	35	12	50.2	59
H (2)	Isotopes	44	15	73.8	57
I	Isotopes	38	16	53.5	54
J	I and C	32	11	35.4	51

*Major portion of exposure received while assigned to the Chemical Technology Division.

Note: Six of the employees who appear in this table are listed in Table 9 as being among the ten employees who have received the highest cumulative dose of total body radiation as of December 31, 1962. The numeral in parenthesis shows the relative position in Table 9.

TABLE 11 PERSONNEL METER DISTRIBUTION AND PERFORMANCE DATA

A. Pocket Meters

1.	Meters distributed	358,525
2.	Non-readable meters	93
3.	Off-scale readings	1,220
4.	Off-scale pairs	87

B. Film Meter Processing Data

1.	Film badge meters (record)	20,700
2.	Film badge meters (non-record)	2,496
3.	Film meters (temporary passes)	38,318
4.	Hand meters, special packets, etc.	5,699
5.	Neutron films developed (not read)	16,089
6.	Neutron films developed (read)	3,236
7.	Films for non-ORNL groups	2,832
	Total Number of Films Processed	89,370

TABLE 12A BIO-ASSAYS ANALYSES - 1962

Analytical Procedure	Number of Analyses
Urine:	
Gross Alpha	3059
Sr^{90}	2800
U	509
TRE (total rare earths)	127
H^{3}	62
Cs^{137}	43
Pu^{239}	14
Ru^{106}	5
Sr^{89}	4
P^{32}	4
Other	91
	6718
Fecal:	
Gross Alpha	68
Sr^{90}	6
	74
GRAND TOTAL	6792

TABLE 12B MEASURABLE RADIOACTIVITY FOUND IN ROUTINE WHOLE BODY MONITORING PROGRAM - CALENDAR YEAR 1962[1]

Isotope	Number People	Highest Quantity Measured (μc)	Maximum Permissible Burden (μc)
Cs^{137}	12	0.36	30
I^{131}	17	0.28	0.7 (thyroid)
Sb^{125}	6	0.16	40
Ce^{144}	2	0.062	20
Ru^{106}	3	0.13	10
Co^{60}	4	0.002	10
Zr^{95}	3	0.005	20
Zn^{65}	1	0.003	60

[1]Information provided by Health Physics Technology Section - B. R. Fish, et al.

TABLE 13 UNUSUAL OCCURRENCES SUMMARIZED FOR THE 3-YEAR PERIOD ENDING WITH 1962

	Yearly Totals		
	1960	1961	1962
Part I. Overall Summary			
1. Recordable events involving personnel exposure below MPE limits and/or requiring little or no clean-up measures following a radioactive contaminant release	27	34	25
2. Events involving personnel exposure above MPE limits and/or requiring special clean-up measures following a release of radioactive contaminants	60	41	30
Totals	87	75	55
Part II. Personnel Exposure Breakdown			
3. Minor events constituting exposures in excess of planned operational exposure limits	9	5	7
4. Major events constituting exposures in excess of FRC limits with work restrictions imposed	1	2	0
Totals	10	7	7
Part III. Area Contamination Breakdown			
5. Minor events requiring special clean-up measures handled by the regular work staff with no appreciable program loss	56	37	28
6. Events involving special clean-up measures that required interdepartmental assistance with minor departmental program loss	2	3	2
7. Major events resulting in the temporary suspension of parts of the Laboratory program	1	0	0
Totals	59	40	30

TABLE 14 UNUSUAL OCCURRENCE FREQUENCY RATE WITHIN THE DIVISIONS FOR THE 3-YEAR PERIOD ENDING WITH 1962

Division	No. of Unusual Occurrences 1960	1961	1962	3-Year Total	Per Cent Lab. Total (3-Year Period)
Analytical Chemistry	4	3	5	12	5.5
Biology	2	1	1	4	1.8
Chemical Technology	17	19	13	49	22.6
Chemistry	3	2		5	2.3
Engineering and Mechanical	5	4	3	12	5.5
Inspection Engineering			1	1	0.5
Electronuclear Research	5	7		12	5.5
Health Physics	1			1	0.5
Instrumentation and Controls	1			1	0.5
Isotopes	14	9	18	41	18.9
Metals and Ceramics	3	5	2	10	4.6
Neutron Physics	2	3	3	8	3.7
Operations	14	12	6	32	14.7
Physics		1	2	3	1.4
Reactor	11	7		18	8.3
Reactor Chemistry	1	1		2	0.9
Solid State	3		1	4	1.8
Thermonuclear		1		1	0.5
Construction	1			1	0.5
Totals	87	75	55	217	

TABLE 15 UNUSUAL OCCURRENCES CLASSIFIED ACCORDING TO THE OPERATING FACILITIES IN WHICH THEY OCCUR FOR THE 3-YEAR PERIOD ENDING WITH 1962

Building or Facility	Number Recorded 1960	1961	1962	3-year Total
2000	2	1	1	4
2001	1			1
2005		1		1
2007	1			1
2528	1			1
3001	3	2	2	7
3005	4	1		5
3010			1	1
3012	1			1
3019	11	16	9	36
3025	3	2	2	7
3026-C	2	1		3
3026-D		2		2
3028	2	2	2	6
3029	2	1	1	4
3031		1		1
3032			2	2
3033	1		2	3
3038			3	3
3042	3	5	3	11
3044	1			1
3500	1			1
3505		1		1
3508	2	1	1	4
3517	6	3	8	17
3550	1		2	3
4500	2	2	1	5
4501	3	5		8
4507		2	3	5
5500			2	2
7500	5	7		12
7700			2	2
9201-2	4	8		12
9204-1	6	1		7
9204-3	4		1	5
9207	2	1	1	4
9213	2	2	1	5
9733-3	1			1
9766			1	1
Misc.	10	6	4	20
GRAND TOTAL	87	75	55	217

TABLE 16 UNUSUAL OCCURRENCES BY DAY OF THE WEEK FOR THE 3-YEAR PERIOD ENDING WITH 1962

Day of the Week	Number Recorded			Percentage		
	1960	1961	1962	1960	1961	1962
Monday	14	14	10	16.1	18.6	18.2
Tuesday	15	13	6	17.2	17.4	10.9
Wednesday	16	14	12	18.4	18.6	21.8
Thursday	11	9	10	12.6	12.0	18.2
Friday	19	15	13	21.8	20.0	23.6
Saturday	9	4	4	10.4	5.4	7.3
Sunday	3	6	0	3.5	8.0	0.0
Totals	87	75	55			

TABLE 17 MONITORING FILMS PROCESSED, 1962

Beta-gamma Personnel Monitoring Films	70,045
Nuclear Track Monitoring Films	19,325
Calibration Films	2,768
Autoradiograms	1,307
Nuclear Track Alpha Plates	24
TOTAL	93,469

TABLE 18 COUNTING FACILITY RESUME, 1962

Type of Sample	Number of Samples Alpha	Beta	Gamma	Unit Total	Weekly Average
Smear Tabs	193,885	203,754		397,639	7646.9
Air Filters	31,296	29,817		61,113	1175.2
Environs Monitoring	246	3,486		3,732	71.8
Water (Waste Disposal Research)	71	112		183	3.5
Threshold Detector Foils			465	465	8.9
GRAND TOTAL	225,498	237,169	465	463,132	8906.3

TABLE 19 BIO-ASSAYS, 1962

I. ORNL Employees			6792
Urine:			
	Routine Procedures	6627	
	Special Handling	91	
Fecal:			
	Routine Procedures	74	
	Special Handling	0	
II. Non-ORNL Employees			590
		TOTAL	7382

TABLE 20 ENVIRONMENTAL MONITORING SAMPLES, 1962

	Sample Type	Type of Analysis	Number Samples
1.	Monitoring network filters	Particles/ft^3, gross beta	1702
2.	Gummed paper fall-out trays	Particles/ft^2, gross beta	1392
3.	CAM filters	Particles/ft^3, gross beta	8000
4.	Rain water	Gross beta	1130
5.	White Oak Dam Effluent	Complete radiochemical	1312
6.	Clinch River water	Complete radiochemical	60
7.	Raw milk	Sr, I	65
8.	Pasture grass	Sr, gamma scan	24
9.	Potable water	Sr,	52
10.	Silt composites	Gross beta, gamma	16
		TOTAL	13,753

TABLE 21 CALIBRATIONS RESUME, 1962

A.	Portable Instruments Calibrated	
	1. Beta-Gamma	3243
	2. Neutron	72
	3. Alpha	571
	4. Pocket chambers and dosimeters	2699
B.	Films Calibrated	
	1. Beta-Gamma	5960
	2. Neutron	122
	TOTAL	6082

TABLE 22 PORTABLE INSTRUMENT INVENTORY, 1962

Instrument Type	Working Inventory 1961	Instruments Acquired 1962	Instruments Retired 1962	Working Inventory 1962
GM Survey Meter	316	44	2	358
Cutie Pie	337	61	58	340
Juno	37	0	1	36
Alpha Survey Meter	101	39	10	130
Thermal Neutron Meter	6	12	1	17
Fast Neutron Meter	18	2		20
TOTAL INVENTORY	815	158	72	901

8.0 FIGURES

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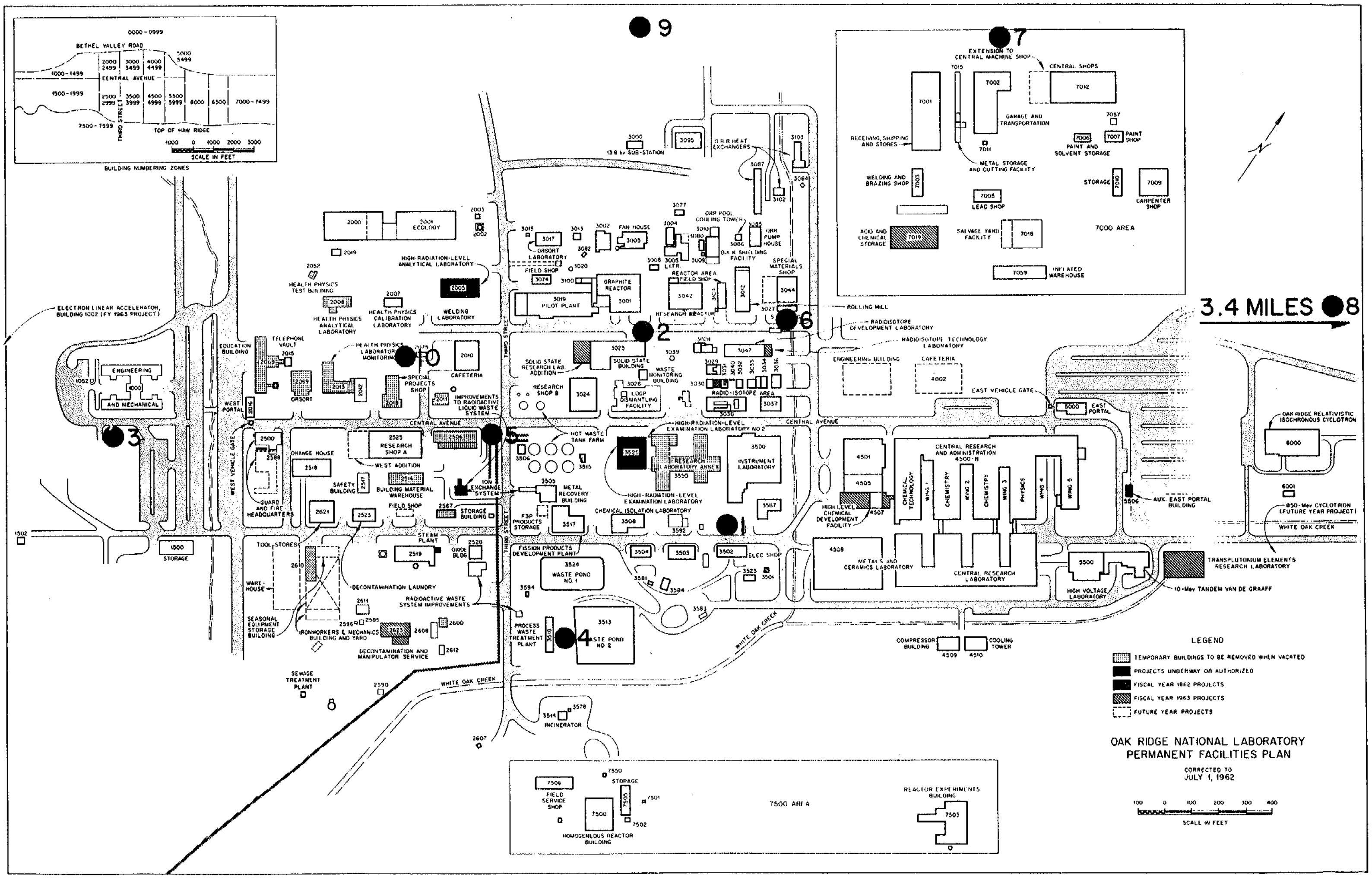

Fig. 1. (Map of) Laboratory Areas Showing the Approximate Location of the Local Monitoring Stations Constituting the LAM Network.

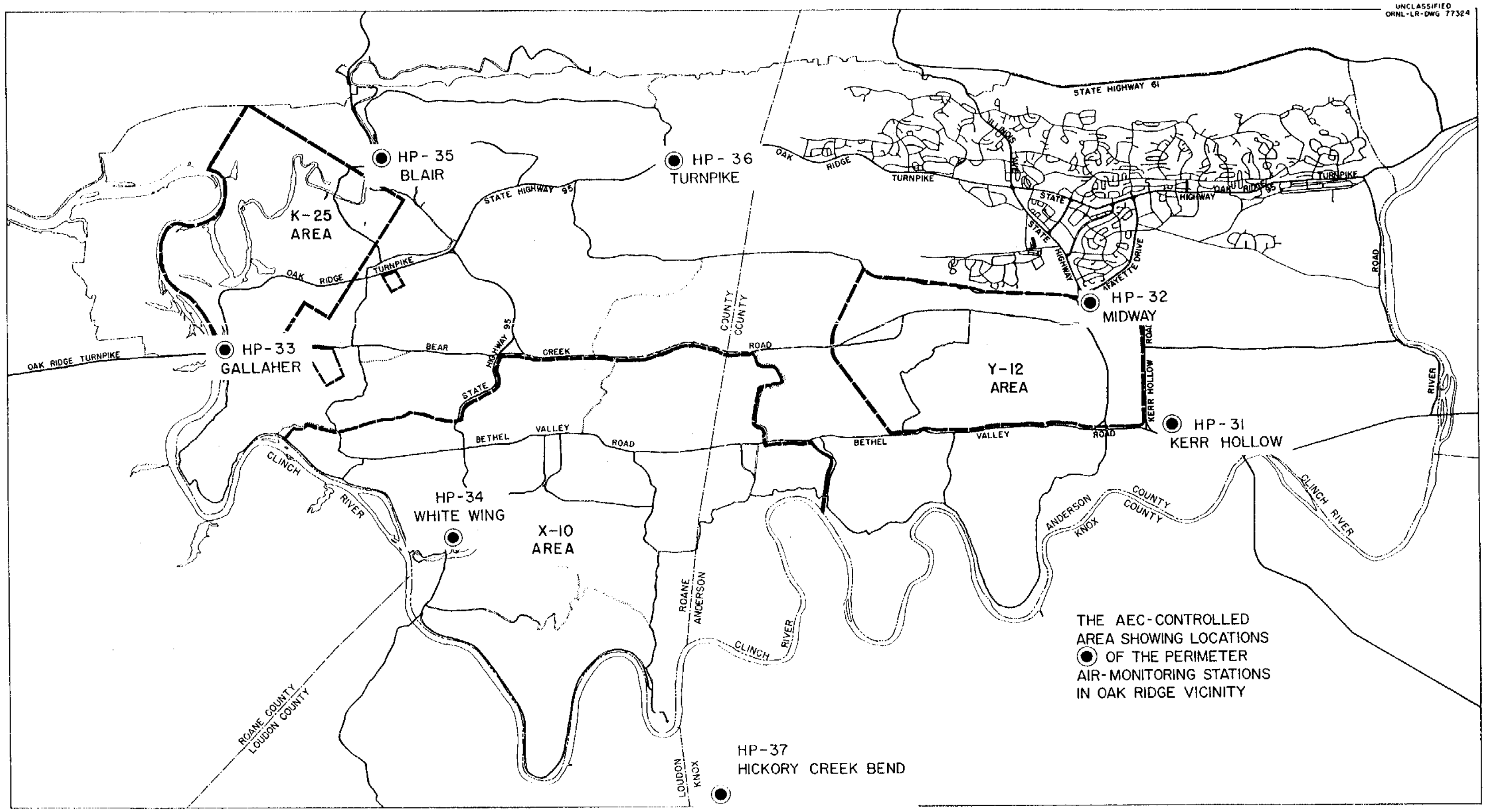

Fig. 2. (Map of) AEC Controlled Area and Vicinity Showing the Approximate Location of the Perimeter Air Monitoring Stations Constituting the PAM Network.

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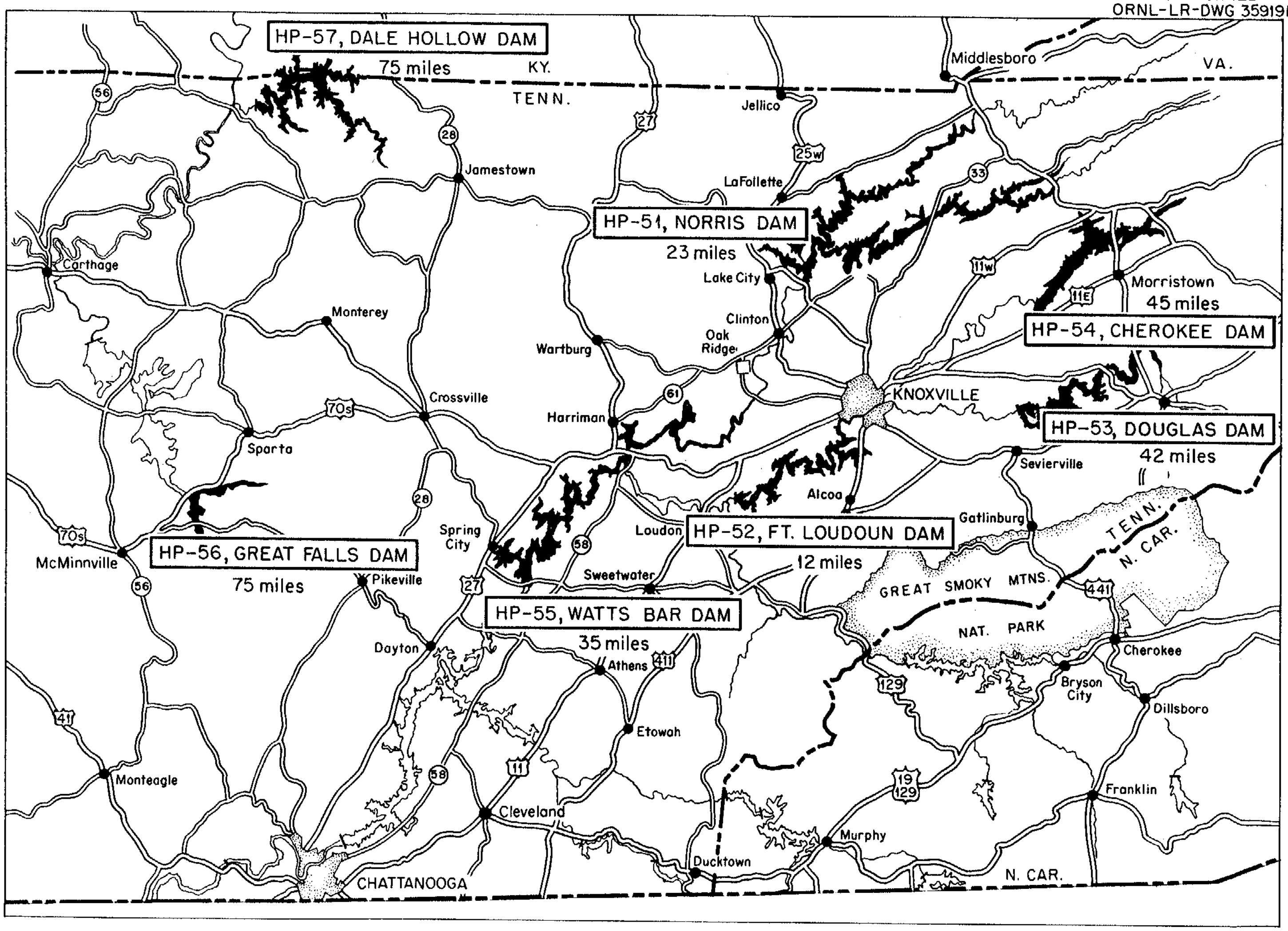

Fig. 3. (Map of) East Tennessee Area Showing TVA and U.S. Corps of Eng. Dam Sites at Which Are Located the Remote Air Monitoring Stations Constituting the RAM Network.

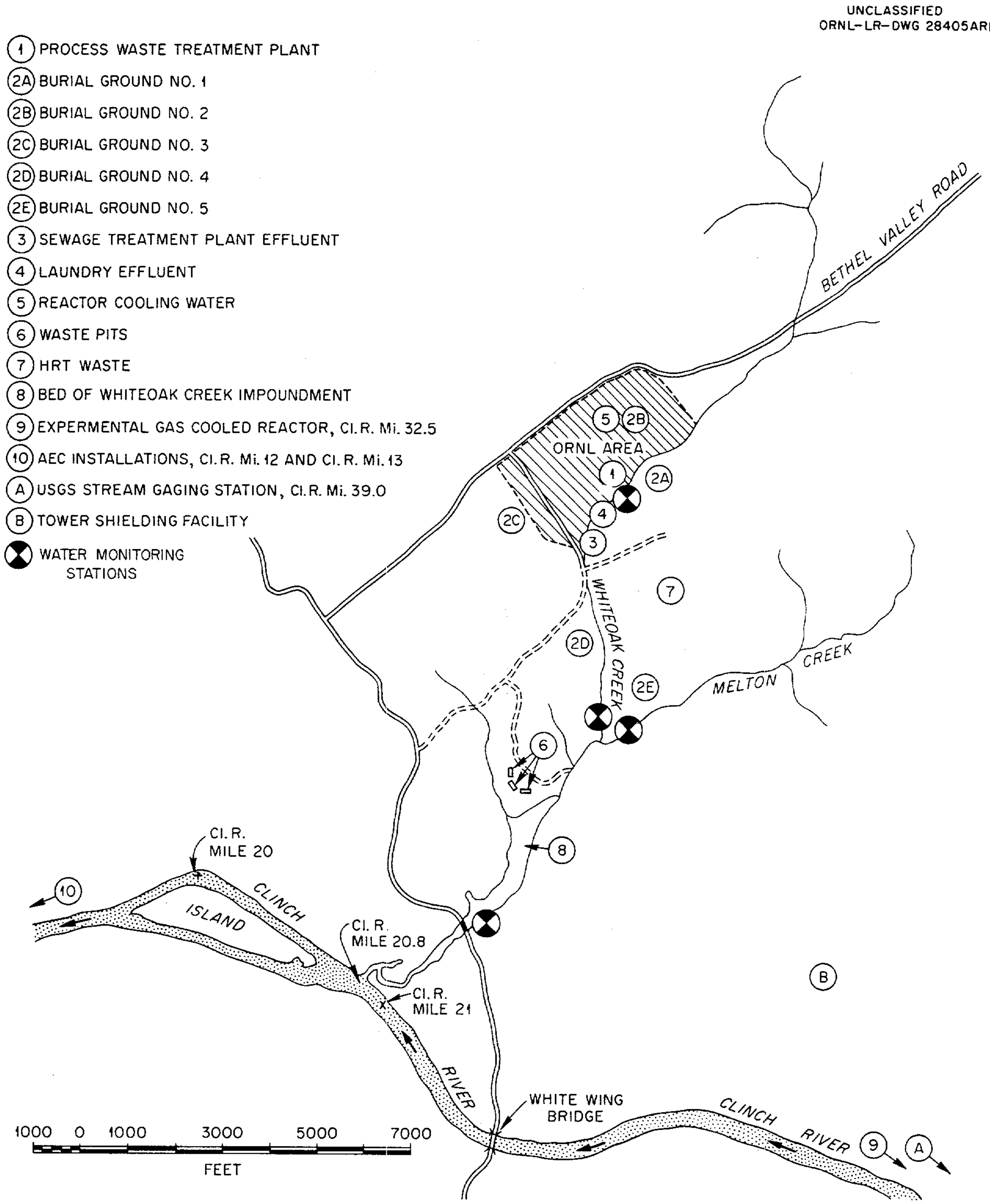

Fig. 4. (Map of) White Oak Creek Drainage Area Showing Water Monitoring Stations in Relation to Potential Waste Releases.

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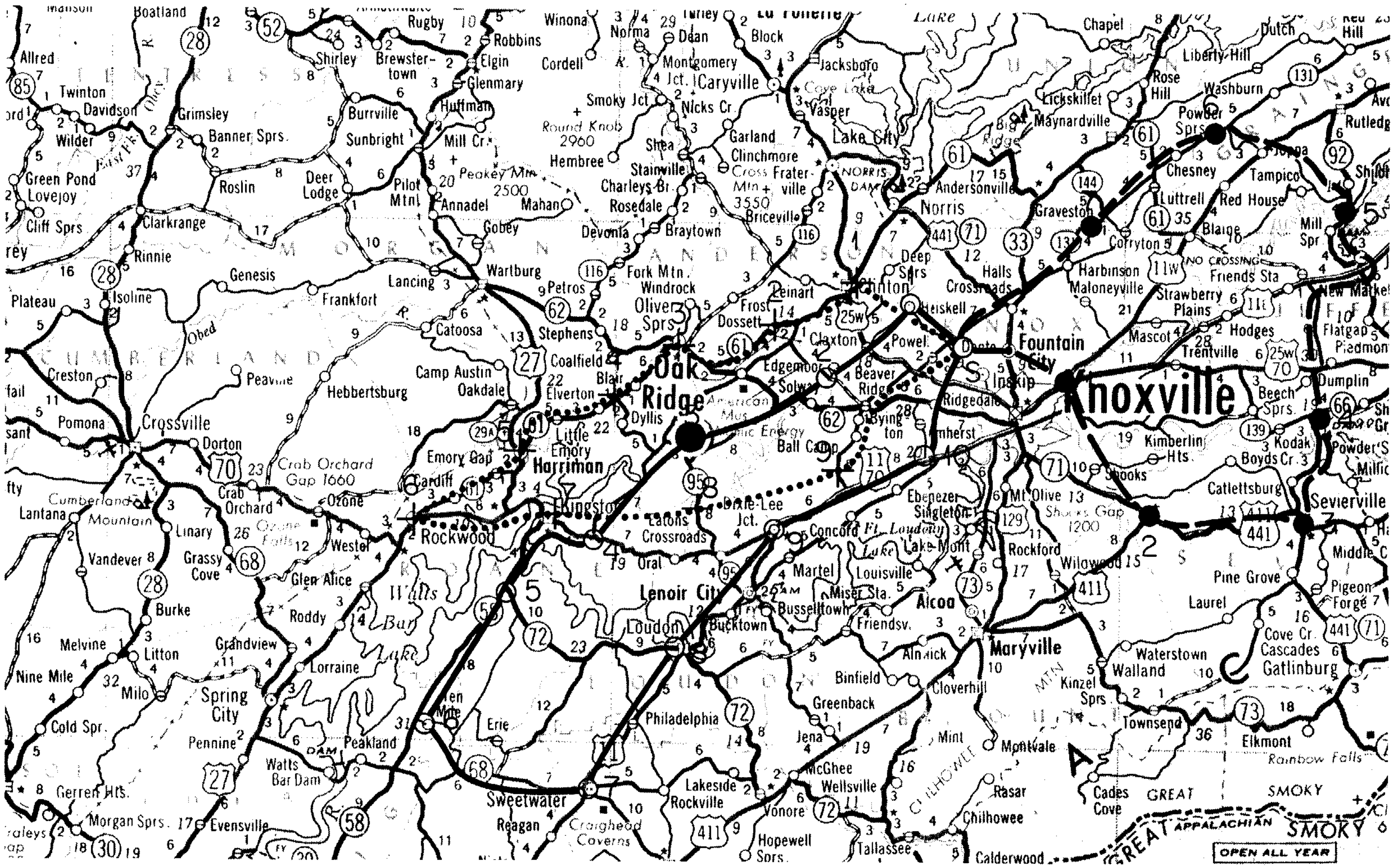

Fig. 5. (Map of) Flight Patterns Used in Aerial Background Radiation Surveys.

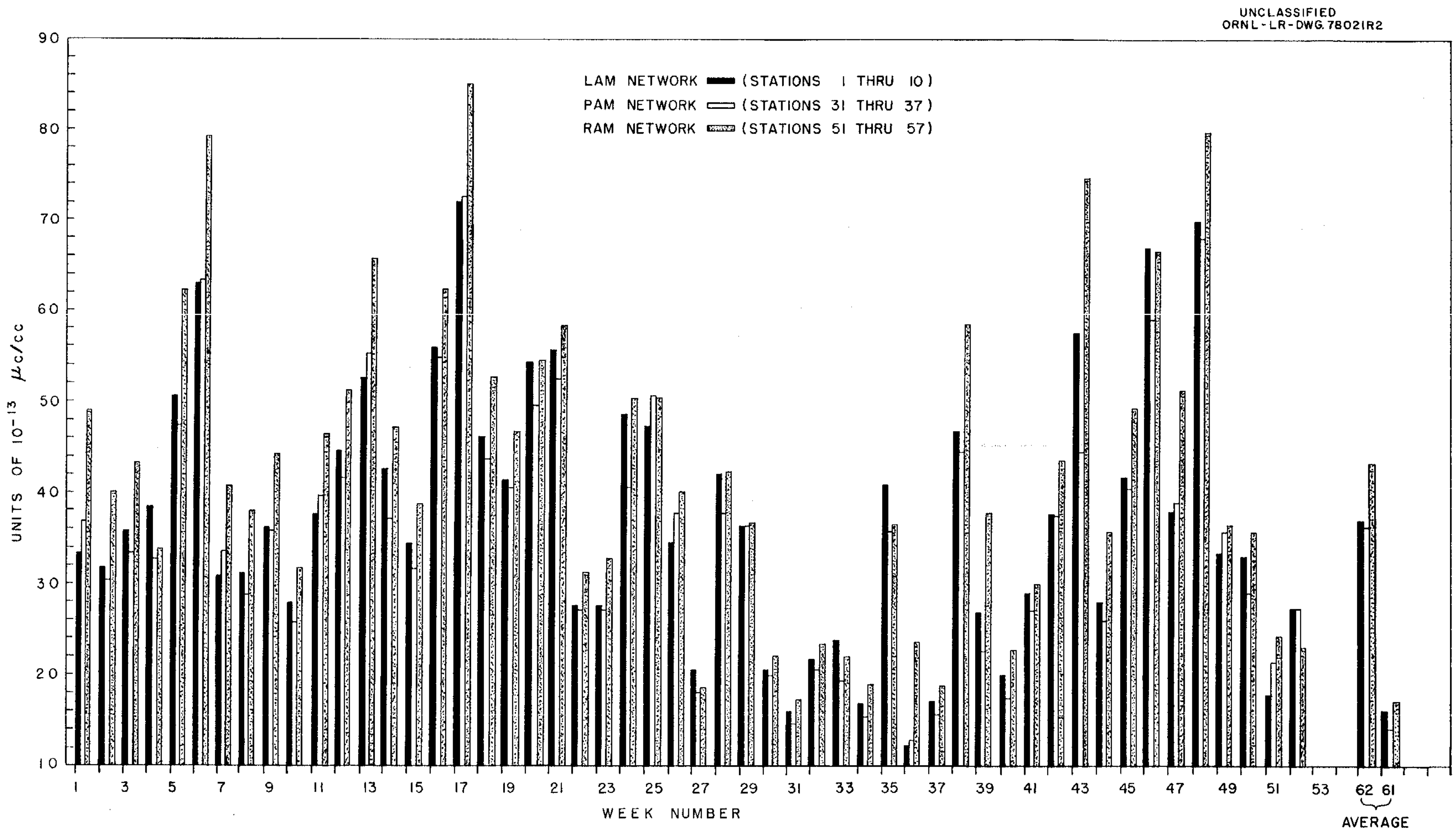

Fig. 6. Concentration of Radioactive Materials in Air as Determined from Filter Paper Data, 1962.

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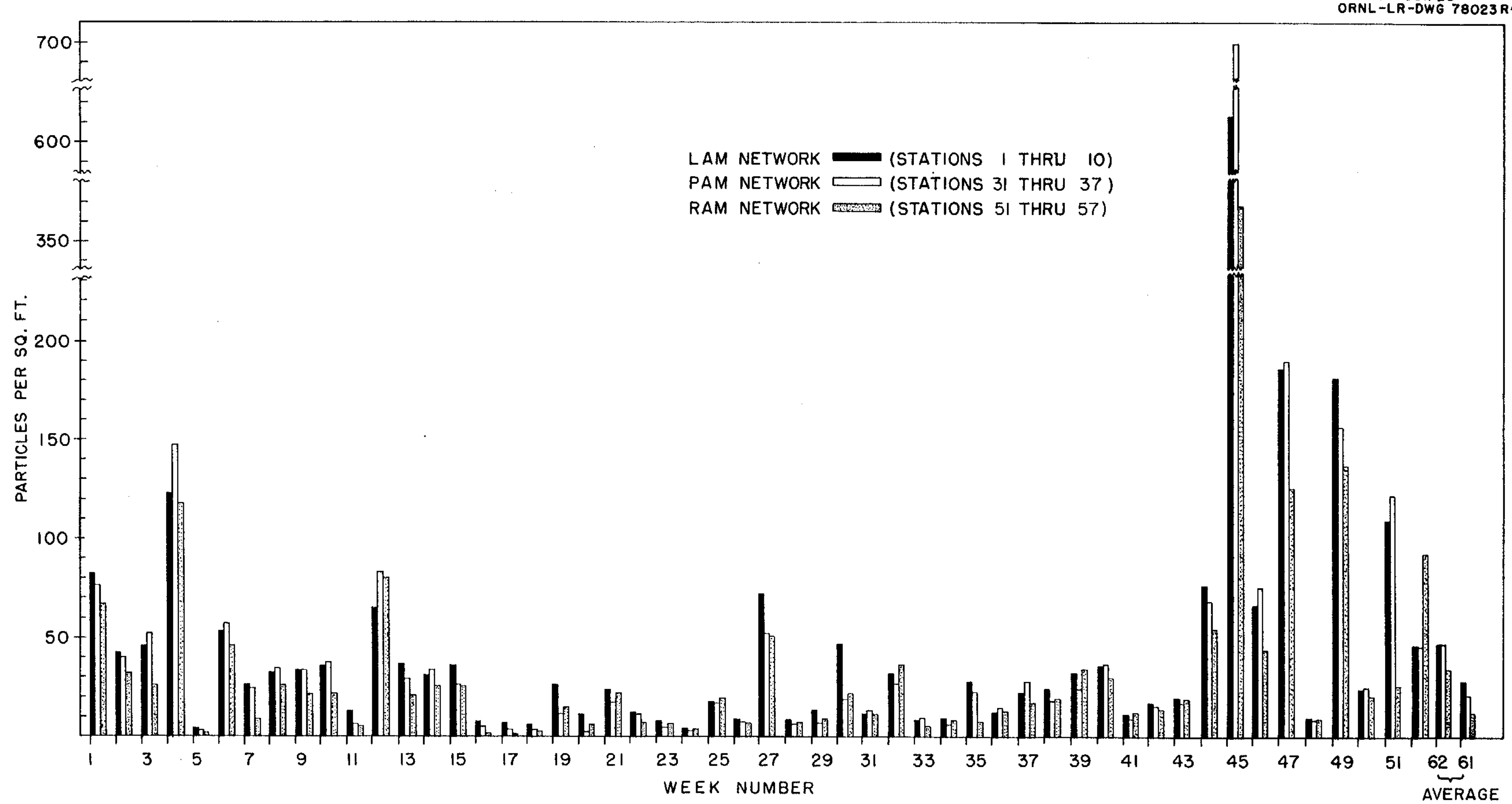

Fig. 7. Radioparticulate Fall-Out Measurements as Determined by Autoradiographic Techniques Using Gummed Paper Collectors, 1962.

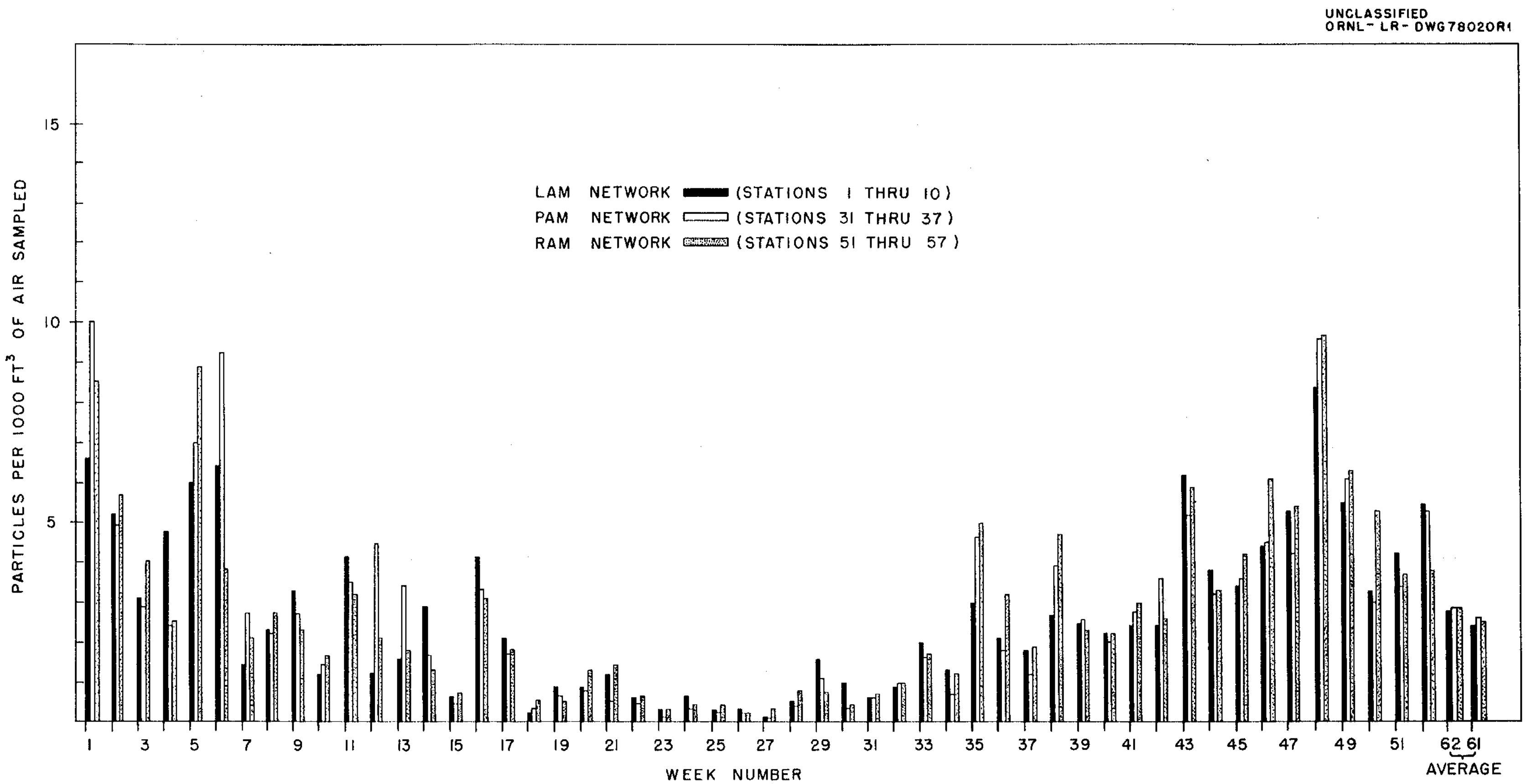

Fig. 8. Radioparticulate Fall-Out Measurements as Determined by Autoradiographic Techniques Using Filters from Continuous Air Monitors, 1962.

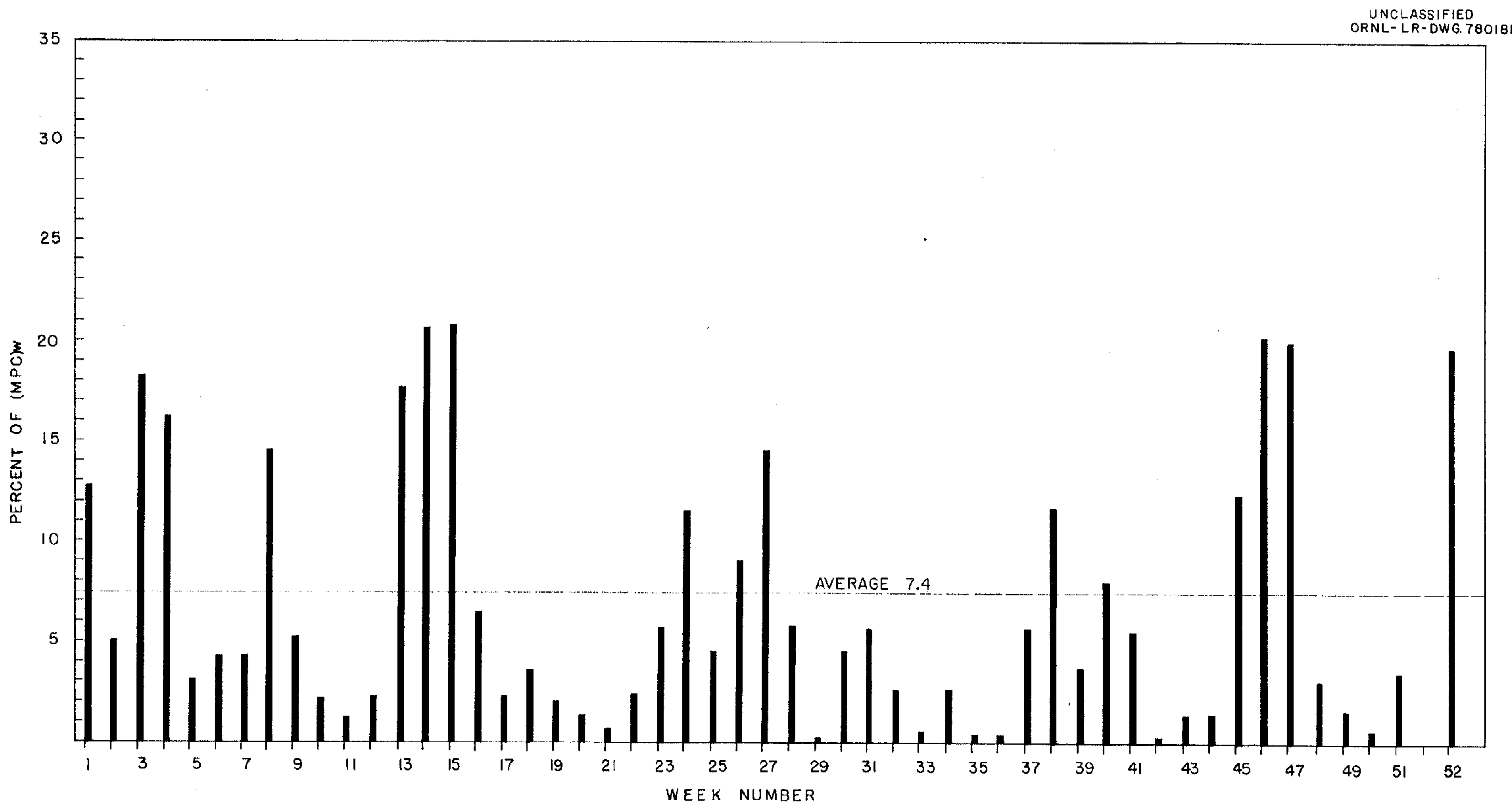

Fig. 9. Estimated Per Cent $(MPC)_w$ of Clinch River Water Below the Mouth of White Oak Creek, 1962.

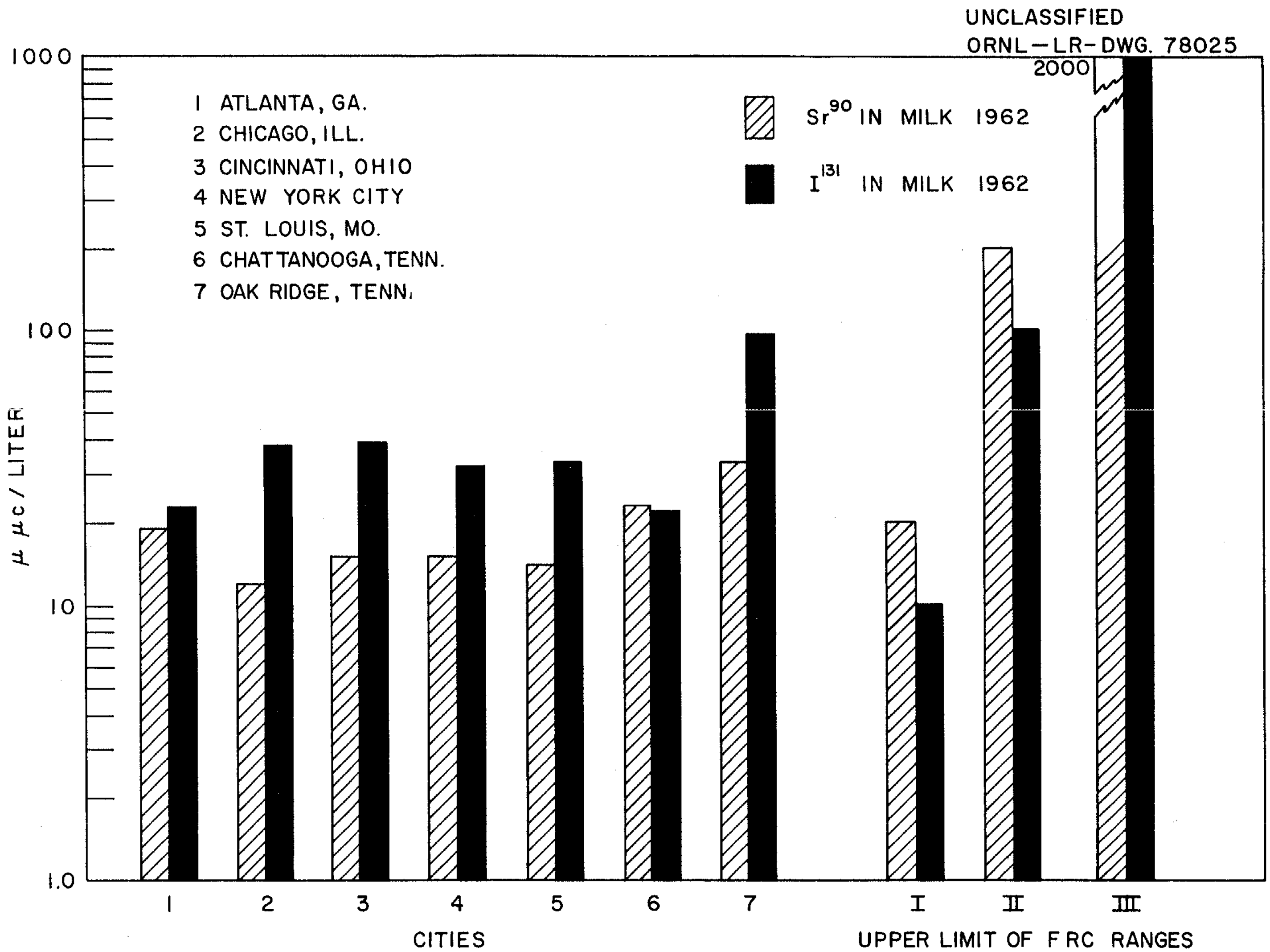

Fig. 10. Sr^{90} and I^{131} Content in Milk Produced in the Oak Ridge Area Compared to the Milk Produced in Other Sections of the U.S. and FRC Ranges of Intake, 1962.

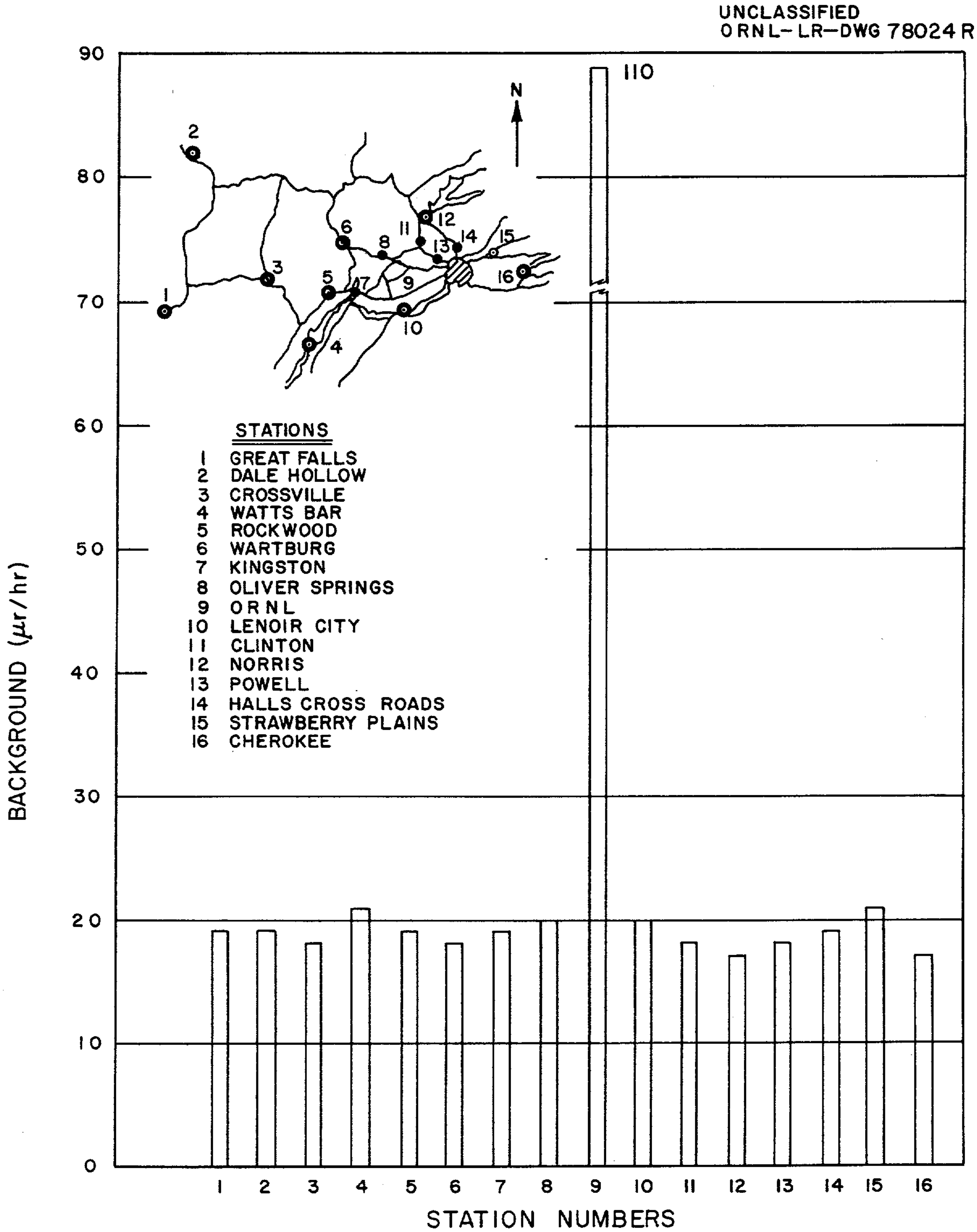

Fig. 11. Radiation Measurements Taken During 1962, 3 ft above the Ground Surface out to Distances of 75 Miles from ORNL.

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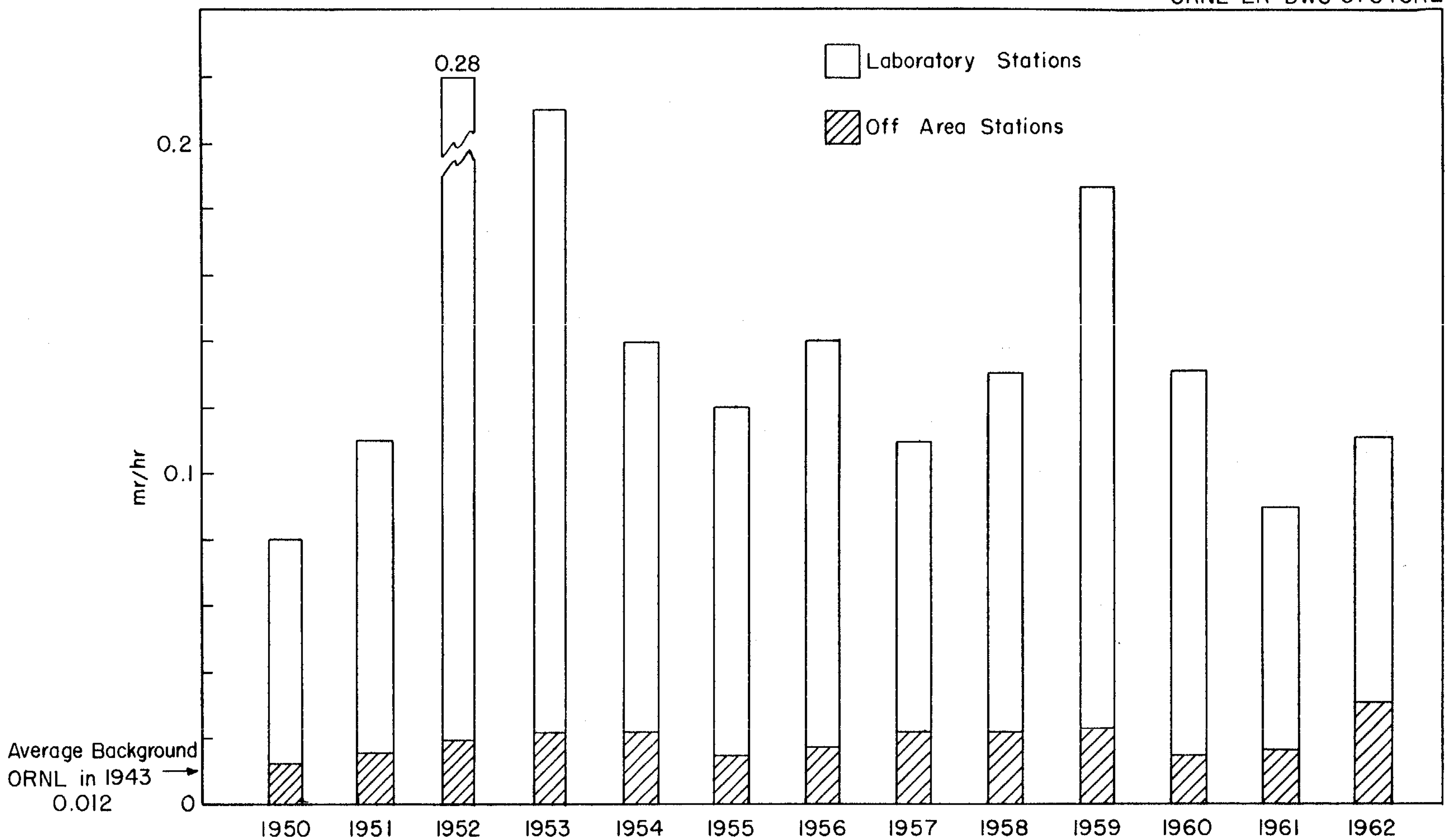

Fig. 12. Radiation Measurements Taken 3 ft Above the Ground Surface at ORNL Compared with Like Measurements Taken Elsewhere within the AEC Controlled Area for the Years 1950-1962.

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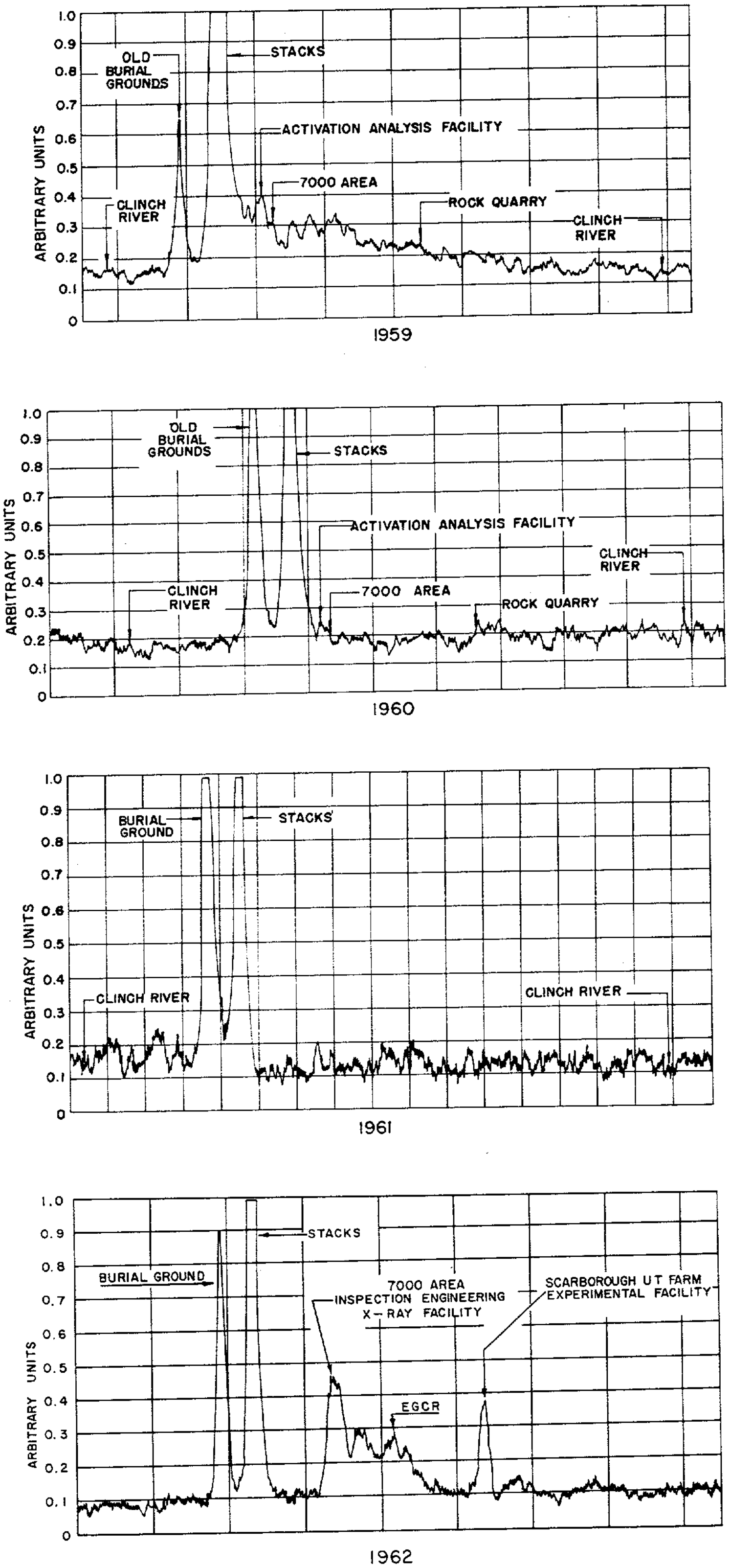

Fig. 13. Radiation Background Profile of the ORNL Area as Determined by Aerial Survey Techniques, 1959-1962.

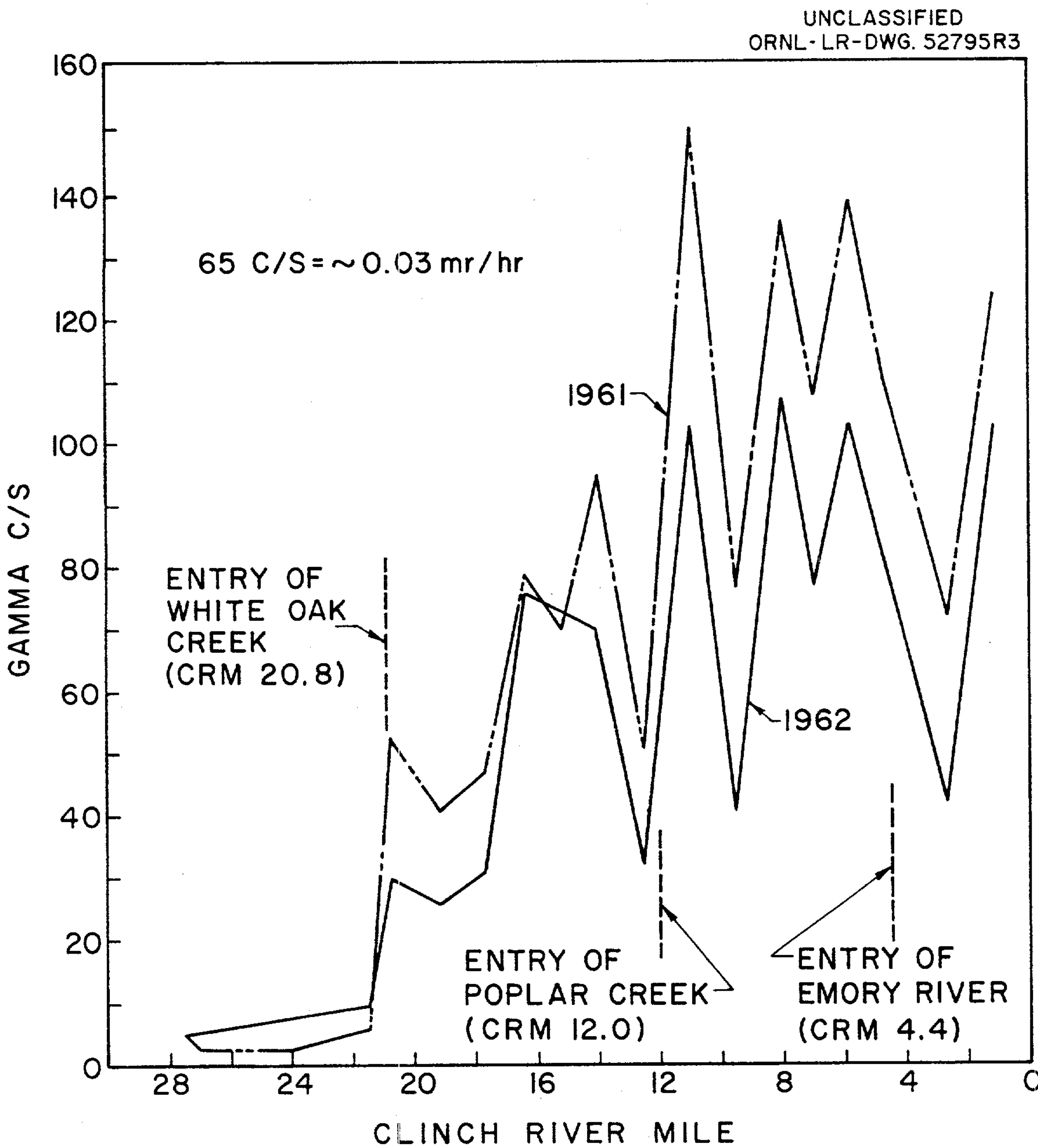

Fig. 14. Gamma Count at Surface of Clinch River Silt.

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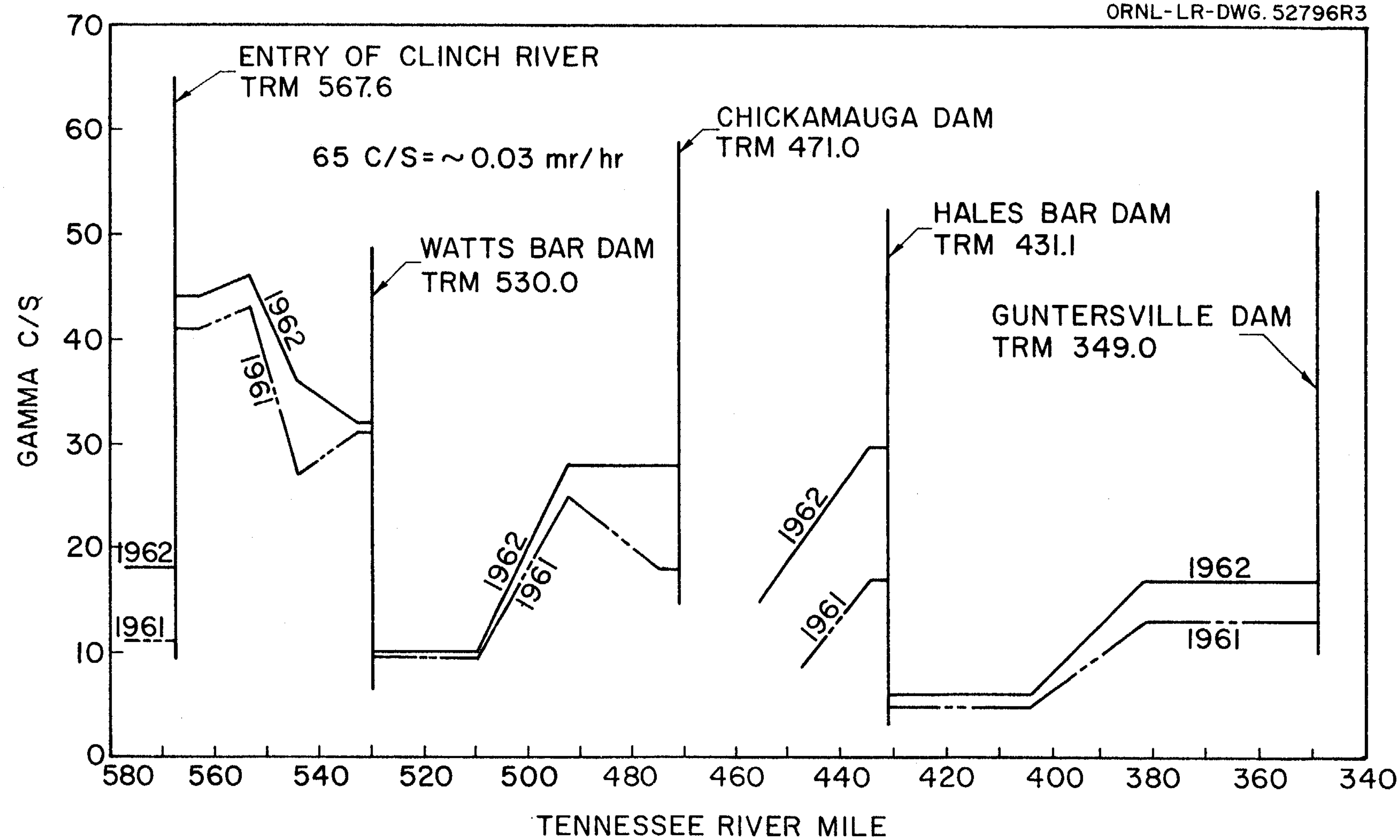

Fig. 15. Gamma Count at Surface of Tennessee River Silt.

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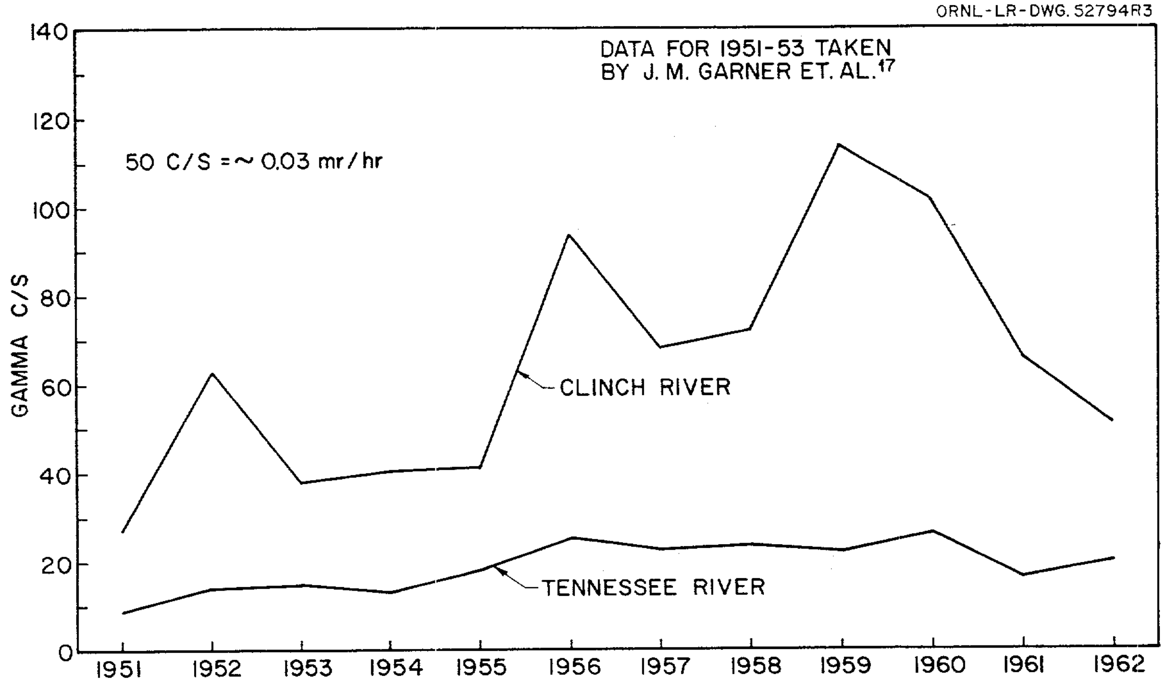

Fig. 16. Average Gamma Count at Surface of Silt, Clinch and Tennessee Rivers, 1951-1962.

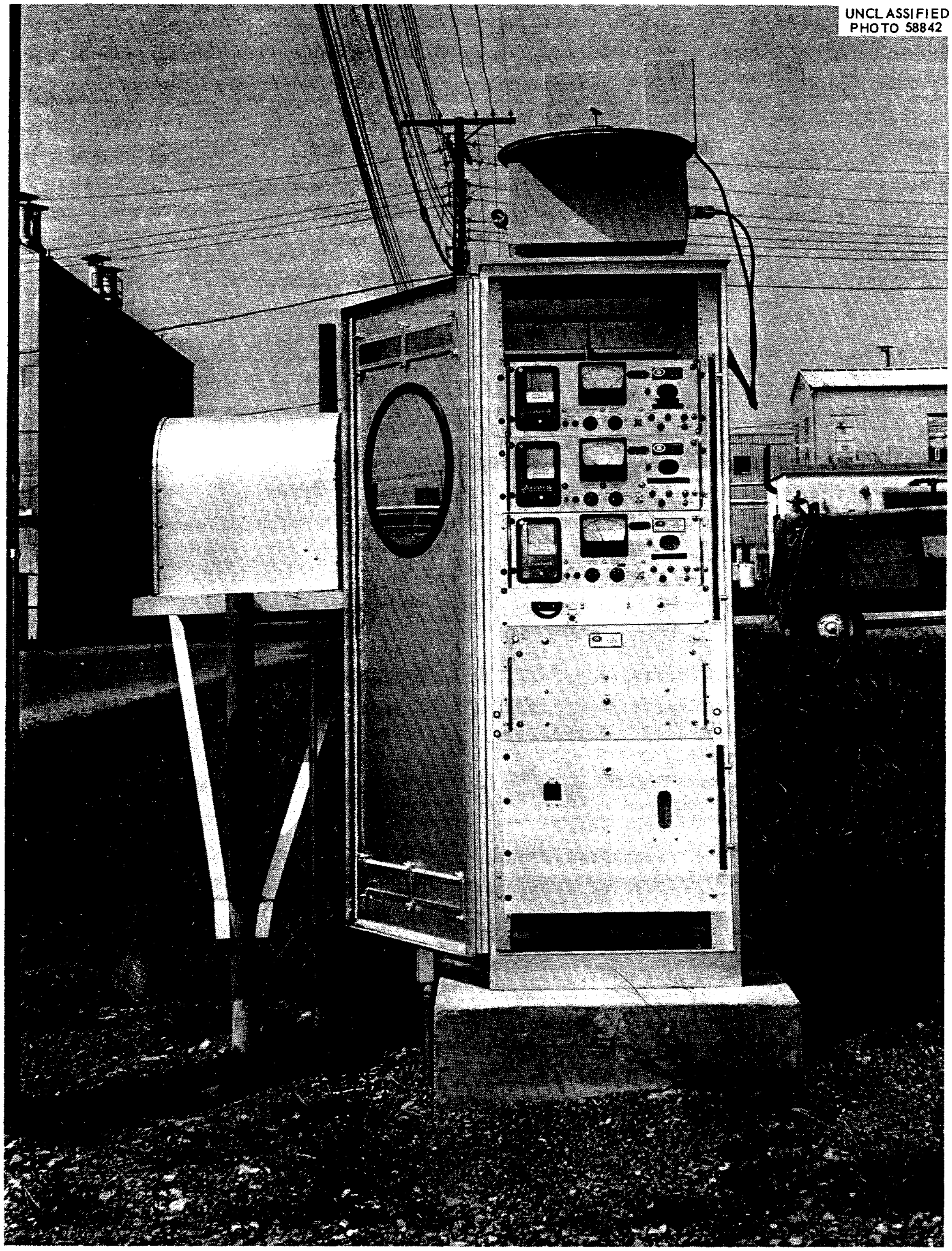

Fig. 17. Local Air Monitor Station.

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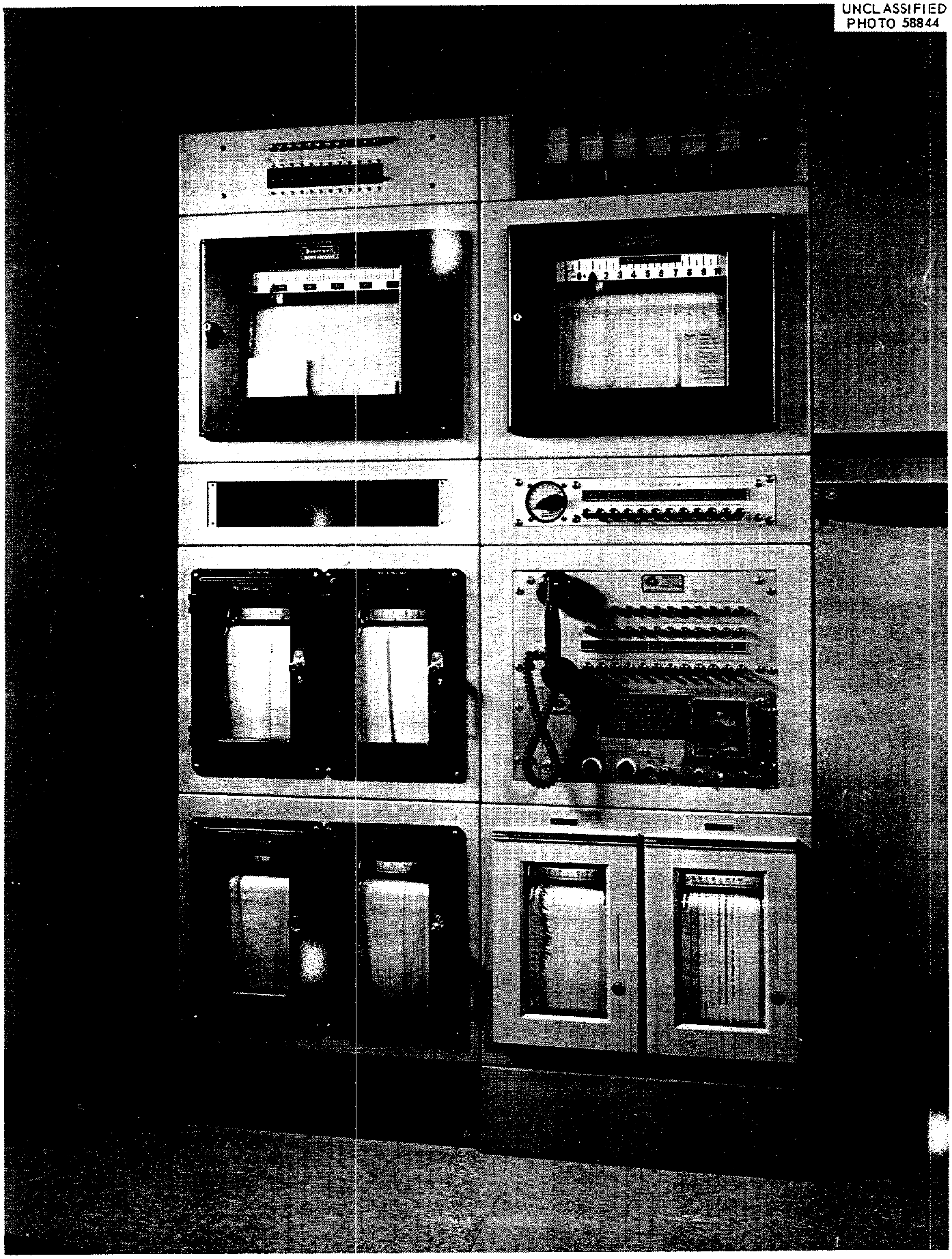

Fig. 18. Environmental Surveillance Readout Panel.

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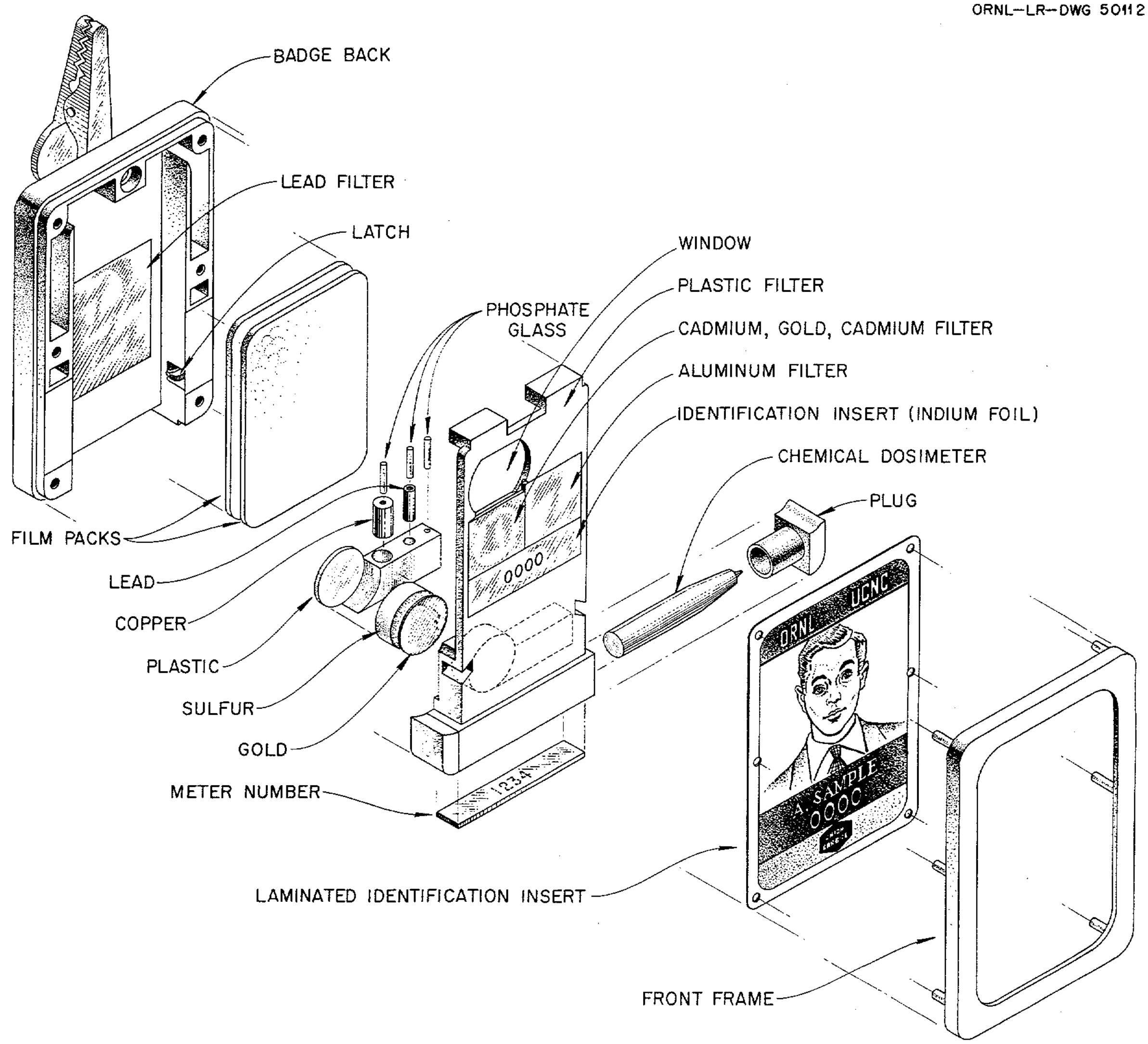

Fig. 19. ORNL Badge-Meter Model II.

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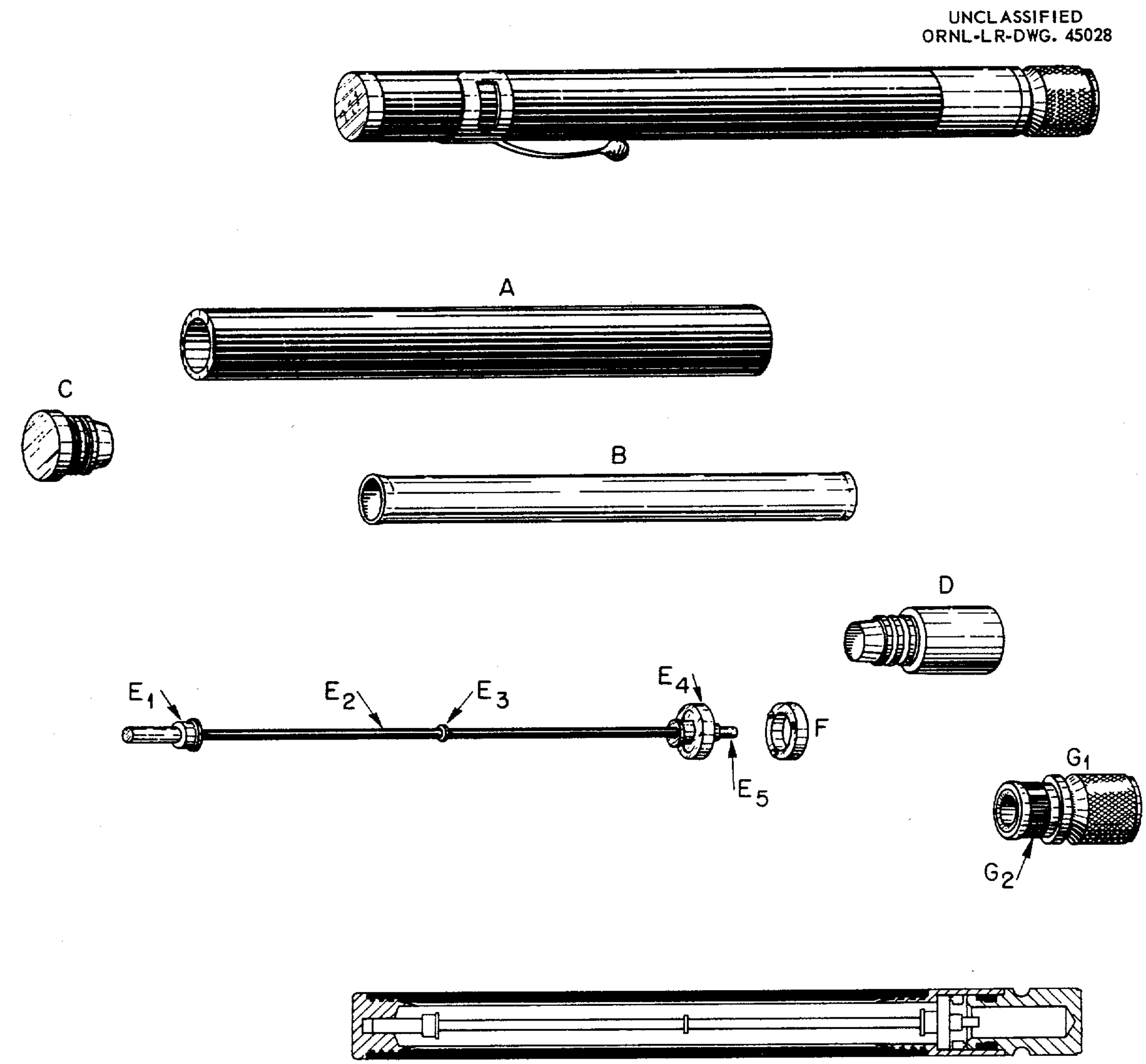
Fig. 20. Typical Temporary Security Passes Equipped with Monitoring Film.



A. LOW ATOMIC NUMBER WALL
B. GRAPHITE-COATED PAPER SHELL
C. ALUMINUM TERMINAL HEAD
D. ALUMINUM TERMINAL SLEEVE
E_1 POLYSTYRENE SUPPORT BUSHING
E_2 CENTRAL ELECTRODE, GRAPHITE COATED
E_3 POLYETHYLENE INSULATING WASHER
E_4 POLYSTYRENE FIXED BUSHING
E_5 ELECTRODE CONTACT
F. RETAINING RING
G_1 ALUMINUM BASE CAP
G_2 POLYETHYLENE FRICTION BUSHING

Fig. 21. Victoreen Pocket Meter, Model 352.

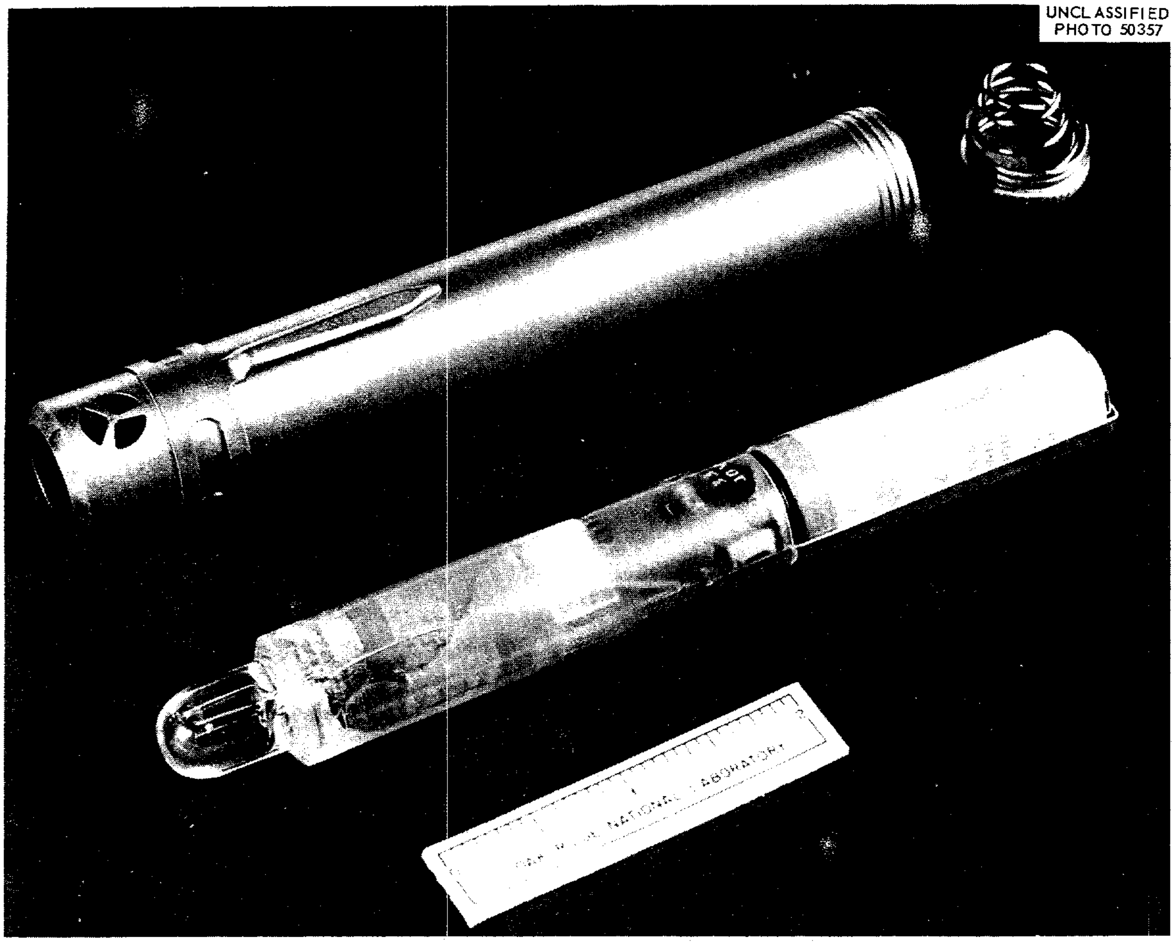


Fig. 22. Personal Radiation Monitor (PRM).

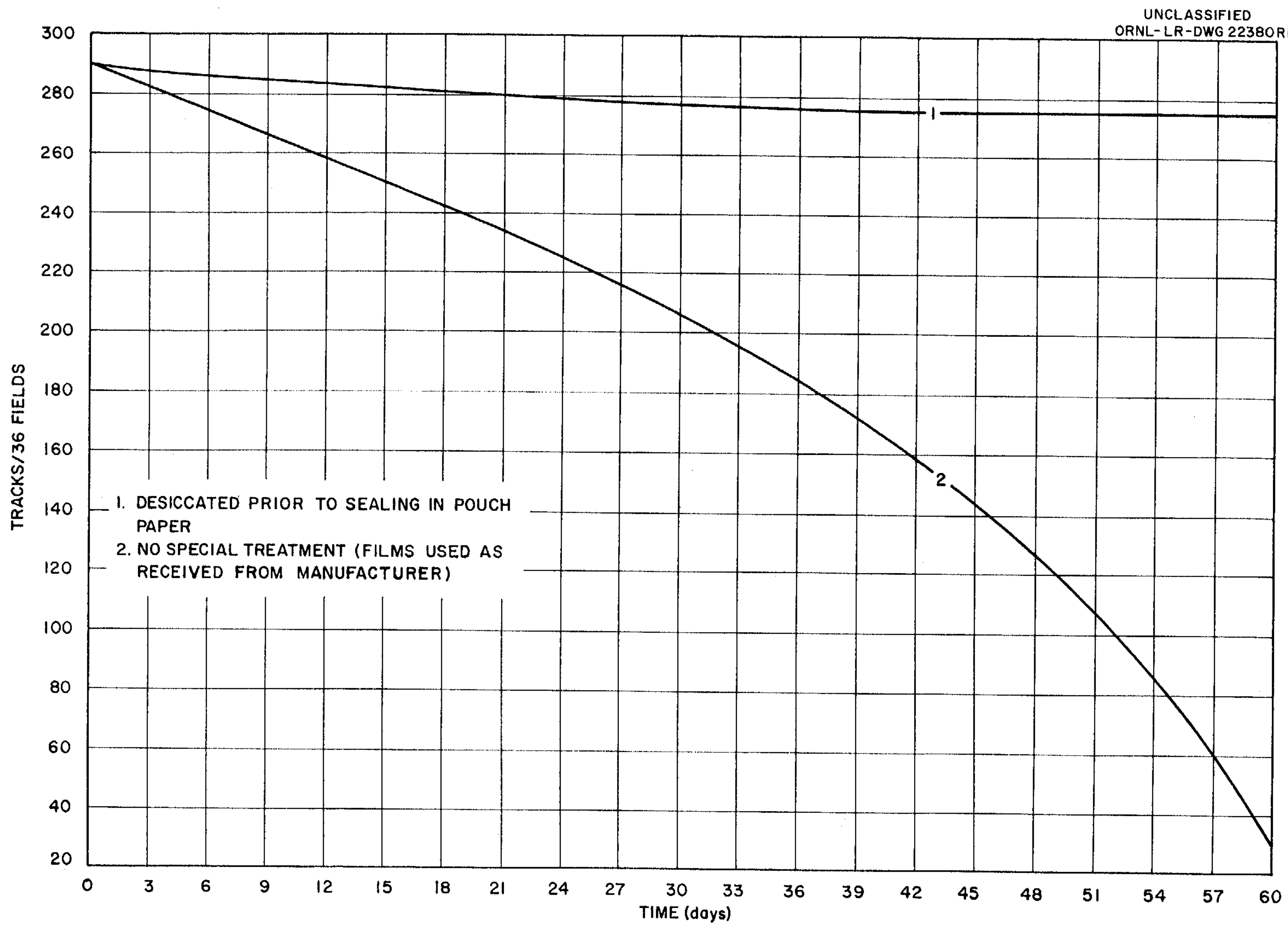


Fig. 23. Effect of Sealing NTA Film in Moisture Proof Paper.

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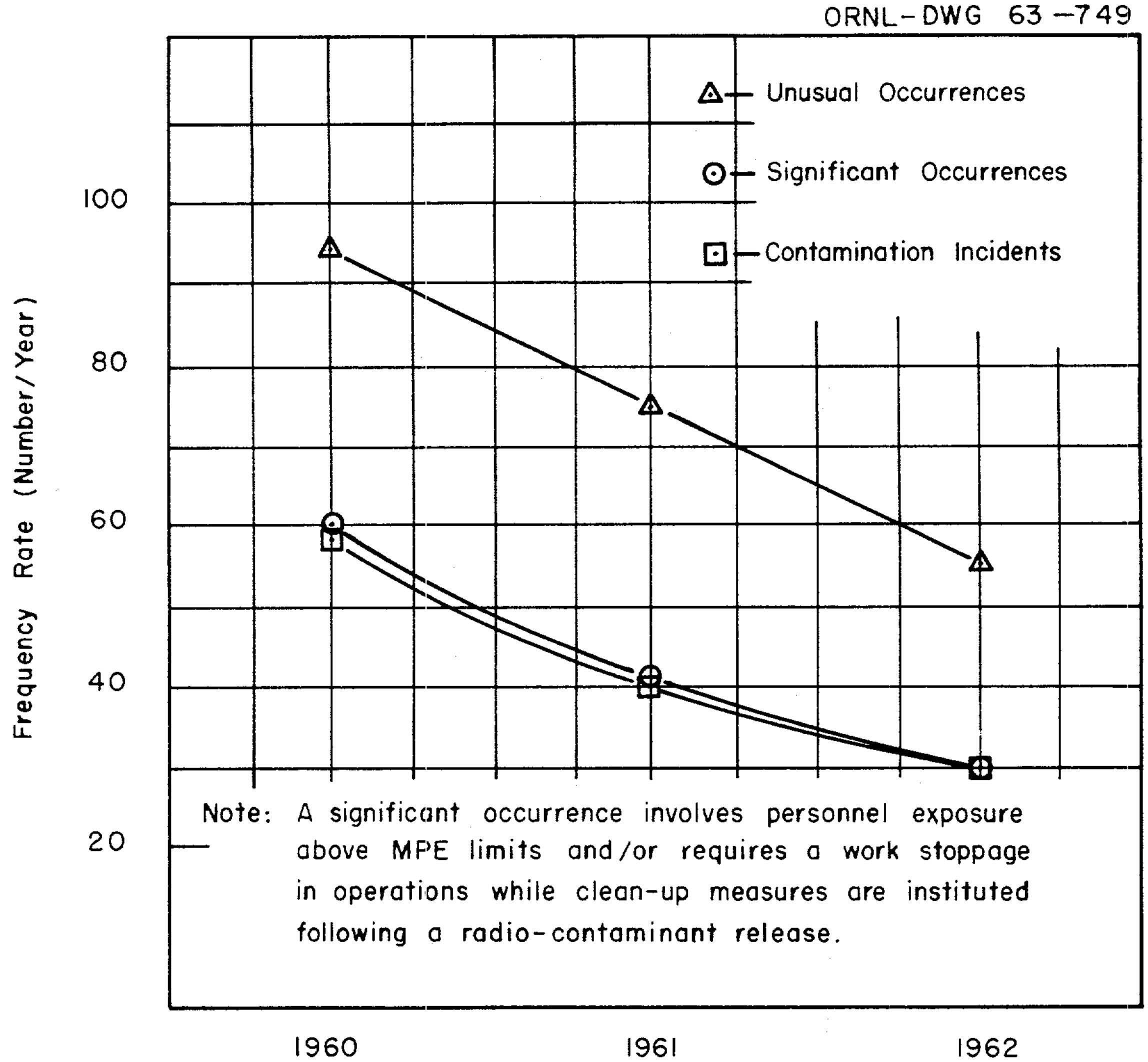

Fig. 24. Unusual Occurrences (plotted for the 3-year period ending with 1962).

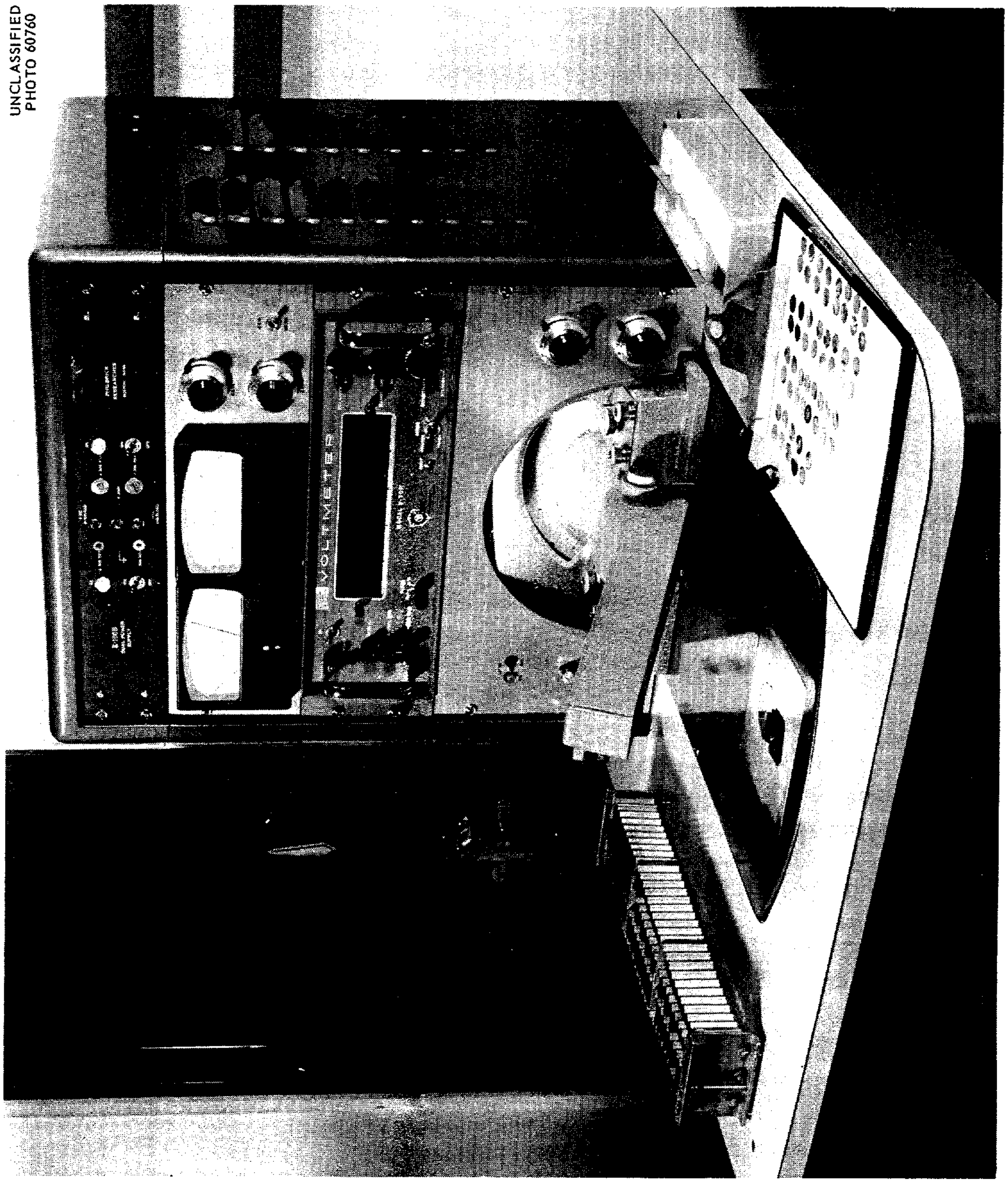

Fig. 25. Film Reader (ORNL Q-2578).

9.0 APPENDIX: THE MERITS OF CATTLE THYROID ANALYSIS FOR THE DETECTION OF I^{131} IN THE ENVIRONMENT[1]

Recent experience tends to show that cattle thyroid analysis is a practical method for the detection of I-131 in the environment and that this method has some advantage over milk analysis where detection is of primary concern.

The cattle thyroid, as obtained from the slaughter house, is placed in a plastic counting planchet and easily preserved by storing in the freezing compartment of a refrigerator. Prior to counting, the thyroid requires no preparation other than removal from refrigeration and storage at room temperature while thawing is accomplished. Gamma counting is done directly using a 4" x 2" NaI (Tl) crystal coupled with a 200 channel analyzer. No significant contribution from gamma emitters other than those associated with the thyroid spectrum has been observed.

In contrast to thyroid analyses, milk has been observed to contain gamma emitters other than those found in I-131. Consequently, either a radiochemical separation must be performed or a rather complex spectrum stripping operation must be utilized. An anion exchange technique[2] gives an average I-131 recovery of about 90 per cent in the 80 to 97 per cent recovery range. The procedure is rather simple and the extremes of the range are acceptable in view of variations found in I-131 concentrations in milk obtained from cows located on the same farm.

As in the case of milk, it has been observed that thyroids taken from different animals located on the same farm differ in I-131 concentration by a factor of 3 or more. Thus, it is important to collect a fairly large number of milk or thyroid samples from a given region in order to get a reasonable average for the I-131 concentrations. During the calendar year 1962, a total of 266 cattle thyroids and 54 milk samples were taken from cattle which grazed in the East Tennessee area and analyzed for I-131. During the period of highest I-131 levels, i.e., June through December of 1962 (Fig. 9.1), a relatively large number of samples were collected on each collection date with the result that data processed during this period were more representative of average I-131 levels than the data derived from spot sampling techniques utilized during the previous five months. The June-December data suggest that the average concentration of I-131 in milk is about eight per cent of the average concentration in cattle thyroids. Consequently, it is necessary to collect about 280 liters of milk in order to equal the total I-131 content in one thyroid (average weight 22.4 grams); also, for equal detection sensitivities, the milk must be reduced about 8000 to 1 in volume.

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[1] Submitted by B. R. Fish, et al., Health Physics Technology Section.

[2] ORNL-3347, pp. 149-152.

While not a substitute for milk sampling since the health physicist is primarily concerned with I-131 intake, the analysis of cattle thyroids is suggested as a valuable extension to an environmental monitoring program in that it provides for a more sensitive detection limit and requires relatively little in the way of laboratory preparation.

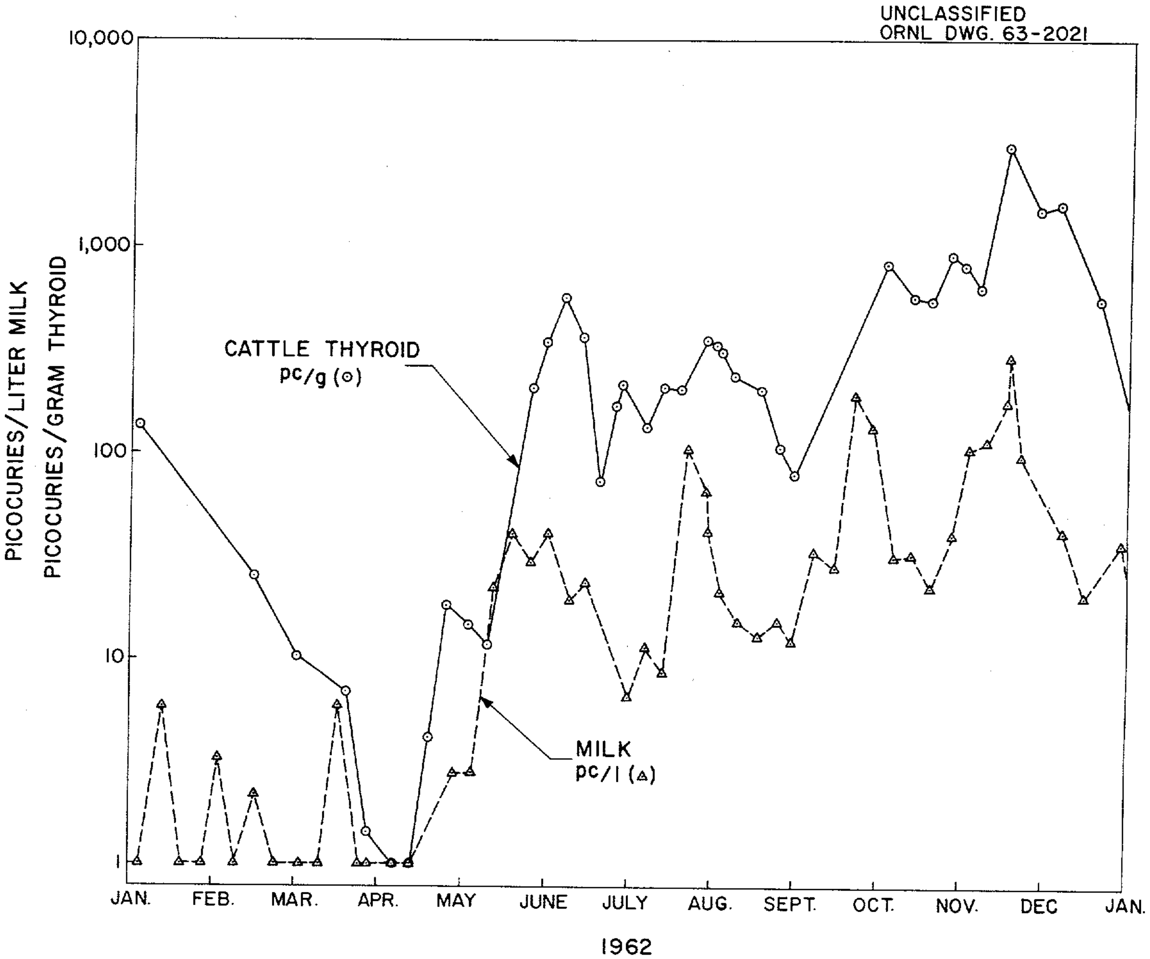

Fig. 9.1 Concentration of I^{131} in Milk and Cattle Thyroids, 1962

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