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PROCEEDINGS
OF THE

HEALTH PHYSICS SOCIETY
ELEVENTH MIDYEAR
TOPICAL SYMPOSIUM
ON RADIATION
INSTRUMENTATION

JANUARY 16 - 19, 1978

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EDITOR

RADIATION

INSTRUMENTATION

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WILLIAM
W.
WADMAN III

PREFACE

Radiation Instrumentation is an essential part of health physics. Operational monitoring is impossible without it. The H.P.S. Eleventh Midyear Topical Symposium addressed the need to exchange state of the art instrumentation information with coverage of laboratory and field applications of it in operational and research areas of health physics.

These proceedings have been published after the meeting, held January 16-19, 1978 to enable the editor the tune to include questions and answers-in essence-peer review, prior to final printing. These proceedings should not be used as the only current reference on the topic, but are meant to serve as one resource to fill source immediate needs. No textual editing of the questions and answers were performed, and as these comments were spoken, may tend to read strangely. It is felt that much of the value of topical meetings comes from exchanges at the end of each presentation, and hence worthy of inclusion.

These proceedings, and the symposium it reports, were made possible by the support of numerous people. Initially, one must recognize the authors, without whose cooperation, this would only appear as a summary.

Principally, however, the Southern California Chapters Board of Directors and Officers made them possible because of the conceptualization of the meeting and selection of the very able General Chair and Committee Chair. Members of the chapter that served the committees performed superbly to the person, to the end, truly a fine team effort.

It has been a privilege to serve as the anchorman in wrapping up the second, highly successful symposium hosted by the Southern California Chapter in less than a decade.

William W. Wadman III

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OPENING SESSION

WELCOME TO SAN DIEGO
Janet Trunzo
Symposium Chair

OFFICIAL OPENING
Dr. John Auxier
President, Health Physics Society

CHAPTER PRESIDENT'S WELCOME
William W. Wadman III

Southern California Chapter President

OPENING OF THE SYMPOSIUM

JANET TRUNZO, GENERAL CHAIR

I'd first like to welcome everyone to San Diego, and especially to the Health Physics Society 11th Midyear Topical Symposium. As you know the topic of this Symposium is Radiation Instrumentation and I think there is a fine program ahead of you for the next 2½ days.

Before I make a few comments about the program, I'd like to discuss a few of the social activities that we have planned for the next few days. This evening we have the annual night out and this is a Luau, a California-type of Luau, for those of you from Hawaii. But I'd like to explain what fun this is going to be. The costs of the tickets include drinks, a complete buffet and entertainment in the form of a show and continuous music from 7 to 11 p.m, the whole evening's entertainment with food and beverages. So I hope that you can all come. Also the spouse activity scheduled for today is the Tijuana tour. I believe it's going to be leaving around 9ish. Tomorrow we have a harbor cruise. Both of them are very entertaining and enjoyable. Tomorrow at noon is the luncheon and we will have a speaker at that luncheon and everyone has a ticket for that with registration.

There are a few changes in the program that I think you should be aware of. Today, Technical Session B, Environmental Monitoring, a paper was not listed that will be given at 11:40 a.m. The name of the paper is "Radiation Measuring Instrumentation for the Uranium Mining Industry" by Robert Gray. Two papers have been cancelled: Tuesday, January 17, Session C, Radiation Detectors, 1:50 p.m., "The Low Temperature Scintillation Properties of Bismuth Germanate and its Application to High Energy Gamma Radiation Imaging Devices". Next page, Session D, one paper cancelled, 4:40 p.m., "A Radiation Detection System for Vehicular Traffic Monitoring". All papers given today will be of 20 minutes duration: 15 minutes for the presentation, 5 minutes for questions, except one session on Portable Instrumentation, the papers will be of 15 minutes duration.

At this time, I would like to introduce to you, the President of the Health Physics Society, Dr. John Auxier.

DR. JOHN AUXIER, PRESIDENT, HEALTH PHYSICS SOCIETY

OPENING ADDRESS

Thank you, Janet. I am happy to be here and I'm happy to add my welcome to San Diego and to the Symposium. I'm especially happy to see so many of the folks that were at the Chicago Symposium, the very first of the topical symposia, and that we have an active participation of what are called the core group of old pros. I know that the local symposium committee, headed by Janet, is especially pleased at the reception of the subject by so many of you in the attendance, which is very good for a topical symposium and which, of course, meant that so many of us had to come through snow and sleet and all that to get out here. My wife is still wondering about where the sunshine is I've been promising her.

The Symposium Committee has done a fantastic job. I believe that they went just a little far out Sunday morning when they arranged the helicopter rescue of a fellow from a fence out behind this motel. He could have have been drowned if he had fallen into the water or from the helicopter. I hope you staged that very carefully, with OSHA in mind.

It's very clear that the subject is pertinent because so many of you are here. The interest and relevance is something that is not clear universally, and it is especially not clear in our minds, looking at the programs of the various funding agencies. During the first ten years of my active participation in dosimetry, which seems a long time ago, the old AEC had a Special Instruments Branch that was a very diverse, well funded organization, which over the years got incorporated into a broader, bigger program, and the instrumentation work gradually got assimilated and disappeared. Today it is very difficult to find a significant program anywhere in any government agency aimed at radiation instrumentation. Perhaps a little facetiously, there have been several persons that have observed that if you wish to have a program in instrumentation, you should pick one in something related to coal - either the effluents from burning or conversion to oil or gas. Even I must agree that there are an enormous number of materials about which we know little and about which we know even less, in some cases, about how to measure. We know almost nothing about the relationship of "dose" of some of these materials to human health.

In any event, radiation instrumentation, from our point of view, is still important and it is something that we'll hear about at this meeting, both in terms of fundamentals and in practice. The program promises us much in both these terms and I'm intrigued with many of the titles and I eagerly await some of the papers. There will be great emphasis on state of the art developments in the last few years. We're all eager to hear that. But of course there will be some superb papers by those who do not always speak disparagingly of the older and more classic instruments. This morning we will get a chance to both laugh and learn at a paper entitled: "Before Transistors, IC's, and All Those Other Good Things", especially when it's delivered by a person like Ron Kathren.

Thank you for being here. I thank the Symposium Committee for the arrangements, and I have warmest best wishes to all of you for a pleasant and a prosperous learning week. Thank you.

CHAPTER PRESIDENT'S ADDRESS

Madam Chair, Mr. President, Officers, members of the Board of Directors, distinguished guests, and colleagues, it is my honor and pleasure as president fo the Southern California Chapter to bid you welcome to Southern California and to the Eleventh Midyear Topical Symposium - Radiation Instrumentation, cosponsored by the Health Physics Society and the Southern California Chapter of the Health Physics Society.

I would like to touch on several areas. First, you are about to embark on an experience in radiation instrumentation knowledge, state-of-the-art knowledge, in the texts of 46 papers to be presented serially over the next three days. These speakers were selected by the program committee based upon the desired content of this meeting. Not all papers submitted were accepted for presentation to you. However, those that were not should feel no shame.

The symposium committees, under the direction of their general Chair Janet Trunzo, have put forth a grand effort to make this meeting successful professionally, technically, and socially. I have no doubt that most of you will agree by the end of this meeting that they have all performed superbly.

I would like to note several unique features about this symposium. First, the entire organization of this meeting was done with help from within the chapter. No outside profesisonal organizations were used. Each committee listed performed all the duties they were given. Second, the initial seed money came from the chapter treasury, not the National Health Physics Society treasury nor one of the associates of the H.P.S. Third, we encouraged submission of papers by the engineering and development staff of our Health Physics Society affiliates and commercial exhibitors. The fourth point is one that "happened", but in the organizational phases, we elected the first woman chair of a Midyear Topical Symposium. To the best of my knowledge, Janet Trunzo was preceded, somewhat, by one other woman in the Health Physics Society, Dr. Elda Anderson, who chaired the 1958 Health Physics Society annual meeting in Berkeley, California, nearly 20 years ago.

In my younger years, I was often admonished with the phrase "you will only get out of it what you put into it". The profession of health physics has many persons with that philosophy. It is a philosophy which, I feel, cannot be stated too frequently. As we look around at other professions, skills and trades, we frequently see and hear the signs of discontent. Such signs may be in some of our work places, but generally, the professional health physicist has faired well. With the basic belief that the health physicist must put something in before he can expect anything out, and by carrying that idea into a daily practice, the health physicist has already established good terms and solid footing with fellow employees and management.

As I face such an impressive audience of "doers", and recall my past 20 years with the many enriching associations and experiences, and the numerous givers, all of which played their part in molding my professional form, I would like to encourage you all not to change from the practice of giving. The full spectrum of human nature fills our ranks. We have our philosophers, our cynics, our seriously devoted and our humorists. But in their own way, your own way, when help is needed, you give. Rarely have I seen comparable acts of helpfulness and sharing in any other profession as I have heard of, as well as experienced, while in health physics. But giving is not the total picture. As proud professionals with an impressive record collectively, and almost in spite of it, our silence has cost us, for we no longer have financial support from government to educate and train the next generation of the needed professionals, technologists and technicians. We have suffered one of the few cases where action has not spoken louder than words. Our critics words. We are faced with the challenge of encouraging new students to invest in their own education in health physics to become our replacements. Then, when educated, we must encourage them to also be the doers, the sharers and the givers of the future. Health physics has a future we must mold.

We must make changes to face some of the future challenges ourselves. Changes such as extending our professional lives beyond our places of employment, and even outside of the Health Physics Society. In essence, to go public. Just because "you gave at work" doesn't carry the weight as a good excuse any more to not give at home in the community. Resting on our laurels during a crisis time of declining confidence in so many things, especially those in the nuclear field, is hardly the way to project a proud professional posture or promote nuclear understanding.

When you realize how many people can spell health physicist and how few can define it, something is wrong.

When you realize some of the obtuse postures that congresspersons take on nuclear issues of legislation, something is wrong.

When you see a community or city struggling to make the correct decision on how to vote on nuclear issues on the ballot - and they lack a reasonable consultant - something is wrong.

What seems to be wrong is the conservative health physicist who doesn't want to get involved. "I don't have all the answers. My expertise is in cesium in Reindeer". Well, within professional circles, where extremely accurate communication is required, that may be a valid posture. But in your home community, who is best qualified to interpret or translate the technical language into peoplese? I'd bet my money on the health physicist who lives in or near that community of searching citizens. Where is that potential interpreter, the reducer of extreme fear and ignorance? In all likelihood, sitting on his fat safety record, uninvolved.

We need to do more tooting of our own horn outside of our own professional circles. We need to become more open, helpful and giving in our home community. As we tend to do literature research to answer questions at the office, so must we also give the same measure for the folks back home. If we are supposed to know what we are doing in the field of radiation safety, it seems to be a well kept secret outside of our workplace.

With the thought of changing for the future, trying different approaches, learning something new, breaking with being fixed-in-one-rut posture, I hope many of you here have come to hear about something you do not feel expert in. Those who possess expertise in one or more areas to be covered in this symposium, I hope you will stay to learn more about areas in which you feel you have little or no interest. In so doing, you may surprise yourselves by finding application in your present work, or have an original idea come from the combination of the new knowledge with your areas of expertise after returning to work next near. One never knows.

In keeping with my own admonition of adhering strictly to speaking time alloted, let me conclude by saying once again, we welcome you to Southern California and the Radiation Instrumentation Symposium, and on behalf of the many symposium committee people, the Southern California Chapter of the Health Physics Society, the Health Physics Society and the Host Hotel and City, we sincerely hope that this will be a very rich and rewarding experience, and one you'll think pleasantly of, often.

I will now change hats to chairman, session A, for invited papers.

TECHNICAL SESSION A

INVITED PAPERS

Chair: Bill Wadman

BEFORE TRANSISTORS, IC'S, AND ALL THOSE OTHER GOOD THINGS: THE FIRST FIFTY YEARS OF RADIATION MONITORING INSTRUMENTATION

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INTRODUCTION

The measurements that led to the discovery of both X rays and radioactivity in the waning years of the 1800's were both made with solid-state devices - devices much different than the transistors, integrated circuits, and their relatives and descendents. Indeed, compared to the discovery of radiation, the transistor is a Johnny-come-lately, having not been discovered until 1948 by John Bardeen, Walter Brattain, and William Shockley at the Bell Laboratories.

The very first measurements, and therefore the very first instruments for the detection of radiation, were made with solid-state devices still in use today: fluorescence and photography. It was Professor Roentgen who noted the pale fluorescence from a screen in his darkened laboratory that led to his intuitive discovery; and Becquerel used photographic plates, sealed in black paper to exclude all light, to detect the emanations from samples of uranium-bearing ores.

Among the earliest attempts at radiation measurement were those of Ernst Dorn, a German physicist who in 1897 attempted to quantify the energy transfer by X rays with an air thermometer, using this commonplace scientific device to measure the heat produced by X-ray absorption in sheets of metal.(1) This was, however, hardly an early instrument, for its use was limited to the laboratory.

Most early measurement was, in fact, made by photographic or fluorescence techniques and was more in the line of detection rather than quantification. Gold leaf electroscopes were used early on, with the time taken to discharge the unit used as a crude method of quantification. There was, however, a definite need for quantification in the medical area, and early on this need was met by chemical dosimeters of several types. In 1902, Guido Holzknecht became the first to define a dose using chemical methods.(2) Using a solid-fused secret mixture of hydrochloric acid and sodium carbonate, Holzknecht correlated the discoloration in the mixture with radiation dose. The dose unit was the H, defined such that 3H produced a mild skin reaction. The dosimeter was called a "Chromoradiometer".

The following year two French physicians, Raymond Sabouraud and Henri Noire', produced a colorimetric chemical dosimeter based on radiation-induced color changes in a pastille of barium platinocyanide.(3) This chemical changes from bright green to dark yellow under irradiation; one S-N unit was found to equal five H units. Both the Chromoradiometer and the Sabouraud-Noire' dosimeters were commercially available by 1904.

The year 1903 also brought with it the first radiation instrument, in a sense. This was the spinthariscope of Sir William Crookes, a device that was simply a lighttight tube with a ZnS screen at one end and an eyepiece at the other.(4) A tiny quantity of radium or other alpha-emitting material could be placed behind the thin screen, and the individual flashes or scintillations from each alpha particle observed and counted. Ah, tedium!

Ionization measurements, for years the basic radiological measurement, were apparently first begun by French physicist Louis Benoist, aided by his Roumanian associate, Dragomir Hurmuzescu, in eary 1896.(5) Together they used a small gold lead electroscope, observing the deflection with a simple mirror and telescope arrangement. The rate of deflection of a simple gold leaf electroscope was used by the Curies in many of their early measurements, as well as by J. J. Thomson, Rutherford, and other early scientists of note.

The state of the art with regard to the first decade of measurements was neatly summarized by Rutherford as Chapter III of his pioneering book, Radio-Activity.(6) In this 22-page chapter, Rutherford notes three methods of measurement: photographic, ionization, and scintillation. Most of the chapter deals with electrical - i.e., ionization - measurements, and almost all of that portion deals with electrometers. Rutherford describes electrometers in detail - construction, applications, and use - noting that they are "The most generally convenient apparatus..." He also describes a crude parallel plate ionization chamber and a cylindrical one in which the radioactivity was, of course, placed inside. Readout of current was with the electrometer, usually a quadrant electrometer, which was effectively a very simple device used to determine the rate of charge produced in the gas in the ionization chamber. There was no amplification, of course, and the power supply was a simple battery.

Indeed, in the first decade after the discovery of X rays and radioactivity, instruments were quite primitive, with most reliance in the field placed on chemicals that showed colorimetric changes and, to a lessor extent, gross observations of fluorescence or photographic effects. But the stage was set, although for the next 35 years it would seem that progress would be very slow.

1905-1930

The quarter-century span from 1905 to 1930 produced numerous interesting and significant advances in the field of radiation instrumentation. Measurement was one thing, but what to measure was another. Ionization seemed to hold the most promise, at least in the laboratory, and to many the need for a radiation unit based on ionization represented a necessary first step. In 1905, Milton Franklin of Philadelphia first proposed such a unit(7), and several others, notably English physicist Charles E. S. Phillips and his French colleague Paul Villard, the discoverer of X rays, followed his example, as did the American physicists S.J.M. Allen and H. K. Dunham.(8,9) By 1908, a formal definition of dose, based on ionization, was reasonably well accepted by at least those involved with X rays.(10) The Curie as the unit of radioactivity was formally adopted in 1912 based on work by Mme Curie herself.

As early as 1907 Rutherford, working with Hans Geiger, a German physicist, then at the Cavendish Laboratories in England, began to use gas-filled tubes for the detection and measurement of radioactivity. They made a cylindrical tube with a central axial wire, applied a potential, and projected alpha particles into the tube.(11) The result was a much larger current than previously obtained, producing a current measurable even with "...an electrometer of moderate sensibility". This appears to be the first practical application of gas amplification, which had been discovered about 1900 by James H. Townsend, a physicist working in J. J. Thomson's laboratory.(12)

The Geiger tube as such was further refined by Geiger in 1912 and 1913 after returning to Germany.(13) His original version was a point counter that employed spherical geometry and used a filar electrometer. In 1928, in conjunction with Walther Muller, he constructed counters with large sensitive areas and with varous fill gases, very similar to the modern day Geiger-Muller tubes.(14) Development of these tubes was made possible by the advances in electronics, particularly with regard to vacuum tube applications, which provided appropriate stable high-voltage power supplies, fast response times, and amplification as well as recording capability. Particularly important in this latter regard was the trigger circuit of Eccles and Jordan(15) devised in 1918, which made use of two triodes in a bistable configuration. Another significant advance was the stable amplifier developed by the Swiss physicist Henri Greinacher in 1924.(16) This amplifier was specifically applicable to pulses produced by ionizing particles in gas-filled detectors.

Geiger, in association with O. Klemperer, also was instrumental in the development of modern proportional counting techniques. In 1928, the same year the modern Geiger-Muller tube appeared, Geiger and Klemperer perfected the technique of differentiating between alpha and beta particles on the basis of pulse size using a proportional counter as a detector.(17)

To the cosmic ray physicists must go the credit for the development of coincidence counting, which has its origins in the late 1920's. Early circuitry for coincidence counting was developed by Bothe and Kolhorster(18) and by Bruno Rossi(19) for studying cosmic rays in 1929. Throughout the 1930's, these circuits were improved upon and served to develop the basic methods for coincidence and anticoincidence counters used in modern day health physics work. Merle Tuve, an early childhood friend of Ernest Lawrence, was also an early developer of coincidence counting equipment.(20)

Instrumentation for measuring X-ray quantity - exposure or dose measurements, as it were - were developmentally slowed by many factors. Not the least of these was the difficulty in agreeing on a standard unit and type of measurement. Ionization, which seemed so promising in 1908 as the means of standardization, was difficult to measure in practice; X-ray quality was largely an unknown. Electronic equipment development was really not begun until the 1920's, and so the pastilles of Sabouraud and Noire' plus the later development by Kienbock(21), who used strips of silver bromide impregnated paper, were used well into the 1920's. Photographic film was also a widely used method of X-ray measurement. Indeed, in 1920, the U. S. Army X-ray Manual stated that only chemical and photographic methods were suitable for measuring X-ray dose.(22)

Indeed, photographic film was early applied to radiation protection measurement, and in a sense is the first health physics instrument. George Pfahler, an eminent Philadelphia radiologist recommended the use of film packets for X-ray and radium workers in 1922; and four years later Edith Quimby, a New York radiological physicist, devised a film badge with filters to minimize energy dependence.(23, 24)

An interesting instrument for measuring X-ray intensity was the photoelectric device developed by a German physicist in 1915.(25) A selenium cell, made from a flat insulator wound spirally with two evenly spaced wires with selenium in between, was connected to a resistance-measuring device such as a Wheatstone bridge. When exposed to radiation, the resistance of the cell drops in proportion to the intensity of the field. Thus, an effective ratemeter was made, using the change in resistance introduced in selenium by X-ray exposure. Unfortunately, this promising instrument was highly energy dependent, and in addition, the selenium showed a tendency to fatique with continuing exposure.

It was in the ionization instrument area that perhaps the greatest progress was made. Although the Roentgen unit was not formally adopted until 1928 at the Second International Congress of Radiology in Stockholm, many investigators were attempting to obtain better ionization measurements and to devise practical instruments for routine measurement. In the late 1920's the goal was well realized, and several such instruments became generally available. Condenser ionization chambers with small chambers were developed in the early 1920's including those by Duane(27); Iser Solomon, a French radiologist(28); and by Otto Glasser in collaboration with Robert Fricke(29), a radiotherapist at the Mayo Clinic. Solomon's devise was to be sold commercially under the name Ionometer, with about 1,500 units ultimately placed into operation. Several other commercial ionometers were available, mostly from German sources in the early 1920's.

In 1927, John Victoreen, at the suggestion of Fricker and Glasser, developed the first American commercial ionization chamber.(30) This chamber used a carbon wall and aluminum electrode and was based on a simple 1 cm^3 chamber charged with a friction wheel. The following year Glasser, in collaboration with U. V. Potmann and V. B. Seitz, made an experimental condenser-type ionization chamber which permitted the chamber to be removed from the electroscope during exposure.(31) This was the famous Glasser-Seitz dosimeter, which was commercially produced by Victoreen beginning in 1930. This venerable instrument survives today in a similar form as the Victoreen R-meter; the first Victoreen R-meters had friction charging and a battery-powered light to permit reading of the calibrated scale. Doubtless many of these are still in use today.

The 1920's also saw the creation of what was probably the world's first and, no doubt for a long time, only portable survey meter. This was an ionization chamber developed by Lauriston Taylor of the National Bureau of Standards in 1929.(32) The instrument featured three interchangeable chambers of various sizes, thus providing "range changes" in a rather different manner than contemporary survey meters.

THE 1930's

In 1930, the discovery of the transistor was only 18 years off in the future, and the age of the vacuum tube was in full swing. To some extent, the decade of the 30's was marked by a consolidation of the accomplishments of the earlier decade; but there were also real gains in the radiation instrument field, sparked by the discovery of the neutron by James Chadwick in 1932, the development of the cyclotron by E. O. Lawrence, and the appearance of artificially made radionuclides. In the electronics area, many new scaling circuits were devised, including scales-of-five and decade scalers. These were relatively complicated and expensive by the standards of the time, and decade sealers, as such, did not become widespread until the late 1940's. Count rate circuits also made their appearance, having the basic design of a large capacitor shunted by a resistor. When the equalized pulses from an amplifier were fed into the capacitor, the leakage current through the shunt resistor was proportional to the count rate and was read directly with a milliammeter or amplifed through a vacuum tube voltmeter.(32A)

Another significant electronic advance was the vacuum-tube electrometer, which used a single thermionic tube with high-grid resistance and low-grid current in an amplifying circuit designed to compensate for small changes in the filament and plate supply voltages.(32B) The vacuum tube electrometer provided distinct advantages over the duant and quadrant and other earlier types in that it had no moving parts and was often easier to stabilize. It made a perfect marriage with the Lauritsen electroscope, a rugged and simple device developed in the mid-1930's by Charles and H. Lauritsen of Cal Tech.(32C) Unlike earlier electroscopes, which were essentially an ion chamber containing two charged gold leaves which converged by gravity as they discharged, the Lauritsen electroscope had but a single moving element, a gold-covered quartz fiber. The electrostatic force on this fiber was balanced by a mechanical force in the form of tension in the charged and hence deflected fiber. Thus, the unit was free of reliance on gravity and could be operated in any position. The fiber could be easily observed by a microscope and its deflection ascertained by a small calibrated scale mounted behind it. Its small size made it very sensitive, and it was widely used throughout the late 1930's and even into the early 1960's.

For counting particles, the Lauritsen electroscope, equipped with an appropriate window, was sometimes used as were some scintillations and the various gas-filled counters: ionization chambers, proportional counters, and Geiger-Muller counters. Of all the particle counters, the G-M was the most common for it was relatively simple, highly sensitive, and was a true pulse counter with great amplification. The 1928 improvements by Geiger and Muller allowed great flexibility in the design of the tube. Virtually the entire cross-section of the tube could be used as the window and, if constructed in the traditional cylindrical shape, a large volume and hence quite sensitive tube could be made. A variety of materials could be used in construction, allowing beta particles or even alphas to be admitted and counted.

Between 1930 and 1940, there were numerous improvements in Geiger tubes, and special processing and manufacturing techniques and special purpose tubes developed. Although throughout most of the decade thousands of Geiger tubes were made in all shapes and sizes, most were not self-quenching - that is, the mechanism in the tube. Quenching was accomplished by a special high "quenching resistance" or an equivalent circuit. The quenching resistance, although simple, resulted in long-time constants, effectively limiting the use of the counter. Simple quenching circuits, using one or two triodes, were devised in the 1930's which reduced the quenching time by a factor of about 100 to as low as 100 μsec, a fairly respectable value for the 1930's.

But the additional electronic circuitry was undesirable from many standpoints: cost, complexity, reliability; and, most of all, the tube still had a long dead time which was largely a result of the long quenching time. Then, in 1937, the Dutch physicist Trost discovered that small amounts of an organic vapor such as ethanol or amyl acetate, added to the fill gas in a G-M tube, produced quenching right inside the tube.(33) These counters eliminated the need for high resistance and hence were termed "fast" counters. The term "fast" was in fact a misnomer, because the use of quenching circuitry, such as simple two-triod multivibrator could effectively reduce the dead time of most tubes to 150-250 μsec, about the same as achieved with the self-quenching variety.

Neutrons presented then, even as today, interesting and special measurement problems. Although it may seem hard to believe, neutron detection in the 1930's was accomplished with virtually all the same types of detectors in use today. Proton recoil counters using a paraffin-lined ionization chamber and a pulse amplifier were among the first neutron instruments, and cadmium-wrapped G-M tubes were also used. Helium-filled G-M tubes operated in the proportional region were developed in the 1930's, and by the end of the decade, the BF_3 counter was in wide use.(34) An interesting application was the use of the Glasser-Seitz condenser R-meter for neutron measurements. When exposed to neutrons, proton recoils, produced primarily by fast neutrons interacting in the Bakelite chamber wall, would enter the sensitive volume, causing the chamber to discharge. The specific ionization was, of course, greater from a proton recoil than a proton; hence, to fast neutrons, there was an appreciable response. This response was calibrated in n units, one n unit being equivalent to the amount of X rays required to produce a comparable discharge - i.e., IR - in a chamber.

The decade of the 1930's was also notable for some rather significant advances in the area of personnel dosimetry. The film badge, which was really a product of the late 1920's was, in a sense, married to the Roentgen in 1928 by Heinrich Franke, a German Engineer who was later to become an executive of Siemens-Reiniger-Werke (SRW), a large German firm. This was a major step, and of particular value to radiologists insofar as X-ray quality was concerned, for many leaders in the field of radiation physics thought it impossible to equate the Roentgen, defined in terms of air ionization, with a chemical effect such as the reduction of silver, which is of course the mechanism of film blackening.

But it was in 1930, two years later, that two Dutch engineers, Albert Bouwers and J. H. van der Tuuk, adapted and expanded the principal of Franke to personnel dosimetry. In their paper they suggested a "tolerance dose" of $7x10^{-6}$ R/sec, or 0.2 R/day, noting how this level could be measured with film.(36) This was the first attempt to tie a film exposure to personnel tolerance dose and, as such, represented a great advance, although it did not find particularly widespread application since in the 1930's health physics as such was still very primitive and personnel dosimetry was at best haphazard.

From the standpoint of instrumentation theory, the 1930's saw the statement of the Bragg-Gray principal(37-39) which, together with the definition of the Roentgen, served as a basis for the design of the first generation of health physics instruments which were to appear during the 1940's. The Bragg-Gray principle, simply stated, notes that if there is a small air-filled cavity in a medium (such as tissue or plastic) that is being irradiated, the ionization-per-unit volume in the cavity will be directly proportional to the ionization-per-unit volume in the medium. The constant of proportionality was simply the ratio of the stopping power for secondary electrons in the medium to that of air. There are, of course, certain conditions that must be fulfilled for the Bragg-Gray principle to hold; for example, the cavity dimensions must be small relative to the range of the secondary electrons produced in the medium, but relatively these can be readily met.

The Bragg-Gray theory actually had its roots in a work by the great British physicist Sir William L. Bragg published in 1912. This book, entitled simply Studies in Radioactivity, formulated the basic relationship between the ionization in a chamber irradiated with X rays and the electron-stopping power of the chamber wall. Bragg, of course, was the senior member of the father-son team that showed that diffraction could be used to determine X-ray wavelength, and who, in 1915, would share the Nobel Prize in physics with his son for their applications of X rays to the study of matter. About 1904, Bragg - for whom the Bragg curve, used to characterize ionization as a function of track length for heavy particles, is named - discovered the discrete energy or range of alpha particles emitted from the nucleus.

It was L. H. Gray who independently restated the findings of Bragg and refined them into the modern version of the Bragg-Gray principle. Gray was a great British radiation physicist whose contributions have had profound impact on dosimetry and radiation biology and in whose honor the new SI unit of absorbed dose, the Gray, is named.

From a practical health physics standpoint, the decade of the '30's was significant in that it saw the development of the first commercial Geiger-Muller survey meter, which featured a glass-walled G-M tube, earphones, and a weight of about 20 pounds. This was the product of the Victoreen Instrument Company, which also produced a portable ion chamber instrument in the late 1930's. The instrument was, perhaps, the first commercial portable survey meter and was powered by a 45-volt "B" battery.(40) The heart of the instrument was a miniature electrometer tube which had been developed for use in hearing aids by a Dr. Ewing, then a professor of physics at Northwestern University. Through the efforts of Dale Trout, twenty of these tubes were obtained; and Trout, along with John Victoreen and George Bell, an electronic technician, devised the instrument which made use of some common materials in unusual ways. The chamber, for example, was a cardboard mailing tube coated with Aquadag to make it conducting. A brass and paper pill box, obtained from a local pharmacy, was used to provide the support for the center electrode, the brass ring of the box being first fitted with an insulator to support the tube.

Victoreen also developed and made several other radiation monitoring instruments during the 1930's, in addition to the survey instruments which closed out the decade and the Glasser-Seitz R-meter which began it.(30) These included the Integron, an electronic integrating ionization chamber and early ancestor of the modern Radocon, which appeared in 1935 and used a static vacuum tube amplifier. The Integron, which went through several iterations in the late '30's, was useful to radiologists for measuring cumulative skin doses of X rays. The 1938 model featured an electrostatic alternator and ac amplifier. Another instrument was the Iometer, which first appeared in 1937 and provided radiologists and others with a measure of beam constancy. And to usher in the 1940's, Victoreen brought forth the Minometer, still available and useful today. The Minometer of 1940 was, as it is today, a modified R-meter chamber for personnel use - in other words, the first pocket ionization chamber. It was of the nonself-reading type, being provided with a readout device that the dosimeter needed only to be plugged into to obtain a large-scale reading of 0-200 mR.

The Victoreen Instrument Company was, in the United States, a pioneer in the development and manufacture of radiation measuring instruments. The company was officially founded in 1928 by 26-year-old John Austin Victoreen, a brilliant, self-taught engineer, to manufacture the Fricke-Glasser X-ray dosimeter. Victoreen, whose formal education included no college, would later win an honorary doctorate in science. The first advertisement of the Cleveland-based company appeared in the September 1928 number of Radiology, offering a "Practical-Flexible" dosimeter, calibrated in the R-unit, that would "eliminate(s) the possibility of X-ray burns". Victoreen was perhaps the first American company to become involved with commercial radiation monitoring instruments. The year following Victoreen's offer of the Fricke-Glasser dosimeter, the Standard X-ray Company of Chicago offered an integrating-type ionization chamber designed to attach to the primary beam part of an X-ray tube. This device, which was successfully marketed for a number of years, was the creation of Robert S. Landauer, Sr., who, among other things, was for many years physicist at Cook County Hospital in Chicago. Landauer's creation was known as the Roentgenometer and was directly connected by a large, metal-shielded cable to a meter readout.(41).

The 1930's also saw the development of the photomultiplier tube, or electron multiplier as it was more commonly called. The European physicist Z. Bay had been experimenting with direct detection of ionizing particles with photomultipliers as early as 1938,(42) and in 1940, J. A. Raichman and R. L. Snyder(42A) developed what is essentially the modern version, which provided very large amplification of individual light flashes and thus paved the way for portable scintillation counters and photon spectroscopy.

Thus, as the 1930's drew to a close, there were a variety of instruments available for the measurement of radiation. There was, however, only a single commercial portable survey meter, and that only newly available, and a number of instruments for measuring exposure in Roentgens such as the Glasser-Seitz R-meter. Devices like the Lauritsen electrometer, although tedious to use, could be applied to area monitoring, and for personnel, film, and pocket dosimeters, the latter in the form of the modified Glasser-Seitz chambers, were also available. There were reasonably good laboratory counters and fast circuits, but compared with what was to be available and in use by the end of World War II, only a half decade hence, the armamentarium, at least from the standpoint of the radiation protectionist, was weak indeed.

THE LAST HALF DECADE

As the 1940's began, there were but six years left in the first half-century following the discovery of X rays and radioactivity. These few years were to yield an extraordinary development surge in the detection and measurement of radiation, largely as a result of the Manhattan District which produced the atomic bomb. Not all development took place in the Manhattan District, however.

Several studies were carried out in the early years of the 1940's of the response of Geiger-Muller tubes that led to some interesting and novel applications. One of these was the directional counter tube developed by H. G. Stever(43), in which the collecting electrode was divided into three equal lengths by two appropriately spaced beads. A particle traversing the entire length of the tube would produce a discharge along the full length of the electrode, and hence a triple-size pulse would be observed with the aid of an oscilloscope. Scattered particles might not traverse the full length and hence produce only a double- or single-size pulse. If the axis of the tube (and hense its collecting electrode) was parallel to the beam or field, the number of triple pulses would be maximal. Conversely, when the tube axis was normal to the beam, the number of triple pulses would be minimized and the single pulses maximized. The Stever tube was, in a sense, a curiosity; but it did fill a need in a simple and direct fashion.

An interesting electronic circuit developed outside of the Manhattan District was the decade or scale-of-ten devised by J. T. Potter.(44) Potter adapted and modifed the Eccles-Jordan trigger circuit of 1918 by a feedback technique which coupled four triggers into an array with four neon glow tubes. The tubes were lit in a coded sequence to indicate the count. Since four triggers would give $2^4 = 16$ counts, the final six counts were cut off from the decade. When the eleventh count was received, the coding system would drive the next decade or the register, as appropriate.

But the major advances were, of course, associated with the Manhattan District and were so numerous that a complete chronicle is far beyond the scope of this brief work. Even if only health physics type instruments are considered, there is a great deal to report. Development went on, sort of in parallel at both Oak Ridge and Hanford, and in many respects the record as to who did what, first or otherwise, is muddled by the feverish activities of those hellion days.

An early development was the development of pocket ionization chambers, used in pairs to measure the daily radiation exposure of personnel. Initially, these were not too rugged, but later developments made them so reliable that false readings were obtained only 0.16% of the time. Film badges were instituted in 1943, each with an open window and 1-mm-thick cadmium shield. The technique of using filters and correlating optical density with dose was greatly refined by Manhattan District health physicists. The film packets used contained two films - one low range (20 mR to 20 R) and a high range (1 R to 400 R). Nuclear emulsions 30 μ thick on a standard film base were worn also, and film rings for monitoring hand doses were also used.(45)

But perhaps the most dramatic and meaningful accomplishments from a health physics standpoint were in the field of portable instruments. By the end of the war, just a few months before the 50th anniversary of the discovery of X rays and radioactivity, no less than 16 portable survey meters, including the one available at the start of the war, were in use.(46) For alpha monitoring, there were three specific instruments: Poppy, Zuto, and Pluto, plus the Zeus, an ion chamber useful for α, β, and photon radiations. The Poppy was a large instrument whose low-level detection was aural - based on pops in the speaker. At 60.5 pounds it was portable, but more in name than in fact. Zuto and Pluto were less sensitive, but weighed between seven and ten pounds. Warmup time ranged from 30 seconds to 90 seconds, and both were battery powered. The Poppy was ac powered.

Eight β-α instruments were also developed. One of these was the Walkie-Talkie, a low-range instrument based on clicks, and the Cutie Pie and Fish Pole. The Fish Pole was an ionization chamber designed for remote monitoring and had the fastest response time - one second - and warmup time. The Cutie Pie was the lightest at four pounds, with a warmup time of 30 seconds; it is, of course, the forerunner of modern Cutie-Pie instruments. Response time was ten seconds, and it featured three ranges - 0-30, 0-300, and 0-2000 mR/hr. Another ion chamber survey meter for β-α went by the name of Paint Pail, so named because a paint pail was in fact what it was housed in.

For neutrons, a compensated ion chamber was devised. Since the device had two chambers, identical except that one was boron lined, it was named Chang and Eng after the famous Siamese twins. The Chang and Eng was also the heaviest of the so-called portable instruments, tipping the scales at 80 pounds. There was even an air monitor developed and given the unlikely, if appropriate, name of "Sneezy".

The saga of health physics instrument development in World War II is a long and convoluted one, and the above serves only to provide some small measure of what was accomplished during those years. Suffice to say that almost exactly a half century after the discovery of X rays, the atomic bomb appeared, and the revolution in radiological monitoring that was the by-product of the bomb began to be revealed. The instruments of that time look, even 33 years later, reasonably modern at first glance; and even some of the names - Cutie Pie, for one - linger on today. The internals, which would be revolutionized again by the transistor and its progeny in the early '50's and beyond, were adequate to the task and provided an earlier generation of health physicists with the measurements on which our profession is based.

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WHAT WE HAVE AND WHAT WE NEED

JOHN HANDLOSER

Looking back, I make a habit of following the wrong people. One time, I followed Hoyt Whipple, who, as many of you know, is a very difficult act to follow. Another time, I followed Dale Trout, and that was impossible. This time I even had something to do with the program and I put myself after Ron Kathren. How do you follow a presentation like that?

This is an opinion paper. Unlike the papers in the other sessions, which are based on fact and hardware, my remarks on what is needed in H.P. instrumentation are based on my opinion, observations and experience. Fortunately, I do not have to have solutions to the problems I will mention. I plan to discuss a number of types of instruments we have, and mention their problems. I also plan to mention some instruments I think we need, but are not yet developed. But first, I can't pass up the opportunity to mention one thing I believe we need in the instrument field today.

From the dim past, I recall three names: Joe Deal, Bob Butenhof, and Dick Johnson. Those of you in the field long enough remember these gentlemen, and others, were part of the instrument branch of the division of Biology & Medicine of the AEC. The branch's tasks included evaluating commercial instruments and cataloging them. I still have (and occasionally refer to) RIB-8 (1954) which technically describes most of the available instruments at that time. This organization helped both the user and the manufacturers. No such organization exists today, and I believe we need one to act as a clearing house for H.P. instrument information, and to impartially evaluate H.P. instruments. One thing I'd like you to bear in mind during this discussion is that as the years pass, the regulations require that we make measurements of smaller and smaller quantities of radiation. Thus, an instrument that was once considered very sensitive may now not have the required sensitivity.

Portable Instruments

The most widely used single instrument in H.P. is the portable GM rate meter. It is often used for two purposes: detecting contamination, and measuring radiation fields. I have a personal objection to its use as a radiation monitor because of its energy dependence if the response is not corrected by shields. I find many such instruments on the market. I also find many GM survey meters on the market which jam at high dose rates and read zero. One should be aware of these factors when purchasing this type of instrument, because good instruments are available.

As a contamination monitor this instrument is very useful. Although the pancake GM counter was once considered very sensitive, it now lacks sensitivity to measure some of the very low levels of contamination now permitted. John Sommers at INEL has studied this instrument's ability to pick up small low level contamination areas. The results indicate it would take about 10 minutes to do a thorough survey of a person's clothing, because the probe must be moved very slowly in order to avoid missing the low level of contamination that is now considered permissible.

This discussion points out an instrument we need. An ultra-sensitive contamination monitor for quickly clearing people and objects for unrestricted release. The Beta/Gamma levels we are discussing are of the order of 1 000 d/m/100cm^2. This is equivalent to about 30 c/m with a pancake GM counter. But, this rides on top of a 50 to 100 c/m background counting rate. It is difficult, if not impossible, to detect this level of contamination during the few seconds the probe is in front of the contaminated spot. You will be hearing from Rupprecht Maushart on one solution to this problem.

Some excellent radiation fieldmeasuring instruments are the portable ionization chamber instrument. Many are accurate and reliable. In general, however, they lack the sensitivity needed for uncontrolled area surveys. If you want to know if your source is producing 100 mrem in 7 days, you must be able to measure 0.6 mrem/h. If you want to know it your source is producing 500 mrem in a year, you must be able to measure .06 mrem/h.

The ionization chamber instruments often lack the true air equivalence to accurately measure scattered radiation from diagnostic x-rays. Most of the instruments are useful for 100 kVp x-rays, but are most inaccurate for scattered radiation from mamography x-ray machines. More and more surveys are being required in this area. This subject will be discussed in a paper by Bruce Horne (Permanente Group, L.A.) and Dick Furgus (ANL) later in this symposium.

Scintillation counters seem to be making a comeback into the gamma measuring field. They used to be used for prospecting, but now, in new packages, they are being used with NaI crystals for low level gamma radiation measurements in the μR/h ranges.

Survey meters for detecting alpha contamination are available, and probably can't be much improved as far as sensitivity is concerned. The sensitivity of even the large area probes is such that an appreciable time is required to accumulate enough counts to make a measurement of low level contamination. Two general types are available: scintillation counters and gas flow proportional counters.

In passing, I would like to mention the problems of measuring tritium. Since it is so difficult to measure, and its use is spreading, we need more instrumentation. Tritium monitors exist on the commercial market today, but I note a number of papers in this symposium dealing with this subject, indicating more development work is necessary in this field.

In general, good Neutron instruments are available which read directly in mrem or mrem/h. The Spherical Moderated Detector is a good example of a practical (somewhat heavy) instrument coming from development work at the National Labs, particularly from the work of Dale Hankins.

But what is lacking at this time is a rem meter for mixed gamma-neutron fields. Absorbed dose measuring instruments are available, but not instruments which measure dose equivalent. With the present state-of-the-art in electronics, it should be possible to devise a detector and rate meter which weights the neutron response with the proper quality factors and adds it to the gamma absorbed doserate. This type of isntrument would be very useful around reactors and around accelerators, if the instrument was able to handle pulsed radiation. You are probably aware that work on such an instrument is going on at BNL (John Baum), Rossi, & EG&G, but no practical survey instrument is presently available.

This leads to a discussion of instruments which are not yet developed, but would be useful in specific situations. I know of two such instruments, but I'm sure you know of many more. The first is a survey instrument for use around the large, high energy accelerators, such as AGS at Brookhaven or Fermilab. The radiation is pulsed with a pulse width of microseconds, and a repetition rate of the pulses every few seconds. The instantaneous doserate is very high during the pulse. What is needed, is an instrument which integrates the dose from the pulses over a practical period of time, continuously divides the stored dose by the integration time, and calculates the average absorbed dose rate. At this type of accelerator it is important to measure the absorbed doserate in tissue, because all types of particles are present. Ultimately, this nonexisting instrument should be improved to read in mrem/h for all types of radiation. Peter Golon and his group at Fermilab are working on this development.

The second instrument requiring development for use in facilities where the limiting exposure is from beta particles, is one which measures both penetrating and nonpenetrating radiation at the same location. Jim McLaughlin at Environmental Measurements Laboratory has constructed a prototype of a dual ion chamber instrument for this type of measurement.

Before leaving the subject of portable instruments, I would like to tell you what I see coming. I believe that soon a whole new family of "smart" instruments will be developed. The Fermilab instrument I just discussed is an example of what I call a smart instrument. With the electronics available today, I can visualize a microprocessor Type electronic package which can automatically handle any type of detector, and present meaningful data. It could also be programmed to handle many special situations. I note a start of this trend in a paper at this symposium by Jimmie Little of Eberline. I also note another new smart instrument to be described by Dave Raymond of MDH.

Installed Instruments

We presently have excellent instrumentation for sample analysis. Gamma Spectrometers, automatic alpha-beta sample counters with ultra low backgrounds, and automatic liquid scintillation counters are available. Perhaps the reason these are available is that other sciences supported their development, and the H.P. use is incidental. At least 6 papers at this symposium deal with this subject. One instrument in this area which is needed is a simple computerized scaler, which with proper programming presents the data in useful terms such as microcuries or d/m. A smear counter scaler which presents the d/m/100cm^2 above background directly would be useful.

Another area which is well instrumented is the effluent monitoring instruments. The power reactor regulations and guidelines have necessitated the development of accurate, sensitive and automatic waste water and exhaust gas monitors. We also seem to have a reasonable good line of remote area monitors and criticality alarms. The use of microprocessors in this type of equipment makes them very versatile. Improvements will undoubtedly be made, but since they are available, I will not dwell on them. A number of papers will be given at this symposium on these instruments.

The instruments available for measuring contamination on people lack the necessary sensitivity. The need for quickly detecting small amounts of contamination on hands, shoes and clothing is real. The presently available portal monitors are excellent for catching gross contamination, but do not have the sensitivity to detect the small amounts required in the field today. I know of two approaches to this problem being worked on or in use. Brice Rich's group at Allied Chemical at INEL has developed a portal monitor employing many pancake tubes and sophisticated electronics which will be described by Cliff Nichols at this symposium, and Bob Stevenson at EG&G Idaho has developed a large area personal monitor which is a gas flow proportional counter. At the present, however, no portal monitor is commercially available which detects releasable amounts of beta-gamma contamination quickly.

Hand and shoe monitors are available, and some are excellent. I believe all the modern hand and shoe monitors should present data as a go - no-go signal, or as a digital display of the net c/m. The factory worker should not be expected to subtract background counting rates.

Personnel Monitoring Instruments

Film dosimeters have been used for personnel monitoring since the beginning, but, as I have heard said, if film didn't exist, it wouldn't ever have been invented for this purpose. TLD is replacing film because it does not have the characteristics of film which are unwanted for dosimetry. The instrument we need now, however, is a good neutron dosimeter. I know work is being done by Art Lucas on this subject at Harshaw, and Joe Cortez will describe some work on this subject being done at LASL.

Air Monitors

Air monitors for breathing air have become quite sensitive and sophisticated. The problem of quickly measuring alpha air contamination from transuranic elements has been solved by the use of solid state detectors and single channel analizers which eliminate the alpha counts from the Radon daughters. However, there still seem to be problems in the field, as is evidenced by the six or seven papers to be presented on this subject.

I'm sure there are many other instrument problems with which I am not familiar. In the past, they have been solved at the national labs or by the instrument industry. This is one area where an organization such as the instrument branch would be useful - locating the problems and finding the source to develop the instrumentation.

FUTURE LEGAL REQUIREMENTS

THOMAS D. MURPHY

I plan to summarize some of the requirements that the NRC currently has. I will hopefully pass on some of the new requirements that we expect will be coming out in the near future. In recent years, the NRC has been placing an increasing emphasis on the concept of maintaining occupational radiation exposures as low as it is reasonably achievable. This concept is not new to those of us in the radiation protection field. Most health physicist have espoused this concept of reducing exposure, or avoiding unnecessary exposure or keeping exposures as low as practical, for many years. What is new, however, is the increased emphasis recently being placed on this concept, not only by such renowned bodies as the ICRP and NCRP, but also by regualtory agencies, such as the EPA and the NRC. Although the NRC has had a requirement in this regulation since 1971, "that exposure should be maintained as low as it is reasonably achievable", it's only been in recent years that this requirement has been emphasized in our licensing, standards development and inspection and enforcement activities. Most of my comments are going to be pertinent and directed towards the commercial reactor plants for a good reason. We have been concerned for a number of years about the slowly rising value of collected dose per occupational radiation exposure at commercial reactor sites. Since 1969, the value has risen from less than 200 man rem per reactor per year to a figure of approximately 500 man rem per reactor per year in 1976. Although the average exposure per individual receiving measurable exposure at reactor plants has remained at approximately 700 mrem per year, the average number of personnel receiving that dose has steadily increased so that it was almost 700 people per reactor per year in 1976. So we at the NRC feel that the challenge exists to lower doses to personnel at reactor plants. To do this, it is imperitive that we are able to make consistent and accurately informed decisions concerning radiation fields and radiation dose. It follows that it is equally imperative that in order to make such decisions, we must rely on radiation measuring and monitoring instrumentation which provide consistent and accurate results. I am going to discuss the NRC role in evaluating radiation monitoring instrumentation and radiation monitoring programs used at our licencees facilities.

First a word about our regulatory authority. Title 10 Part 20 requires in addition to limiting exposure to personnel, that the licensee provides for personnel monitoring, that the licensee evaluates radiation hazards and measures concentrations into radioactivity in air and also that the licensee measures, when appropriate, radioactivity levels in the body. Although, instrumentation requirements are not explicitly called out in our regulations, the implicit requirement for reliable and accurate measurements exist. We have augmented our regulations by spelling out acceptable methods of measuring radiation and radioactivity in effluents in our licensees' plants as well as for personnel. We have done this by providing criteria and regulatory guides, we have included information in our reactor licensing standard review plans, we have included information in our branch technical positions contained in those review plans, and we have built-in requirements in our reactor plants standard technical specifications. As we go to each of the specific technical areas, I am going to point out what guidance exists now and what activity presently is taking place in NRC with regard to additional guidance pertinent to radiation monitoring instrumentation that we as health physicists or instrument designers should be aware of in the future development and use of instrumentation.

Let me discuss the area of personnel monitoring instrumentation. The commissions' requirements for personnel monitoring are spelled out in Part 20, Section 202, and in addition, we have amplified those requirements, specific to instrumentation, in Regulatory Guide 8.3, 8.4, and recently issued Regulatory Guide 8.14 on personal neutron dosimeters. Those guides endorse, with certain comments, ANSI standards, specifically ANSI Standard 12.7 on film badge performance criteria, ANSI Standard N 12.5 on the performance specifications for direct reading and indirect reading pocket dosimeters for X and gamma radiation and ANSI Standard N 319 on personal neutron dosimeters. These guides and standards outline our position on acceptable performance criteria for personnel monitoring is that the NRC is currently involved in a pilot test study at the University of Michigan to test personnel dosimetry performance. Through the study, our office of standards development is seeking to provide a test situation for the proposed ANSI Standard N 716 which is entitled "the criteria for testing personnel dosimetry performance". That standard was developed by the working group 15 of the Health Physics Society Standards Committee. Our contractor, the Unviersity of Michigan expects to be ready to start accepting dosimeters from dosimetry processors sometime this spring and all processors are invited to participate in that test study, and we hope that this is going to give the dosimetry processors an opportunity to find out differences in their processing techniques and interpretations from the test for laboratory radiations. The contract is also going to develop a comprehensive manual describing how the program will be conducted after the pilot study is completed.

Now let me discuss in plant, area radiation, and airborne radioactivity monitoring. Again, the requirements are called out for measurements of areas and airborne radioactivity in Part 20. These requirements for reactors have been augmented and clarified in Regualtory guide 8.8 in our standard review plans, chapters 11 and 12, and in our standard technical specifications for light water reactors. These documents provide guidance on Nuclear Society, Health Physics Society Standards Committee, working group 6.8 has drafted a standard for area monitors which is currently progressing to the concensus review process. This standard will provide criteria and information on the selection designed and placement of radiation detectors and other components for the area radiation monitoring systems for commercial reactors. The standard considers location of detectors, indicators, recording devices, range of measurement, sensitivity, accuracy, precision, alarms, power sources, environmental qualifications and calibration methods and frequencies. The NRC has participated in the development of that standard. We anticipate that that standard will provide acceptable criteria to the NRC for area radiation monitoring systems for reactors. In another development with respect to area monitoring, in our office of standards development again, we are working on a task to consider a rule change which is to specify the frequency in which portable area survey instruments should be calibrated. Such a change will probably be accompanied by a new regulatory guide providing approaches in instrumentation calibration acceptable to the staff.

Now, currently, our acceptance criteria for airborne radioactivity monitoring is spelled out in our Standard Review Plan, chapter 12, and we expect in the near future to change one of those criteria to require that airborne radioactivity monitoring systems be capable detecting at least 10 mpc-hrs inhalation exposure instead of the current requirement which is 1 mpc/hr inhalation exposure. I want to note that this is a change which raises the lower limit of detection for airborne radioactivity monitoring systems which is a little bit different from the direction of most of our changes.

I want also now talk a little bit about environmental and effluent monitoring. The commission outlined its requirements concerning radioactivity in effluents to unrestricted areas in section 106 of Part 20. These requirements have been augmented and clarified in regulatory guides, 1.21 for measuring, evaluating and reporting radioactivity in solid wastes and releases the radioactive materials and liquid and gases that effluent from light water reactors, and the new regulatory guide 4.13 on performance testing and procedural applications for thermoluminescent dosimetry for environmental applications. The recently issued radiological assessment branch technical position for environmental radiological monitoring and the newly developed effluent and environmental radiological monitoring standard technical specifications provide guidance on gaseous and liquid effluent radioactivity monitoring criteria. These will include information on sensitivity, alarm set points and location of monitor. Our branch technical position for an acceptable environmental monitoring program provides information on types of samples to be monitored for various pathways, sampling and collection frequency, types of analysis. An understanding of these criteria should be useful to you in the design of future radiation monitoring instrumentation. There are wide differences in the lower limit of detectability for various pathways which point out the wide range of instrumentation requirements and monitoring capability needed for effluent and environmental monitoring programs. It is the intent, also of that branch technical position, to require our licensees or licensee contractors to participate in the EPA laboratory intercomparison studies program or its equivalent. That's the program better known as the crosscheck program. That is a requirement that we are building into the effluent and environmental monitoring as a quality assurance measure. Also, of particular to you who in the uranium processing industry, is a recently issued regulatory guide on a measurement, evaluation, and reporting of releases of radioactive material in liquid and airborne effluents from uranium mills. This guide provides lower limits of detection for analysis of air particulate samples at 1% of the Part 20 limits, and provides for a lower limit of detection for stack effluent sampling and 10% of the Part 20 limits, and it provides for a lower limit of detection for liquid samples of 1% of the Part 20 limits.

Finally a word about accident and emergency monitoring. Regulatory guide 1.97 discusses this instrumentation for light water cooled reactor power plants to assess plant conditions during and following an accident. It provides recently issued guidelines of methods acceptable to the staff for complying with the commission's requirements in Part 50 to provide instrumentation to monitor plant variables and systems during and following postulated accidents in nuclear power plants. That guide required detailed safety analysis to be made with regard to ranges, location, accuracy, and the length of time required to provide an operator with accident information after the accident occurs. With regard to fixed monitoring instrumentation used to follow the course of an accident, the staff is modifying its standard review plan acceptance criteria, to provide that instrumentation be placed in locations shielded from the extremely high dose rates predicted as a result of the design basis accident. Let me say, that's for radiation monitoring instrumentation. An acceptance criteria will provide that applicants demonstrate that the instrumentation is calibrated to and will measure the range up to the maximum radiation fields expected in an accident. By placing instrumentation in shielded locations and most likely outside of the primary reactor containment, it can be expected that such instruments will survive the radiation damage and other environmental conditions expected in an accident.

I have tried to provide you with some information on recent and projected changes to our regulations and regulatory guidance which will affect instrumentation requirements for NRC licensees. It is our hope that these requirements reflect our desire to provide consistent and accurate decisions which respect the reducing dosage to personnel.

TECHNICAL SESSION B
ENVIRONMENTAL MONITORING I

Chair: Jim McLaughlin

A PROGRAMMABLE AUTOMATIC ALPHA-BETA AIR SAMPLE COUNTER

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ABSTRACT

A programmable automatic alpha-beta air sample counter was developed for routine sample counting by operational health physics personnel. The system is composed of an automatic sample changer utilizing a large silicon diode detector, an electronic counting system with energy analysis capability, an automatic data acquisition controller, an interface module, and a teletypewriter with paper tape punch and paper tape reader. The system is operated through the teletypewriter keyboard and the paper tape reader, which are used to instruct the automatic data acquisition controller. Paper tape programs are provided for background counting, Chi^2 test, and sample counting. Output data are printed by the teletypewriter on standard continuous roll or multifold paper. Data are automatically corrected for background and counter efficiency.

INTRODUCTION

Several years ago, a review of the routine air sample counting program in the Hanford Engineering Development Laboratory (HEDL) showed a need for improvement and modernization. An average of 100 air samples were being counted each day on two manually-operated counters, with all of the counting data being transcribed by hand. The counting workload was about eight man-hours per day. In view of the ready availability of automatic sample changers, improved solid state electronics and strong, reliable printers, it appeared that significant improvements could be made in manpower input and recording of counting data.

Discussions between operational health physics and instrument development personnel resulted in identification of all of the basic elements of a desirable counting system, and a general description of its operating functions. The desired functions were as follows.

1) Count alpha and beta radiation simultaneously.

2) Separate gross beta, gross alpha, and plutonium counts and place in individual channels.

3) Allow radon (and its daughters) counts in the gross beta and gross alpha channels, but exclude them from the plutonium channel.

4) Count up to 100 samples in one counting sequence.

5) Provide background subtraction.

6) Perform the Chi^2 counter test automatically.

7) Calculate the sample results and print out in μCi per cc.

8) Print output data on continuous roll or multiple fold paper 8-1/2 inches wide.

9) Modular construction for serviceability.

Since we did not know of any readily available system with all of these characteristics, we decided to develop such a system.

DISCUSSION

For ease of servicing and providing replacement parts, we decided to utilize commercially-available components to the maximum extent possible. However, it was recognized that some in-house electronic component development and construction would be necessary to provide necessary interface functions. The system, shown in Figure 1, is composed of an automatic sample changer, a large silicon diode detector, an electronic timer, four scalers, four single channel analyzers, a linear amplifier, an automatic data acquisition controller, an interface module, and a teletypewriter with paper tape punch and paper tape reader. The interface module and the sample changer electronics were designed and constructed by HEDL personnel. The other electronic components are standard, commercially available items. With the exception of the sample changer electronics, all of the electronic components are modular and are contained in a single cabinet holding two standard NIM bins with their power supplies. The sample changer and teletypewriter are commercially available models. However, the sample changer has some physical modifications in addition to the new electronics. A block diagram of the system is shown in Figure 2.

The system is operated through the teletypewriter keyboard and the paper tape reader, which are used to instruct the automatic data acquisition controller. ASCI II machine language is used. Paper tape programs are provided for background counting, Chi^2 test, and sample counting. Because the memory of the automatic data acquisition controller is divided into two memory banks with no access directly across the memory boundary, a separate tape containing necessary calculational subroutines is provided for loading this information into the upper memory.

For the purposes of this sample counting program, the particle energy spectrum was divided into four channels as follows.

1) 100 KeV to 2 MeV for gross beta activity

2) 2 MeV to 3 MeV - set aside as a test channel

3) 3 MeV to 8 MeV for gross alpha activity

4) 3.75 MeV to 5.20 MeV for plutonium activity

The sequence of operations starts with loading the subroutines into the upper memory through the tape reader. Each of the other tapes is divided into two parts. The first part prints the heading onto the paper through the teletype. The second part is loaded into the memory of the automatic data acquisition controller. The first of these tapes used is the background tape. This operates the counter for a ten-minute background count. The average one-minute count is printed out for each of the four data channels and the average two-minute count for each channel is stored in the controller memory. The next tape used is the Chi^2 test. This operates the counter for a group of 20 one-minute counts and analyzes the data for statistical variations. The statistical analysis data and the counting efficiency for each channel are printed out. The current counting efficiency is also stored in the controller

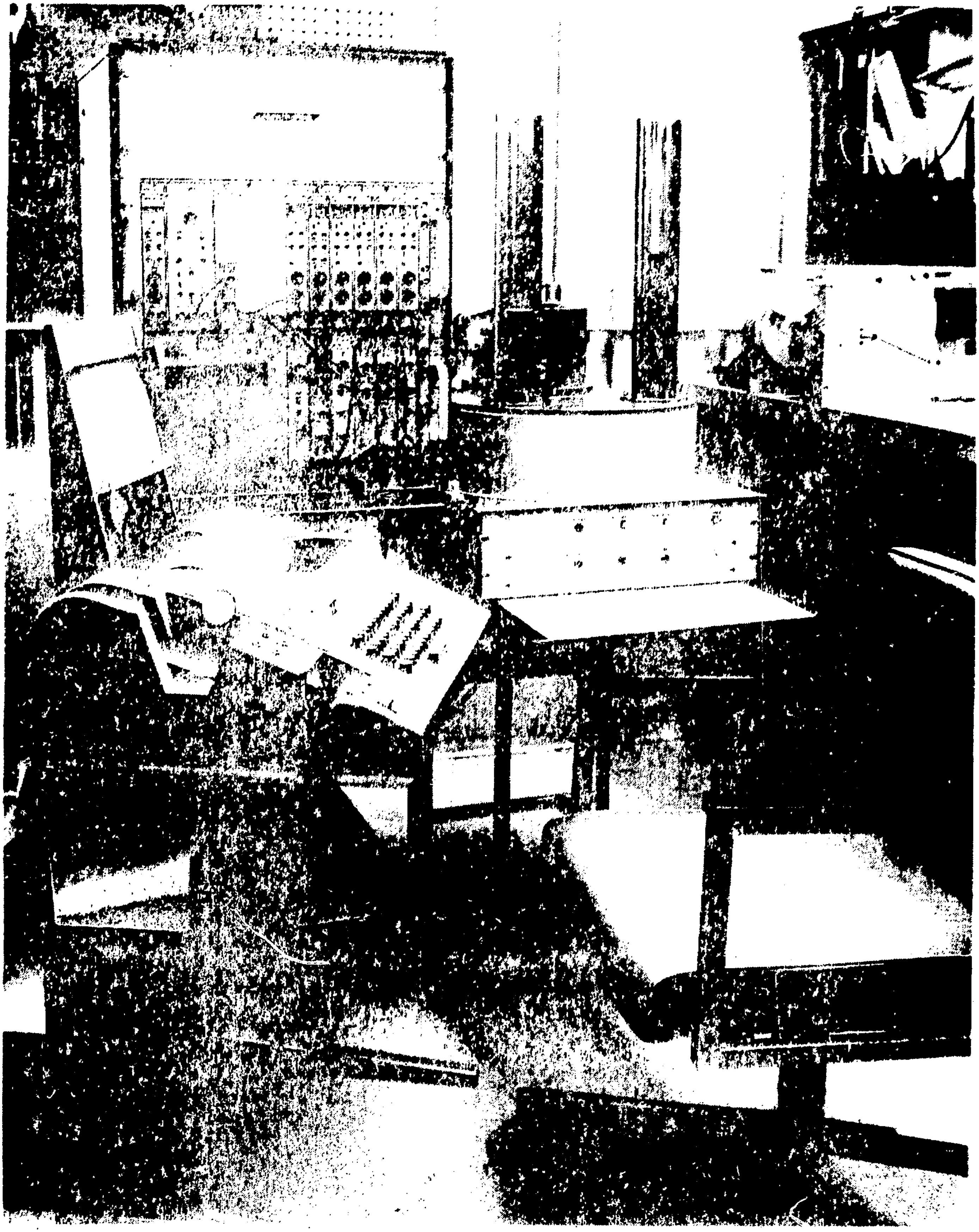

Figure 1. HEDL 200 Area Air Filter Counting System.

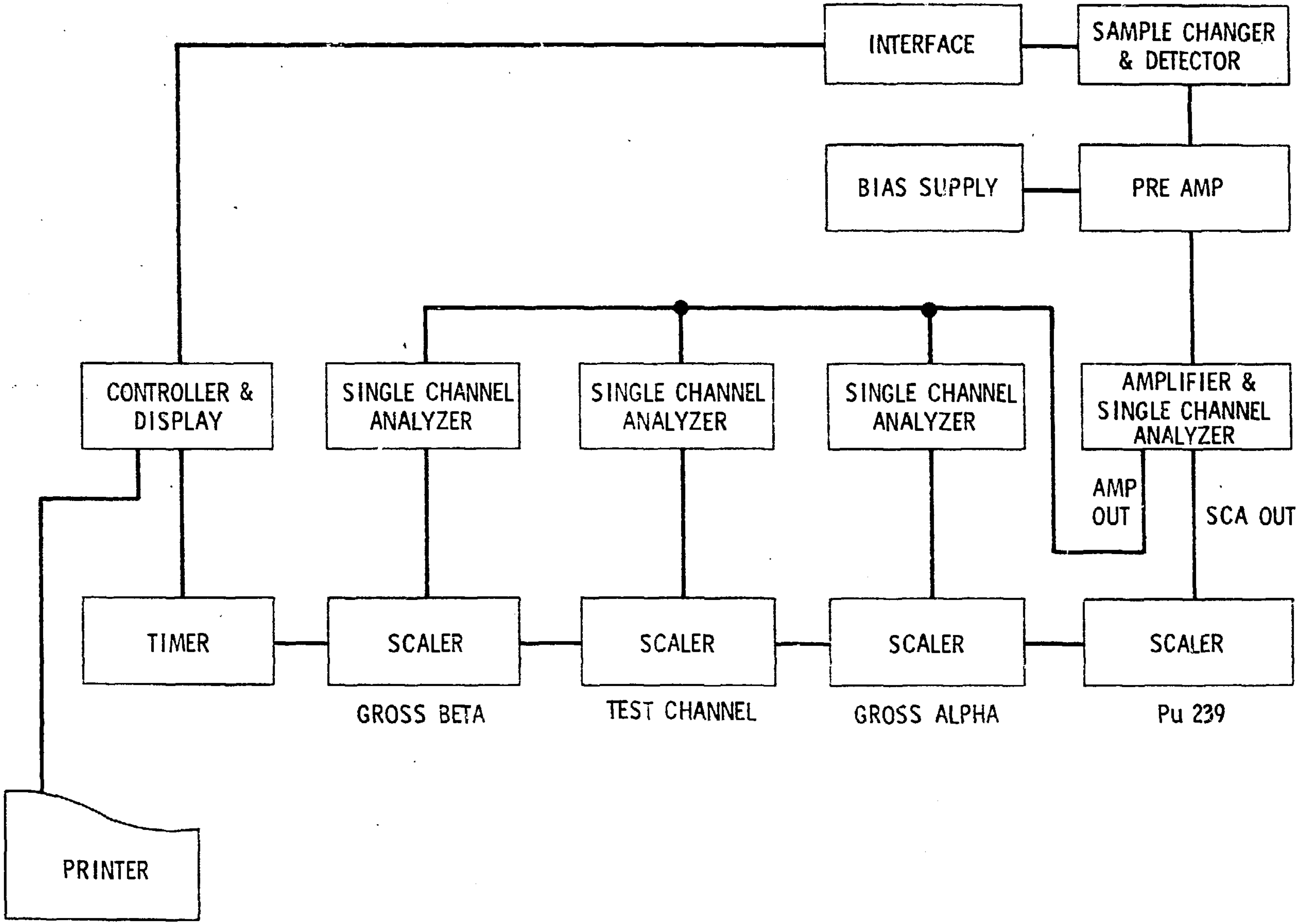

FIGURE 2. HEDL 300 AREA AIR FILTER COUNTING SYSTEM BLOCK DIAGRAM

memory. The final tape used is the sample counting tape. This operates the counter for a two-minute count interval for each sample, changes the samples, and prints out the net count (less background) and the calculated results in μCi per cc for beta, alpha, and plutonium. When the samples are all counted, the automatic sample changer restacks them in their original counting order.

The counter is lightly shielded, with a copper cylinder about 5/8" thick around the detector. However, it exhibits a low background in gamma fields below 1 mR/hr. Counting efficiencies are typically about 25% for beta, 20% for alpha, and 10% for plutonium alpha. These are all based on total source activity (4π geometry).

CONCLUDING REMARKS

From the beginning of its operation, this counting system has performed all of the desired functions. It is programmable, and once it is started, it performs the counting, calculation, and printing operations automatically. Based upon a year's experience, the system is reliable, stable, rugged and relatively trouble-free. Output data are consistent and easily scanned for high or unusual results. In those cases, sample recounting is simple. Instrument condition is monitored more closely because the Chi^2 test is performed daily, rather than at infrequent intervals as before. This also provides improved assurance of data reliability.

The only difficulties which have been encountered are damage to one detector, electrical line noise, and mechanical failure of the changer. However, all of these problems have been relatively easy to correct. Currently, the new counter is handling the entire counting workload with about three man-hours of work per day. System capacity is greatly increased, and a second 100-sample counting run can be carried out on the same day with an additional workload of only about one man-hour.

Question (unidentified):

Is there anything done to reduce the problems of contamination? What are the beta and alpha backgrounds?

Answer (W.P. Howell):

There is no special provision to protect the detector from contamination. We haven't had a serious problem with that - it's been cleaned once, that I know of, by washing with alcohol. Typical beta backgrounds are on the order of about 20 counts/minute, and typical alpha backgrounds in the gross alpha channel are about 8 counts a minute. Plutonium channel background is about 2 counts a minute.

AN ON-LINE MONITOR FOR THE DETECTION OF AEROSOLS CONTAINING URANIUM AND THE TRANSURANICS*

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ABSTRACT

An extremely sensitive on-line monitor for alpha-emitting aerosols containing uranium and the transuranics has been developed. If the only alpha activity present were that associated with the decay of these aerosols, then monitoring for significant concentrations would be straightforward. There is, however, a naturally occurring source of alpha activity from the decay of radon and thoron daughter products. These background levels may be several orders of magnitude greater than the maximum permissible airborne concentration, MPC_a, of the aerosol that is being monitored.

The monitor that is described here incorporates discrimination against background activity that is based on two factors. Firstly, the background activity is associated with small-sized dust particles, while the aerosol to be monitored has most of its activity associated with large-sized particles. Secondly, the alpha energies from the two sources are different. The monitor utilizes virtual impactors to concentrate the large-sized particles into a flow that is a small fraction of the intake flow. The flow containing the large particles is then passed through a filter to collect the particles. The filter is viewed by a solid-state detector for alpha spectroscopy. The monitor is capable of detecting less than one MPC_a of soluble aerosols containing plutonium, 0.002 pCi/l, in less than one-half hour.

INTRODUCTION

Throughout present and proposed nuclear fuel cycle activities and in research activities hazardous levels of alpha-emitting aerosols containing uranium and the transuranics may be encountered. The aerosols include those containing: the transuranics, with emphasis on the alpha-emitting plutonium isotopes; natural and slightly-enriched uranium, which are encountered in the light water reactor fuel supply; and ^{233}U and ^{232}U and its daughter products, which are encountered in the thorium-^{233}U fuel cycle. In order to insure that the operational areas of facilities handling these materials are safe and to prevent the release of these materials to the environment, an extremely sensitive on-line monitor is required.

Research performed under the auspices of the U. S. Department of Energy.

If the only alpha activity present were that associated with the decay of these aerosols, then monitoring for significant concentrations would be straightforward. There is, however, a naturally occurring source of alpha activity -- the alpha activity from the decay of radon and thoron daughter products. The pertinent components of the radon decay chain are shown below.

$$\begin{array}{llllll} {}^{222}\text{Rn} & \xrightarrow{\alpha} & {}^{218}\text{Po} & \xrightarrow{\alpha} & {}^{214}\text{Pb} & \xrightarrow{\beta^-} \\ 3.82\text{ d} & 5.49\text{ MeV} & 3.05\text{ m} & 6.00\text{ MeV} & 26.8\text{ m} & 0.67\text{ MeV} \\ (\text{Radon}) & & (\text{RaA}) & & (\text{RaB}) & \end{array}$$

$$\begin{array}{lllll} {}^{214}\text{Bi} & \xrightarrow{\beta^-} & {}^{214}\text{Po} & \xrightarrow{\alpha} & {}^{210}\text{Pb} \\ 19.8\text{ m} & 3.26\text{ MeV} & 164\ \mu\text{s} & 7.69\text{ MeV} & 22.3\text{ y} \\ (\text{RaC}) & & (\text{RaC}') & & (\text{RaD}) \end{array}$$

Typically, the radon concentrations are in the range of 0.1 to 1.0 pCi/l; the thoron activity is usually 10 to 100 times less. The 40-hr occupational MPC_a for soluble aerosols containing the alpha-emitting plutonium isotopes is 0.002 pCi/l, which is several orders of magnitude less than the background activity.[1]

At the present time there are two types of on-line monitors: 1) the filter solid-state detector monitor[2,3] and 2) the annular impactor solid-state detector monitor.[4] The first monitoring system continuously draws air through a filter, which is viewed by a solid-state detector. The most sensitive system of this design samples air at approximately 100 l/min and utilizes an automatic background subtraction scheme to correct for background events that enter the isotope window. Such background subtraction techniques are mandatory for this design. If a window is set for alpha particles from the plutonium isotopes (a typical window would be from 4.8 to 5.5 MeV), then approximately 25% of the 6.00 MeV alpha particles from the decay of RaA fall within this window. The main drawbacks of this system are: all the background activity is collected on the filter, which means that large corrections must be made for the background; and all the flow is passed through the filter, which limits the flow rate if reasonable pressure drops are to be maintained.

The annular impactor solid-state detector monitoring system takes advantage of the fact that the background activity is predominately attached to small-sized dust particles, while the aerosol to be monitored has most of its activity associated with large-sized particles. An annular impactor is used to selectively impact the large-sized particles on the greased surface of a solid-state detector. The monitor samples at 283 l/min. The system has good sensitivity and excellent background rejection capability. The monitor has, however, a number of drawbacks: the dust buildup on the detector and subsequent worsening of the resolution after a fairly short time even in relatively clean environments, the operational inconvenience of having to coat the detector surface, the lack of any small-particle sensitivity, and uncertain changes in the collection efficiency as a function of dust buildup.

THE VIRTUAL IMPACTOR SOLID-STATE DETECTOR MONITOR

The on-line monitor that is described in this paper, the virtual impactor solid-state detector monitor, combines the best features of the two systems described above. Two stages of virtual impactors concentrate the large-sized particles into a flow that is a small fraction of the intake flow. The flow containing the large-sized particles is then passed through a filter to collect the particles. The filter is viewed by a solid-state detector for alpha spectroscopy.

Figure 1 shows a schematic view of a virtual impactor stage. Air is drawn through the inlet jet at a volumetric flow rate Q_0. Flow conditions are set such that the flow in the receiving tube, Q_1, is a fixed fraction of Q_0, usually about one-quarter. The receiving tube is usually separated by a jet diameter, D_1, from the outlet of the jet. To obtain minimum collection of the particles on the receiving tube walls, the receiving tube diameter, D_2, is about one-third larger than the jet diameter. A few representative streamlines for nominal flow conditions are indicated on the figure. Let us consider the particle trajectories of particles that are located near the streamlines that do not enter the tube. Small particles are able to follow the streamlines and do not enter the receiving tube; large particles are not able to follow the streamlines in regions of rapidly changing curvature and thus enter the receiving tube.

Although virtual impactors have been recently incorporated into a number of aerosol monitors, i.e., the EPA's dichotomous virtual impactors for monitoring of airborne particulate matter,[5] the aerosol collection characteristics as a function of arbitrary operating parameters were not known well enough to allow the design of the on-line monitor described here. Therefore, a detailed investigation, including both experimental and analytical studies, was undertaken on the aerosol separation characteristics of virtual impactors. This work will be described elsewhere; only some representative experimental results are shown here. Figure 2 shows the collection efficiency for a given geometry and flow conditions as a function of aerodynamic particle size. The test aerosol was a liquid monodisperse DOP aerosol, which was generated with a vibrating orifice aerosol generator. The curve labeled "minor flow" shows the percent of the incoming particles that exit from the receiving tube, while the one that is labeled "receiving tube" shows the percent that are collected on the interior walls of the receiving tube near the tip. For zero particle size, the percentage of the particles entering the receiving tube is simply the same as the percentage of the flow that enters the tube. In the limit of large particle size, all the particles exit from the receiving tube. In the transition region there is some collection on the walls of the receiving tube. Tests indicate that when the aerosol is a solid rather than a liquid, this collection is considerably reduced, and for the conditions shown in Fig. 2 the maximum is less than a few percent. The explanation is straightforward. A liquid aerosol on striking a wall will stick, while a solid aerosol has a high probability of rebounding from the wall.

The results of the single-stage virtual impactor study were used in designing the multistage monitor that is shown schematically in Fig. 3. The monitor uses two stages of virtual impactors in order to reduce the percentage

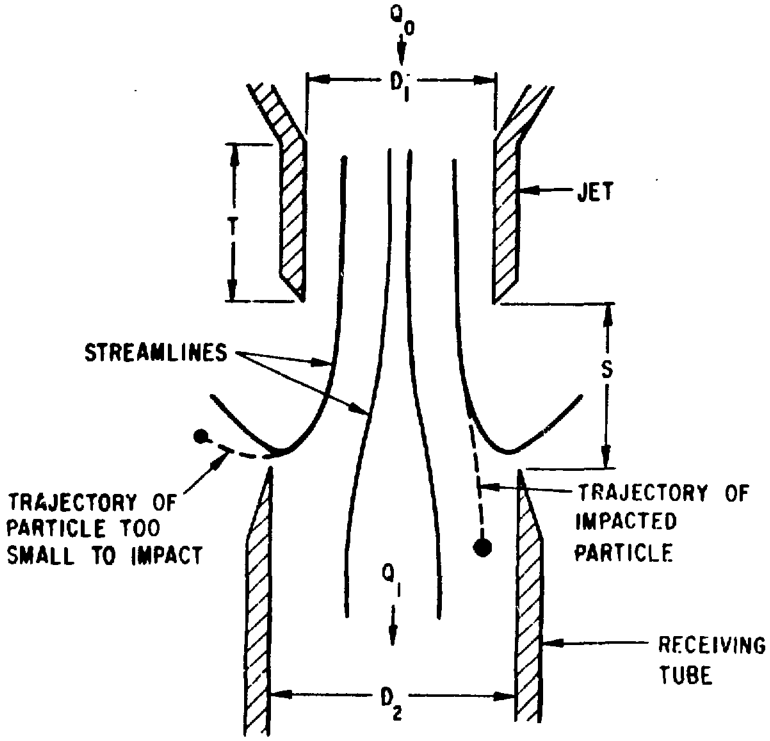

Fig. 1. Schematic of a virtual impactor with the critical parameters indicated.

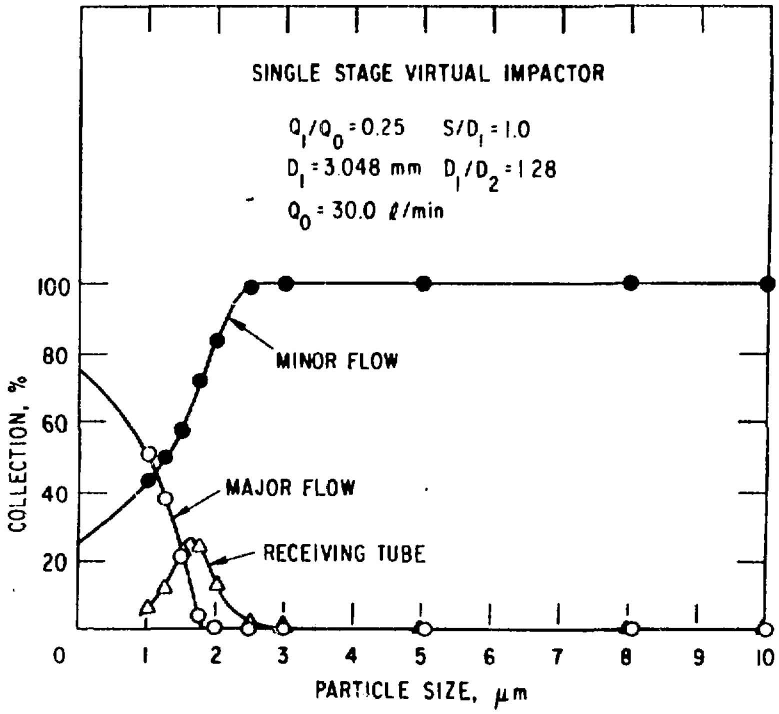

Fig. 2. Collection characteristics of a single-stage virtual impactor as a function of aerodynamic particle size.

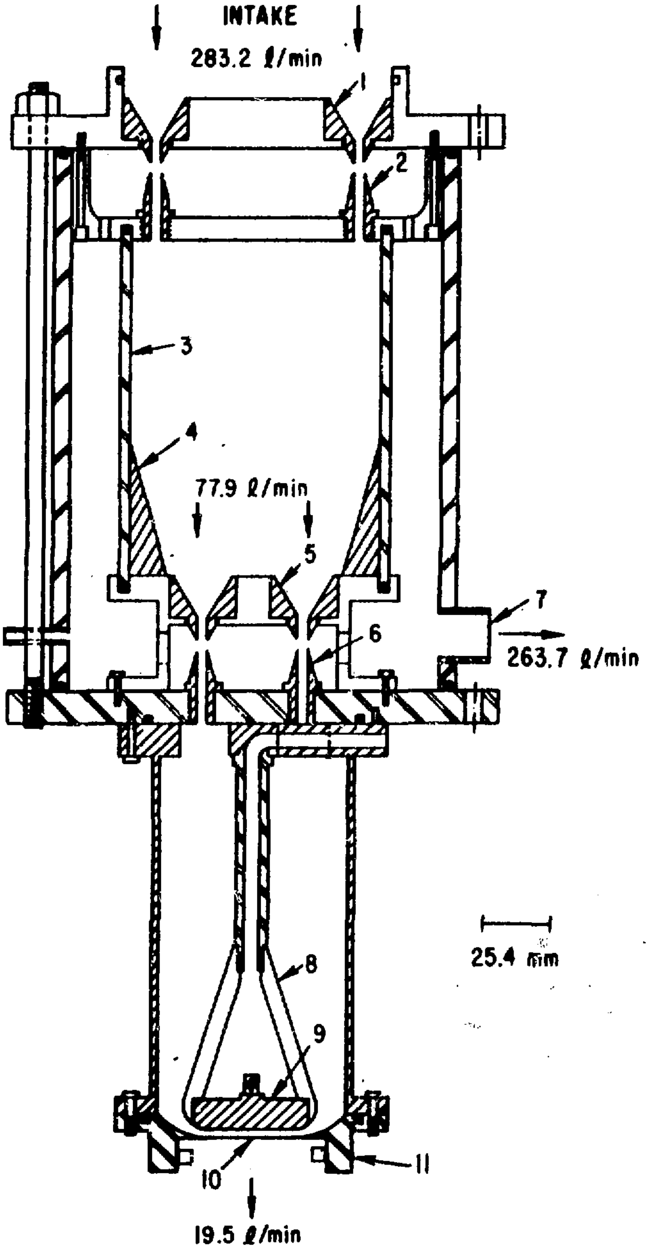

Fig. 3. Detailed drawing of the virtual impactor solid-state detector on-line monitoring system. Some of the key components are: 1) first-stage jets, 2) first-stage receiving tubes, 3) transition region cylinder, 4) transition region cone, 5) second-stage jets, 6) second-stage receiving tubes, 7) small-particle outlet, 8) detector holder, 9) detector, 10) filter and 11) filter holder.

of fine-particle collection and to concentrate the large-sized particles into a small flow. In order to provide good sensitivity, it is necessary to sample at rather high flow rates. For the monitor shown in the figure the sampling rate is 283 l/min. The first stage consists of nine virtual impactors with $Q_1/Q_0 = 0.275$ and $D_1 = 3.05$ mm. The second stage consists of three virtual impactors with $Q_1/Q_0 = 0.25$ and $D_1 = 2.78$ mm. It is necessary to use multiple virtual impactors for each stage in order to keep the Reynolds number in the jet flow low enough to avoid turbulent flow. For the specified flow conditions, approximately 7% of the initial flow is passed through the filter. Figure 4 shows the collection efficiency for the monitor measured with a DOP aerosol. The cut point, i.e., the aerodynamic diameter at which there is 50% collection on the filter, is 1.5 μm. The aerodynamic diameter is referenced to a unit density sphere. For a plutonium oxide aerosol, an aerodynamic diameter of 1.5 μm corresponds to a geometric diameter of approximately 0.4 μm. For very large particle sizes there is less than 100% collection efficiency on the filter because of intrastage losses and collection on the second-stage receiving tubes. Ways to eliminate these losses are currently being studied.

The major flow is monitored by measuring the pressure drop across the inlet and small-particle outlet. The flow through the filter is measured with a flowmeter and the pressure drop across the filter is used to determine the amount of dust buildup on the filter. Because the large particles in the aerosol are concentrated within a low flow, the sample may be collected on a small diameter filter without having an excessive pressure drop. The pressure drop is 80 mm Hg for a 47 mm Millipore AA filter. For a solid-state detector with 800 mm^2 area, the efficiency for alpha particle detection is 30%.

The attractive features of this monitoring system are: the deposit is uniformly spread out over the filter which reduces the mass buildup per unit area; the most recently collected particles are closest to the detector so that the alpha particles do not have to pass through the old dust deposit; the monitor has a fixed reduced collection efficiency for small particles consistent with the requirement of good background reduction, but not so small that an aerosol composed of fine particles would not be detected; the large particles are concentrated into a small flow so that a small diameter filter can be used with the resulting high efficiency for alpha particle detection; the only normal maintenance is filter changes; and collection and monitoring occur simultaneously.

SYSTEM OPERATION

Operational tests have been carried out for the monitor using modular electronic components and a small multichannel analyzer. A dedicated system using a microprocessor is under development and is briefly described in the next section. Figure 5 shows the pulse-height spectrum for a collection time of one hour with the monitor in a reactor cell where the radon daughter product activity was about 1.5 pCi/l. The RaA and RaC' peaks are clearly seen, as well as a smaller ThC' peak. In the radon decay chain the activity from RaA usually is the greatest source of interference if one is looking for alpha particles from uranium and the transuranics. In the thoron decay chain the interference is from ThC, which emits alpha particles of 6.05 and 6.09 MeV.

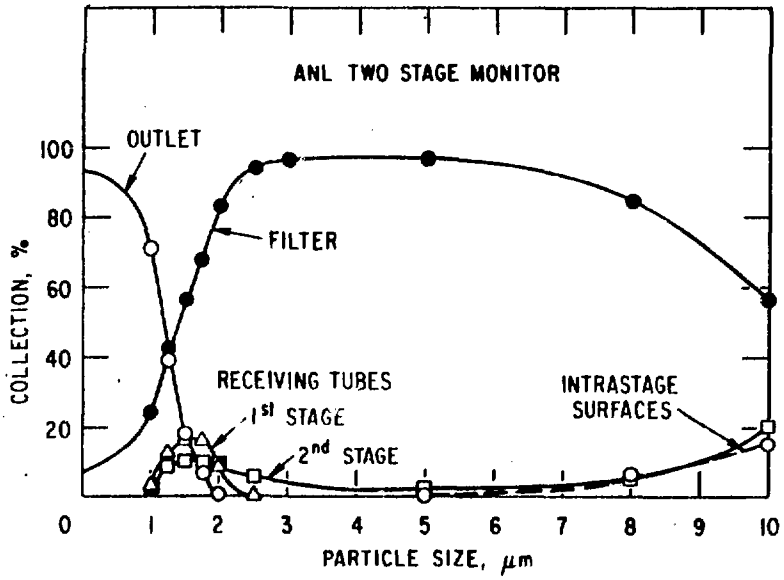

Fig. 4. Collection characteristics of the virtual impactor solid-state detector on-line monitoring system as a function of aerodynamic particle size.

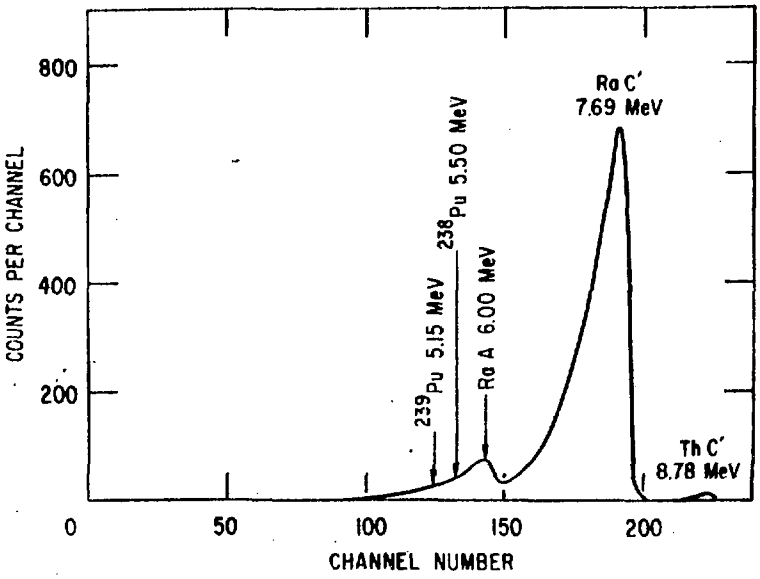

Fig. 5. Alpha spectrum of the radon and thoron daughter products collected on the filter of the virtual impactor solid-state detector on-line monitoring system.

On the figure are also shown the locations at which the peaks would occur if ^{238}Pu and ^{239}Pu were present. In the tests that have been conducted to date, only 7% of the background activity has been collected on the filter, which is consistent with the assumption that the background is predominately attached to fine particles.

The sensitivity of the monitor is a function of: the intake flow rate, the collection efficiency, the resolution of the spectroscopy system, the energy settings of the window, the nature of the background and the nature of the aerosol which is to be monitored. For the calculations of the sensitivity which appear below assume that all the background activity is attached to very small particles. Furthermore, assume that all the alpha activity from the aerosol to be monitored is associated with particles larger than the cut point, and that all these particles are collected on the filter. Studies on plutonium aerosols have indicated that in almost all cases the majority of the activity is associated with particles having an aerodynamic diameter greater than 1.5 μm.[6] Assume a step function release of the aerosol which reaches the monitor at the beginning of the sampling period. An energy window is set to detect alpha particles of a given energy. The background activity on the filter is assumed to have reached equilibrium. The number of background counts in the energy window for a given sampling interval is

$$C_b = A_{RaA} V \varepsilon_f \varepsilon_d f_{wb} t \qquad (1)$$

where A_{RaA} is the equilibrium activity on the filter associated with RaA, V is the volumetric sampling rate, ε_f is the fine-particle collection efficiency, ε_d is the detector efficiency for alpha particles, f_{wb} is the fraction of the RaA alpha particles with energies within the window and t is the sampling time interval. The number of counts in the window from the aerosol that is being monitored for a given sampling interval is

$$C = \lambda n V \varepsilon_d f_w t^2/2 \qquad (2)$$

where λ is the decay constant of the alpha emitter, n is the concentration in air, f_w is the fraction of the alpha particles with energies within the window and the other terms are the same as defined above. Note that the counts are proportional to the time squared. Table I presents the results for four different types of transuranic releases for a RaA background activity of one pCi/l for sampling times of 15, 30 and 60 minutes. For each type of release the concentration is set at one 40-hr occupational MPC_a. The two columns that are labeled "Pu/Background" and "Pu/3σ" indicate the sensitivity. The first of these simply shows the ratio of plutonium counts to background counts. Tests that have been conducted to date indicate that it is possible to determine almost within statistics the number of background counts falling within the transuranic window by multiplying the counts in a RaA window by a constant. This technique may be used as long as the curium isotopes which emit alpha particles of about 6.00 MeV are not present. The column labeled "Pu/3σ" shows the ratio of the plutonium counts to three standard deviations in the inferred background counts in the transuranic window. An alarm could be activated with very few false alarms when this ratio exceeds one and when certain statistical conditions are met on the number of counts in the transuranic window and background window. The first type of release assumes that the only important

TABLE I. Sensitivity of the Virtual Impactor Solid-State Detector On-Line Monitoring System for One MPC$_a$ Releases of Various Alpha-Emitting Aerosols

Isotope Concentration pCi/l	Window MeV	Collection Time Min.	Pu Count	Background 1 pCi/l of RaA Background Count	 Pu / Background	 3σ Background	 Pu / 3σ
^{239}Pu	4.910 - 5.230	15	56	76	0.7	26	2.2
0.002		30	225	153	1.5	37	6.1
		60	900	306	2.9	52	17.3
^{238}Pu	5.250 - 5.580	15	56	132	0.4	34	1.6
0.002		30	225	265	0.8	49	4.6
		60	900	529	1.7	69	13.0
Pu + Am	4.910 - 5.580	15	56	194	0.3	42	1.3
0.002		30	225	388	0.6	59	3.8
		60	900	776	1.2	84	10.7
LWR[a]	4.910 - 5.580	15	139	194	0.7		
0.009		30	557	388	1.4		
		60	2230	776	2.9		

[a] One MPC$_a$ for LWR discharge has the following concentrations in pCi/l: Pu + Am, 0.0016; ^{242}Cm, 0.0063; and ^{244}Cm, 0.0011.

transuranic is ^{239}Pu, which emits alpha particles of 5.11 and 5.16 MeV. The energy window has been set to detect alpha particles from this isotope. This situation occurs at weapons facilities and at research facilities which are utilizing plutonium that is almost entirely composed of ^{239}Pu. The second release assumes that the only important transuranic is ^{238}Pu, which emits alpha particles of 5.46 and 5.50 MeV. This situation arises at facilities fabricating isotope sources. The third release assumes that the only important transuranics are plutonium and americium isotopes. The americium isotopes emit alpha particles with energies between those from ^{239}Pu and ^{238}Pu. This composition aerosol may be present in a reprocessing plant after the curium has been separated into the high-level waste stream and in fuel fabrication facilities. The fourth release is that associated with LWR discharge fuel and contains curium isotopes. For this aerosol one MPC_a corresponds to 0.009 pCi/l. In this case one cannot infer the background counts in the transuranic window because of interference from the curium isotopes. In all cases, especially those in which background subtraction can be used, the sensitivity is quite good. Even better ratios are obtained for the other alpha-emitting aerosols because the MPC_a's are greater.

DEDICATED ELECTRONICS SYSTEM

A dedicated electronics system for the virtual impactor solid-state detector on-line monitoring system is under development. A block diagram of the system is shown in Fig. 6. The system is controlled by a microprocessor. The system is designed to accommodate four monitoring heads. The output from the solid-state detector goes to a charge-sensitive preamplifier and then to an amplifier. The amplifier output goes to three single channel analyzers, SCA's. Two of the SCA's are set to detect alpha particles from the aerosol to be monitored. One SCA might be set for the plutonium isotopes, while the other might be set for the uranium isotopes. The third SCA is set to detect alpha particles from RaA. The output from the SCA's go to 16-bit preset counters. After a preset number of counts have been reached in a given counter, the contents of the three counters for that channel are read into the central processing unit, CPU. The CPU performs background subtraction and determines whether an alarm should be activated. The CPU also monitors general system operation. Interrupts are generated by low intake flow, an out-of-range pressure drop across the filter or an AC power failure; the appropriate alarms are activated. Each channel has three multipurpose displays. During setup and calibration the displays are used to read out the preset values of the counters and to display the contents of the counters. During normal operation the displays are used to read out the contents of the counters and to indicate for the prevailing background conditions, the MPC_a level that will generate an alarm. During alarm conditions the displays are used to read out the contents of the counters and to indicate the magnitude of the release in terms of MPC_a for an assumed set function release. The system also contains a real time clock. An RS-232 data link is provided to communicate with a printer or remote terminal.

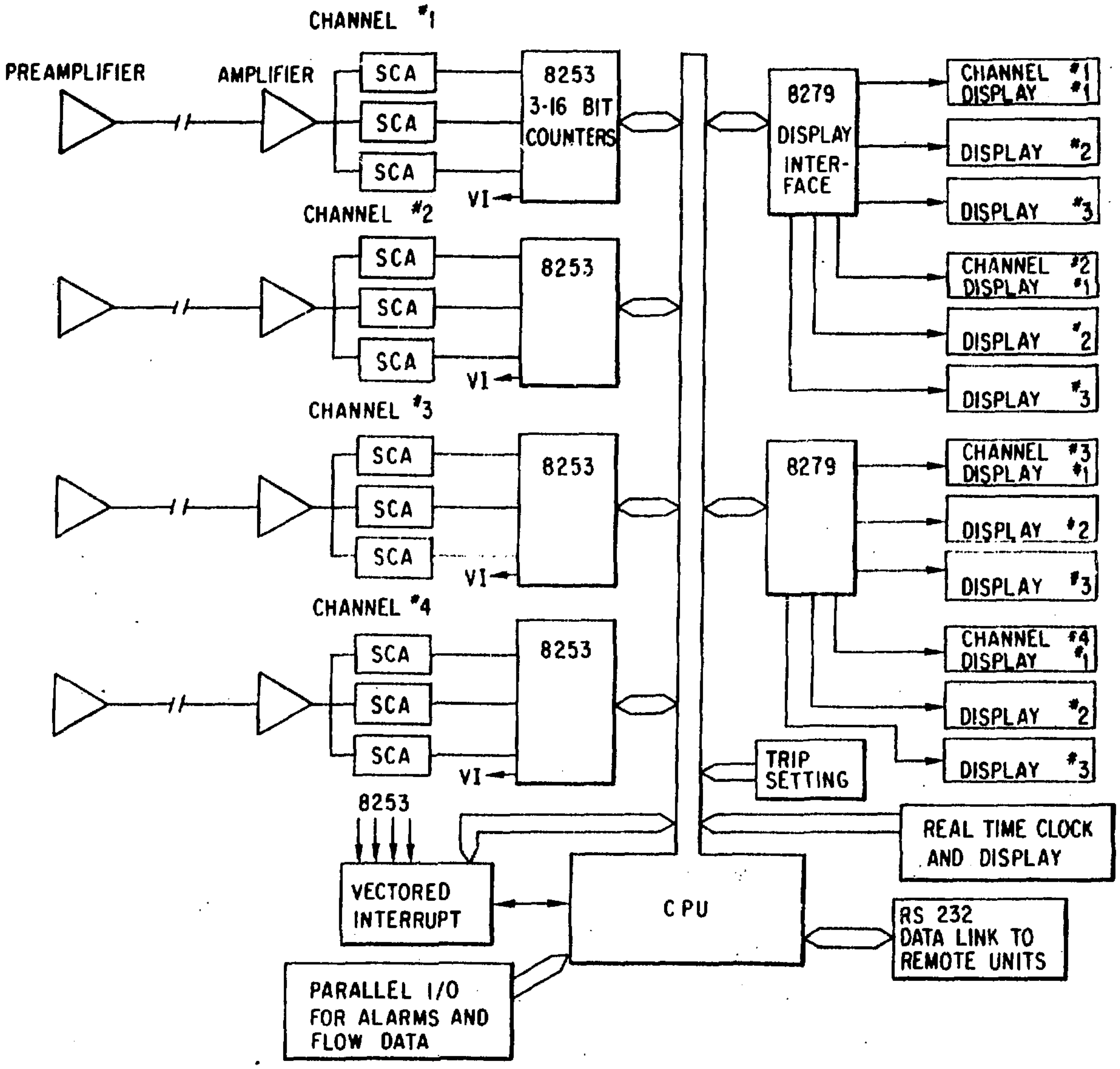

Fig. 6. Block diagram of the dedicated electronics system for the virtual impactor solid-state detector on-line monitoring system.

CONCLUSION

The virtual impactor solid-state detector on-line monitoring system for aerosols containing uranium and the transuranics combines simplicity with high reliability and sensitivity. It is capable of readily detecting one MPC_a of plutonium in less than one-half hour. The system may be applied to both operational area and field monitoring.

ACKNOWLEDGMENT

The author is indebted to Mr. Donald J. Keefe and Mr. William McDowell of the Electronics Division at Argonne National Laboratory for designing the dedicated electronics system for the monitor and to Ms. Vivian Kafalenos of Macalester College for aiding in the measurement of the collection efficiency of virtual impactors.

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Question (unidentified):

I gather you have done some studies to determine typically how much the plutonium or uranium aerosols are lost because of smaller particle size. Do you have any numbers for that?

Answer (Thomas J. Yule):

It would depend on the process in this type of thing. There is a paper by Edinger in Health Physics in which a number of plutonium oxide and carbide aerosols are looked at. If they're not chemically generated, usually in excess of 70% of the activity is associated with aerodynamic particle sizes greater than 1.5 microns. I've done some tests in which I've generated uranium oxide aerosols just taking uranium and creating the uranium oxide, putting it in a nebulizer, putting it on a filter using track recorders getting very nice looking log normal distributions of the particle size, and in that particular case it looks like it's still better than 70% of the activity would be above this cutoff. You could make the cutoff somewhat lower if you want, you probably wouldn't want to go less than maybe a half a micron, or .3 microns, or something like that, because then you'd need very many nozzles and the system gets very complicated. It's sort of a tradeoff between how much activity you're going to use, and how much complication you want to build on the system. We've used them at Argonne for about 6 or 7 years, a system similar to this, the annular impactor solid state detector system that I'm talking about. And it's been quite effective in picking up the releases when we've had releases. This I feel combines the best features of that system with the simple solid state detector filter system.

MEASUREMENT OF AIRBORNE CONCENTRATIONS OF RADON-220 DAUGHTER PRODUCTS BY ALPHA PARTICLE SPECTROMETRY*

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ABSTRACT

The decay of naturally occurring uranium-238 and thorium-232 produces radon-222 and radon-220 isotopes which can escape into the atmosphere. If these radon gases become concentrated in air, their daughter products may present an inhalation hazard to man. The airborne concentrations of radon-222 can usually be measured very accurately in the presence of normal airborne concentrations of radon-220 and its daughters.

In contrast, the measurements of the airborne concentrations of radon-220 daughters are usually complicated by the presence of radon-222 and its daughters even at normally occurring airborne concentrations. The complications involved in these measurements can be overcome in most situations by using an alpha particle spectrometer to distinguish the activity of radon-222 daughters from that due to radon-220 daughters collected on a filter.

A practical spectrometer for field measurements of alpha particle activity on a filter is discussed. Also discussed are methods that can be used to make measurements of either radon-220 or radon-222 daughter concentrations in air, or simultaneous measurements of the airborne daughter concentrations of both radon isotopes.

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*Research sponsored by the Department of Energy under contract with Union Carbide Corporation.

INTRODUCTION

Over a period of about five years we have made extensive measurements of radon-222 daughter concentrations in air using a modification of an alpha particle spectrometry technique suggested by Martz et al. (1). In our radon-222 daughter measurements, the concentrations of RaA (Po-218), RaB (Pb-214), and RaC (Bi-214) are calculated from one count of the 6.00 MeV alpha particle activity of RaA and two counts of the 7.69 MeV alpha particle activity of RaC' (Po-214) collected on a filter. Various aspects of these measurements and some results have been discussed elsewhere (2-5).

Recently we have used a modification of an alpha particle spectrometry method suggested by Duggan (6) to measure the radon-220 daughter concentrations of ThB (Pb-212) and ThC (Bi-212) in air as well. It is impossible to resolve the activity on the filter due to the 6.05 and 6.09 MeV alpha particles from ThC and the 6.00 MeV alpha particles from RaA, but the activity can be partitioned between RaA and ThC using the 8.78 MeV alpha particle activity from ThC' (Po-212). In our radon-220 daughter measurements, the concentrations of ThB and ThC in air are calculated from two counts of the 8.78 MeV alpha particle activity of ThC' collected on a filter.

METHOD OF MEASUREMENTS

Our alpha particle spectrometer shown in Fig. 1 utilizes a modified gas flow proportional counter to house a 450 mm^2 silicon diode (3). Normally this type of detector is used in a vacuum, rather than air, to improve the resolution of the measurements. The need for vacuum equipment in our measurements is eliminated by flowing helium through the chamber at atmospheric pressure. This provides considerable gain in resolution compared to counts of filter activity made in air and eliminates filter handling complications encountered when using a vacuum (2). It is not necessary, for example, to cycle the bias voltage on the diode because there are no changes in pressure. The resolution of our spectrometer for an air sample collected at a flow velocity of about 50 cm per sec with a membrane (Metricel, Gelman GN-6) and glass-fiber (Acropor, Gelman An-450) filter having a median pore size of 0.45 μ are compared with a more porous filter (Whatman grade 4) in Fig. 2. Either the membrane or the glass-fiber filter with a median pore size of 0.45 to 0.80 μ allows easy resolution of the 6.00 and 7.69 MeV alpha particles emitted by RaA and RaC', respectively. At flow velocities of up to 100 cm per sec, these filters are normally more than 99% efficient in collecting aerosols (7).

Figure 3 shows several counts of RaA, RaC', ThC, and ThC' alpha particle activity collected on a filter. Instead of count rates used by Martz et al. (1) and Duggan (6), we use counting intervals of 10 min or more to improve the accuracy and sensitivity of the measurements. Our counting intervals of time are always measured from the termination of air sample collection. Air sampling periods of 5 to 15 min are normally used in our measurements followed by a 2 to 12 min count of the 6.00, 7.69, and 8.78 MeV alpha particle activity from RaA, RaC', and ThC', respectively. Note that the 2 to 12 min count of the 6.00 MeV alpha particle activity from RaA in Fig. 3 contains a small contribution from the 6.05 and 6.09 MeV alpha particles from ThC. Alpha particles having these energies are emitted in about 36% of the ThC decays, while 8.78 MeV alpha particles are emitted in the remaining decays. Thus, the 8.78 MeV alpha particle count from 2 to 12 min can be multiplied by 36/64 or 0.56 to obtain the ThC contribution to the 2 to 12 min count of RaA activity on the filter.

A second RaC' count from 15 to 30 min plus the first 2 to 12 min counts of the RaA and the RaC' alpha particle activity on the filter and a computer program written in BASIC are used to obtain the airborne concentrations of the radon-222 daughters. This program, designated as RPCON4, will calculate the concentrations of RaA, RaB, and RaC in air and will estimate the accuracy in these calculated concentrations for variable air sampling rate, air sampling time, counting time, and detection efficiency of the alpha particle spectrometer used to measure the activity on the filter (4). A modification of this program, designated as THOR2, is used to obtain the airborne ThB and ThC daughter concentration of radon-220 from the first 2 to 12 min count of the 8.78 MeV alpha particle activity of ThC' plus a second ThC' count of 10 min or more starting at least 200 min after the termination of the air sample collection. The concentration of ThB daughters in air is determined primarily from the second count of ThC' activity on the filter, while the concentration of ThC daughters in air is determined primarily from the first count of ThC' activity on the filter. These two counts are, however, very complex mathematical functions of the air sampling time, the counting times, and the activity ratio of ThC:ThB daughters in air.

METHOD OF ANALYSIS

The THOR2 computer program is used to calculate the concentrations of ThB and ThC in air and to estimate the accuracy in these calculated concentrations as follows. First, the two counts, c_1 and c_2, of the 8.78 MeV alpha particle activity are related to the number of ThB and ThC atoms, n_1 and n_2 respectively, on a filter at the end of a sampling period by the matrix equation

$$C = 0.64\ g\ L\ N \qquad (1)$$

or

$$\begin{vmatrix} c_1 \\ c_2 \end{vmatrix} = 0.64\ g \begin{vmatrix} l_{11} & l_{12} \\ l_{21} & l_{22} \end{vmatrix} \begin{vmatrix} n_1 \\ n_2 \end{vmatrix} , \qquad (2)$$

where capital letters are used in Eq. (1) to represent the matrices shown in Eq. (2). The elements l_{ij} of the matrix L are

$$l_{11} = \lambda_2(e^{-\lambda_1 t_{1s}} - e^{-\lambda_1 t_{1e}})/(\lambda_2 - \lambda_1) - \lambda_1(e^{-\lambda_2 t_{1s}} - e^{-\lambda_2 t_{1e}})/(\lambda_2 - \lambda_1) ,$$

$$l_{12} = e^{-\lambda_2 t_{1s}} - e^{-\lambda_2 t_{1e}} ,$$

$$l_{21} = \lambda_2(e^{-\lambda_1 t_{2s}} - e^{-\lambda_1 t_{2e}})/(\lambda_2 - \lambda_1) - \lambda_1(e^{-\lambda_2 t_{2s}} - e^{-\lambda_2 t_{2e}})/(\lambda_2 - \lambda_1) ,$$

$$l_{22} = e^{-\lambda_2 t_{2s}} - e^{-\lambda_2 t_{2e}} , \qquad (3)$$

where t_{1s} and t_{1e} are the starting and ending times of the first counting interval, t_{2s} and t_{2e} are the starting and ending times of the second counting interval, and λ_1 and λ_2 are the decay constants of ThB and ThC, respectively. In Eqs. (1) or (2), the factor 0.64 is the number of 8.78 MeV alpha particles emitted in the decay chain of either a ThB or ThC atom, and the factor g is the detection efficiency or geometry factor of the spectrometer used in measuring the 8.78 MeV alpha particle activity on a filter.

Next, a relationship between the number of ThB and ThC atoms on a filter at the end of a sampling period of time, t, and the concentrations of ThB and ThC atoms in air, q_1 and q_2, respectively, is established by the matrix equation

$$N = v\ K\ Q \tag{4}$$

or

$$\begin{vmatrix} n_1 \\ \\ n_2 \end{vmatrix} = v \begin{vmatrix} k_{11} & k_{12} \\ \\ k_{21} & k_{22} \end{vmatrix} \begin{vmatrix} q_1 \\ \\ q_2 \end{vmatrix} \quad , \tag{5}$$

where

$$\begin{aligned} k_{11} &= (1 - e^{-\lambda_1 t})/\lambda_1 \quad , \\ k_{12} &= 0 \quad , \\ k_{21} &= (1 - e^{-\lambda_2 t})/\lambda_2 + (e^{-\lambda_2 t} - e^{-\lambda_1 t})/(\lambda_2 - \lambda_1) \quad , \\ k_{22} &= (1 - e^{-\lambda_2 t})/\lambda_2 \quad , \end{aligned} \tag{6}$$

and the factor v is the air sampling rate used in the measurements. Equations (1) and (4) can be combined to obtain

$$C = 0.64\ gv\ L\ K\ Q \tag{7}$$

and inverted to yield

$$Q = \frac{1}{0.64\ gv}\ K^{-1}\ L^{-1}\ C = \frac{1}{0.64\ gv}\ M\ C \quad . \tag{8}$$

The computer program THOR2 calculates the matrices K and L, inverts these to obtain the matrices K^{-1} and L^{-1}, and multiplies these together to obtain the matrix M for the sampling and counting times of the measurements.

Finally, the elements m_{ij} of the matrix M are used to calculate the concentrations of ThB and ThC in air by Eq. (8) and to estimate the accuracy in these calculated concentrations by

$$\frac{s_{q_i}^{2}}{q_i^{2}} = \frac{\sum_j m_{ij}^{2} s_{c_j}^{2}}{\left(\sum_j m_{ij} c_j\right)^{2}} + \frac{s_v^{2}}{v^{2}} + \frac{s_g^{2}}{g^{2}} \quad , \qquad (9)$$

where s_{c_j} = the standard deviation of the two counts c_1 and c_2,

s_v = the standard deviation of the air sampling rate,

s_g = the standard deviation of the detection efficiency,

s_{q_i} = the standard deviation of the calculated concentrations q_1 and q_2 of ThB and ThC, respectively.

The THOR2 program, which is available on request, automatically uses standard deviations of $(c_1)^{1/2}$ and $(c_2)^{1/2}$ for the two counts of the 8.78 MeV alpha particle activity on the filter. Table 1 shows an example of the data input and output of this program.

DISCUSSION

An example of the accuracy and sensitivity of our radon-220 daughter measurements in an area with a ventilation rate of about two air changes per hour is given in Table 2. These data are for a spectrometer with a detection efficiency of 0.25 and an air sampling rate of 12 l/min. The air sampling time is 15 min followed by 2 to 12 min and 200 to 220 min counts of the 8.78 MeV alpha particle activity from ThC' on the filter. Estimates in Table 2 of the standard deviation in the calculated concentrations of ThB and ThC in air are based only on the uncertainties in the two counts of the ThC' activity on the filter. These uncertainties dominate the accuracy of the measurements at very low concentrations of ThB and ThC in air, while the accuracy of the measurements at very high concentrations is determined mainly by the uncertainties in the air sampling rate and detection efficiency of the spectrometer. The uncertainties in the air sampling rate and the detection efficiency are difficult to evaluate but with care are usually in the range of 2 to 5%. While our measurements provide information only on the ThB and ThC concentrations in air, they can be used in some instances to obtain additional information on the ventilation rate and thereby the airborne concentrations of ThA (Po-216) and radon-220 (6).

The methods discussed here have been used to make simultaneous measurements of the airborne concentrations of radon-222 and radon-220 daughters in areas contaminated with thorium-232 and in several buildings contaminated with raffinates from the processing of thorium ore. Recently, airborne concentrations of radon-219 daughters as high as, or higher than, the normally more abundant airborne concentrations of radon-220 and radon-222 have been measured in a building contaminated with raffinates from uranium ore processing. A method developed for the simultaneous measurement of the airborne daughter concentrations of all three radon isotopes will be discussed elsewhere (8).

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8. P. T. Perdue, R. W. Leggett, and F. F. Haywood, "A Technique for Evaluating Airborne Concentrations of Daughters of Radon Isotopes," Proceedings of Third International Symposium on the Natural Radiation Environment, Houston, Texas, April 23-28, 1978.

Table 1. Example of data input and output of THOR2 computer program

```
THOR2


SAMPLE DESCRIPTION ?  EXAMPLE

THE AIR SAMPLING TIME IN MIN IS ?  15

THE AIR SAMPLING RATE IN LITERS/MIN AND THE
     UNCERTAINTY IN THIS QUANTITY IN PERCENT
     ARE ?  12, 2

THE GEOMETRY FACTOR OF THE DETECTOR USED TO
     COUNT THE SAMPLE AND THE UNCERTAINTY IN
     THIS QUANTITY IN PERCENT ARE ?  0.25, 5

COUNTS OF 8.8 MEV ALPHA ACTIVITY FROM TH-C'
     IN THE TIME INTERVALS T1 AND T2:
     COUNTS IN T1= ?  262
     COUNTS IN T2= ?  1048

STARTING TIMES TS AND ENDING TIMES TE IN MIN OF
     THE INDICATED COUNTING TIME INTERVALS ARE:
     TS,TE OF T1= ?  2, 12
     TS,TE OF T2= ?  200, 220


     CONCENTRATIONS OF THORON DAUGHTER RADIONUCLIDES IN AIR


     THORON DAUGHTER       CONCENTRATION IN AIR
     RADIONUCLIDE    ATOMS/LITER   PCI/LITER         SIGMA

         TH-B        2.08E+03      1.02E+00    +-    6.3%
         TH-C        5.88E+01      3.03E-01    +-   11.6%
```

Table 2. Example of the accuracy and sensitivity of radon-220 measurements in an area with a ventilation rate of about two air changes per hour[a]

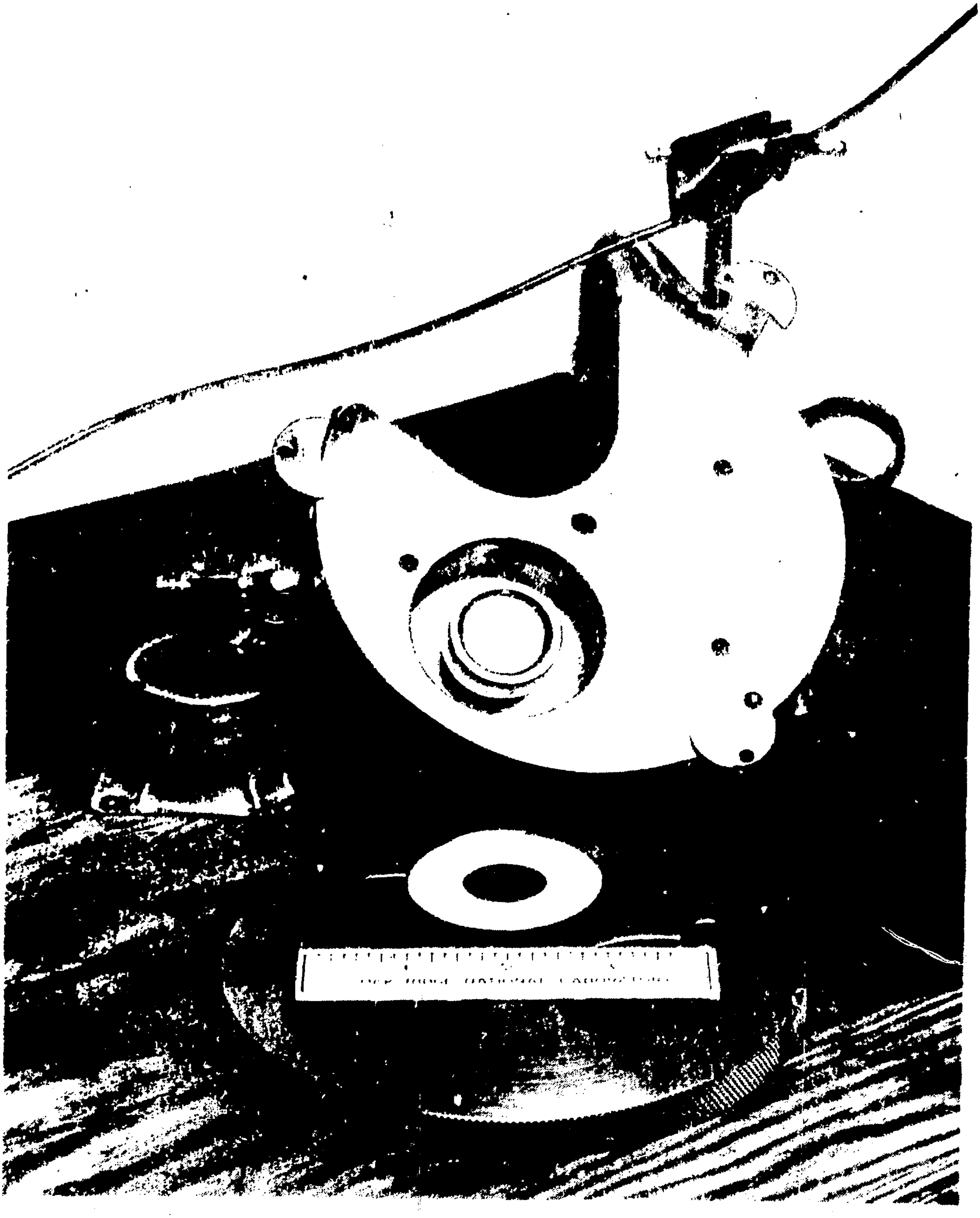
Radon-220 daughter concentrations in air in pCi l^{-1}		Standard deviation in radon-220 daughter measurements in percent[b]	
ThB	ThC	ThB	ThC
10.0	3.0	1.0	3.2
3.0	1.0	1.9	5.9
1.0	0.3	3.3	10.2
0.3	0.1	6.0	18.5
0.1	0.03	10.3	32.6
0.03	0.01	19.0	57.3

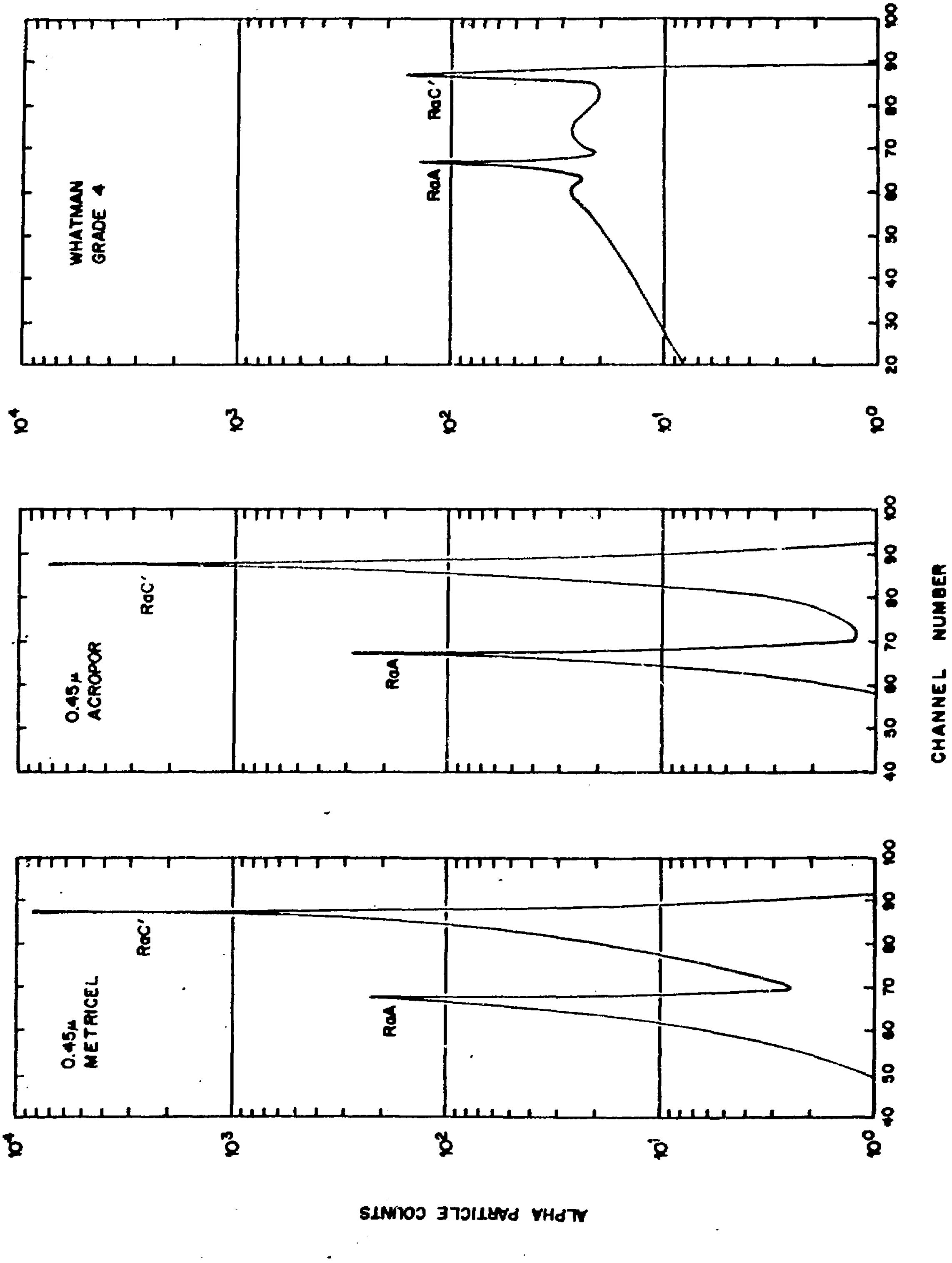
---

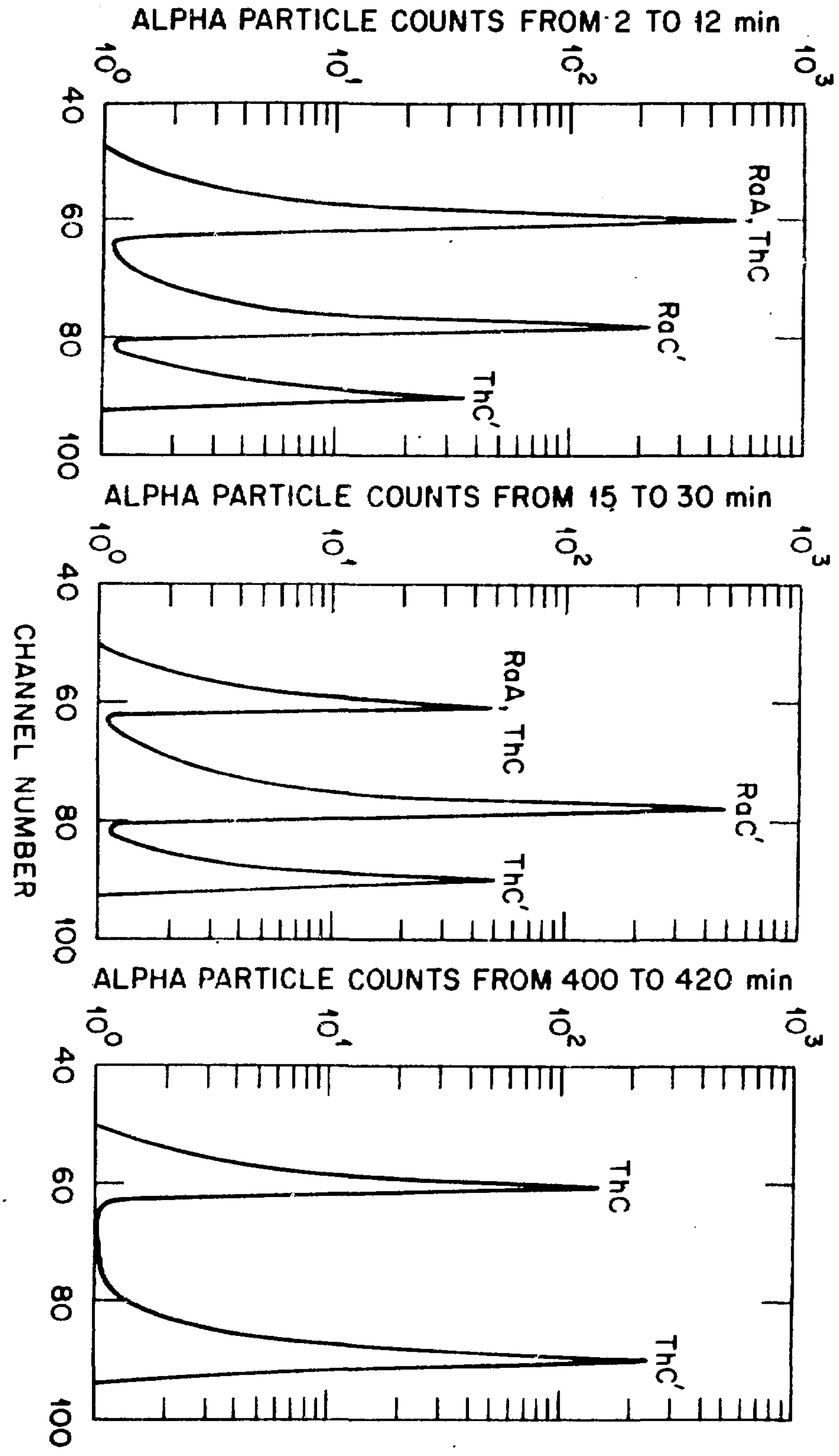
[a] These data are for a spectrometer with a detection efficiency of 0.25 and an air sampling rate of 12 liters per minute. The air sampling time is 15 min followed by a 2 to 12 min and a 200 to 220 min count of the 8.78 MeV alpha particle activity from ThC' on the filter.

[b] These standard deviations are based only on the standard deviations in the counts of alpha particle activity on the filter.

LIST OF FIGURES




0.45μ METRICEL
RaA
RaC'
0.45μ ACROPOR
RaA
RaC'
WHATMAN GRADE 4
RaA
RaC'
10⁰
10¹
10²
10³
10⁴
40
50
60
70
80
90
100
20
30
CHANNEL NUMBER
ALPHA PARTICLE COUNTS


ALPHA PARTICLE COUNTS FROM 2 TO 12 min

ALPHA PARTICLE COUNTS FROM 15 TO 30 min

ALPHA PARTICLE COUNTS FROM 400 TO 420 min

INSTRUMENTATION FOR THE URANIUM MINING INDUSTRY

by Robert A. Gray
Eberline Instrument Corporation

Introduction

Radiation detection instruments have been used for more than three decades within the uranium mining industry. They have played a prominent role in exploration and ore grade control and as health physics devices for mill and mine safety as well as environmental monitoring of areas surrounding mines and mills. The purpose of this paper is to discuss several new developments in instrumentation for exploration and mining processes, health physics devices for protection of workers, and environmental monitoring devices to comply with Federal and State Regulatory guides for the industry.

Micro R Meter, Eberline Model PRM-7

The Micro R Meter consists of a portable count rate meter with a one-inch dia. by one-inch thick NaI crystal as the detecting element. The detector and all associated electronics are contained within the instrument, assuring that the operator will require only one hand for monitoring. Four ranges of operation are provided with top scales of 25, 50, 500 and 5000 μR/hr. The unit also has its own built-in audio system. This instrument serves as an excellent portable exploration tool, a survey meter to monitor mill tailings, mills, equipment and areas surrounding mills and mines. A self-contained light which illuminates the meter allows the operator to monitor in dark areas such as concrete basements constructed from mill tailings. The extreme sensitivity of the Micro R Meter makes it an ideal portable multi-purpose gamma monitor. Its major disadvantage is energy dependence of the NaI crystal, which must be considered for highly accurate exposure rate measurements. (See Figure 1.)

Radon Daughter Working Level Counter, Eberline Model PS-2/SPA-1

The Model PS-2 Portable Scaler in combination with the Model SPA-1 detector assembly is used to evaluate the concentration of radon daughters in uranium mines. A small portable air pump is used to draw an air sample through a filter which is then measured for alpha activity. The instrumentation has been designed to comply with the standard approved method for measuring working level (WL) as specified by the American National Standards Institute (ANSI N7. la-1969) This instrumentation also can be used as an "instant" working level meter, with WL determined ten minutes after entering an area where radon daughters may be a problem. The procedure in this case is the Rolle method described in **Health Physics**, Volume 22, pages 233-238, March 1972. This method has the advantage of providing on-the-spot evaluation of WL with a maximum conversion error of 10-12% regardless of the disequilibrium conditions. A one-minute alpha count taken with the PS-2/SPA-1 eight minutes from the mid-time of sampling is directly proportional to WL. The Rolle method has the added advantage for being extremely sensitive when used with the PS-2 scaler and changes in external gamma radiation levels have no effect on precision or accuracy of measurement. Furthermore, the measurement of WL made by this method can be verified by counting the filter again thirty minutes later by the ANSI procedure. (See Figure 2.)

Radon Gas Detector, Eberline Model SAC-R5

The Model SAC-R5 is a 5-inch dia. scintillation detector in a light tight enclosure for the detection of Radon-222 alpha particles and radon daughter products. It may be used with a variety of counting instruments such as Eberline Models PS-2, MS-3 or MS-2.

The SAC-R5 may be used to count scintillation cells such as the Eberline SC-3, SC-5 or SC-6; or it may be used with standard Lucas cells. It may also be used to count alpha emissions form filter papers and swipe samples when used with a separate sheet of ZnS scintillation material.

The scintillation cells (SC-3, SC-5 and SC-6) are used to collect gaseous samples. The cells are made of transparent plastic and the inner walls are lined with silver-activated zinc sulfide scintillation material. Two valves are provided on top so that gas can flow through the cell when collecting the sample or purging the cell. (See Figure 3.)

Alpha Contamination Monitor, Eberline Model RM-19/AC-3

The Model RM-19/AC-3 Alpha Contamination Monitor is ideal as a personnel frisker at mill sites and other areas subject to alpha contamination. It is a small, extremely versatile, alarming count rate meter operated from a Gel-Cell battery which is float charged when the unit is plugged into an AC line. Four meter scales are provided (500, 5k, 50k, and 500k counts per minute full scale) and a speaker with volume control provides aural indication of the count rate. The alarm controls are located on the front panel.

A PUSH-TO-SET alarm switch allows the alarm point to be located on the meter directly and an ALARM SET control allows the alarm point to be adjusted from zero to greater than full scale. When actuated, the alarm does not affect the meter reading and is indicated by a red light and a high frequency tone on the speaker.

The detector high voltage is adjustable over a wide range by a rear panel control. This allows setting the proper operating voltage for maximum detector efficiency. The high voltage setting may be viewed on the meter by actuating the PUSH-TO-READ high voltage switch. Circuitry in the RM-19 is all solid-state. The top cover is easily removed for access to internal components. Rear panel connectors are provided for an external scaler and a 50 μA DC recorder. The particular combination is well suited to personnel monitoring in mills and monitoring of clothing and equipment. (See Figure 4.)

Radon Gas Monitor, Eberline Model RGM-1

The Model RGM-1 Radon Gas Monitor is a complete, portable system for continuous measurement of radon gas. The instrument contains an Eberline SAC-R5 Detector and SC-6 Scintillation Cell for radon gas detection. A pump is included for continuous gas flow and digital recorder is used to record the data for unattended operation. The electronics section contains a variable high voltage supply, charge sensitive input amplifier, a light emitting diode six-decade scaler and a timer. All electronic circuits are solid-state with extensive use of integrated circuits to enhance reliability.

The instrument is housed in an aluminum carrying case for ease of portability and may be operated unsheltered under average climatic conditions. The carrying case is insulated, with a thermostatically controlled ventillation fan for cooling in the summer. A thermostatically controlled heater is provided inside the digital recorder to help prevent freeze-up in the winter. Additional protection may be required during extremely cold winter months. (See Figure 5.)

Regulated Air Sampler, Eberline Model RAS-1

Air sampling for radioactive particulate material is extremely important in dry ore or product handling areas. Exhaust stacks from these areas also require periodic monitoring for airborne radioactive particles. Proposed regulatory guides also specify continuous sampling on the site perimeter. The Model RAS-1 pump is ideal for these applications. Filter samples thus collected may be analyzed for gross alpha using a SAC-4 alpha counter. They are then composited for subsequent fluorometric determination of uranium and radiochemical determination of ^{230}Th, ^{226}Ra and ^{210}Pb. The most limiting radionuclide in ore dust is ^{230}Th. The most limiting radionuclide in yellow cake (U_3O_8) areas is usually natural uranium.

The RAS-1 is also used to monitor at the plant perimeter for airborne radioactive particles. In this case, a weatherproof housing (Eberline WPH-1) is required. For this type of environmental monitoring, the pump is allowed to run continuously with the filter changed weekly. (See Figure 6.)

Portable Ore Grade Evaluator, Eberline Model PIT-1000

The Portable Ore Grade Evaluator is a versatile, reliable and extremely rugged ore grade control instrument. The unit consists of an electronics section with a meter read-out in $\%U^30^8$. Recessed controls allow the operator to adjust response time, calibrate to known ore values and check the power source. The detecting element consists of a 24-inch probe with a shock mounted G-M tube enclosed (30 mg/cm^2 stainless steel wall thickness). With this probe the instrument can be used to survey ore cars, ore trucks, stock piles, underground mines and open pit mines. The probe is easily unscrewed from the electronics section and a rugged semi-flexible cable connected to the probe and the electronics thus providing the operator with a unit which will monitor horizontal, vertical and angular drill holes in underground and open pit mines. Further, a sample drawer which utilizes a thin end window G-M detector can be connected to the electronics in lieu of the standard probe. This combination allows for on-the-spot sampling of drill cuttings or other ore samples. (See Figure 7.)

Gamma Energy Analyzer, Eberline Model PRM-5-3/SPA-3

The Gamma Energy Analyzer is a Portable Pulse Meter with the Eberline Lin-Log Meter presentation which displays a continuous linear meter reading from 0 to 500,000 counts per minute without range switching. This Count Rate Meter is used in combination with a Scintillation Detector Assembly (2 inch x 2 inch NaI crystal) and functions as a three-channel gamma energy analyzer. Each of the three channels may be set to distinguish specific energy spans. When exploring for uranium ore, one channel is usually set to measure gross gamma. Another channel is usually set to detect 1.76 MeV gamma radiation which is typical of uranium ore. A third channel is usually set to detect 2.62 MeV gamma radiation typical of thorium ores. When a radiation anomaly is detected, the source can thus be identified as to whether it is uranium or thorium by switching channels. (See Figure 8.)

Portable Ore Analyzer, Eberline Model PS-2/HP-210M/SH-4

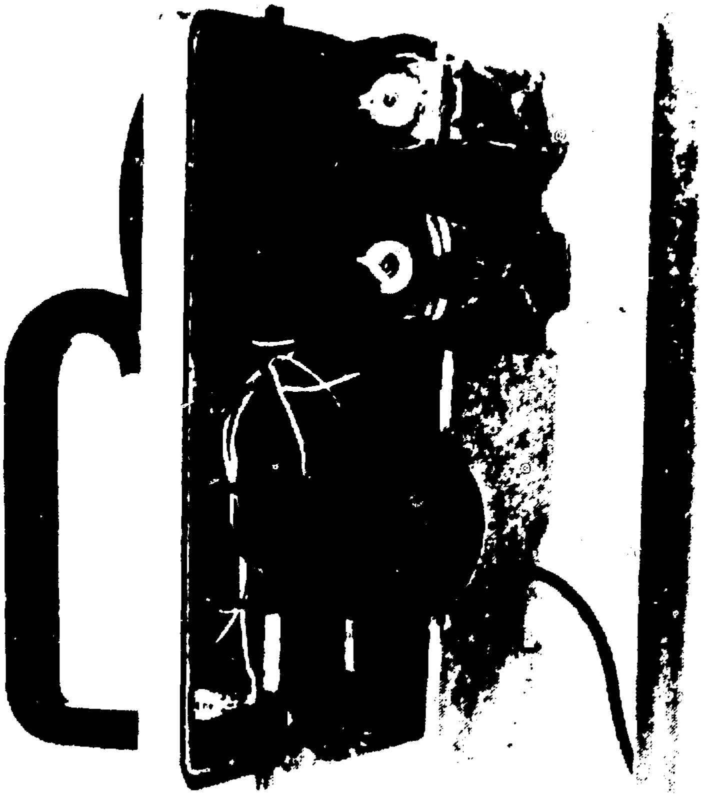
The Portable Ore Analyzer consists of a portable scaler in combination with a pancake G-M detector assembly and sample drawer with holder.

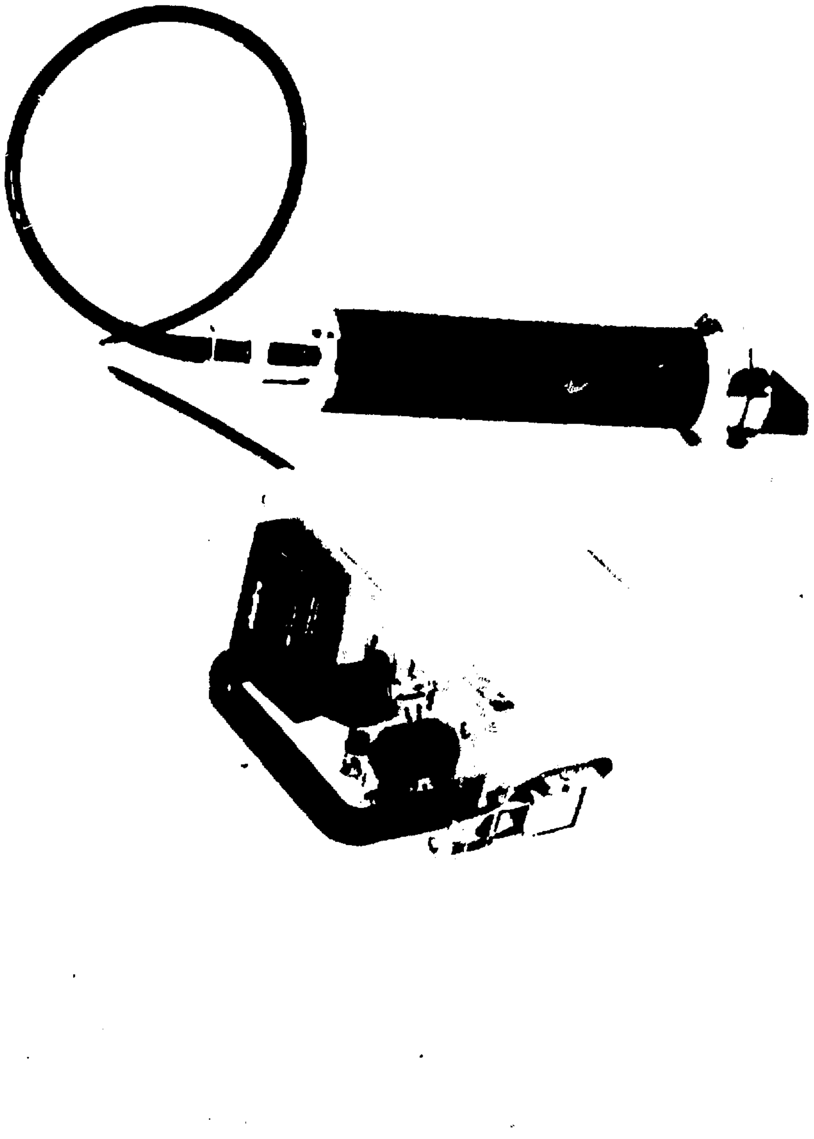
This system is designed to determine %U_3O_8 for uranium ore which is known to be in equilibrium or for ore in which the fractional equilibrium remains constant. The portable scaler is compact, ruggedized and waterproof. It can be used in areas where AC power is not available. All circuits are solid-state with extensive use of integrated circuits to enhance reliability. The NiCd batteries and battery charger are supplied as standard equipment. The detector assembly has a high density tungsten shield to allow for counting samples in a radiation field. Analyzing of drill cuttings or other ore samples can thus take place on the spot rather than having to take them to an assay station or laboratory.

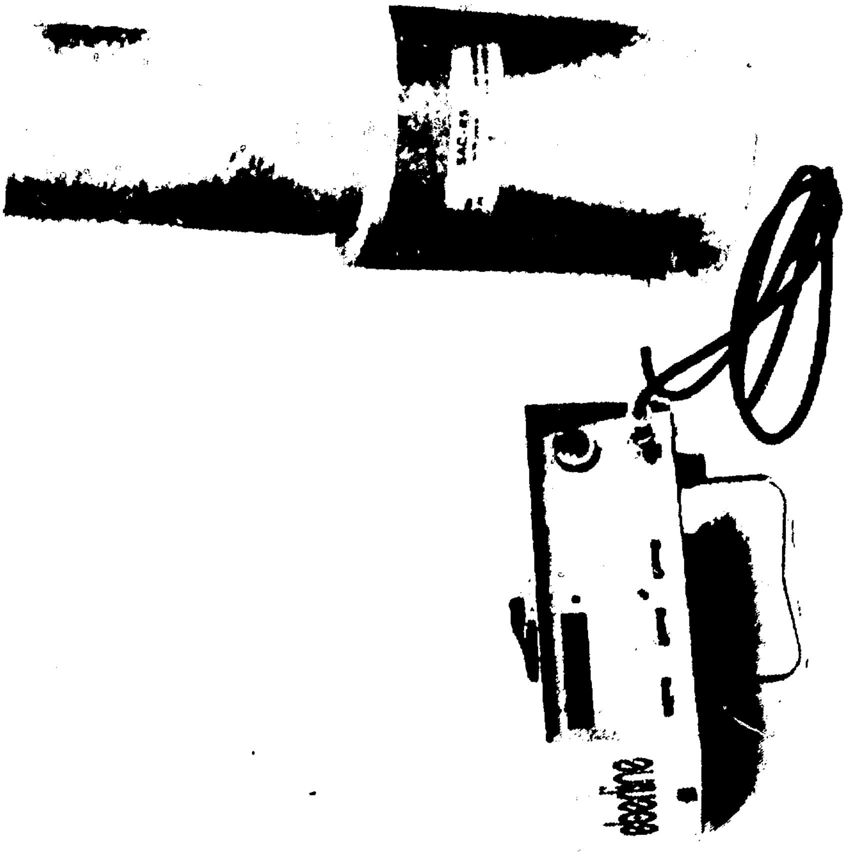
Other Significant Developments

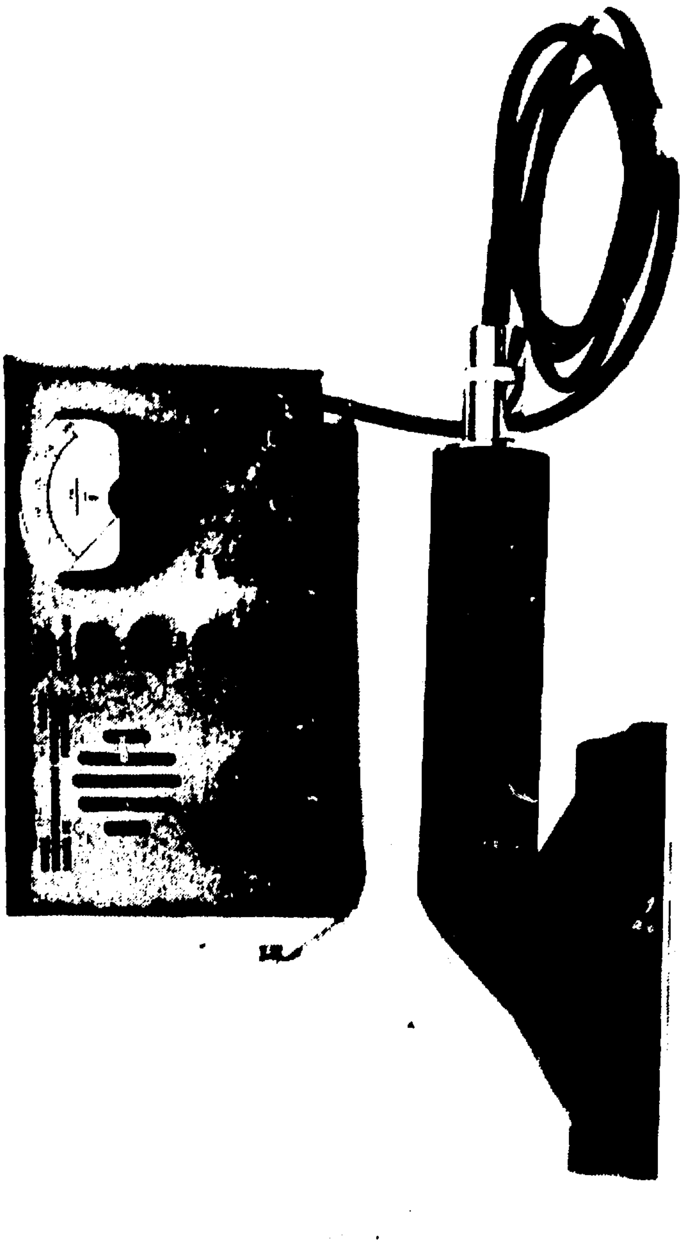
(a) Continuous Radon Progeny and Gas Monitor (see Program Report "Workshop on Methods for Measuring Radiation in and Around Uranium Mills," Vol. 3, No. 9, pp. 349-358). (See Figures 9 and 10)

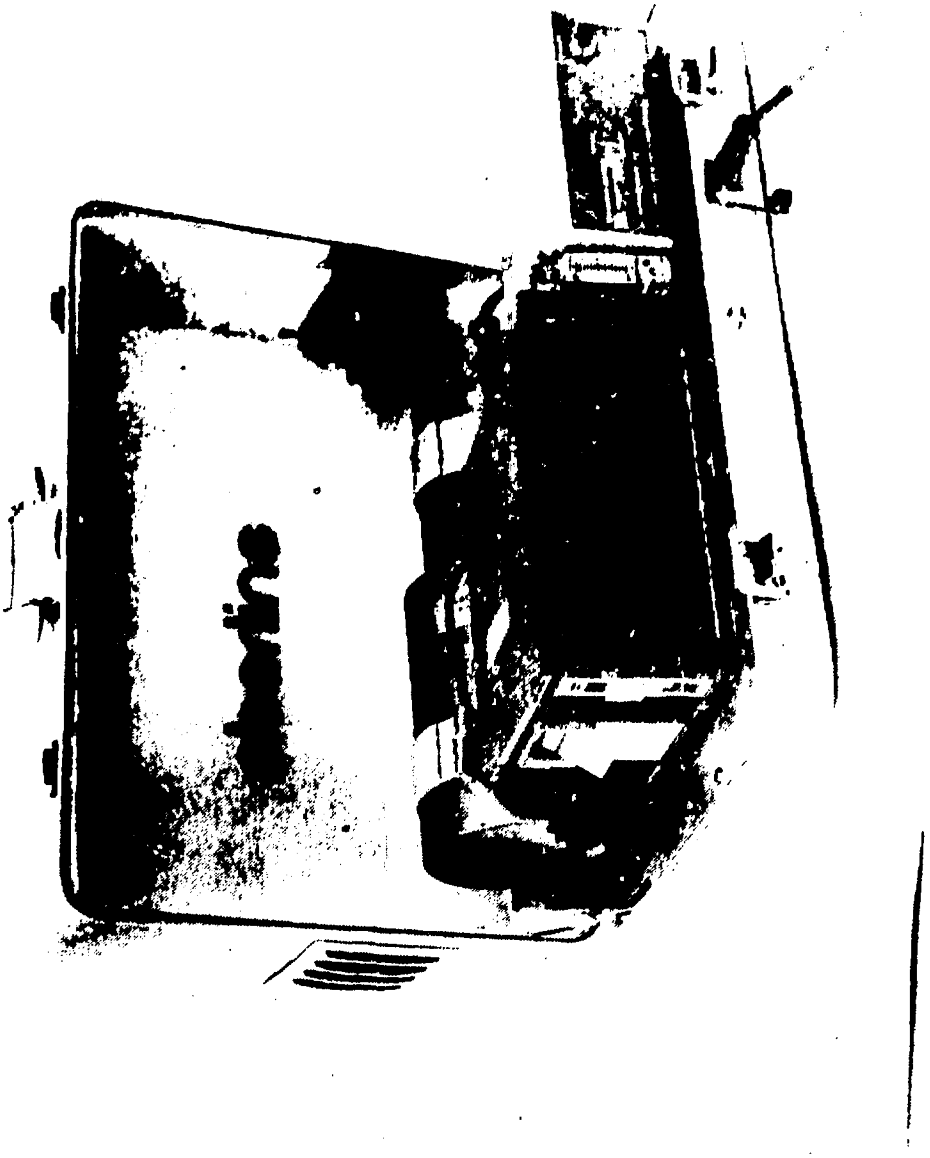
(b) The RASCAL Portable Ratemeter Scaler (see "A Completely New Approach to Portable Instruments," J. Little, afternoon session of this Symposium, January 18, 1978). (See Figure 11.)

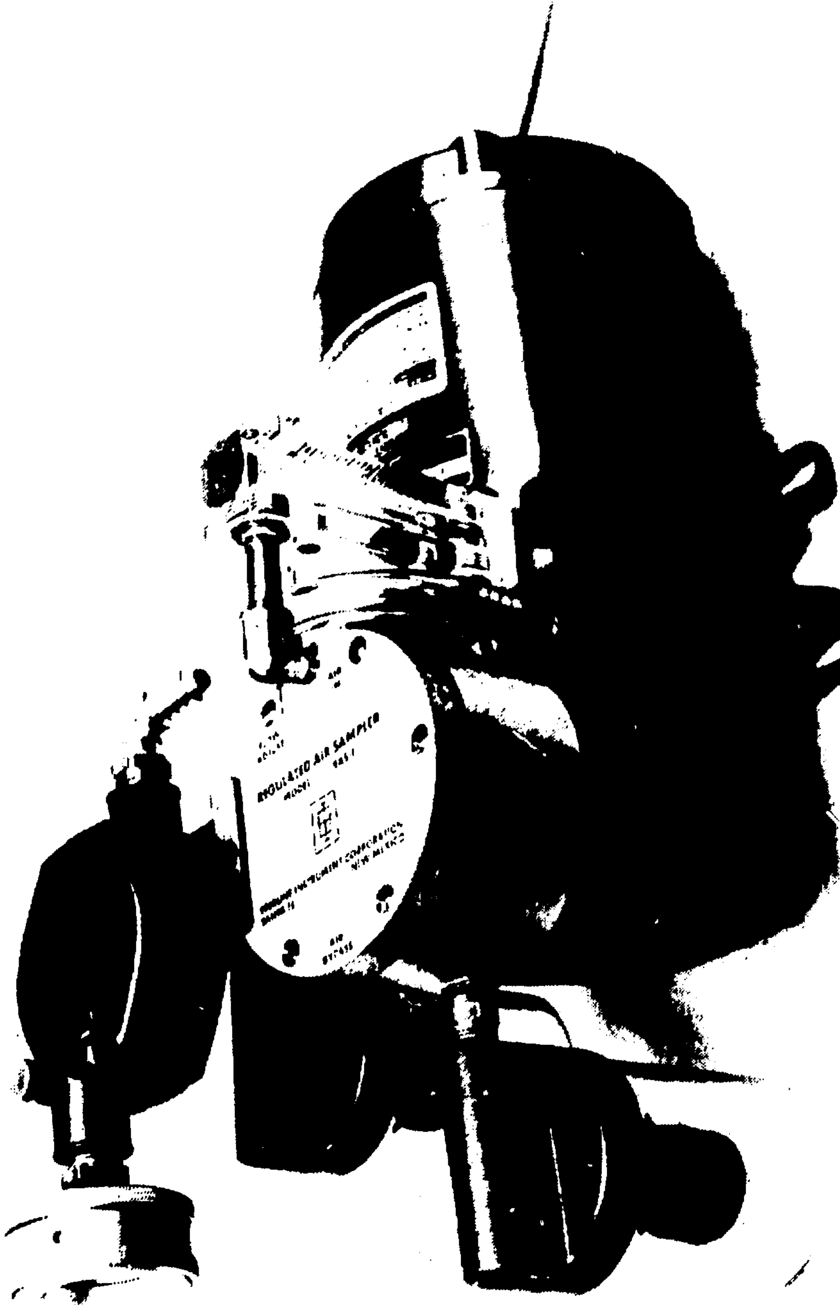








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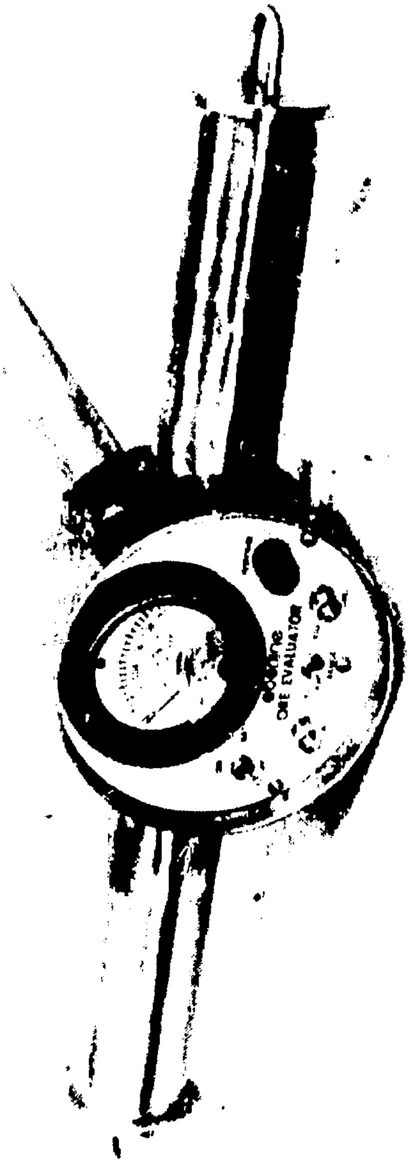
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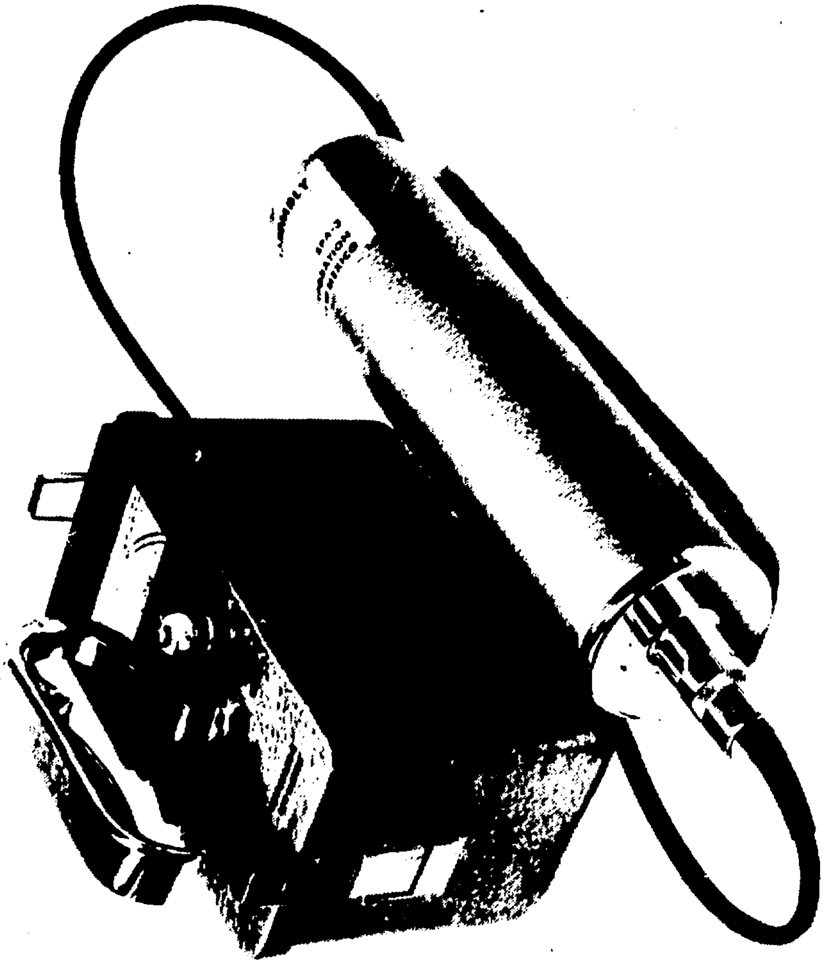

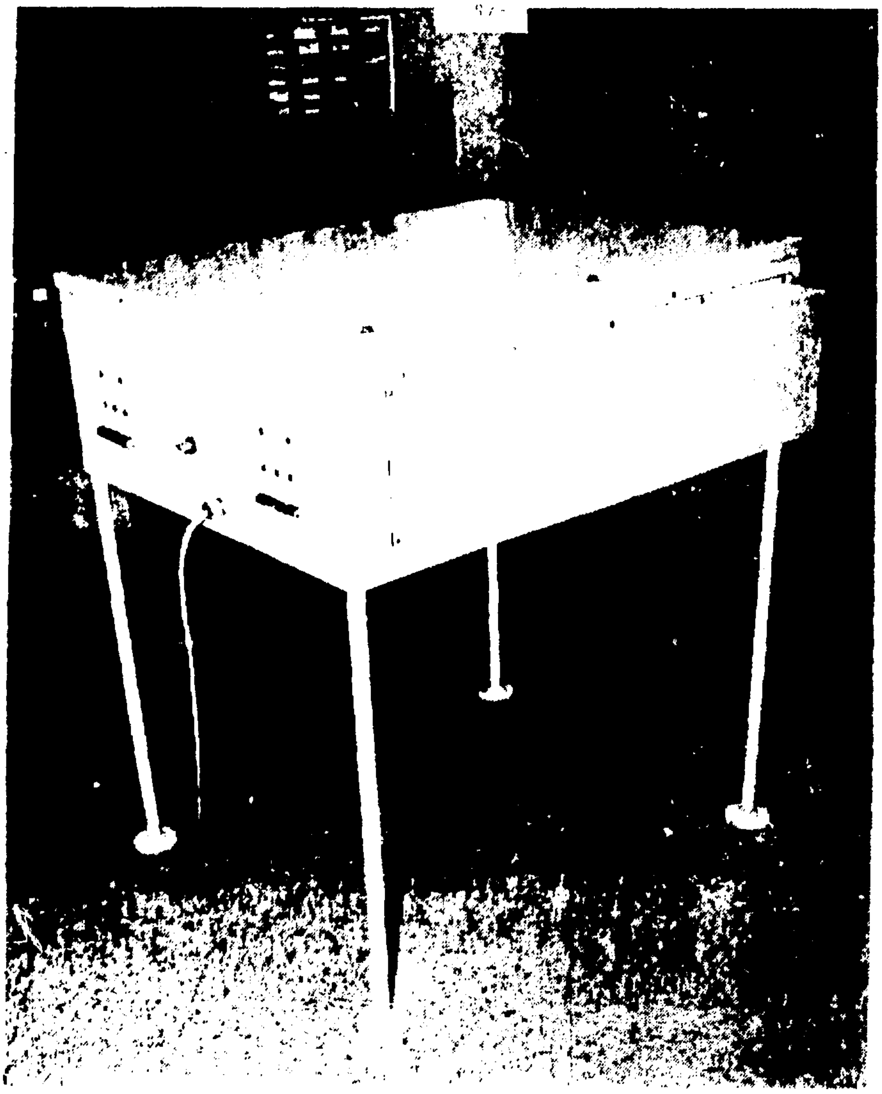

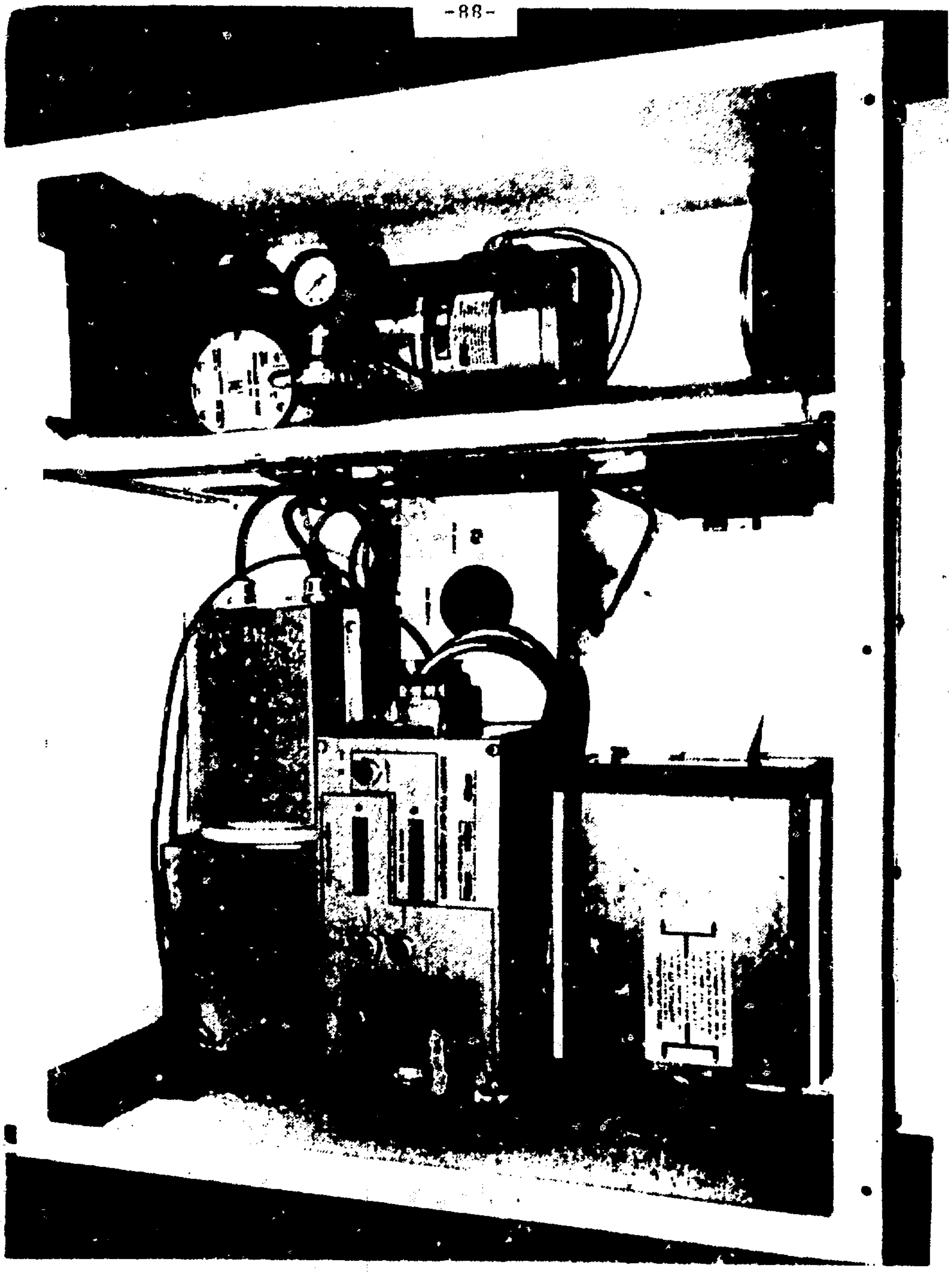

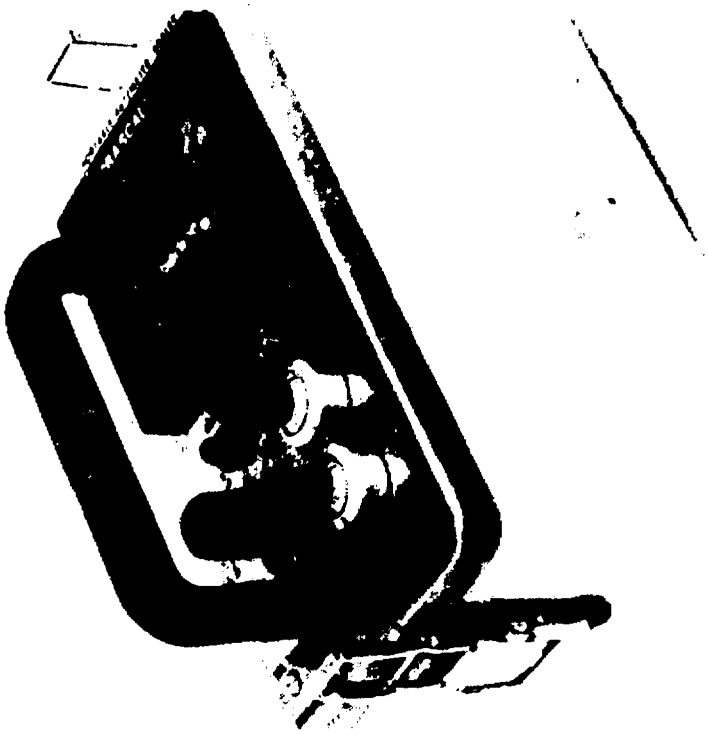

TECHNICAL SESSION C

RADIATION DETECTORS I

Chair: Art Lucas

IMPROVED GERMANIUM WELL DETECTORS

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INTRODUCTION

Germanium well detectors have been made for some years using the Ge(Li) process; they have been described by several authors[1-5]. These detectors have generally suffered from having a layer of inactive germanium lining the holes in the detector crystals. This layer limited response to low energy gamma-rays and increased the background due to scattering. The energy resolution has usually been quite poor compared to that of conventional Ge(Li) detectors. A metal-semiconductor surface barrier contact offers a substantial improvement. Such a contact, used with thin window planar semiconductor detectors, contains minimal inactive material, causing virtually no attenuation or scattering; in addition, it is a quiet low-leakage-current contact.

DESCRIPTION

Figure 1 shows the detector schematically. There is a hole through the germanium crystal, in this case 16mm diameter to clear the 10mm sample tube in the cryostat shell. The surface of the hole is a gold evaporated surface barrier contact similar to that used on many thin-window planar germanium detectors. The low energy response is limited by the .5mm wall thickness of the sample tube, generally made of aluminum.

In the Ge(Li) case, the crystal is first drifted, mounted and tested as a conventional coaxial Ge(Li) detector. This assures that the well detector has a reasonable probability of success. The hole is then machined by spark cutting and any excess mechanical damage is lapped away. The outside lithium contact is re-diffused followed by chemical etching and evaporation of the gold inside contact. The crystal is then clean-up drifted. Before final etching and mounting of the detector the contact is stripped and remade. This last cycle generally must be repeated at least once before adequate current vs. voltage characteristics are achieved.

For high purity detectors, the procedure is similar except that steps pertaining to drift and pre-testing are omitted. To date, it has usually been more difficult to get good leakage current characteristics with high purity detectors than with Ge(Li)'s. This is thought to be due to crystal damage in the machining process. To ensure that the high purity detector can be warmed to room temperature and recooled without deterioration, it is baked at a moderately elevated temperature and cycled between room temperature and 77° K several times.

The surface barrier contact is DC coupled to a cooled FET preamplifier. The FET assembly is mounted on the cold finger next to the detector.

Figure 2 is the low energy portion of spectrum of a mixed source prepared from SRM 4254 with ^{241}Am added.[8] It shows strong clean response at several X-ray energies, particularly at 22 keV, probably Ag X-rays from ^{109}Cd. It also shows strong sum-coincidence peaks from ^{139}Ce decay, 166 keV summing with La X-rays. Such gamma-gamma or gamma-X-ray summing is characteristic of well detectors. It complicates the spectra but also provides additional information for isotope identification.

Figure 3 is an expanded spectrum of the same source. The arrow indicates the ^{139}Ce sum-coincidence doublet. In addition, gamma-gamma summing is seen for ^{60}Co and ^{88}Y. In the case of ^{88}Y, a small triple sum-coincidence (gamma-gamma-X-ray) is seen.

Figure 4 is the same spectrum except that a 0.1mm Pb foil was wrapped around the source. The gamma-gamma sum peaks remain but the X-ray sum peaks are gone. This technique can sometimes clarify a spectrum.

Table 1 gives characteristics of three typical well detectors, showing the range of results that are obtained. The 73 cm^3 detector gave 19% relative efficiency before machining, and the 98 cm^3 gave 27%. The in-well 59.5 keV efficiency is almost independent of detector size, while the 662 keV efficiency is roughly proportional to the active volume.

There is sometimes a significant difference in the 1.33 MeV resolution measured with source in the well and outside. In all Ge(Li)'s seen so far, the in-well measurement is worse while in the only high purity case the in-well is markedly better. This is attributed to differences in hole vs. electron trapping properties.

Peak-to-Compton (P/C) ratios measured outside the well are comparable to those for conventional coaxial detectors of similar resolution and 25 cm efficiency. In the well P/C is about 40% higher for ^{60}Co and 70% higher for ^{137}Cs.

To see the effect of source position on sensitivity, the count rate was measured as a function of source height in the 52 cm^3 high purity detector. The detector crystal is 3.8 cm long with its top at about 3.2 cm from the well bottom. The results for three energies are shown in Table II. The relative count rate is over 80% for about the bottom 2.5 cm. However, it is clear that if precise quantitative determinations are sought, source position must be carefully controlled.

SUMMARY

Germanium well detectors with metal surface barrier contact are comparable for general use with conventional germanium coaxial detectors. They offer very high sensitivity, the highest presently available for some cases[6], for sources of suitably small volume.

ACKNOWLEDGEMENT

The author wishes to thank C. Sanderson for discussion and help in source preparation,

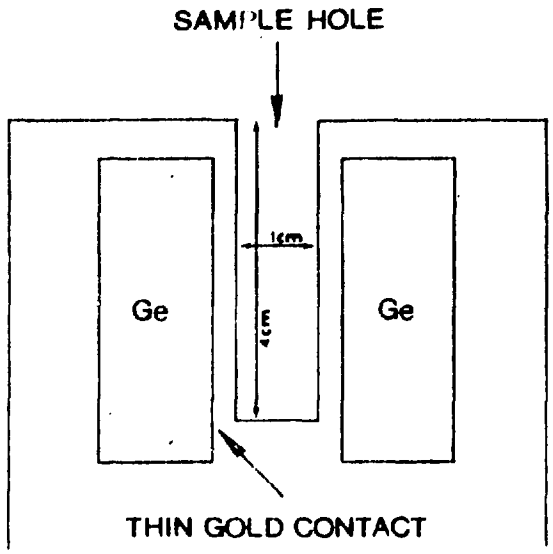
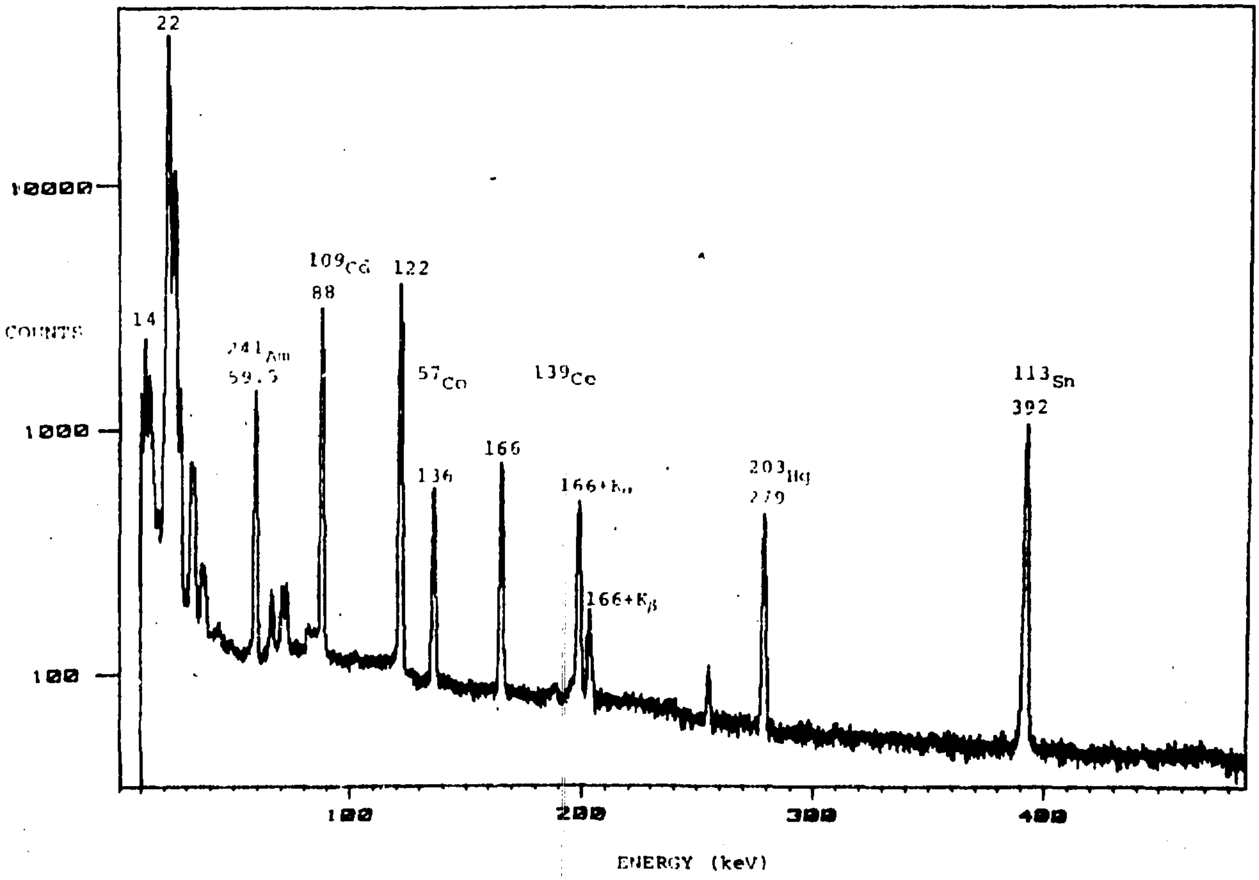


Fig. 1▲ Schematic Diagram of Germanium Well Detector with Metal Surface Barrier Contact

Fig. 2▼ Low Energy Portion of Mixed Source Spectrum



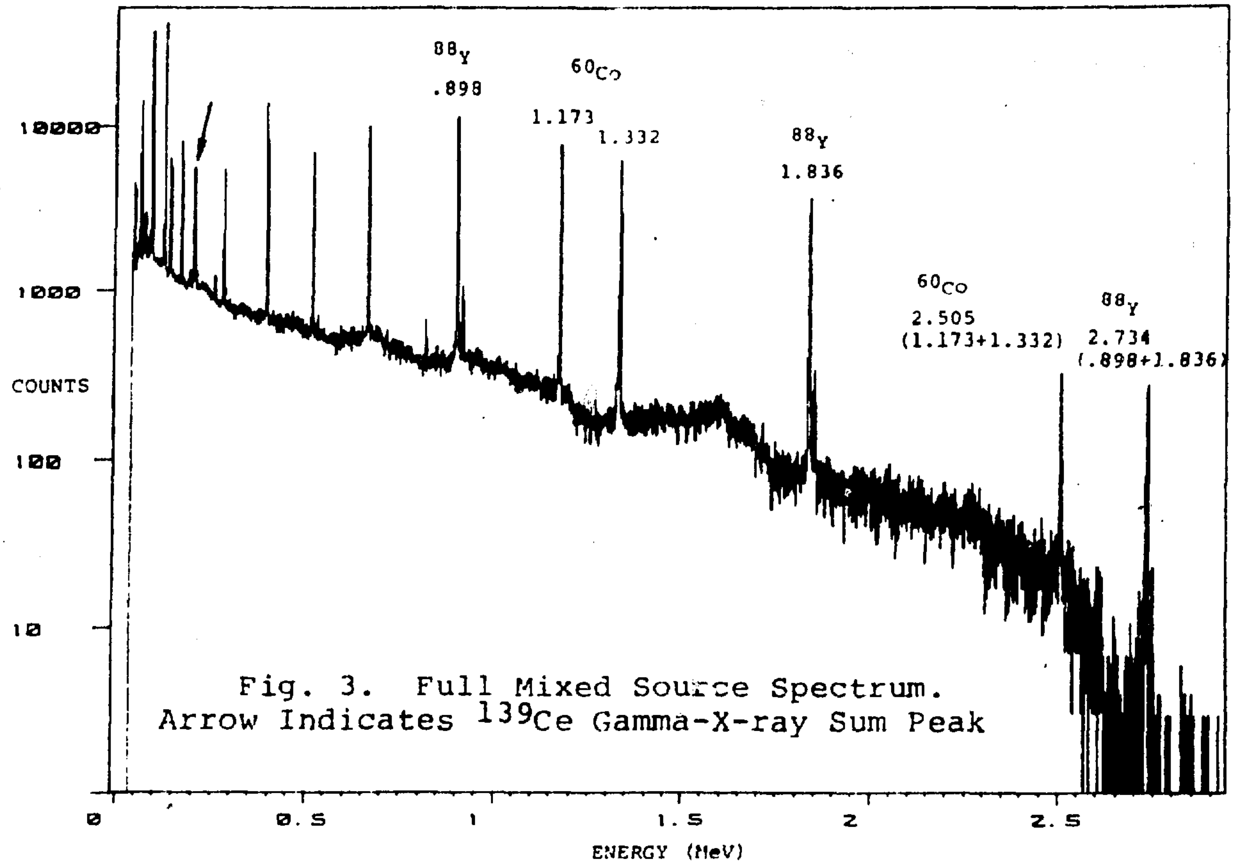


Fig. 3. Full Mixed Source Spectrum. Arrow Indicates ^{139}Ce Gamma-X-ray Sum Peak

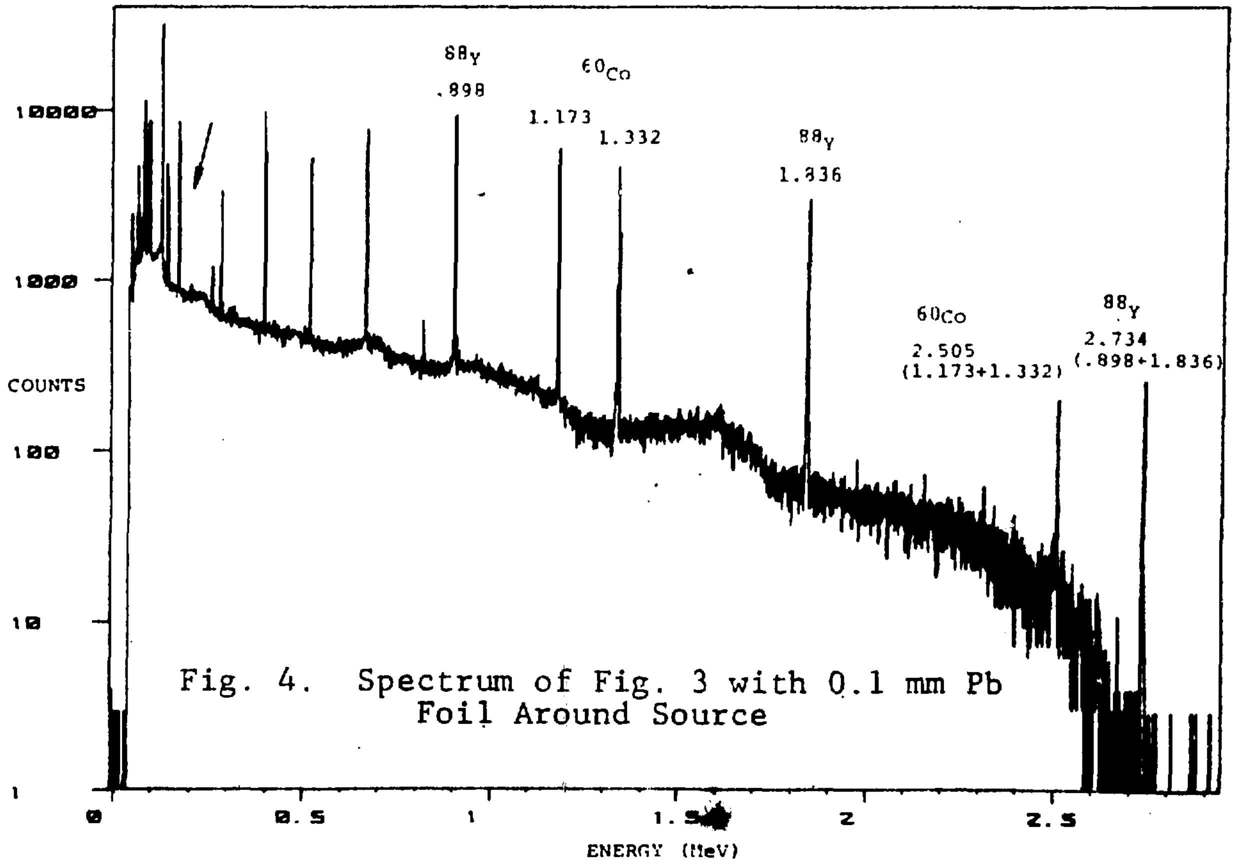


Fig. 4. Spectrum of Fig. 3 with 0.1 mm Pb Foil Around Source

TABLE I

CHARACTERISTICS OF SOME GERMANIUM WELL DETECTORS

Estimated Active Volume	73 cm^3	98 cm^3	52 cm^3
Ge(Li) or High Purity	Ge(Li)	Ge(Li)	High Purity
25 cm Efficiency$^{(a)}$	13.2%	20.1%	8.9%
FWHM in Well @ 1.33 MeV	1.96keV$^{(c)}$	2.85keV$^{(d)}$	2.15keV
FWHM External Source @ 1.33 MeV	1.91keV$^{(c)}$	2.55keV	2.44keV
FWHM External Source @ 122 keV	.99keV$^{(c)}$	1.13keV	1.02keV
Efficiency in Well ^{137}Cs 662 keV	.10$^{(c)}$	.144$^{(d)}$	.080
Efficiency in Well ^{241}Am 59.5 keV	.72$^{(b,c)}$	.84$^{(d)}$	.74$^{(b)}$

a. Relative to 3" x 3" NaI @ 1.33 MeV

b. Not corrected for sum coincidences

c. Ref. 6

d. Ref. 7

TABLE II

RELATIVE COUNT RATE VS. SOURCE HEIGHT FROM BOTTOM OF WELL

	Energy		
Height(cm)	22 keV	88 keV	Sum of 1.33+1.17 MeV
3.3	.26	.36	.52
2.8	.61	.58	.72
2.3	.87	.86	.82
1.8	.99	.96	.99
1.3	1.00	1.00	1.00
0.8	.97	.94	1.00
0.3	.89	.88	.90

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A REVIEW OF LYOLUMINESCENCE DOSIMETRY AND A NEW READOUT METHOD USING LIQUID SCINTILLATION TECHNIQUES

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Introduction

The emission of light which accompanies the dissolution of certain previously irradiated substances in water was reported in 1895 by Wiedeman and Schmidt (Wi-95) who termed the phenomenon "lyoluminescence" (from "lyo" meaning "dispersion" or "dissolution"). They discovered that alkali chlorides irradiated with cathode rays and dissolved in water produced the light emission during dissolution. A similar effect was observed in 1959 by Ahnstrom and Ehrenstein (Ah-59) who found persistent luminescence from the dissolution of glucose irradiated with gamma rays and fast neutrons. They found the relationship between dose and light intensity to be linear in the 60 to 200 megarad range. Subsequent studies have included those of Eriksson (Ek-62) who investigated the quenching effect of oxygen on the emission of light when irradiated NaCl was dissolved in water, and studies by Ahnstrom who also investigated lyoluminescence in NaCl (Ah-66).

In 1973, Atari and co-workers at the University of Birmingham (England) suggested lyoluminescence as possible basis for radiation dosimetry (At-73a). They initially made a survey of suitable materials including NaCl, NaBr, KBr, KI, and CsCl. The NaCl was found to be the most sensitive and was used in subsequent studies to examine other parameters affecting lyoluminescence response. They found it possible to measure doses between 1 and 10^7 rads with 5% accuracy using water as a solvent.

NaCl, with an effective atomic number of 16, is quite different from the atomic number of tissue and thus is in many respects undesirable as a dosimeter material. Accordingly, the Birmingham group studied more tissue equivalent materials that exhibited lyoluminescence (At-74a). These included several monosaccharides (glucose, xylose, and mannose). The minimum dose that could be detected with these sugars was about 100 rads.

To improve the sensitivity, Atari and his co-workers enhanced the light output by utilizing the chemiluminescent reactions of luminol (3-aminophthalhydrazide) with oxidizing agents. They obtained a light enhancement of about 10^6 times, with best results being obtained with trehalose dihydrate. The response was fairly linear between 1 and 10^6 rads.

Other materials which more recently have been found to exhibit lyoluminescence include other organics such as carbohydrates, amino acids, nucleic acids and proteins. Some twenty-two amino acids have been evaluated (Th-76; Et-77a) for lyoluminescence response following exposure to ^{60}Co gamma rays. The sensitivities varied over three orders of magnitude, with the five most sensitive acids being glutamine, glutamic acid, threonine, valine, and phenylalanine. A linear dose response was observed between approximately 10^3 and 10^6 rads. Since amino acids, nucleic acids, proteins, and carbohydrates are all essentially tissue equivalent, their potential as radiation dosimeters presents an extremely attractive possibility, not only for gamma dosimetry, but for neutrons and mixed fields as well.

Readout Techniques

The lyoluminescence detection method used by the British researchers (At-73a) is a relatively simple one. A borosilicate glass cell containing the solvent is placed above the window of a photomultiplier tube inside a light-tight box. The irradiated samples are added to the cell by remote control and stirred automatically. The signal from the PM tube is fed to a DC amplifier and from there to a voltage-to-frequency converter and scaler. The light yields are recorded as number of counts per unit sample weight. The amount of sample used varied between 2 and 15 mg depending on the material used; when water was the solvent, the volume used was 5 ml.

The shape of the light decay curve indicated the presence of two or more diffusion controlled reactions differing in time scale. Because of its greater intensity and direct relation to dose, only the short lived component was measured. For this component, the light intensity appeared to peak at approximately .1 second and then die off, at first quickly and then more gradually. For this reason the light output was integrated over a three second period (At-73a; At-74a).

In an alternate read-out method used by K. J. Puite (Pu-77), a glass cup covered with a light-tight rubber membrane and containing 15 mg of the sample was placed above the closed shutter of a PM tube. After the shutter was opened, 4 ml of distilled water was injected through the membrane. In this method the light output was integrated over ten seconds (Pu-77).

Mechanisms of Light Emission

For alkali halides or other inorganic materials, the mechanism causing lyoluminescence is believed to be the reactions of trapped electrons from F centers (Ah-66). It is postulated that dissolution in water results in the hydration of the electron-hole pairs created by radiation, and in turn these pairs recombine with the emission of light.

It has been shown (Lo-61) that in organic materials such as saccharides, trapped free radicals are involved in the lyoluminescence process in a manner analogous to the reactions of the trapped electrons from the F centers in the alkali halides. According to Atari and Ettinger (At-73b) when solid saccharides are irradiated, some of the energy is stored in the form of stable free radicals. When the material is dissolved, the free radicals can react freely in the solvent. Possible reactions may include recombinations of free radicals, reactions with dissolved oxygen, or reactions with the solvent resulting in the emission of luminescence. For example, in carbohydrate lyoluminescence, Buchan and Ettinger (Bu-75) postulate the reaction

$$\dot{R} + H_2O \rightarrow RH + \dot{O}H$$

as the mechanism involved. In this case, the free radicals interact directly with the solvent (water) to yield hydroxyl radicals. The light emission is thought to result from reactions of the hydroxyl radicals with organic material in solution. Other such mechanisms have been proposed, and the literature on chemiluminescence includes several detailed discussions on the chemical reactions involved for both aqueous solutions and for solutions containing sensitizers such as luminol (Wh-63a, Wh-63b, Bo-75, Ma-73, Mc-73, Bo-68, Sc-76, He-70)

Radiometric Characteristics of Lyoluminescence Dosimeters

The materials whose lyoluminescence response have been studied by various investigators are summarized in Table 1. The dose ranges over which the response was studied as well as the linearity (dose vs. response) is included whenever such information was provided. The energy dependence has been studied in only a few cases, and these are indicated. It is not clear in most of the cases cited whether or not the low value of the range studied is a detection limit. For many of the studies cited, the dose rates were not specified. The lyoluminescence response may be dose-rate dependent, although Ettinger and his coworkers (Et-77a) found no dose rate dependence for saccharides in the range of 3 rad/min to 18.5 Krad/min. Other variables which may be involved are discussed in the next section.

Although most of the studies on lyoluminescence dosimetry have involved the use of gamma radiation (usually ^{60}Co or ^{137}Cs sources), the potential of this method for neutron dosimetry should not be ignored. Ahmstrom and Ehrenstein (Ah-59) observed that the irradiation of crystalline glucose produced 5 to 7 times more luminescence than the same dose of ^{60}Co gammas. Ettinger et al. (Et-77a) have pointed out that saccharides, because of their atomic composition, approximate soft tissue for neutrons and mesons. They are somewhat deficient in hydrogen, however (a content of about 7% vs 10% for tissue) which may be of concern in the 100 eV to 10 MeV range. On the other hand, valine, with a hydrogen content of 9.4% may be very suitable for neutron dosimetry. For this reason, Ettinger's group investigated the response of valine relative to ^{60}Co gammas for 2.5 MeV, 7.5 MeV, and 14 MeV neutrons. They found relative responses 0.8 in the 7.5 and 14 MeV range, and 0.5 at 2.5 MeV.

Puite (Pu-77) studied the neutron response for mannose. He found that the lyoluminescence per rad relative to ^{60}Co gamma rays was 0.24 for neutrons in a degraded fission spectrum, 0.45 for 5.3 MeV neutrons, and 0.60 for 15 MeV neutrons.

Factors Affecting Lyoluminescence Response

In conducting lyoluminescence research or in using lyoluminescence for practical dosimetry, attention must be given to controlling a number of factors which may affect the light output from the sample. Based on the available literature, a summary of these factors is given in Table 2. The table includes only those factors associated with sample treatment and readout, and does not include dose, dose rate, energy or LET factors which must also be considered as for any dosimeter system. A brief discussion of the factors listed in the table is given here. For details, the reader should consult the references listed.

Solvent Factors

In studies on the alkali halides, Atari and Ettinger (At-73a, At-74a, At-75) used distilled water and TlCl solution as solvents. Compared to water, the fluorescent TlCl solution enhanced the light yield 200 times for a 500 krad dose. The degree of enhancement decreased with decreasing dose, dropping off linearly from 10^3 down to 1 krad, and fading to zero enhancement below 1 rad. Thwaites and co-workers (Th-76) indicate that for irradiated amino acids, the light yield can be extended by using a solution of NaOH instead of distilled water. This enhancement is probably related to the pH of the solutions.

The pH of the solvent affects the dose-response curve by influencing the yields of reactive products that cause luminescence. Ettinger et al. (Et-77a) indicate that small variations in pH, not exceeding ± 0.2, do not change the yield by more than 1 to 2%. Atari and Ettinger (At-74a) showed that the slope of the dose-response curve is dependent on the pH of the solution for trehalose·$2H_2O$ phosphor. In the studies on glucose by Ahnstrom and Ehrenstein (Ah-59) it was found that the intensity of luminescence was a function of the pH of the solution. These investigators found a twenty-fold increase in intensity after alkalinization of their water solutions with NaOH. Similar results were reported by Buchan and Ettinger (Bu-75) who found that all the carbohydrates that they studied showed light enhancement in alkaline solutions. Glucose monohydrate was found to be the most sensitive to pH changes.

With respect to oxygen concentration in the solvent, Ettinger et al. (Et-77a) indicate that in their studies with saccharides, variations in the oxygen concentration close to the value of equilibrium with air have no observable effect on the light yield. In a related study, Atari and Ettinger (At-73b) compared the effect of oxygen or nitrogen dissolved in water with freshly distilled water. They found that water saturated with nitrogen had a quenching effect in some cases but not in others, while water saturated with oxygen has an enhancing effect in some cases and a quenching effect in others.

Sensitizer Factors

A number of investigators have shown that significant enhancement of light yield can be obtained through the use of sensitizers. Atari and Ettinger (At-74a, At-74b) sensitized their solutions using luminol (3-aminophthalhydrazide) plus Na_2CO_3 and chlorohemin dissolved in water. With this solution, they found an increase in the lyoluminescence response by a factor of 10^6 for glucose and a factor of 20 for trehalose·$2H_2O$. The use of the sensitizer extended the detection limit of the system down into the millirad range, whereas with water alone, the lowest dose that could be detected was 400 rads.

In another study Ettinger et al. (Et-77c) indicated that lucigenin (N,N-dimethyl-9,9-biacridinium dinitrate) is a sensitizer with properties similar to luminol. They also found that the light yield is dependent on the concentration of the sensitizer for both luminol and for lucigenin. For luminol solutions with trehalose monohydrate they obtained light enhancements of about a factor of 10 when the luminol concentration was 10^{-5} mg/g, and enhancements of about 10^4 when the concentration was 10^{-1} mg/g. The concentration also affects the background caused by the self-glow of the sample. This self-glow is also a function of concentration and hence, sensitivity limits are related to the sensitizer concentration. For lucigenin, light enhancement increased by a factor of about 10^4 at concentrations of 10^{-8} mole/ml but then decreased as the concentration was increased further.

Sensitization can also be achieved by the presence of certain impurities in the solution. Ahnstrom and Ehrenstein (Ah-59) found that light intensity could be increased by the addition of peroxides to the solutions, such as hydrogen peroxide, benzoylperoxide, or dioxan. They observed a ten-fold increase in luminescence intensity by this method.

Sample Factors

In the study on alkali halides (At-75) the lyoluminescence properties of different NaCl samples appeared to depend on the origin of the samples. For six samples of NaCl of different origin, light yields differed by factors of up to five for 100 krad doses. These variations were attributed to differing amounts of impurities and differing heat treatment histories. It was observed that pre-irradiation annealing (600°C for 5 hours) or crushing enhanced the response.

A study of pre-irradiation annealing of NaCl showed that although higher temperatures improved sensitivity, faster fading of the luminescence also resulted.

The response of monosaccharides has been found to remain stable for a period of seven months when stored at room temperature. Samples of trehalose heated to 80°C for two hours after irradiation displayed no change in light output and heating to 60°C for 60 hours gave a decrease of only 18% (At-74b). On the other hand, in the amino acid study (Et-77b), valine, typical of the amino acid group, was found to have lost 20% of its response over a one year period when stored in a desiccator at room temperature.

Humidity is an important factor when the materials are hydroscopic. Atari and Ettinger (At-73b) found that samples of glucose, xylose, and mannose stored 16 hours in a relative humidity of 60% lost 20%, 40%, and 50% of their luminescence, respectively.

For most saccharides, exposure to daylight for 48 hours, or to a tungsten lamp, had no affect on the response (At-73b). For trehalose dihydrate, exposure to direct sunlight for a period of 5 hours produced a loss of 10% in light yield. Exposure of UV light gave an increase in lyoluminescence output for the saccharides (At-73b). A similar situation was observed by Ettinger et al. (Et-77b) for proteins and nucleic acids.

When these materials were exposed to polychromatic UV radiation in the range of 180 to 370 nm. the light emitted on dissolution was found to be proportional to the energy fluence of the UV light. Proteins are as also being sensitive to the UV component of daylight. Bleaching by visible light was observed in pepsin but not in trypsin.

The temperature at which dissolution takes place is reported by Atari et al. (At-73b; Et-77a; Et-77c) to have a marked effect on the light yield. The change is of the order of 2 to 4% per °C.

The fact that impurities in the sample can affect light emission during dissolution was considered by Atari and Ettinger (At-75). In their studies with alkali halides they found large differences in light yields due to the affect of impurities. They indicate that the most common impurities for the alkali halides are OH^-, Cu^+ and O_2^- ions. Of these, they found the Cu^+ ion to have the greatest effect, with light emission enhancement of more than one order of magnitude resulting from dissolving the NaCl powder in water containing a 10^{-4}M concentration of Cu^+.

Atari et al. (At-73a) have indicated that particle size is an important factor in affecting lyoluminescence output for NaCl. In their work, they controlled particle size by sieving, although the actual sizes they used are not indicated. Buchan and Ettinger (Bu-75) have shown a dependence of light yield on the particle size for lyoluminescence of xylose in water. Samples studied ranged from about 50 to 350 μ in size, with light yields being greatest for the larger particles.

Liquid Scintillation Lyoluminescence Readout

Recently we initiated studies at Purdue University on lyoluminescence dosimetry. The first requirement was, of course, to build or obtain a readout device. Readers such as those described by Atari and co-workers seem to be fairly straightforward and relatively simple to construct. We also considered various modifications for thermoluminescence readers which might make them suitable for lyoluminescence work. We have, however, discarded both of these possibilities in favor of utilizing another device, one which is found in nearly every facility that uses radioactive materials: the internal liquid scintillation spectrometer. We knew, for example, that chemical luminescence (chemiluminescence) can be detected with such a device. Indeed, chemiluminescence is often a source of counting interference when assaying radioactive samples. Furthermore, in the past 10 years there has been developed a relatively large number of analytical techniques which utilize induced chemiluminescence in the assay of biological materials. For example, Stanley and Williams (St-69) described a method for the assay of ATP down to as little as 10^{-12} mole by using a liquid scintillation spectrometer to measure the bioluminescence produced by ATP with firefly luciferase.

Subsequently, the use of such instruments has proven to be applicable not only for the assay of ATP and related substances, but also for flavine and pyridine nucleotides (Sc-76). The use of scintillation counters for the study and assay of other biochemical reactions involving oxidases and oxygenases in addition to luciferases has also been reported (Sc-76). Thus, the literature provides some well established techniques for quantitating chemiluminescence by means of liquid scintillation techniques. Accordingly, we elected to use this type of instrument for our lyoluminescence work.

The immediate practical problem that we faced was that of how to mix the sample phosphor and the solvent and then be able to count immediately. Lyoluminescent light emission as a function of time following dissolution is well known and occurs within a few seconds of dissolution. It was clear that if the sample and solvent were manually mixed and then put into the counting position of any typical liquid scintillation spectrometer, the main light emission would be lost.

It was necessary to develop a method for bringing the sample and the solvent together for dissolution to occur after the sample holder was in the counting position. To accomplish this, a unique source holder was developed by modifying a standard glass liquid scintillation counting vial. The item is illustrated in Figure 1. Basically, the solvent is contained in an outer cylinder and flows into an inner cylinder. This flow is initiated when the vial is placed into the counting position of the liquid scintillation spectrometer. There is a time delay before the solvent reaches the sample which is located in a cup about half-way up in the inner cylinder and held in place by standard injection needle.

After the liquid floods the inner cylinder to the level of the sample cup, dissolution occurs and the lyoluminescence output is measured. Currently we are considering some modifications and simplifications of this method, which we believe will improve the handling and the reproducibility of samples.

In the preliminary studies which we report here, the samples were counted using a Packard Model 2002 "Tri Carb" liquid scintillation spectrometer. Because of the nature of the lyoluminescence process, in contrast to liquid scintillation counting of radioactive samples, the scintillation spectrometer must be operated with the coincidence circuit off. The reasons for this are similar to those involved in chemiluminescence procedures, namely that the light emitted consists of isolated photons and not bursts of photons as is the case in scintillation counting of radionuclides. This matter is discussed in detail in the literature (St-69, Sc-70, Sc-76). The instrument gain was set at 100%, and two energy channels were selected for operation with the lower "red" channel spanning settings 83-88 and the upper "green" channel at settings 100 to 106. Additionally, a count-rate meter was coupled to the output of the amplifier to provide a visual indication of the light burst when it occurred.

Based on the work reported in the literature, our preliminary studies were carried out with trehalose, which is a non-reducing disaccharide found in young mushrooms. The samples, in 10 mg amounts, were subjected to four doses from ^{60}Co gamma rays: 0.23 Rad, 1.15 Rads, 5.75 Rads and 28.75 Rads. These doses were based on free air measurements with a precision ionization chamber. The exposure rate involved was 3.7 R/min.

The solvent used was a luminol solution at a concentration of 1.24×10^{-2} g/l and a pH of 11.75. The luminol was used as obtained from the vendor (Sigma Co.) and no additional purification was done. After irradiation, each trehalose sample was placed in the inner cup of a gamma counting vial which was sealed with parafilm and placed in a liquid scintillation vial containing 15 ml of luminol. The inner cup will not be flooded until the parafilm seal is punctured. No effort was made to control the oxygen content and temperature besides the ambient conditions of the luminol solutions. The Tri-Carb was also kept at ambient conditions.

The vials were allowed to sit in the dark overnight. The vials were then counted for half a minute intervals until three succeeding intervals gave reproducible background counts. The vials were then brought out of the counting chamber, the seal was punctured and the vial was then sent back down into the counting chamber. It took approximately 15 seconds for the cup to flood by which time the count was already underway.

At the luminol concentrations used, some chemiluminescence is apparently induced by the flowing and flooding of the cup by the solution. Thus the background has to be evaluated using unirradiated sugar blanks with background counts taken under flow conditions. Also an ordinary tritium source was used to check the possible change in efficiency of the Tri-Carb.

For the exposures specified, three different parameters were measured: the total net counts in the lower red energy channel, the maximum count rate on the rate meter, and the channels ratio (green channel counts/red channel counts). The results are summarized in Figures 2, 3, and 4. The spread or error on the data for total counts vs. dose (Fig. 2) is sufficiently large for the relatively low doses used, that one could fit the points with either a straight line or a curved line. It should be recognized that these data are quite preliminary and we expect to be able to refine them considerably as we develop the technique. Fig. 2 can be fit with a regression model as follows:

$$D = b \ln X + a$$

where D is the dose in rads, X is the total counts in the lower "red" energy channel, and the coefficients are $a = -64.063$ and $b = 6.993$. The correlation coefficient for this model is $r^2 = 0.994$.

Using the count rate, which probably corresponds to a peak height for a lyoluminescence "glow curve", a straight line relationship with dose is observed (Figure 3). The regression model in this case is

$$\ln D = d \ln Z + \ln c$$

where D is the dose in rads, Z is the maximum count rate, and the coefficients are $c = 3.479 \times 10^{-4}$ and $d = 2.0037$. This relation has an $r^2 = 0.960$ correlation coefficient.

It is observed that as the count rate increases, there is a shift in the relative number of counts appearing in each energy channel. This relationship is shown in Figure 4, and it appears that the channels ratio may itself be used as a method of evaluating the dose. The regression model on the channels ratio is:

$$\ln D = b_1 Y + b_0$$

where D is the dose in rads, $Y = \frac{1}{n}\sum_{i}^{n} \frac{(\text{counts green channel})_i}{(\text{counts red channel})_i}$

for n indicating number of replicates, and the coefficients b_0 and b_1 are $b_0 = 7.798$ and $b_1 = 7.528$. For this model, the correlation coefficient is $r^2 = .994$.

In addition to the preliminary work with trehalose reported here, we have initiated investigations of the lyoluminescence of a number of other compounds as measured with the internal liquid scintillation spectrometer. One particular compound not previously reported which produces lyoluminescence is vitamin C (ascorbic acid). Details on its response will be reported elsewhere.

DISCUSSION

Lyoluminescence dosimetry appears to have excellent potential as a useful method for personnel monitoring or for other types of dosimetric measurements. It shows great promise as a neutron dosimeter, and this area should be given special attention.

It is interesting to note that one material used commonly in thermoluminescence studies, LiF, has been reported to exhibit lyoluminescence also (We-60). Consideration might be given to evaluating mixed fields in terms of the relative response of this material (thermoluminescence output vs. lyoluminescence output) to gammas and neutrons.

The use of liquid scintillation spectrometers for measuring lyoluminescence needs additional study. However, it is clear that the method can be used and would be readily available to many health physicists both in applied health physics areas and in research.

Table 1. Summary of lyoluminescence materials studied by various investigators.

Ref.	Material	Solvent System	Dose Range Studied (Gamma)	Energy Dependence	Saturation Dose	Linearity (Gamma Rad.)
Ah-65	NaCl	Water	1 rad to 10^6 rads		10^5 rads	Linear to 30 rads; response proportional to square of dose between 30 & 1000 rad; Logarithm above 10^3 rads
At-73a	NaCl	Water	1 rad to 10^7 rads			Sublinear
At-74a	Trehalose	Water	10^{-1} to 10^6 rads		10^5 rads	Linear to 10^5
At-74a	Trehalose	Luminol	10^0 to 10^6 rads			Approx. linear
Et-77c	Trehalose	Water	10 to 10^6 rads			Linear to 10^5 rads
Et-77c	Trehalose	Lucigenin	10 to 10^6 rads		10^5 rads	Approx. linear to 10^5 rads
Ah-59	Glucose	Aqueous alkaline solution	60 to 170 Mrads			Luminescence intensity directly proportional to dose
At-73b	Glucose	Water	0.5×10^3 to 5×10^6 rads		10^5 rads	Approx. linear to 10^5
"	Xylose	Water	0.5×10^3 to 5×10^6 rads		" "	" " " "
"	Mannose	Water	0.5×10^3 to 5×10^6 rads		" "	" " " "
Pu-77	Mannose	Water	5 rad to 10^5 rad	Independent from 0.1 to 1.25 MeV	10^5 rads	Linear to 300 rad; supralinear above 300 rads
Et-77a	Glucose$\cdot H_2O$	Water	10 rad to 10^6 rads	Independent from 0.1 to 1.25 MeV	300-450 Krads	Linear below 1 Krad

Ref.	Material	Solvent System	Dose Range Studied (Gamma)	Energy Dependence	Saturation Dose	Linearity (Gamma Rad.)
Et-77a	Mannose	Water	10 rad to 10^6 rads		80-90 Krads	Linear below 1 Krad
"	Sucrose	"	10 rad to 10^6 rads		80-100 Krads	" " " "
"	Trehalose·$2H_2O$	"	10 rad to 10^6 rads		250-400 Krads	" " " "
"	L-Valine	"	10 rad to 10^6 rads		> 600 Krads	" " " "
Et-77b	Glutamine	Water	10^3-10^6 rads			Approx. linear
"	Glutamic acid	"	"			" "
"	Threonine	"	"			" "
"	Valine	"	"			" "
"	Phenylalanine	"	"			" "
"	(15 other amino acids)	"	"			" "
Th-76	L-glutamine	Water	10^3-10^6 rads			Approx. linear
"	L-threonine	"	"			" "
"	L-glutamic acid	"	"			" "
"	L-valine	"	"			" "
"	L-phenylalanine	"	"			" "
"	(5 other amino acids)	"	"			" "

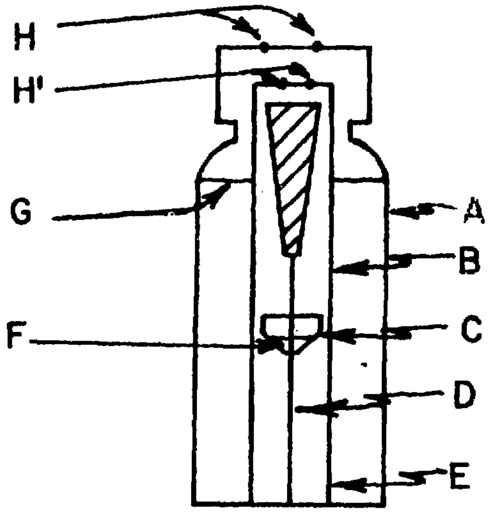
Ref.	Material	Solvent System	Dose Range Studied (Gamma)	Energy Dependence	Saturation Dose	Linearity (Gamma Rad.)
Et-77b	Enzymes:					
"	Lysozyme	Water	10^3-10^6 rads			Not given
"	Alkaline phosphatase	"	"			Not given
"	Trypsin	"	"			Linear to 200 Krad
"	Pepsin	"	"			Not given
Et-77b	Nucleic acids	Water	10^3-10^6 rads			Linear to 400 Krad
Et-77b	Antibiotics:					
"	Streptomycin	Water	10^3-10^6 rads			All responses in form of saturated exponential leveling off at 300 Krads
"	Gentamycin	"	"			
"	Oxytetracycline	"	"			
We-60	NaCl	Water; PPO; TlCl	Up to 200 Mrads			
"	LiF	Water	" " " "			
"	KI	Water	" " " "			
"	Glucose	Water	" " " "			
"	Saccharose	Water	" " " "			
"	Alginic acid	Water	" " " "			
"	Polystyrene	Xylene/ POPOP	" " " "			

*Saturation dose indicates a peak in response with subsequent decrease in light yield at higher doses.

Table 2. Factors affecting lyoluminescence response.

Solvent Factors	References
Type	At-73a, At-74a, At-75
pH	Ah-59, At-74a, Et-77a, Bu-75
Dissolved oxygen concentration	At-73b, Et-77a
Volume	Et-77a
Impurities	At-74b
Sensitizer Factors	
Type	At-74a, At-74b, Et-77c
Concentration	Et-77c
Impurities	Ah-59
Particle size	Bu-75
Sample Factors	
Purity	At-75
Heat treatment history	At-75
Storage temperature	At-74b, Et-77b
Storage time after exposure	At-74b, At-74b
Exposure to light	At-73b, Et-77b
Exposure to humidity	At-73b
Temperature at dissolution	At-73b, Et-77a

Fig. 1. Counting vial for lyoluminescence samples.



A. Liquid scintillation vial
B. Plastic gamma counting vial
C. Plastic cup
D. 18 gauge needle
E. 23 gauge needle hole
F. Trehalose sample
G. Luminol solution level
H. Hole in cap of liquid scintillation vial
H'. Hole in cap of gamma counting vial

Fig. 2 Lyoluminescence response (counts vs dose) of trehalose in luminol solution

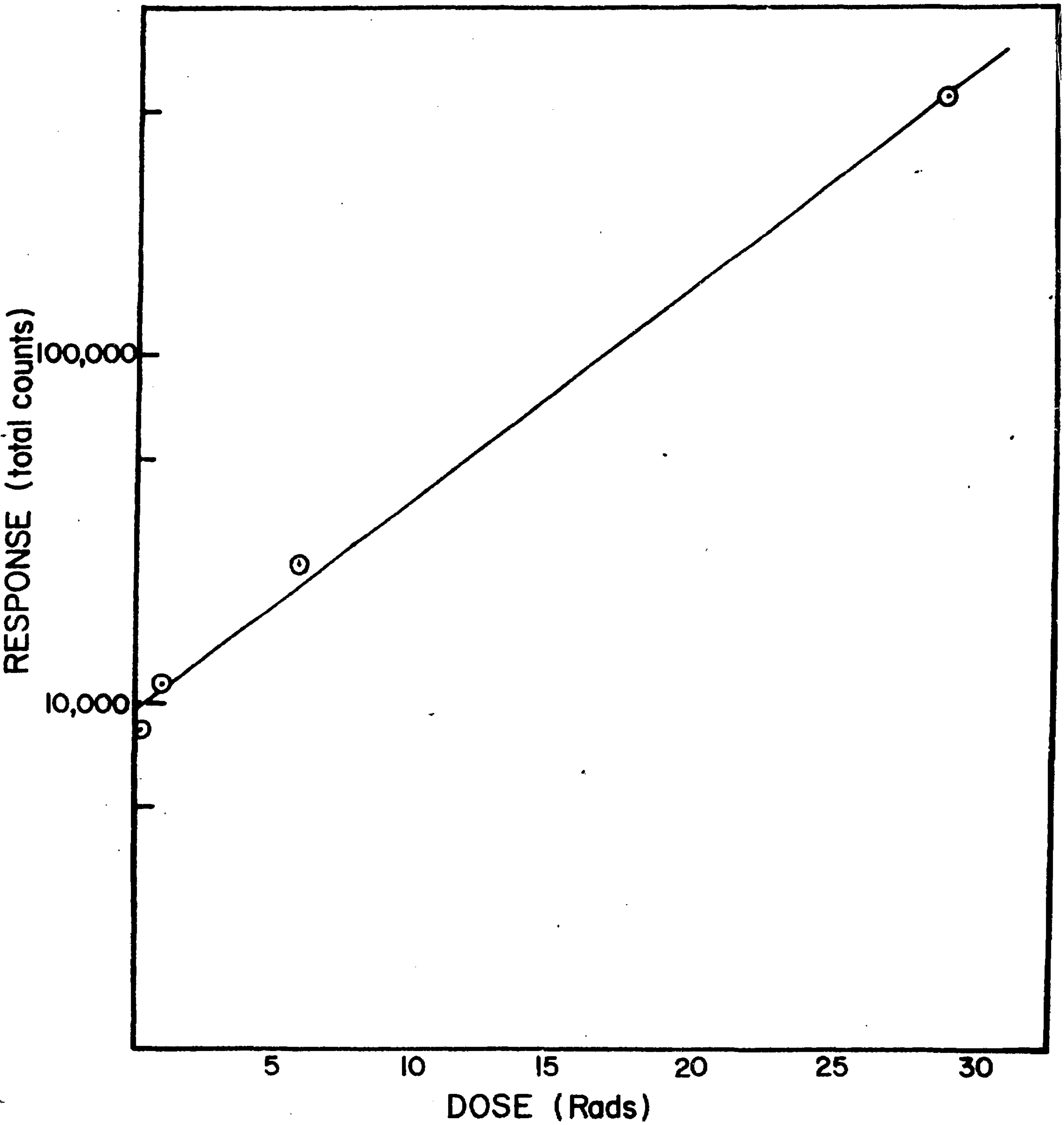
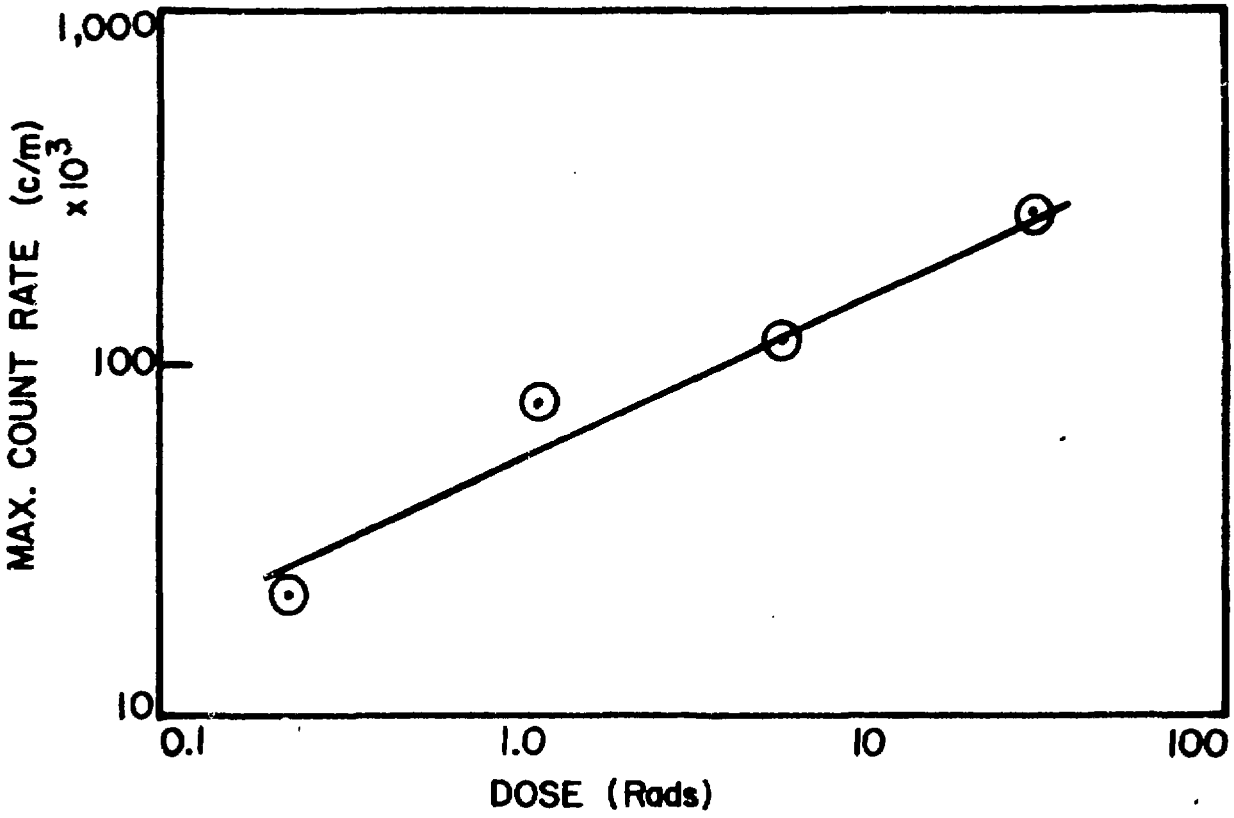


Fig. 3. Lyoluminescence response (max. count rate vs. dose) of trehalose in luminol solution.



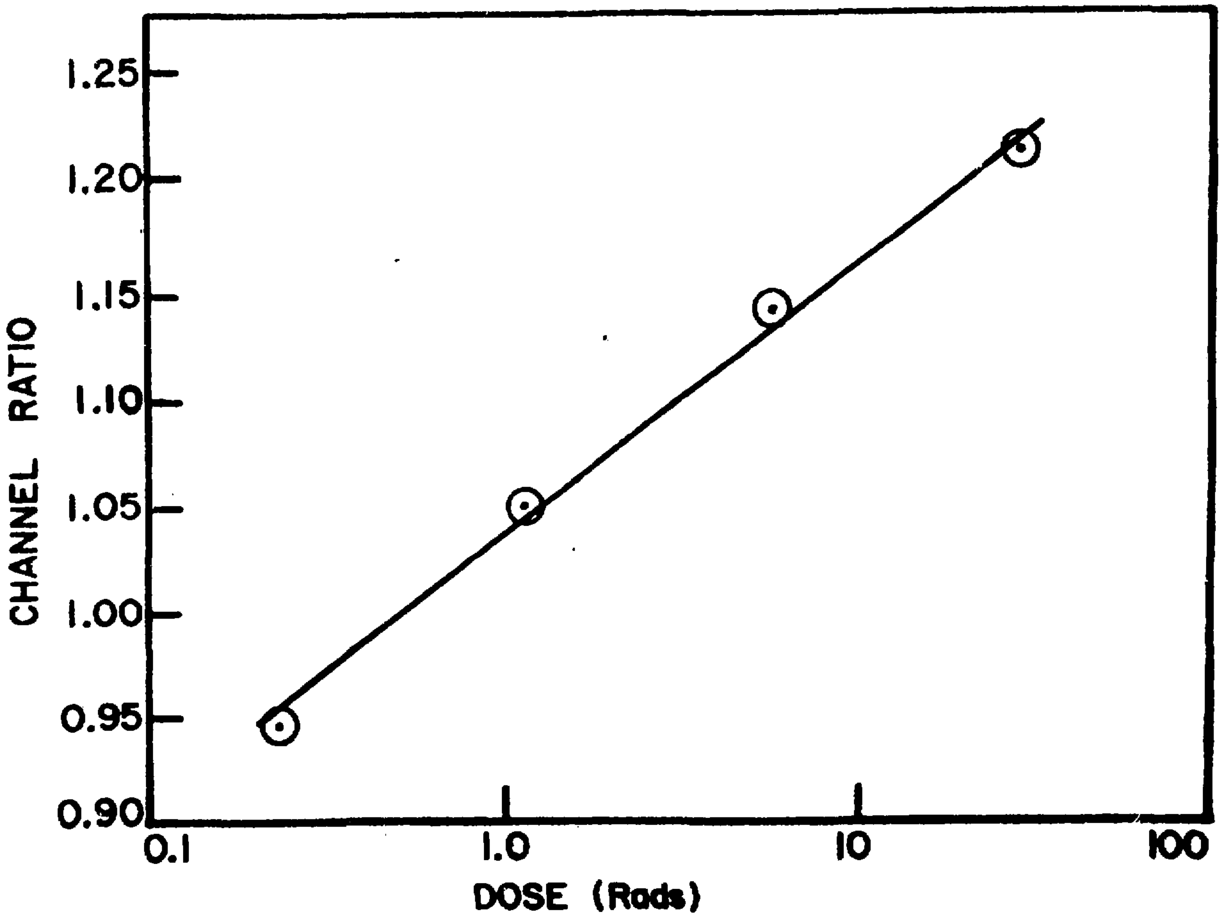


Fig. 4. Lyoluminescence response (channels ratio vs. dose) of trehalose in luminol solution.

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Question (Gene Tochlin):

I think I heard you correctly when you said that the neutron to gamma response for glucose was five, is that correct?

Answer (Paul L. Ziemer):

Yes. That was the report given by Anstrom in one of the early papers - he has two papers in the literature, one was '59 and the other was '65. I don't recall off the top of my head which it was, but he reported a factor of 5 times greater response per equivalent dose, which is rather surprising - the work of Edinger and others, later, it is a factor of less than one in each case.

Question (Tochlin):

Is that on a rad or a rem basis?

Answer (Ziemer):

That's on a rad basis. Clearly work needs to be done - the data are rather fuzzy in the literature, but it does at least indicate some promise.

Question (Skabble):

Is that with a phantom the five to one ratio, or without a phantom? What kind of sample size?

Answer (Ziemer):

I don't know the answer to that, Ken. In both of those cases where it was reported, the literature just in a brief sentence gives the number, it doesn't give information around the sample conditions. I do have those papers with me for those of you who are interested in this - you can see exactly what they say. But they do not mention whether or not a phantom was involved in these.

Question (Gerrard Wong):

Is there an estimation of the disolving time?

Answer (Ziemer):

The disolution apparently occurs very rapidly as evidenced by the response curve, or the light output per unit time with a great deal of the disolution occurring in the first tenth of a second or so. Now, one also has to give consideration probably to the geometry of the situation - whether or not the crystals or the material has ready access to the solution. The European groups have used mixing devices to make sure that there's been good surface contact. Right now, in our case, we've just used the flooding phenomena which may actually prolong this process a little bit.

Question (Wong):

Would that interfere with the tenth of a second response which your material is based on - it seems to me it is a one tenth of a second as a response that you take for the lyoluminescence.

Answer (Ziemer):

Yes. One is integrating the light over a somewhat longer time period - basically you are looking at the area of that curve typically.

VORP, AN IMPROVED DETECTOR SYSTEM FOR REMOTE MEASUREMENTS

By

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ABSTRACT

The Voltage Output Radiation Probe (VORP) is used as the detector to monitor radiological data on underground nuclear tests at the Department of Energy (DOE) Nevada Test Site.

The VORP was designed to convert the current output of a Neher-White ionization chamber to a voltage output. There are several advantages in having a voltage output device rather than a low current probe in the field. The VORP makes it easier to protect against transients induced in the field wires. It also allows the use of redundant (parallel) data acquisition equipment without having to duplicate each field probe.

The electronics was designed to operate off of a self-contained 1.35-volt battery and was fabricated using thick film hybrid circuit technology.

INTRODUCTION

Since 1963, when the Limited Test Ban Treaty was signed by the United Kingdom, the Soviet Union, and the United States, nearly all the U. S. nuclear tests have been conducted deep underground at the Nevada Test Site (NTS). During the conduct of the test event, individuals are not allowed to remain in the immediate vicinity of the test location; hence, the command, control, diagnostic and radiation detection systems are monitored remotely, at a control point that is 5 to 15 km distant. All tests are conducted under the control of the Department of Energy (formerly Energy, Research, and Development Administration, formerly Atomic Energy Commission) which establishes criteria for an extensive radiation monitoring system designed to determine that no radiation has been released from the event. If a release should occur, these remote monitoring systems would also be used to document the magnitude and duration of the radioactive gas and debris released.

For tests sponsored by the LASL, the Laboratory is required to field the close-in (<1 km) portion of this system. This is done by an array of remote radiation detectors typically consisting of one detector at the emplacement hole location, eight detectors equally spaced on a circle that is 100 m to 200 m radius from the emplacement hole, and eight detectors equally spaced in a circle that is 300 m to 400 m radius from the emplacement hole. The exact spacing is chosen by Laboratory personnel to give the best balance between maximum sensitivity, maximum chance of surviving the event ground shock, and ability to give useful data under various meteorological conditions and wind directions. Requirements for this portion of the system have evolved throughout the years, the most important being wide range, simplicity, and ruggedness. The dynamic range has to cover dose rates from 1 mR/h to 1000 R/h, with an accuracy of ±15%; the detector itself has to operate remotely for periods of days, in an arid desert environment where diurnal temperature swings of 30°C, with temperature extremes of -20°C in winter and +50°C in summer, are typical. After each test, the remote system is dismantled, recalibrated, and moved to the next test location. Figure 1 shows a typical emplacement hole location; the detector is housed in the steel pipe nipple strapped to the central pipe. Figure 2 shows a typical arc location. Placement and maintenance of the detectors and associated readout equipment is performed by the Environmental Sciences Department of Reynolds Electrical and Engineering Company.

LONG-LINE CURRENT-DRIVEN SYSTEM

The type of detector that has been used on LASL-sponsored events has remained unchanged; it is a NEHER-WHITE ionization chamber with an enclosed electrometer tube. The current output of the plate circuit of the electrometer tube is approximately proportional to the logarithm of the incident radiation flux; this feature allows the wide range response of from 1 mR/h to 1000 R/h without complicated switching circuits at the remote probe. Remote readout of the probe surrent output was originally done by linking the plate circuit to the readout module at the control point, with 5 to 15 km of ordinary wire pairs of both field and multi-conductor wire. A micro-ammeter in the readout module then indicated the radiation levels. The scale of the micro-ammeter was marked in the three decades from 1 to 1000, and a single switch was used to insert a shunt resistor so that the meter read either 1 to 1000 mR/h or 1 to 1000 R/h.

Some disadvantages of this system were that each detector package required an individual pair of wires from it to the control point, the current mode system did not allow for redundancy in readouts and the long wire length made the system very susceptible to transient noise and stray currents.

EARLY MULTIPLEXED SYSTEM

In mid-1970, a major improvement was introduced when the Electronics Division of LASL designed and fabricated the RAMMS or Remote Area Monitoring and Multiplexing System. For this system, the remote probes remained the same, but the field wires from each probe were routed to a central location at the ground zero area. At this central location, each current signal was individually converted into a 0 to 5 volt signal and then time multiplexed into a frequency modulated (FM) audio signal. This audio signal could then be transmitted to the control point on a single wire pair or by a very high frequency (VHF), ultra high frequency (UHF) or microwave radio link. In actual practice, two separate signals from a redundant multiplexer were sent. This system had the field advantage that the average length of wire between the remote probe and the current-to-voltage (I to V) converter and multiplexer was reduced to less than 1 km; the plate power supplies, previously at the control point, were now at the multiplexer and could be individually set and electronically stabilized and, once the signal had undergone the I to V conversion, redundant features could be introduced. Figure 3 shows the prototype RAMMS remote module at a typical LASL event.

At the control point, after the individual voltages had been discriminated from the FM signal, they were fed through an analog to digital converter into a central processing unit and could be manipulated and displayed in a variety of formats.

This system still suffered from the disadvantage of weak currents flowing in field wires, variable field wire resistance affecting the current and plate circuit balance, lack of good transient protection, and physical separation of the plate circuit voltage source from the electrometer tube.

THE VORP

These problems have been solved by the development of the Voltage Output Radiation Probe (VORP) and the Radiation and Data Acquisition System (RADAS) by the Electronics Division of LASL. The VORP converts the current output of the NEHER-WHITE chamber to a voltage output. It was designed to operate from the existing 1.35 volt filament batteries, using thick film hybrid circuit technology. All field adjustments at the probe are eliminated and excellent temperature stability and transient protection is achieved. The physical dimensions are such that it will fit in place of a 30 volt battery used for the ionization chamber shell potential in the present LASL probe package. Figure 4 and Figure 5 illustrate the VORP packaging and placement in the probe package. The general schematic of the VORP is shown in Figure 6.

The amplifier circuitry provides an excitation voltage of +5.3 (±0.02) volts for the plate supply of the ionization chamber and converts the current of the chamber to a voltage. Diode D1 and amplifier No. 3 provide for the temperature-stable voltage; the plate supply has an excellent temperature coefficient of ±0.01%/°C, which is a necessity for the di-urnal temperature swing of the NTS climate. Amplifier No. 1 converts a 0 to 200 microamp signal from the probe to a 0 to -5 volt signal; amplifier No. 2 is configured as an inverting lowpass filter to provide a 0 to +5 volt output. This output is used to drive the lines between the probe and the ground zero multiplexer location.

Resistors R12 through R15 and diode D2 provide protection from transients of up to 1200 volts for 1 ms or 1.2 watts-sec. Resistor R16 is used to drop the voltage across the filament to about 0.95 volts. The unit has a total power dissipation of less than 10 milliwatts for the VORP and 10 milliwatts for the chamber filament.

A significant feature of the VORP is the DC-to-DC converter that operates from the 1.35 volt filament batteries and provides +30 volts for the shell potential and ±10 volts for the amplifier circuitry. The design is a basic over-driven, push-pull, transformer-coupled transistor oscillator that is designed to oscillate at about 5 kHz. Figure 7 is a detailed schematic of the DC-to-DC converter. The ±10 volts is developed by the primary-secondary wiring (N11:N2 and N12:N2) turns ratio of the torroidal core transformer. Two voltage-doubling circuits are used to develop the +30 volt output.

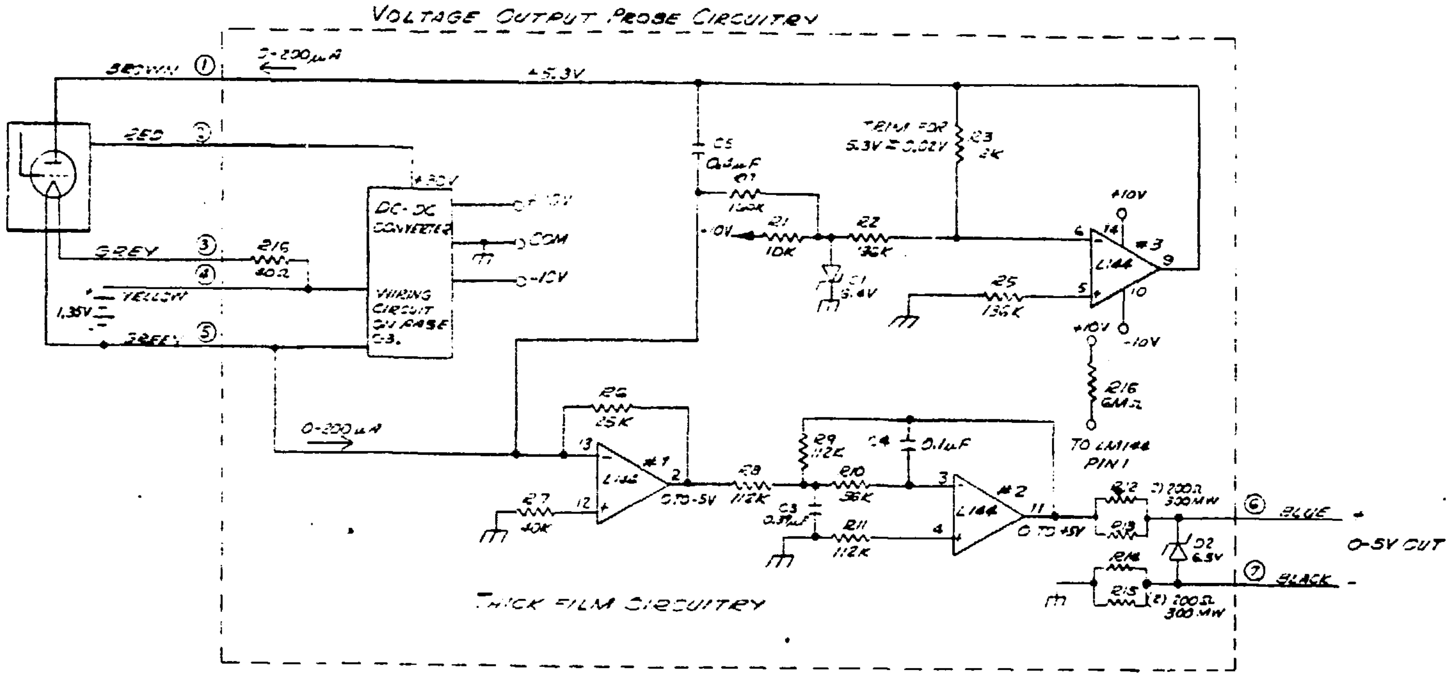
CONCLUSION AND RESULTS

Although a voltage output technique is not especially new, the VORP is custom-designed towards a specific system. The use of the hybrid circuit amplifier, torroidal core DC-to-DC converter, and thick film packaging makes it uniquely suitable to use in remote probes.

A typical NEHER-WHITE chamber circuit will have a current characteristic which varies with radiation, as shown in Figure 8. The output is not exactly proportional to the logarithm of the radiation flux but contains the "S" shaped deviation typical of these chambers. One measure of the VORP is that it provides a faithful transform of this shape. Figure 9 is the o to +5 output voltage for this chamber; it is identical in shape to the current input. When the data is digitized at the control point, the inherent non-linearities of the probe output are easily handled by the computer programs.

The VORP system has been in use on LASL events at NTS since 1974 with unqualified success. It has measurably improved the consistency and reliability of our data collection system.

FIGURES



NOTE:

1. D1 1N4525A MOTOROLA
2. D2 1Z6001 TRANSTECTOR SYSTEMS
3. R16 40Ω RESISTOR -50MW, R12 THRU R15 200Ω -300MW, ALL OTHER 10MW

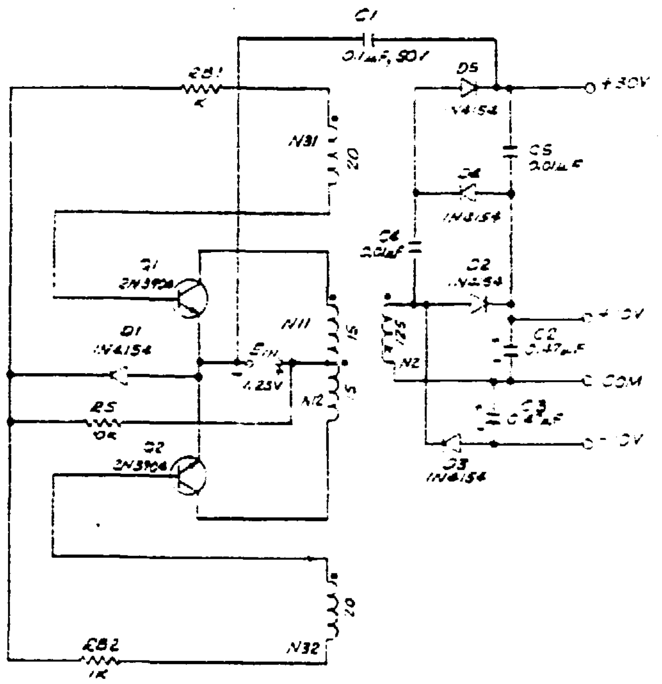
B		REVISION PER DCR No. 1124		PJR		
A		ORIGINAL ISSUE	–	–	–	–

los alamos scientific laboratory of the University of California

TITLE
MODEL 79 VORP
VOLTAGE OUTPUT RADIATION PROBE
AMPLIFIER SCHEMATIC

	M. ERICKSON	9-27	E-5
	ARM	9-16-76	E2
	GR	3-4-76	E2
		9-27	E5
			E-5
			E-2

4Y-168379 C-1



TRANSFORMER WINDING INSTRUCTIONS

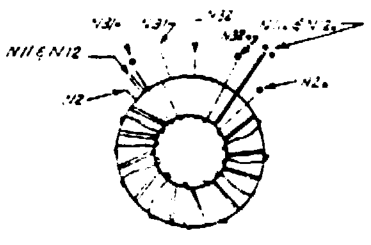
1 USE TORROID CORE #266CT125, MATERIAL 3E2A, FERROX CUBE CO.
2 WIND WITH #40 WIRE, WITH HEAVY FORMVAR INSULATION.
3 WIND ACCORDING TO INSTRUCTIONS AND DIAGRAM.

1ST.
PRIMARY WINDING WOUND BIFILAR (TOGETHER), N11 AND N12 EACH 15 TURNS USING 270 DEGREES OF TORROID.

2ND.
FEEDBACK WINDINGS WOUND ON REMAINING (90 DEGREES, AS SHOWN IN DIAGRAM) N31 AND N32 EACH 20 TURNS.

3RD
SECONDARY WINDINGS WOUND ON TOP OF PRIMARY WINDINGS N2 = 125 TURNS.

NOTICE:
MARK ALL WINDING ENDS ACCORDING TO COIL NUMBER AND POLARITY. (IE N12., N12, N2., N2) ECT.



DETAIL "A"

los alamos scientific laboratory of the university of california

MODEL 79 CRP
VOLTAGE OUTPUT RADIATION PROBE

DC TO DC CONVERTER SCHEMATIC

R. BACKSON 8-27-74 E3
ARM 9-3-74 E2
GR 9-4-75 E2
R. BACKSON 8-27-74 E3

4Y-68379

C-3

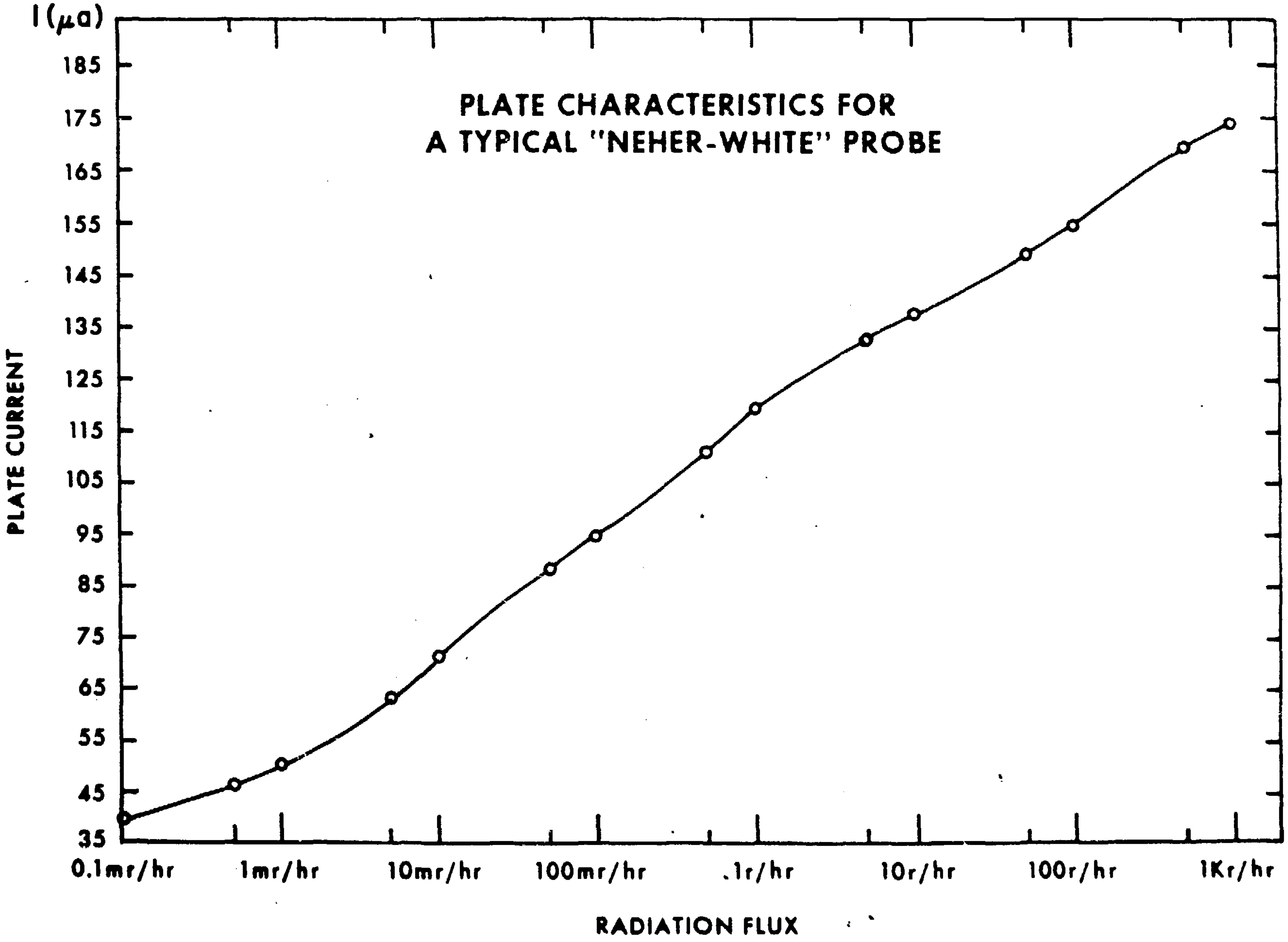
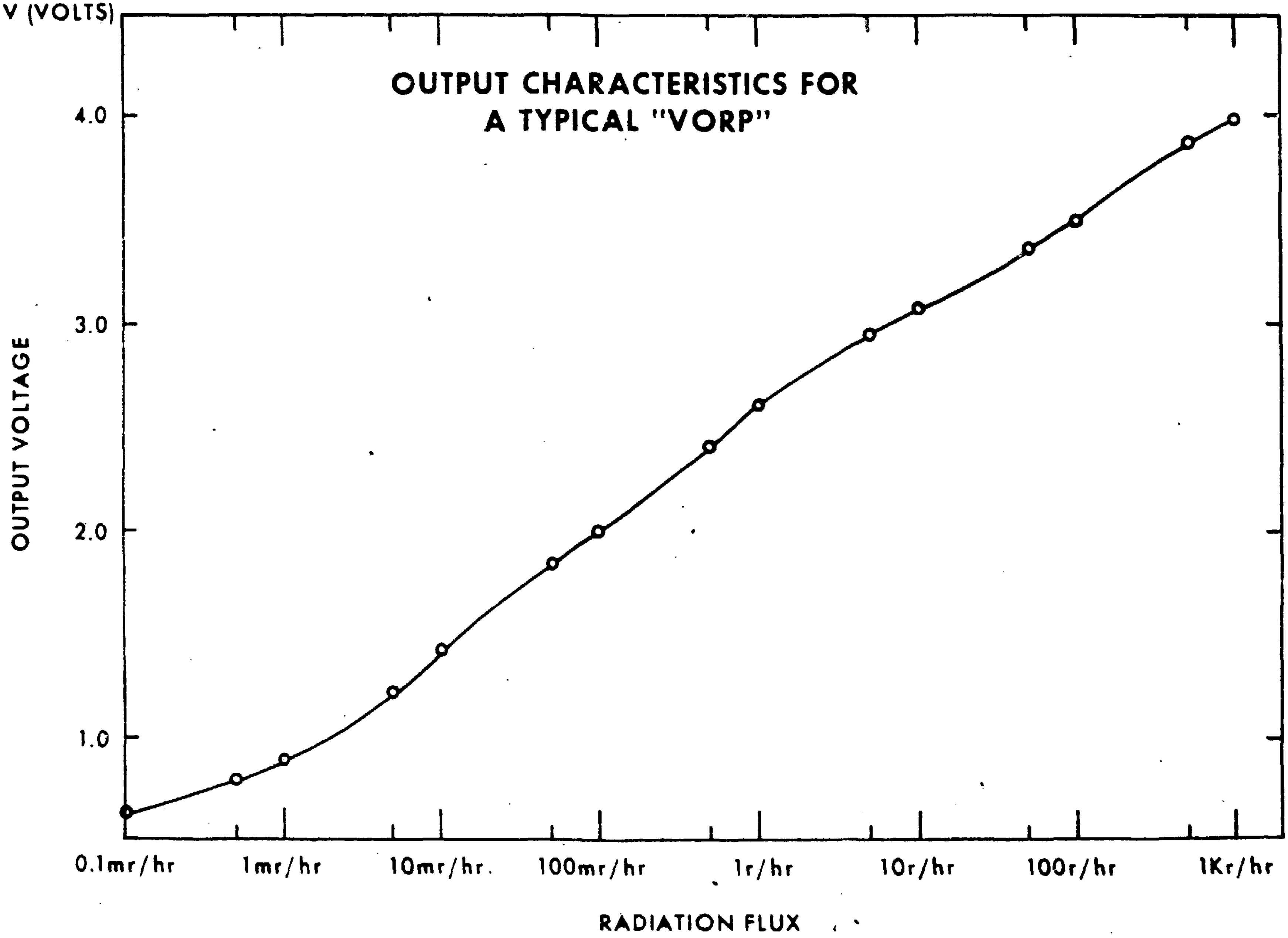
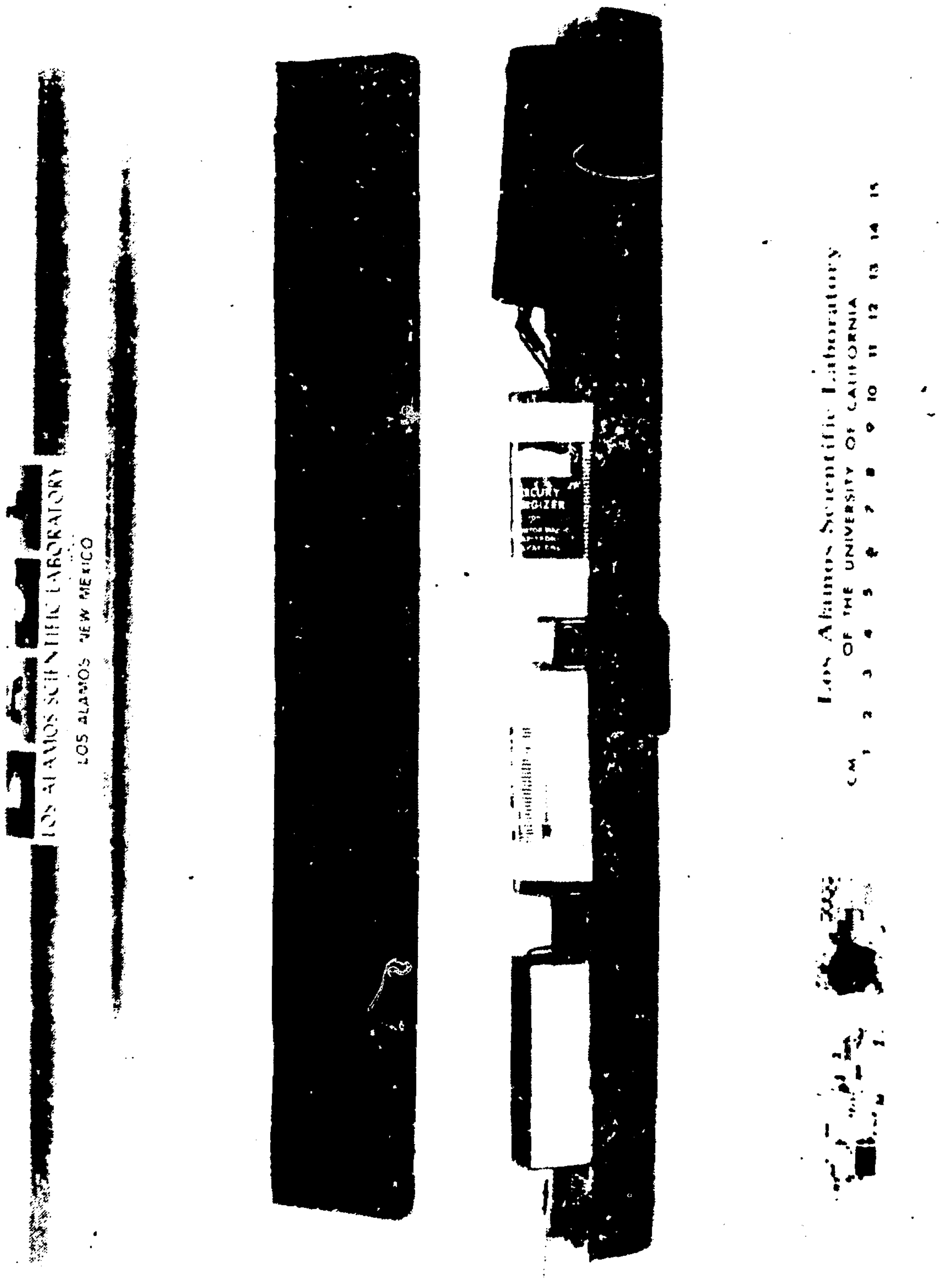
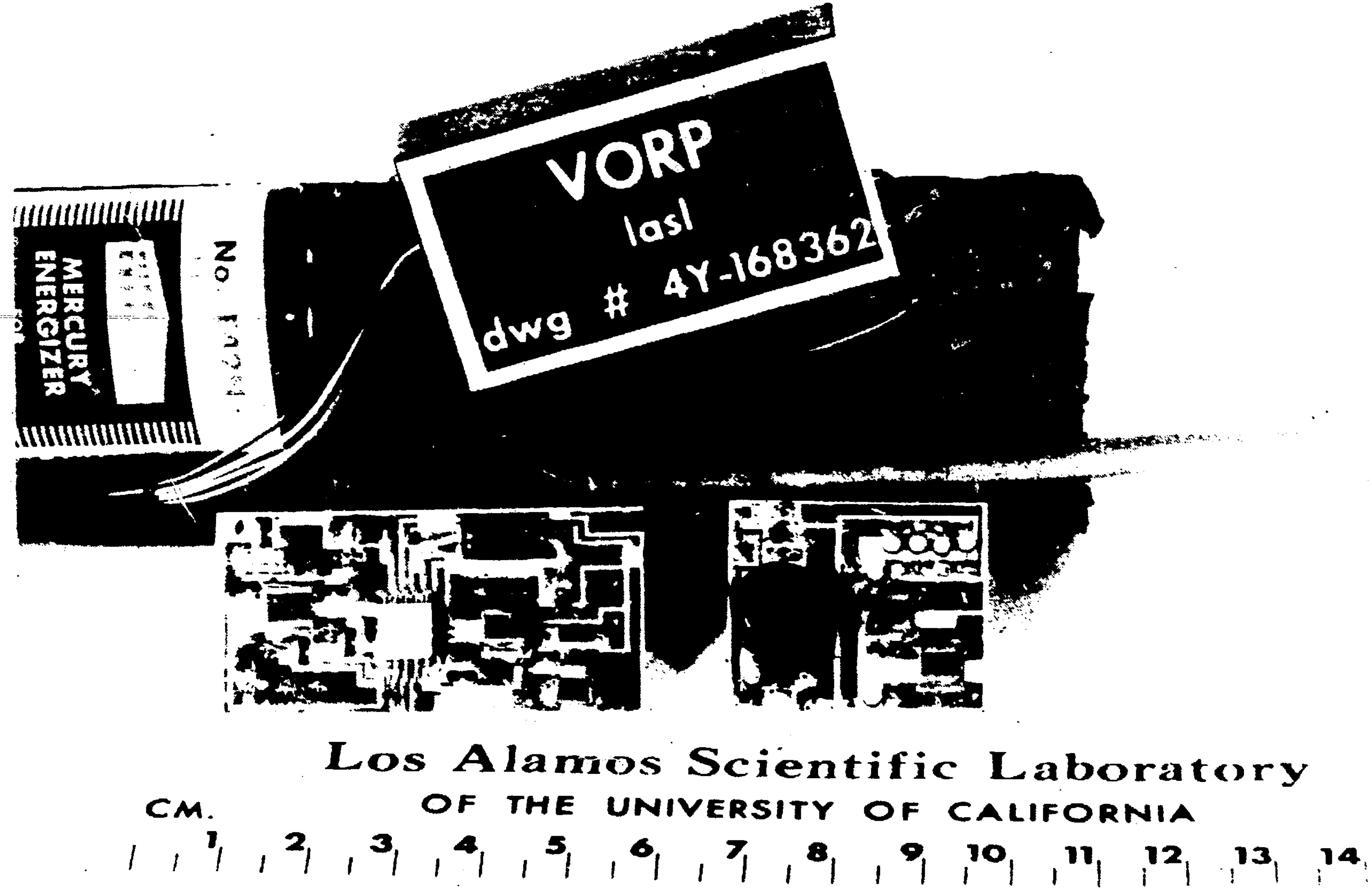
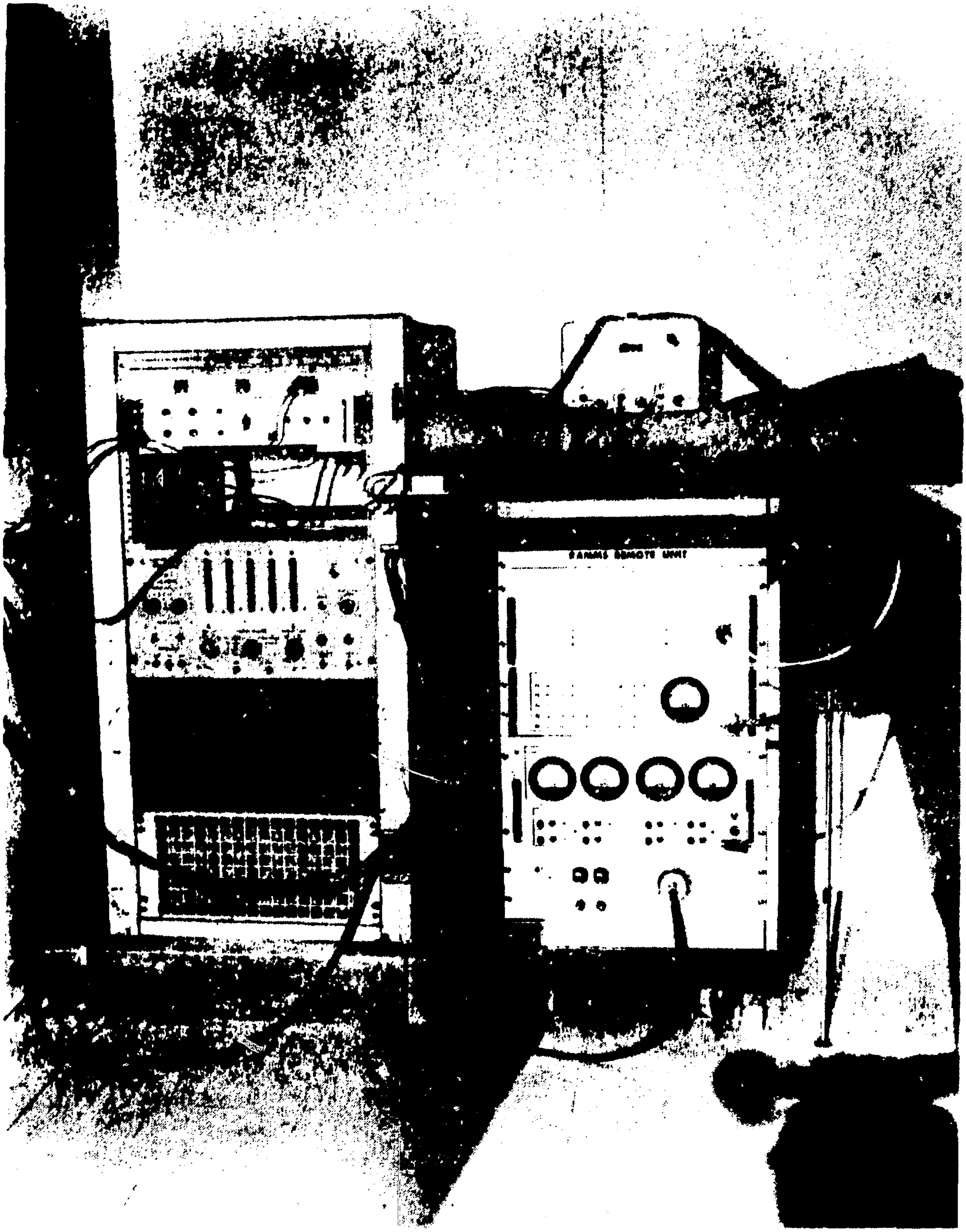


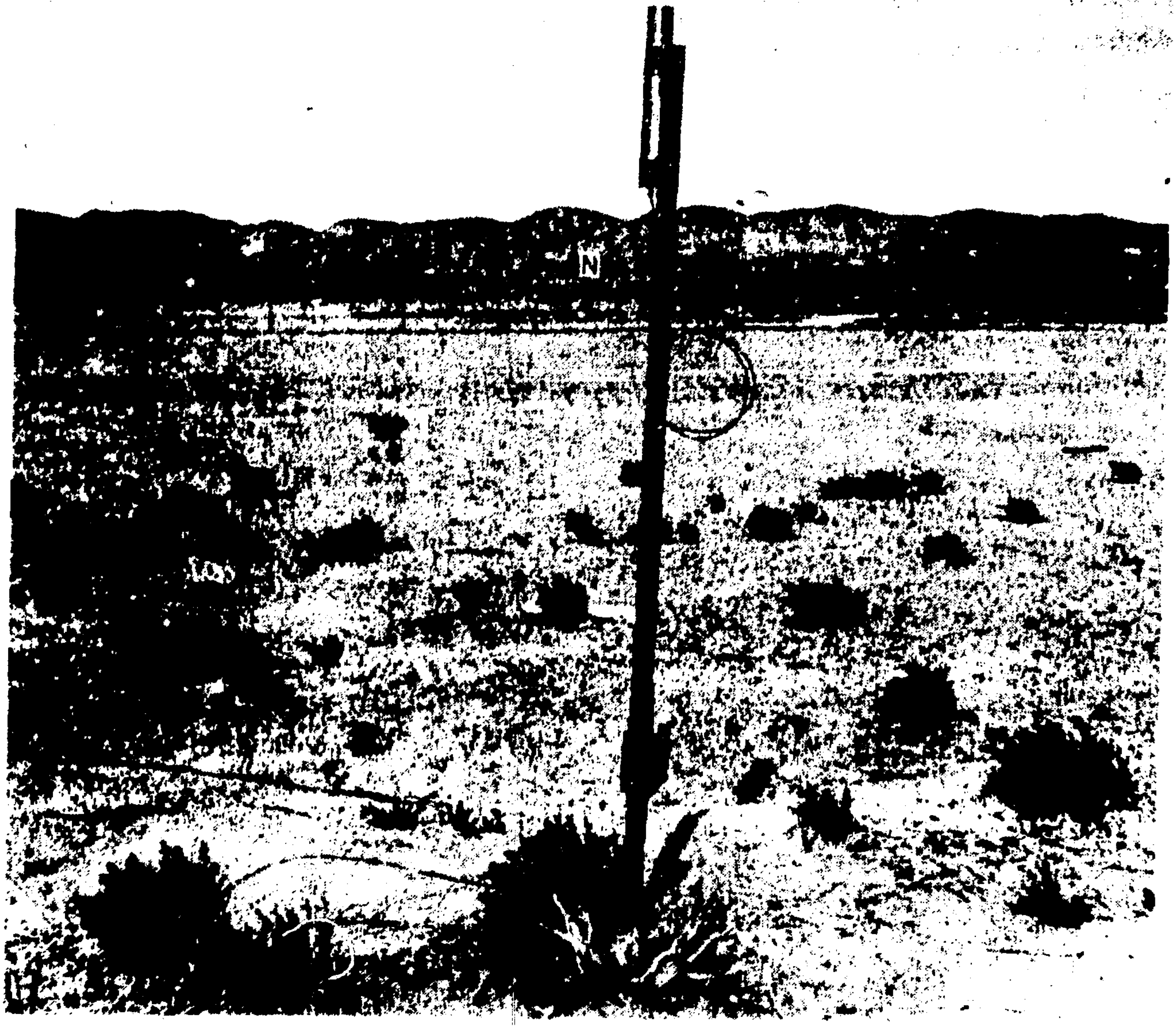
PLATE CHARACTERISTICS FOR
A TYPICAL "NEHER-WHITE" PROBE
I (μa)
PLATE CURRENT
185
175
165
155
145
135
125
115
105
95
85
75
65
55
45
35
0.1mr/hr
1mr/hr
10mr/hr
100mr/hr
1r/hr
10r/hr
100r/hr
1Kr/hr
RADIATION FLUX

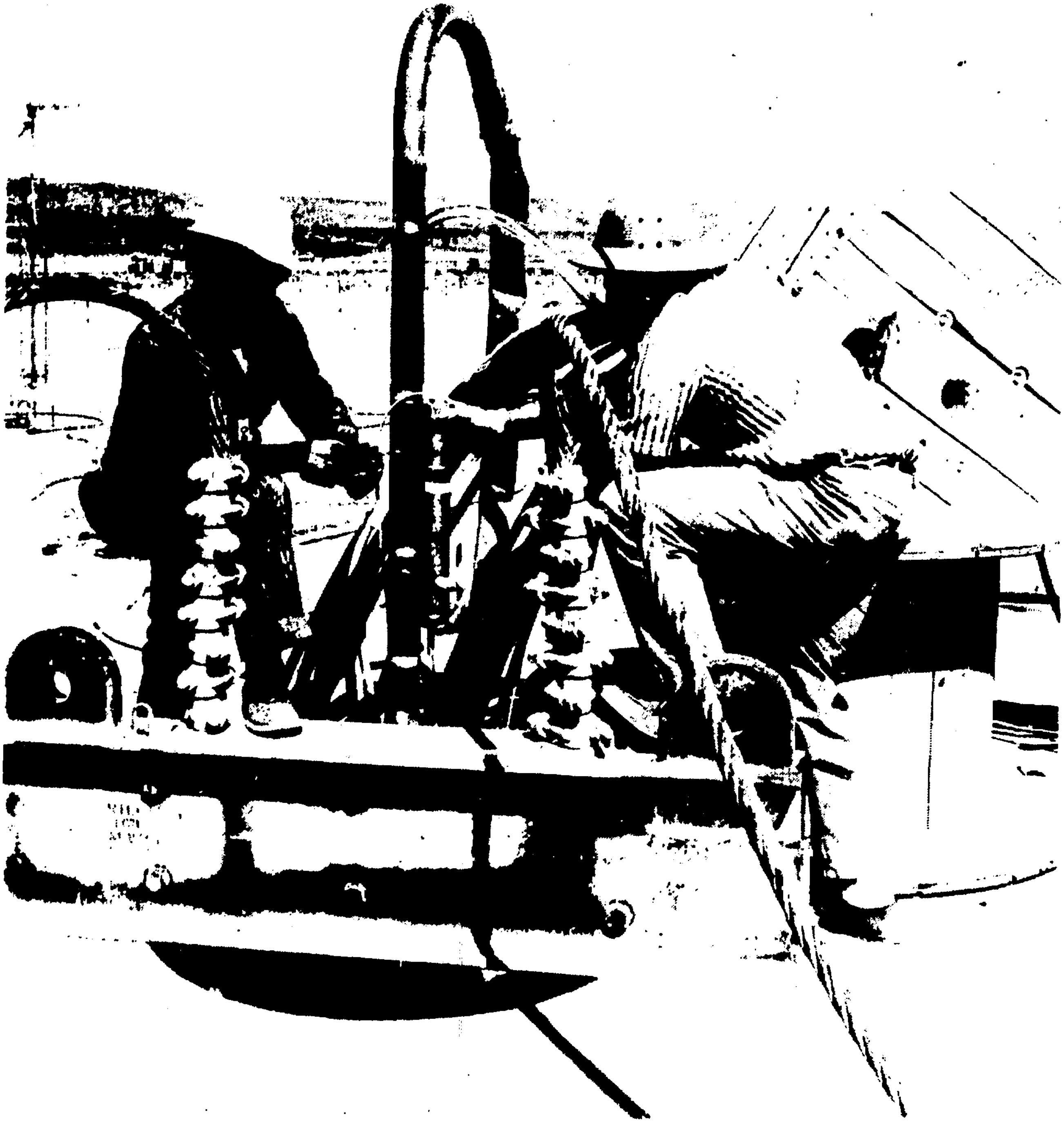

OUTPUT CHARACTERISTICS FOR A TYPICAL "VORP"
V (VOLTS)
OUTPUT VOLTAGE
4.0
3.0
2.0
1.0
0.1mr/hr
1mr/hr
10mr/hr
100mr/hr
1r/hr
10r/hr
100r/hr
1Kr/hr
RADIATION FLUX


LOS ALAMOS SCIENTIFIC LABORATORY
LOS ALAMOS NEW MEXICO
Los Alamos Scientific Laboratory
OF THE UNIVERSITY OF CALIFORNIA
CM 1 2 3 4 5 6 7 8 9 10 11 12 13 14 15


VORP
lasl
dwg # 4Y-168362
MERCURY ENERGIZER
No.
Los Alamos Scientific Laboratory
OF THE UNIVERSITY OF CALIFORNIA
CM.
1 2 3 4 5 6 7 8 9 10 11 12 13 14


REMOTE UNIT


N



Question (Jerry Engstrom, RECO):

Do you have any accoustical sensitivity problems for that instrument?

Answer:

If we do, it hasn't been noticed. Most of them are hanging from the wall and they don't follow the vibration. Eventually it apparently settles down. You can walk up to an instrument and tap it and no apparent problems, so if they're there you don't know it.

Question (James McLaughlin, D.O.E., New York):

Do you have a commercially available electrometer system in this, or is it homemade, and if so, do you have a technical report on that aspect only?

Answer:

There's no thing, really, as an electrometer package, and there's a 1 FET amplifier, which is a commercially available transistor, that you say in the picture in the little brass cage, and that's in the report that's labeled--it's an MEM 520. My comment that that particular FET manufacturer's rating, I think, says gate leakage less than 10^{-12}. We've found that this is more like there typical rating is 10^{-14}. We've used hundreds of them, and as far as I know we haven't found any that weren'e close to that point. We've never selected any.

TECHNICAL SESSION D

AREA MONITORING

Chair: Jack Coogan

ION CHAMBER AREA MONITOR FOR LOW LEVEL SCATTERED X-RAYS

BY Richard W. Fergus & McLouis J. Robinet

Presented at

Health Physics Society Eleventh Midyear Topical Symposium

on

RADIATION INSTRUMENTATION

January 17-19, 1978

San Diego, California

An economical, high confidence instrument was developed to fill the need for radiation area monitors in laboratories using low energy x-rays such as in x-ray diffraction work. The instrument is designed to detect increases in background around such equipment caused by scattered radiation. Exposure rates close to the open port of x-ray tubes used for x-ray diffraction studies are typically the order of 10^3 to 10^6 R/minute, but a few meters away the scatter associated with a potentially serious malfunction or an improper procedure might change the normal background by less than a few tenths of an mR/hr.

Direct monitoring is not practical because these beams may occur in many different locations. Many situations result from equipment malfunction or operator misadjustment, and cannot be anticipated with certainty. Therefore, an indirect monitoring method has been used which is based on detecting the scattered radiation from these beams from a single monitor location (see Typical Application Fig. 1). Various measurements of a typical situation indicated that the monitor should alarm on 0.1 mR/hr of 10 Kev x-rays for detection of possible hazardous conditions.

An economical monitor has been assembled with a polyethylene bottle "ion chamber" and appropriate electronic package. The chamber wall density thickness is 150 mg/cm^2 with a 25% attenuation to 10 Kev x-rays. Since these monitors are not subject to physical abuse, the chamber construction has minimal strength. In most installations, the monitors have been hung from the ceiling with a gimbal mounting arrangement. The bottle cap, which is bolted to the monitor cabinet, becomes the mounting for the chamber shell (bottle) and the center collector. The collector is a self-supported stiff wire attached to the center of a BNC connector and mounted in the bottle cap center. This connector is insulated from the cabinet by the bottle cap so that the connector shell can be biased at a guard voltage to reduce leakage effects. A conducting surface can be formed on the bottle inner surface by either vaporizing aluminum or hand-rubbing graphite into the polyethylene. The aluminized surface on the chambers currently in used is 1 to 4 microns thick. The conducting surface is connected to the cabinet or circuit ground.

The circuitry can be divided into several sections as shown in the Block Diagram (fig. 2). The chamber amplifier is "floated" at the chamber bias voltage. A buffer amplifier stage is used to translate the chamber output to ground reference. Two comparator circuits sense this amplifier output to generate alarms from high level and transient or rate-of-rise conditions. Visual, audible, and external alarm indications are provided. Power requirements are supplied from two regulated sources. The chamber bias supply also has controlled turn-on and turn-off rates to protect the input circuitry from chamber charge and discharge currents.

An enhancement mode metal oxide field effect transistor (MOSFET) is used for the chamber amplifier. The common source circuit (see Simplified Input Diagram - fig. 3) serves both as a current amplifier and bias source for the chamber. In this configuration, the gate and drain voltages are equal at the operating point (no ion chamber current). This point is selected for a zero temperature coefficient gate-to-source characteristic - typically 5 volts gate-to-source voltage and 0.4 ma. drain current. With 100 volts bias supply, the resulting chamber bias is about 95 volts. The relatively high load resistance (R7) required to determine the operating point also provides an open loop gain in excess of 100. In the closed loop condition, the feedback current thru R1 will tend to maintain the gate voltage constant. Collected ion current will tend to lower the gate voltage and the MOSFET will respond with a positive drain voltage change until the current thru R1 is equal to the chamber current. Although chamber capacity effects are reduced by the feedback action, the parasitic capacitance (about 3 picofarad) across the feedback resistance (10 gigiohms) produce a 30 second input time constant. The static stage performance is essentially determined by the feedback resistance (R1). Since the gate voltage is maintained constant, a guard voltage is provided with a simple adjustable voltage control (R4). The current thru R7, which also represents the bias voltage, is returned to the audible annunciator driver. Absence of this current will sound the annunciator indicating bias supply failure.

The buffer amplifier (A2) translates the chamber amplifier output from the bias voltage level to a ground reference. A current mode operational amplifier, which generates a voltage output from the difference between the two input currents, provides this function. Essentially, this stage is a balanced input amplifier which responds to the difference between the bias supply voltage and the MOSFET drain voltage as sensed by the currents thru R5 and R6. With feedback current thru R12 and adjustable balancing currents from circuitry not shown in the simplified diagram, the buffer output can be set to an appropriate voltage with reference to ground. Only the MOSFET drain voltage changes as caused by input ion chamber current will be reflected by the buffer output voltage.

Referring to the Functional Rate-of-rise Diagram (fig. 4), comparator (A3) switches to an alarm state when a positive change of input level is detected. Positive feedback thru R29 - C8 maintains the alarm output for a minimum time period. For slowly changing input levels (background) this circuit is inactive. For instantaneous changes, the circuit operation simplifies as shown. The actual operation is somewhere between these conditions. Assuming a step function input change, the chamber current is integrated by the effective shunt capacitance (about 3 pfd.), amplified by A2, and differentiated by the capacitor (C7). The differentiated current is compared with the constant current thru R26 to determine the alarm condition. With the alarm current constant set for 1 μa. and a one gallon chamber the nominal sensitivity of the rate-of-rise is about 0.1 mR/hr. Test results indicate that the level increase must be sustained for at least 3 seconds for this sensitivity.

Three annunciator functions are driven by both the high level and rate-of-rise comparators. Internal lamps and Sonalert provide local warning. An isolated relay contact set is available for external circuitry. This relay is normally operated by the current thru the extinguished internal alarm lamps thereby providing some fail safe features against power or lamp failure. A circuit has been included which allows for a temporary inhibit of the external alarm circuit. When this feature is activated by a momentary push button, the external relay contact set is maintained normal for about 10 minutes regardless of the alarm condition.

All instrument power is supplied thru an isolation transformer. The bias supply has a controlled turn-on turn-off feature to limit chamber charge-discharge current thru the input amplifier and thereby protect the MOSFET. Referring to Simplified Bias Supply Diagram (fig. 5), when power is first applied, the shunt regulator (Q5) will be driven by the operational amplifier to maintain the C12 current equal to the R37 current. As a result, the C12 voltage will change at a constant rate until the zener (D9) starts to conduct. An equilibrium is reached when the zener current is equal to the R37 current. The turn-on time is about 90 seconds. The bias supply load resistances and C12 form a time constant which provide a slow bias supply decay when power is removed.

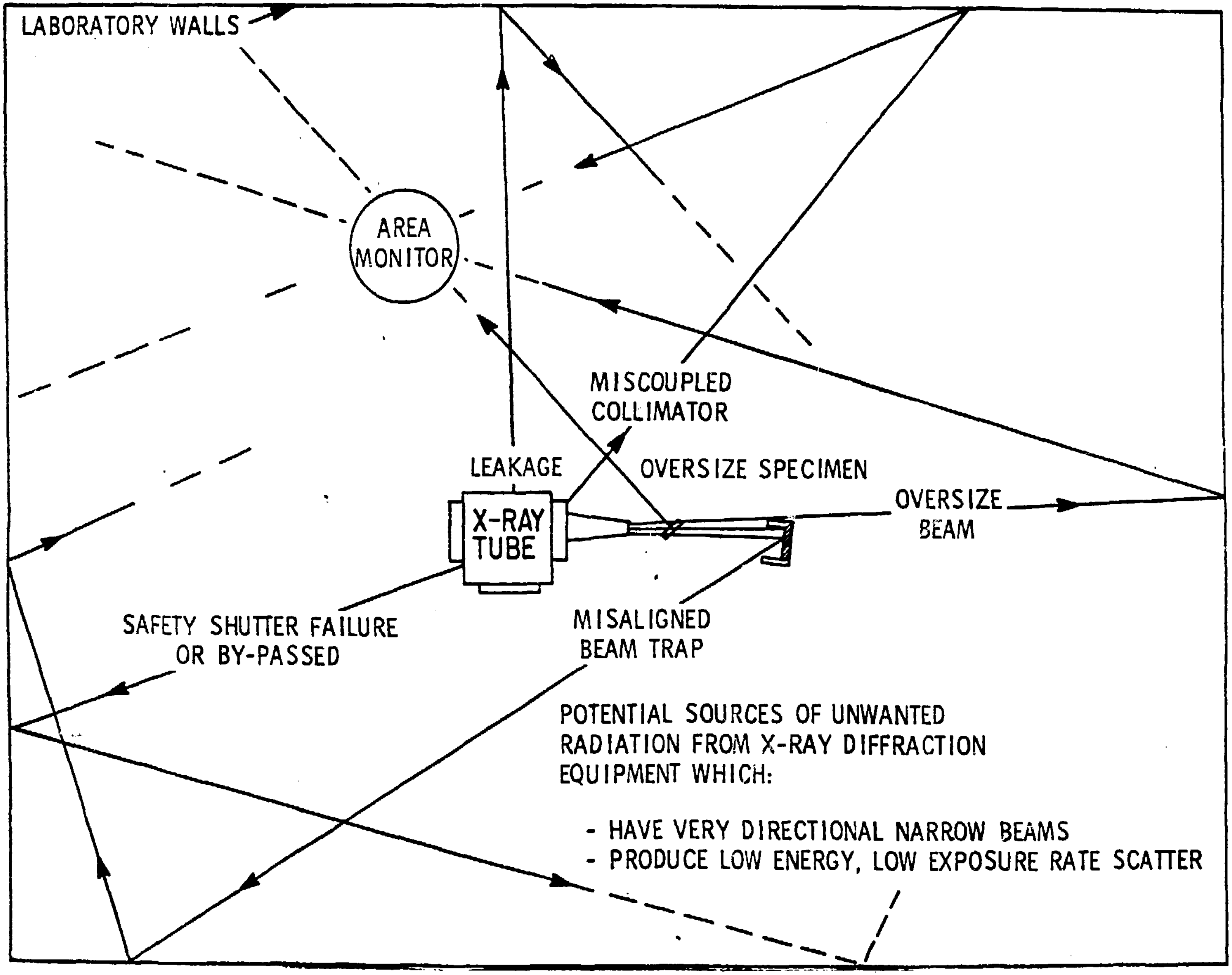
As shown in the Monitor Schematic (fig. 6), a semiregulated source for the low voltage supply is provided by C4, C5, D2, and D3. The negative cycle capacitor (C3) current is conducted by the forward diode action of the zener (D3) while the positive cycle current charges C4 thru D2 until the zener voltage of D3 is reached. Any excessive positive cycle current is conducted by the zener action of D3 thereby maintaining the C4 voltage relatively constant. This voltage source is regulated by the series pass transistor (Q2) and amplifier (IC1b). An equilibrium is attained when the zener (D1) current and R19 current are equal. The regulated output is the power source for all operational amplifiers, including this amplifier, and various reference levels throughout the circuitry.

The Monitor Schematic (fig. 6), describes the complete circuitry that is contained on the circuit board. Circuitry contained within the various dotted areas can be deleted without affecting the basic operation of the monitor if those options are not required. Very seldom are all options used in a monitor so the monitor is labeled with an appropriate subscript to indicate options included: H for 100 volt bias supply, R for rate-of-rise alarm, X for external alarm contacts, I for external alarm timed inhibit, and M for panel meter.

The initial monitor adjustment is somewhat unconventional and can be accomplished by the following procedure. If the panel meter option, (M1 in fig. 6) has not been included, a 100 μa test meter should be connected to J1. The monitor power is applied with the chamber disconnected. During this turn-on time, the alarm control (A) is set to the maximum CW position and the zero control (Z) can be adjusted to maintain an on-scale meter reading. The circuitry has stabilized when the meter reading does not change for a one minute period. The first power-on stabilization may take up to an hour. After the circuitry has stabilized, the guard control (G) is adjusted for zero voltage differential between the guard control arm and the MOSFET drain. Each adjustment will disturb the input charge distribution which must be allowed to recover before the next adjustment is attempted. After the guard voltage has been adjusted, the zero control can be set for the appropriate meter reading---zero or 10 μa on the panel or test meter respectively. The monitor power is removed and, after several minutes for the bias supply to decay, the chamber is connected. The chamber should never be connected or disconnected with the monitor power on or immediately after removing power. Several minutes after power is reapplied, the panel meter can be calibrated with an appropriate radiation field with adjustment of the range control (R).

When setting the alarm level with the alarm control (A), the circuitry must stabilize at the alarm level field to assure the rate-of-rise function is not generating the alarm output. The rate-of-rise sensitivity is not adjustable but can be modified by changing R26. The value shown has been selected emperically for the maximum practical sensitivity of 0.1 mR/hr. Smaller values of R26 will decrease the sensitivity. A high alarm level of 0.3 mR/hr has been found satisfactory for all units installed to date.

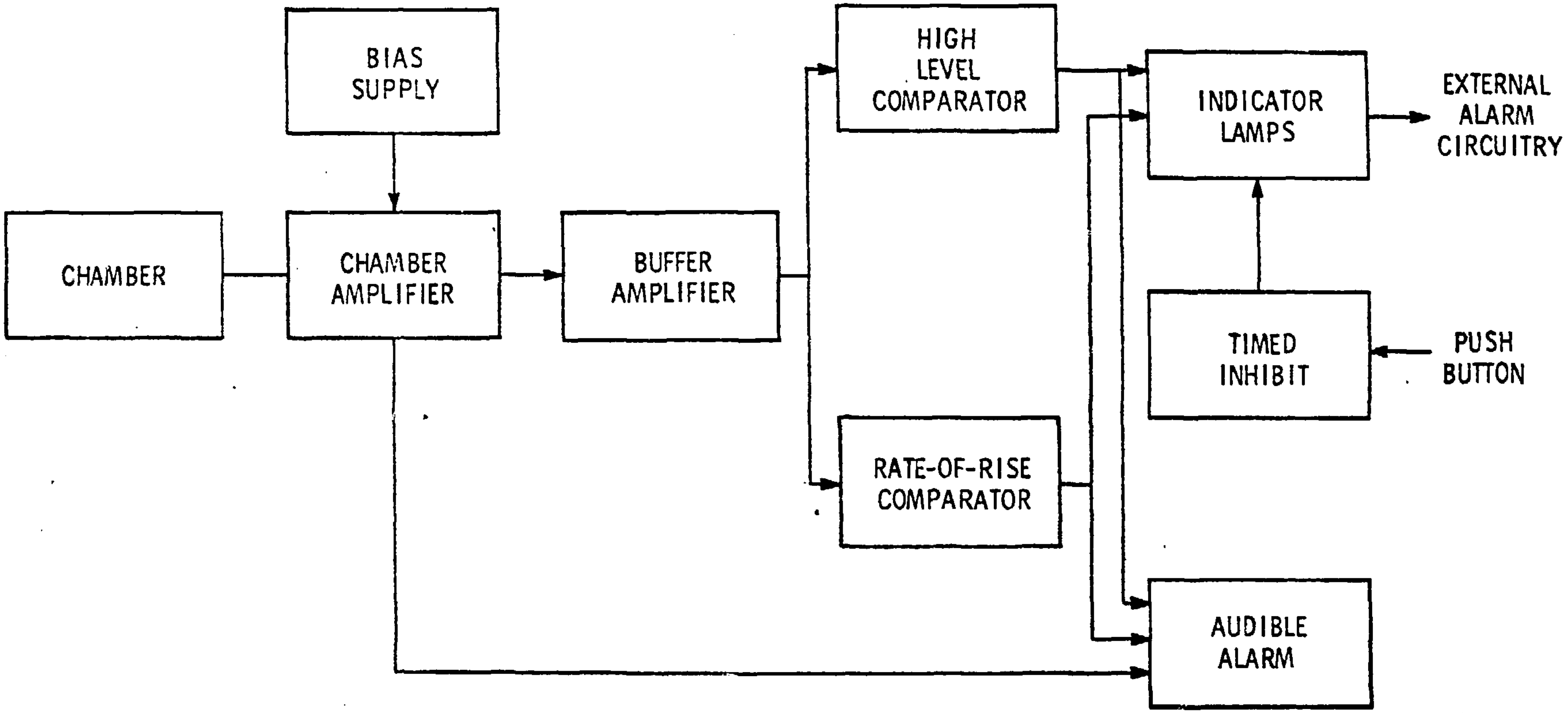
Thirty-five units are in service at this time and have performed to the design criteria. Some of the units have operated for over 4 years without attention. They have been valuable in catching poor techniques and are now well accepted by the users. Although the monitor was developed for x-ray diffraction installations, it has features which have been useful for other area monitoring applications.



POTENTIAL SOURCES OF UNWANTED RADIATION FROM X-RAY DIFFRACTION EQUIPMENT WHICH:

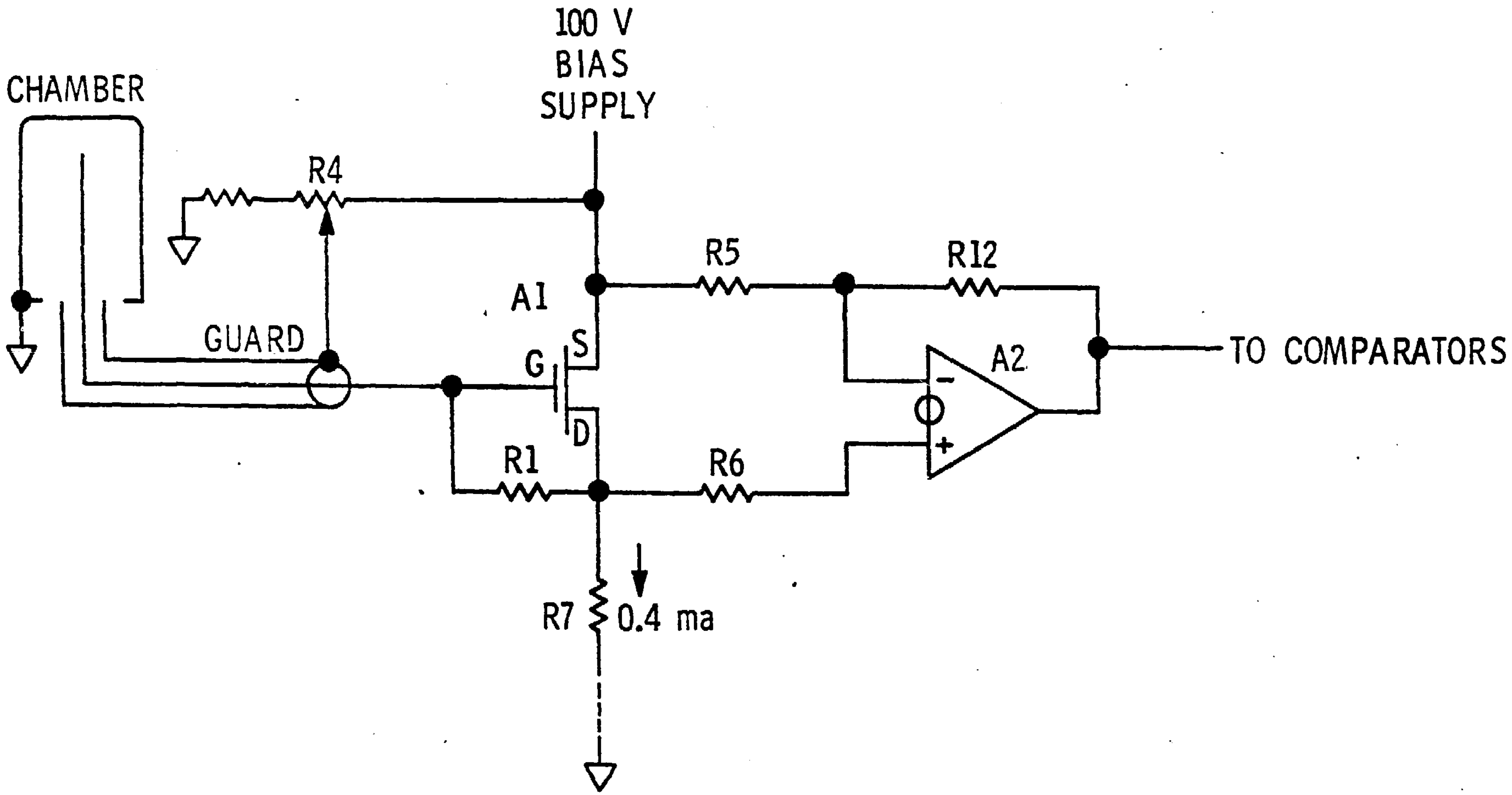
- HAVE VERY DIRECTIONAL NARROW BEAMS
- PRODUCE LOW ENERGY, LOW EXPOSURE RATE SCATTER

FIGURE 1 - TYPICAL APPLICATION



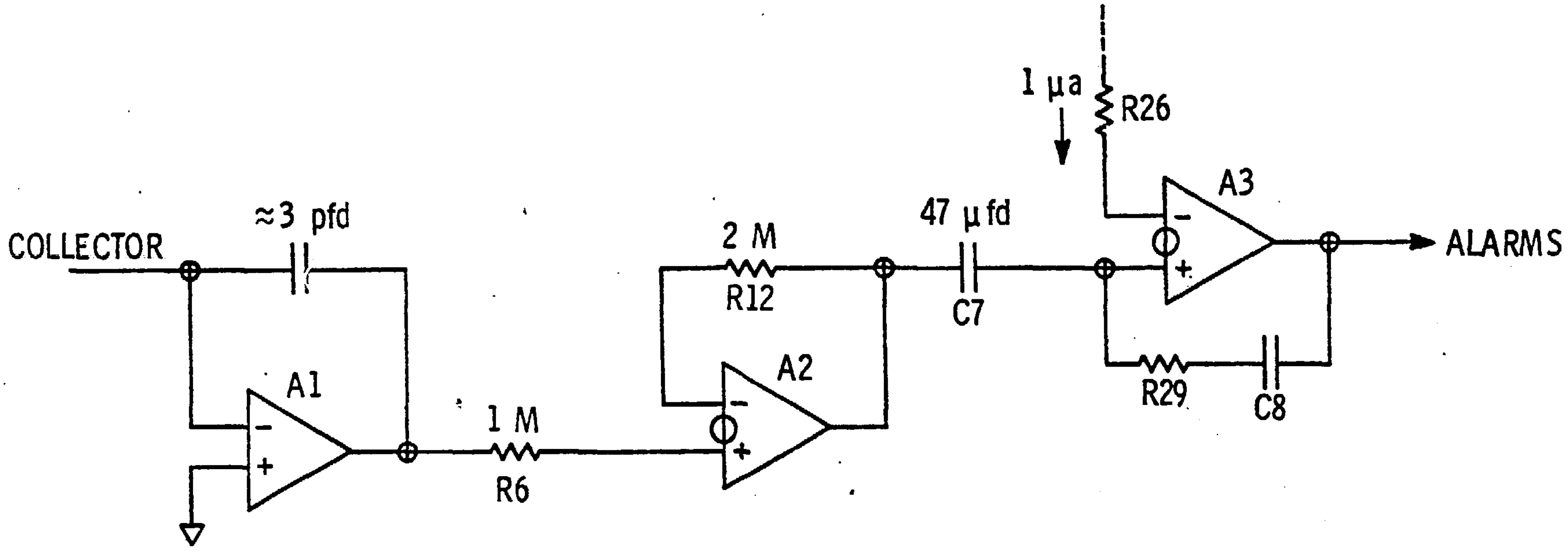
BLOCK DIAGRAM

FIGURE 2



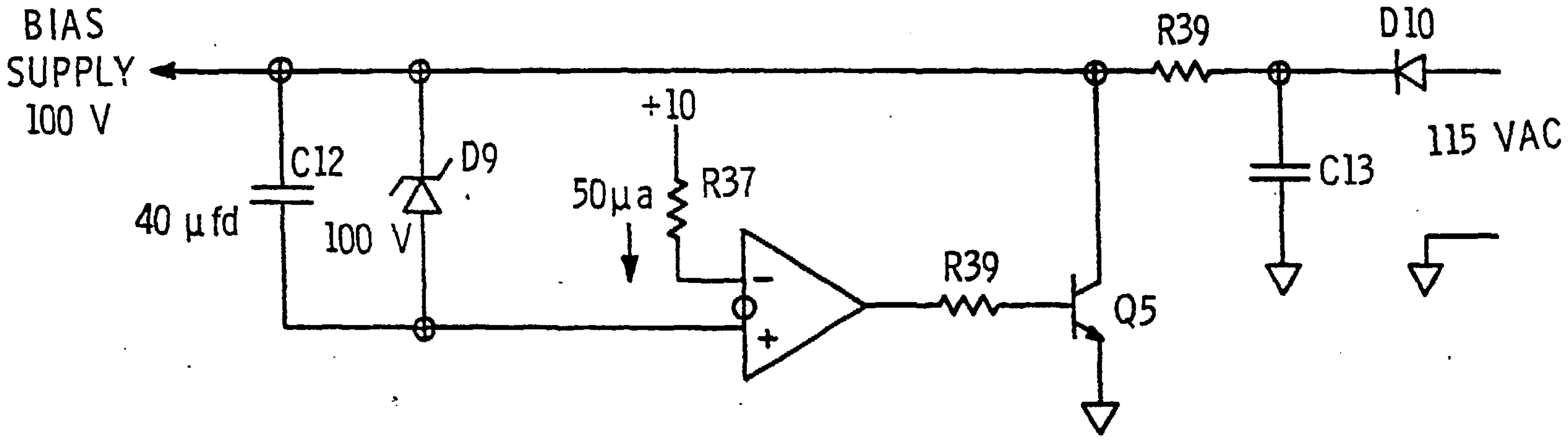
SIMPLIFIED INPUT CIRCUITRY

FIGURE 3



FUNCTIONAL RATE-OF-RISE DIAGRAM

FIGURE 4



SIMPLIFIED BIAS SUPPLY DIAGRAM

FIGURE 5

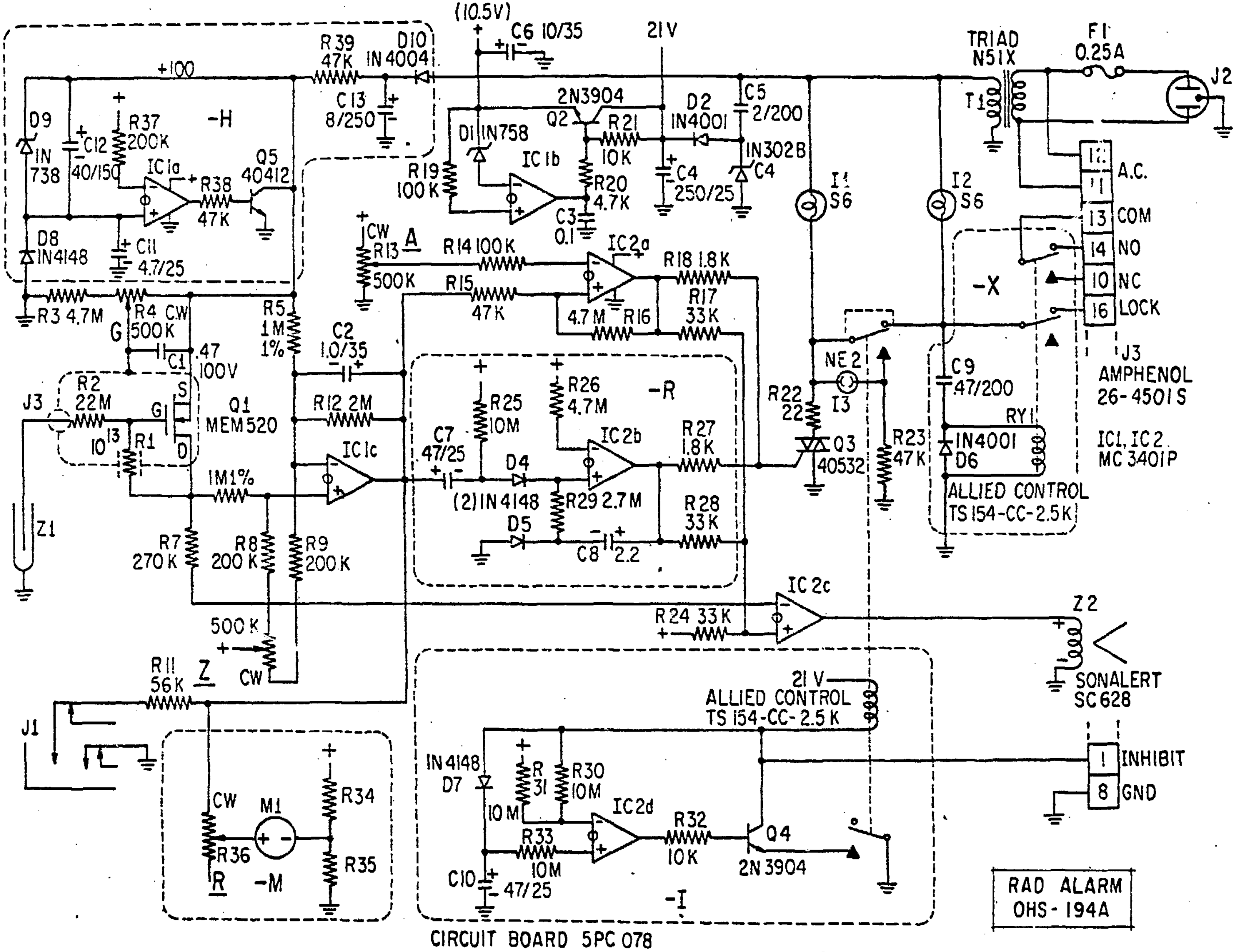


FIGURE 6 - MONITOR SCHEMATIC

A Beam Radiation Monitoring and Protection System for AGS Secondary Beams

Gerald S. Levine

Safety and Environmental Protection Division, Brookhaven National Laboratory,
Upton, New York

Abstract

A commercially available radiation monitor utilizing a scintillation detector has been modified for charged particle beam monitoring. The device controls access to secondary beams of the AGS and limits beam intensity.

The system is briefly outlined and operating experience with many units over a span of several years is described.

I. Introduction

This paper will describe the modifications of and operating experience with a commercially available radiation monitor, utilizing a scintillation detector, for charged particle beam monitoring. The device controls access to secondary beams of the AGS and limits beam intensity.

II. Rationale

A high energy accelerator has a complex of experimental beams (see Fig. 1) for reasons of economy and efficiency. At the Brookhaven National Laboratory 30 Gev Alternating Gradient Synchrotron (BNL AGS) Radiation Safety Guidelines have been adopted which attempt to maximize access by experimenters with a minimum of potential hazard.

For those cases where beam lines do not ordinarily present a hazard, but where dose rates may exceed radiation protection standards when changes in the beam unavoidably occur, the Guideline calls for an active security system, i.e., one which senses increase of beam intensity or dose rate.

Figure 2 illustrates a portion of the General Guideline for AGS Radiation Security System. Columns 2 through 4 define the instantaneous dose rate, beam intensity and steady state dose rates for that radiation classification. The next two list the enclosure required. The last two columns specify the mandated minimum security system and its purpose. Several elements of the Guideline have been omitted for clarity of the illustration. They are Access and Potential Radiation Under Abnormal Conditions, which modify the level of control.

III. Technical Requirements

The minimum beam density to be monitored is about 1000 particles per square centimeter per second over an area of approximately 1 to 100 square cms. A dynamic range of at least three orders of magnitude with a time constant no longer than one second is necessary. The frequent presence of magnetic fringe fields and the stipulation that a minimum of scattering materials be introduced into the beam line are further considerations.

Three levels of response; failure, warning, and alarm are recommended. Operation with the detector at a distance from the circuitry, proven reliability, fail safe circuits, modular construction, and compatability with external equipment connection are desirable features.

Adaptation of a commercially available instrument or design of a custom system are alternatives which have a mix of advantages and disadvantages. The urgent need for a workable scheme and the acute shortage of engineering effort dictated that the first alternative be chosen if at all possible, despite inevitable compromises of performance. Extensive experience with the NMC GA-2TO indicated that minor modifications would make it suitable for this application.

IV. The Solution

The Nuclear Measurements Corp. GA-2TO is a transistorized gamma alarm with a scintillation detector (see Fig. 3). A dust-tight case encloses the entire instrument, though the detector may be operated remotely at distances as great as several hundred feet. The instrument range is five decades. It has three adjustable levels of annunciation; failure at the low end, alert at an intermediate signal level, and alarm at the high level. Relay contacts, lights and a bell provide control, visual, and audible indication of these states. A light source is incorporated in the detector to provide a continual check of system operations. The monitor employs negative feedback to control detector high voltage, giving a quasi-logarithmic response.

The following alterations and additions are made:

A scintillator and light-pipe tailored to a particular experiment replaces the usual scintillator button. A long signal cable is added so that the detector can be installed at the appropriate location in the beam line.

The control relay currents are connected to the AGS security system to drive "Beam On" indicators and interlock beam operations.

Figure 4 shows the portion of the High Energy Unseparated Beam containing the beam monitor. The instrument case is mounted on the shield wall at the extreme right of the photograph. Immediately to the left of it is the security system control box with which it interfaces. A "Beam On" indicator appears in the center of the picture, under the beam pipe.

The detector, an RCA 6655 photomultiplier, the signal cable with connectors, a 2 inch wide by 24 inch long lucite light pipe, and a 2 inch x 4 inch x ½ inch NE-102 scintillator are mounted horizontally at the left. The elements are cemented together with an epoxy, which provides good light transmission and mechanical durability. All are wrapped with household grade aluminum foil to provide internal reflection, and finally covered with black vinyl plastic electrical tape to exclude any light from external sources.

Set up involves the following steps:

1. Installing the detector at the appropriate location in the beam line
2. Mounting the instrument enclosure and connecting it to the security system.
3. Checkout to establish
   - (a) normal and fault radiation levels,
   - (b) establishment of alarm levels,
   - (c) verification of required operations.

V. Experience

I will review our experience gained over about three years with six units.

A properly operating system performs the required functions very satisfactorily. Difficulties arise when beam signal is not signficantly greater than the area background.

Operational problems fall principally into the following categories:

1. The occasional necessity for locating the photomultiplier in the fringe field of a magnet.

2. Until recently, winter 1977-78, three different groups were involved in the total process, modifications by the Safety Instrumentation Shop, installation by the accelerator experimental area support division, and calibration and test by the accelerator safety section. Communications and coordinations were overly complicated, resulting in poor service, and consequently, diminished performance. The system is now a basic element of the experimental services provided the experiment and as such receives full attention by the accelerator department, which performs all but the checkout procedures.

3. A significant number of photomultiplier failures has occurred. This problem has not been restricted to units at BNL. The mode of failure, appearance of the tube and circuit operation lead me to believe that we are burning up the photocathodes. The feedback scheme employed to reduce gain for large signals masks the presence of damaging light levels at the photocathode. The tube replacement frequency is acceptable so the problem is being ignored.

VI. Summary

The immediate need to provide beam radiation monitoring in secondary external beams of the 30 Gev AGS has been satisfactorily met by adaptation of a commercially available instrument at reasonable cost ($1100/unit) and a minimum of engineering effort. The most significant difficulty with this solution has been the lack of clear definitions of responsibilities betwen user and technical service groups.

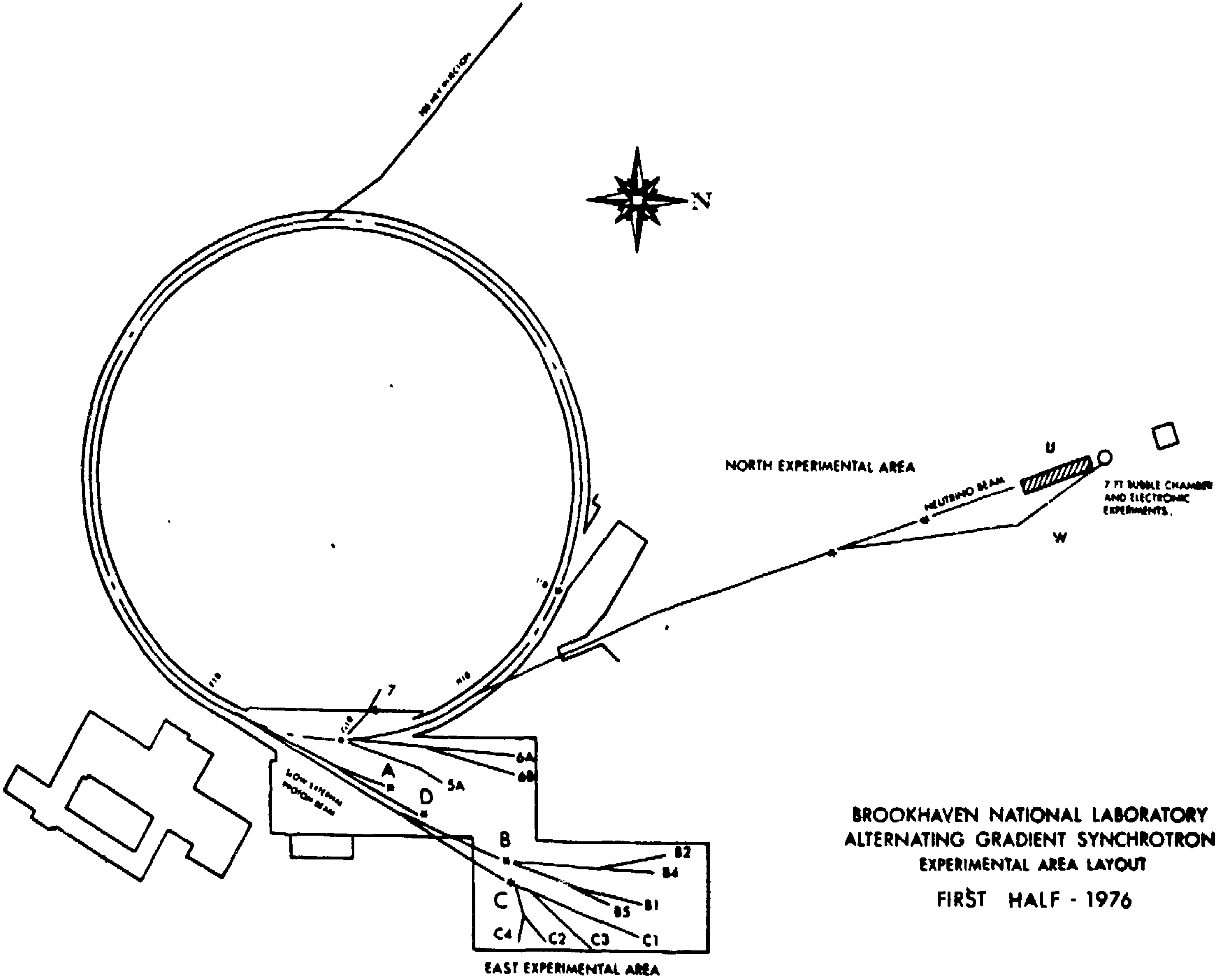

200 MeV INJECTION
N
NORTH EXPERIMENTAL AREA
NEUTRINO BEAM
U
W
7 FT BUBBLE CHAMBER
AND ELECTRONIC
EXPERIMENTS.
7
6A
6B
5A
A
D
SLOW EXTERNAL
PROTON BEAM
B
B2
B4
B1
B5
C
C4
C2
C3
C1
EAST EXPERIMENTAL AREA
BROOKHAVEN NATIONAL LABORATORY
ALTERNATING GRADIENT SYNCHROTRON
EXPERIMENTAL AREA LAYOUT
FIRST HALF - 1976

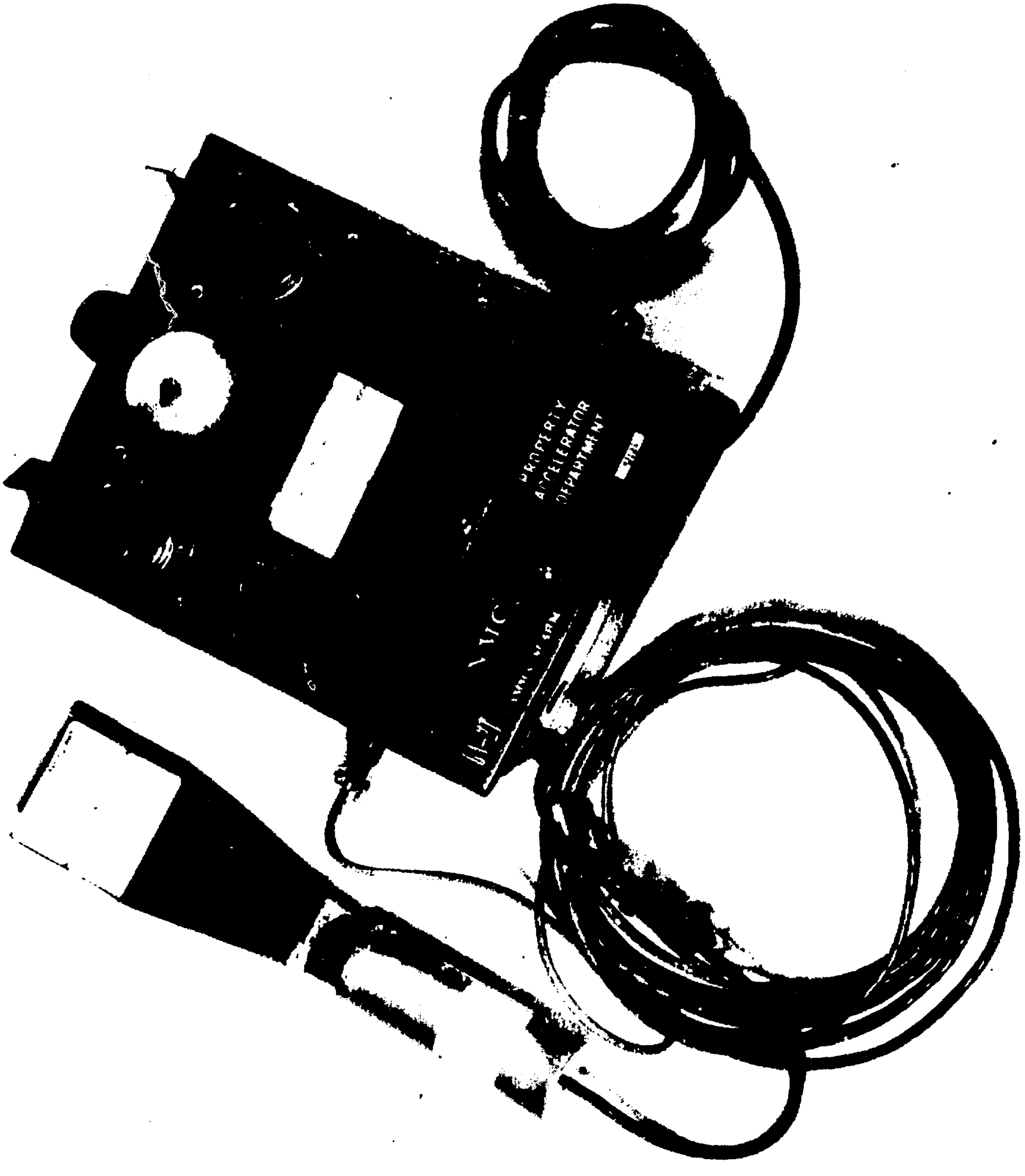
Figure 1

GENERAL GUIDELINE FOR AGS RADIATION SECURITY SYSTEM

CLASSIFICATION & APPLICATION

Area	Allowable Radiation			Enclosure		Minimum Security System (2)	Purpose of Security System
	Area R/hr	Beam p/sec/cm²	Small Source Residual @ 3" R/hr	Barrier	Gate		
Extremely High Radiation	>300	>3x10^{7}	>3000	Impregnable	Primary	**A—Hardwire, Failsafe, Dual**	Preventing Access or Beam
	<300	<3x10^{7}	<3000	Fully Enclosed	Primary	A—Hardwire, Failsafe, Dual Not Specified (3)	Controlling Access or Beam
High Radiation	<30	<3x10^{6}	<300	Walls/ Fixed Fences	Secondary	B—Hardwire, Failsafe E—Active, Failsafe A—Hardwire, Failsafe, Dual	Controlling Access or Beam }Preventing Rise to These Levels
	<3	**<10^{5}**	**<30**	**Fences/ Barriers**	**Locks**	**E-Active, Failsafe,** **E-Active, Failsafe,**	**Controlling Access and /or Beam or Warning**
Radiation	0.1	<3000	<0.3	Noticeable	None	H—Active G—Dual, Active C—Active, Failsafe, Dual A—Hardwire, Failsafe, Dual	Alarm on Radiation }Preventing Rise to These Levels
Control	0.005	<50	<0.005	None	None	None H—Active, C—Active, Failsafe, Dual A—Hardwire, Failsafe, Dual	}Preventing Rise to These Levels

Figure 2


PROPERTY
ACCELERATOR
DEPARTMENT

Question (Bill Wilkdon, Argonne National Laboratory):

There must be times when you want to operate with the target room unoccupied, and high radiation cells in the target room. Is that true?

Answer:

Yes, that's true, but generally these are not in areas like that. Those are caves with gates and locks.

Question (Wildon):

How do you determine whether the target room is occupied or unoccupied?

Answer:

If it's the kind where the hazard is great enough, there is actual physical control on it. The slide, if you see all of it, shows that any time it's possible for anyone to occupy a place like that, and administrative controls--just clear the in area. This system only exists in a place where an intermediate level of hazard could occur--that is, in 99% of the time people could be working there and sometimes the hazard would increase to the point where, in an eight hour day, they would get daily exposure. Not instantaneously. So, in cases where the hazard is great enough, there's just locked gates and clearing procedures. That's another one of the types of controls spelled out. Only at the very lowest end of that table did they have active systems like this. Did I answer your question?

Question (Wilkdon):

Do you have any sort of start up--any alarm that's required to be sounded when you start up? Procedures like that?

Answer:

Oh, yes. The areas I'm talking about mean that people go in and clear them, lock them, and then the beam comes on.

A UNIQUE RADIATION AREA MONITORING SYSTEM*

P. C. Murphy, EG&G, Inc., Las Vegas, Nevada

and

G. C. Allen, U.S. Department of Energy, Las Vegas, Nevada**

I. INTRODUCTION

A change in the way equipment is fielded for the nuclear weapons test program at the Nevada Test Site by Lawrence Livermore Laboratory resulted in a new facility being built to accommodate this operational change. As a result of the new facility, officially designated the North Las Vegas Facility (NLVF), the radioactive source range used in the calibration of weapons test radiation detectors was also relocated to NLVF.

As the design of the NLVF was being finalized, the EG&G Radiation Safety Committee suggested a review of existing radiation area monitoring systems (RAMS) to assure that the new facility would incorporate the most advanced features assuring safe operation and maintaining occupational radiation exposures As Low As Practicable (ALAP) program. The RAMS system used in the old facility was reviewed and compared with currently available technology with the decision being made that it should be replaced even though it was functioning as designed. We concluded a new system would offer improved operational and safety characteristics.

Criteria for a new system were generated by those responsible for safety and operation of the source ranges. Basically, the specifications required dual-door interlocks and prominently displayed, easily readable digital displays of radiation levels installed in both the control room and the radiation area itself. One of the door interlocks was to be controlled by ionizing radiation intensity; it was considered a go-no-go system since its function was not to serve in the display mode but only to respond to a preset radiation level. Alarms were to be both visual and audible. A visual indication at the control panel was to be incorporated indicating an interlock or alarm had been bypassed or that a high radiation level was present in a source range. A malfunction indicator was to be incorporated that not only indicated a malfunction, but also disabled the malfunctioned source.

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*Work performed for U.S. Department of Energy under Contract No. EY-76-C-08-1183.

**Work performed by author while employed by EG&G, Inc.

Health Physics Instruments, Inc., of Santa Barbara, California, responded to these criteria, designing and manufacturing four custom systems as described below.

General Description

The Remote Area Monitor System is designed to monitor four radiation areas with two independent systems in each area. Each system consists of power supplies, four ionization chamber detectors, and four analog and digital circuits. The first system, designated the go-no-go (GNG) system, controls the warning beacons, horns, annunciation panel, and interlocks. It can be set to trip at radiation levels from 5 to 500 mR/h. The second system, designated the Digital system (DIG), presents a quantitative dose rate indication at the console and in the radiation area with large seven-segment LED numerals. The systems have their own fuses, power supplies, and logic boards although they are mounted in the same main chassis.

The main purpose of this system is to provide redundancy in the control of the radiation areas and to provide fail-safe operation to the extent practical. The redundancy is provided by the two independent systems plus the normal electrical interlock system tied to the radiation-producing device and prevents its being operated if the predetermined set of criteria are not met. All circuits for warning devices and source interlocks are operated in a fail-safe mode. In the event of a power or circuit failure, warning devices will be activated and the source interlock will be in a safe condition. Thus, the radiation-producing device will be inoperative if the monitor fails or is turned off. The only interface between the two systems is the malfunction circuit.

In addition to providing circuit fail-safe operation, each system periodically holds its data and sends through a test signal from the detector. If the system fails to respond to this signal, a malfunction is noted by a warning light; and if the go-no-go system is at fault, the radiation-producing device is disabled. In the DIG system, this check occurs at each data update time or about every three seconds. In the GNG system, the self-check is initiated every 20 seconds.

II. COMPONENTS

A. Main Chassis Description

The main chassis consists of all power supplies, power distribution, logic, and controlling elements and circuits. All other units are connected to it through ten 19-lug barrier strips on the rear panel. AC power switches, fuses, test points for all power supplies, and switches to disable the warning beacons are mounted behind a locked front panel. The top and bottom panels are removable and the back panel is hinged for servicing. AC power is provided for each circuit through two master switches. Main fuses for each system are provided.

B. Detector Unit Description

The detector is a 1-liter ionization chamber with 1/8-inch lucite walls. It is sealed in an aluminum can and filled with argon to one atmosphere pressure. The varactor bridge electrometer is mounted inside the sealed can, as is the high-megohm resistor and the zeroing magnetic switch. Only low-impedance signals are brought out of the detector can through a sealed nine-pin plug. The chamber is electrodaged in the inside surfaces and has a 1/2-inch-diameter polystyrene center electrode which is electrodaged. A guard ring prevents excessive leakage of the teflon insulator. The high-voltage insulator is part of the lucite base. The chamber cannot be opened. All chambers are identical and operate in the rate mode with a 5 x 10^{10} ohm feedback resistor.

The chamber is mounted on a 6-inch x 6-inch x 6-inch cabinet with both front and back removable. The back is designed to mount on a 4-inch x 4-inch electrical box. A circuit board is mounted on one side and consists of a comparator circuit in the case of the GNG system or an analog-to-digital converter in the case of the DIG system. The adjusting controls are available from the front after the front cover is removed. In both systems, the electrometer zero and gain controls are the important adjustments. The gain control sets the trip level of the GNG system and adjusts the overall calibration of the DIG system.

The comparator system (GNG) is designed with hysteresis through a feedback loop to prevent hunting. The voltage signal from the electrometer is compared with a fixed voltage. By changing the gain on the input signal, the trip level is changed.

The ADC is basically a voltage-to-frequency converter. A train of pulses, the repitition rate of which is proportional to the electrometer output voltage, is fed back to the logic circuits, where it is scaled for a predetermined time interval, converted to dose rate, and presented digitally.

III. OPERATION

With both systems operating, the readouts, horn, and light and annunciator panels should indicate the correct status of all systems. The truth table for the GNG system is:

	Radiation High	Radiation Safe
Door Open	Horn ON Beacon ON Source interlock OPEN	Horn OFF Beacon OFF Source interlock OPEN

	Radiation High	Radiation Safe
Door Closed	Horn OFF Beacon ON Source interlock CLOSED	Horn OFF Beacon OFF Source interlock CLOSED

The door open, closed and Radiation High, Safe LEDs will follow the action of the door and detector. The beacon off LED will actuate from the beacon off switch behind the locked front panel of the master unit. This disables only the rotating beacon in the designated room.

IV. MALFUNCTION INDICATOR

A malfunction LED will come on whenever a malfunction is detected. A failure of any power supply, except the annunciator power supply, or any digital logic within the detector and counting chain will turn the malfunction LED on. If the annunciator power supply fails, all relays will open, turning on the horn and light, dropping out the source, and turning off the annunciator panel lights. On a GNG malfuntion, the source interlock will open. If this feature is not desired, it is possible to change it so the source will not drop out.

SUMMARY

The system has been in use for 1 1/2 years at the EG&G NLV facility. It has functioned as designed. There have been no major problems encountered in its use; the only ones, minor in nature, were IC failures that occcurred when the system was initially put into operation.

The unique features of this system are involved in the use of digital design techniques in its fabrication, the use of go-no-go redundant safety interlock system, and the self-testing feature that assures proper system operation or a positive indication if not functioning properly.

TECHNICAL SESSION E

ENVIRONMENTAL MONITORING II

Chair: Robert Alexander

Introduction by Robert Alexander (Nuclear Regulatory Commissions):

One can find guidance on radioactive effluent monitoring in Regulatory Guide 1.21. We plan to update that guide this year and to incorporate changes resulting from amendments in Appendix I of Part 50 of our regulations. For those not familiar with those regulations, Appendix I provides numerical guides for design objectives and limiting conditions of operations to meet ALARA criteria for light water cooled power reactor effluents. Standard technical specifications that will implement the requirements of Appendix I are now being completed by the NRC staff. General guidance on the environmental monitoring for nuclear power plants is provided in regulatory guide 4.1. More specific guidance on this subject, as contained in regulatory guide 4.8 titled "Environmental Technical Specifications for the Nuclear Power Plants" which was issued for comment in December 1975. Final issuance of this guide has been delayed pending resolution of questions of treatment of non-radiological water quality issues in the technical specifications and development of the seven of standards or model environmental technical specifications. However the guidance on radiological and environmental monitoring that is in the initial version of regulatory guide 4.8 has been revised in response to comments from both the public and the staff and to reflect the requirements of appendix I. Pending final issuance of regulatory guide, this currant guide is obtained in a branch technical position recently issued by the radiological assessments branch of our office of nuclear reactor regulation. Most of the changes from the guidance given in the initial version of regulatory guide 4.8 are in the form of deletions from the environmental monitoring program. The major deletions are as follows: All Sr^{89} and Sr^{90} analyses have been deleted. Collections and analysis of sediment and indicator organisms have been deleted. Meat and poultry sampling and analysis has been deleted, and broadleaf vegetation collection and analysis has been deleted except when milk samples are not available.

Guidance on radiological effluent monitoring for uranium mills is provided in respiratory guide 4.14 which was issued for comment in June of last year. The title of this guide is "Measuring and Evaluating and Reporting Radioactivity in Releases of Radioactive Material in Liquid and Airborne Effluents from Uranium Mills." This guide is current being revised in response to comments. Guidance on environmental monitoring outside restricted areas for these mills is now being developed by his staff. This guidance probably will be issued initially in the form of a branch technical position.

The next two guides I'm going to mention apply to all types of facilities for which environmental monitoring is required by the NRC. Respiratory guide 4.13 is entitled "Performance, Testing, and Procedural Specifications for Thermon Luminescence do Simetry - Environmental Applications". At the response of public comment revision one of this guide was issued in July of last year. This guide endorses an ANSI standard which specifies minimum acceptable performance of TLD's used for environmental measurements, outlines methods to test for compliance, and provides procedures for calibration, field application and reporting. The scope of this guide is limited to TLD measurements of the total X- gamma-ray exposure in the environment. However, the primary interest of the NRC is in the contribution of the licensee's facility of this total exposure. Therefore the NRC is supporting a study to be done this year by the Environmental Measurements Laboratory of DOE, which was formerly the Health and Safety Lab. This study will evaluate methods for subtracting background exposure in order to obtain a net exposure attributable to the licensee's facility. The need for quality assurance is implicit in all requirements for effluents and environmental monitoring. Regulatory guide 4.15 presents the elements of a quality assurance program that should be provided by each organization that performs radiological effluent or environmental monitoring required by the NRC. This guide was issued for comment late last month. It's title is Quality Assurance for Radiological Monitoring Programs, effluent strains within the environment. I wish all of you who are interested in this subject to send in your comments on this guide.

In conclusion I would like to emphasize that the NRC welcomes comments on its guides, and carefully considers them in guide revisions. If you want to know how to obtain copies of our guides, please contact me after this session and I'll let you know.

RADIOACTIVE GASEOUS EFFLUENT MONITORING UNDER HIGH HUMIDITY CONDITIONS

BY
RUSSELL S. ROBERTS
AND
STEVEN J. FERNANDEZ
ALLIED CHEMICAL CORPORATION
IDAHO FALLS, ID 83401

Prepared Under The Auspices of The
United Stated Department Of Energy

ABSTRACT

A continuous monitoring station for sampling HTO, I-129, Sb-125, Ru-106 and radioactive particulates in calciner off-gas has been designed and installed in the Waste Calcining Facility at the Idaho Chemical Processing Plant. The sample off-gas is approximately 85°C and has a dewpoint of 65°C with NO_x present at a concentration of about 10,000 ppm. The high humidity in the sample off-gas presents a formidable problem in using such conventional sampling media as filter papers, silica gels, etc. A new drying process called permeation distillation is used in the design of Perma Pure Dryers which continuously removes water vapor (and HTO) while preventing particulates and all gaseous components from being removed from a sample stream. A Perma Pure Dryer has been installed in the calciner off-gas sampler and the results of the process are reported. Radioactive particulates are removed from the dry sample off-gas stream and continuously monitored using a NaI(Tl) detector.

INTRODUCTION

Industries associated with the Nuclear Power Fuel Cycle have been criticized by both private and federal agencies for increasing the levels of radioactivity already found in the environment. A very real concern of these agencies is the impact that these nuclear industries have upon the environment due to the radiotoxicity of long-lived fission products such as I-129. Therefore, reliable and effective monitors are required to ascertain the quantities and types of radionuclides released to the atmosphere from the various nuclear industries.

Present federal regulations require that high-level radioactive wastes generated by the reprocessing of spent nuclear fuel be converted to a stable, dry solid and stored until a permanent disposal area is found. The Waste Calcining Facility (WCF) at the Idaho Chemical Processing Plant (ICPP) is operated by Allied Chemical Corporation (ACC) and has been routinely solidifying these wastes since 1963. The process off-gas generated by the calcination process passes through a series of clean-up systems including cyclone separators, a venturi scrubber, and HEPA filters. Prior to 1975, this process off-gas was discharged through a 250 ft. stack. Since that time, an extensive Atmospheric Protection System (APS) consisting in part of a condensor, demistor, superheater, prefilter and a series of HEPA filters, has been installed which further cleans the calciner off-gas before discharge to the atmosphere.

The WCF provides a unique opportunity to construct a sampling station which continuously monitors the radioactive material present in the vessel process off-gas stream before discharge to the APS. It is the purpose of this paper to describe a continuous monitoring station which quantitatively measures the radionuclides present in the highly corrosive calciner process off-gas.

HISTORY

The present off-gas monitoring station at the ICPP is the third one that's been installed since 1963. Each modification has produced an improvement and is consistent with the philosophy of developing state-of-the-art sampling equipment in the field of Health Physics and environmental and effluent monitoring techniques. Because of the high humidity associated with the process off-gas, the first two approaches used to sample the radioactivity in the stream consisted of heating the entire sampling station to a temperature of approximately 180°F to prevent condensation in the lines and on the sample filter. Nevertheless, the filter would often become wet and break apart. Quantitative results were also not posssible with these stations because of the possible wicking effects across the cellulose filter. Another problem present was the high temperature required which made sample changeout very difficult.

The present design of the system involves the quantitative drying of the sample off-gas through a process called permeation distillation. This process eliminates many of the previous problems that plagued the other off-gas monitoring stations.

Some general requirements of a process off-gas monitoring station are that it must be:

1. Physically rugged.

2. Reliable with a minimum amount of maintenance.

3. Simple in operation with an ease in sample changeout and capable of continuously monitoring for particulate radioactivity with the versatility to monitor or sample for a number of radionuclides such as I-129, H-3, etc.

4. Accurate in monitoring activity with the ability to quickly ascertain levels so as to allow immediate corrective action if needed.

5. Capable of withstanding:

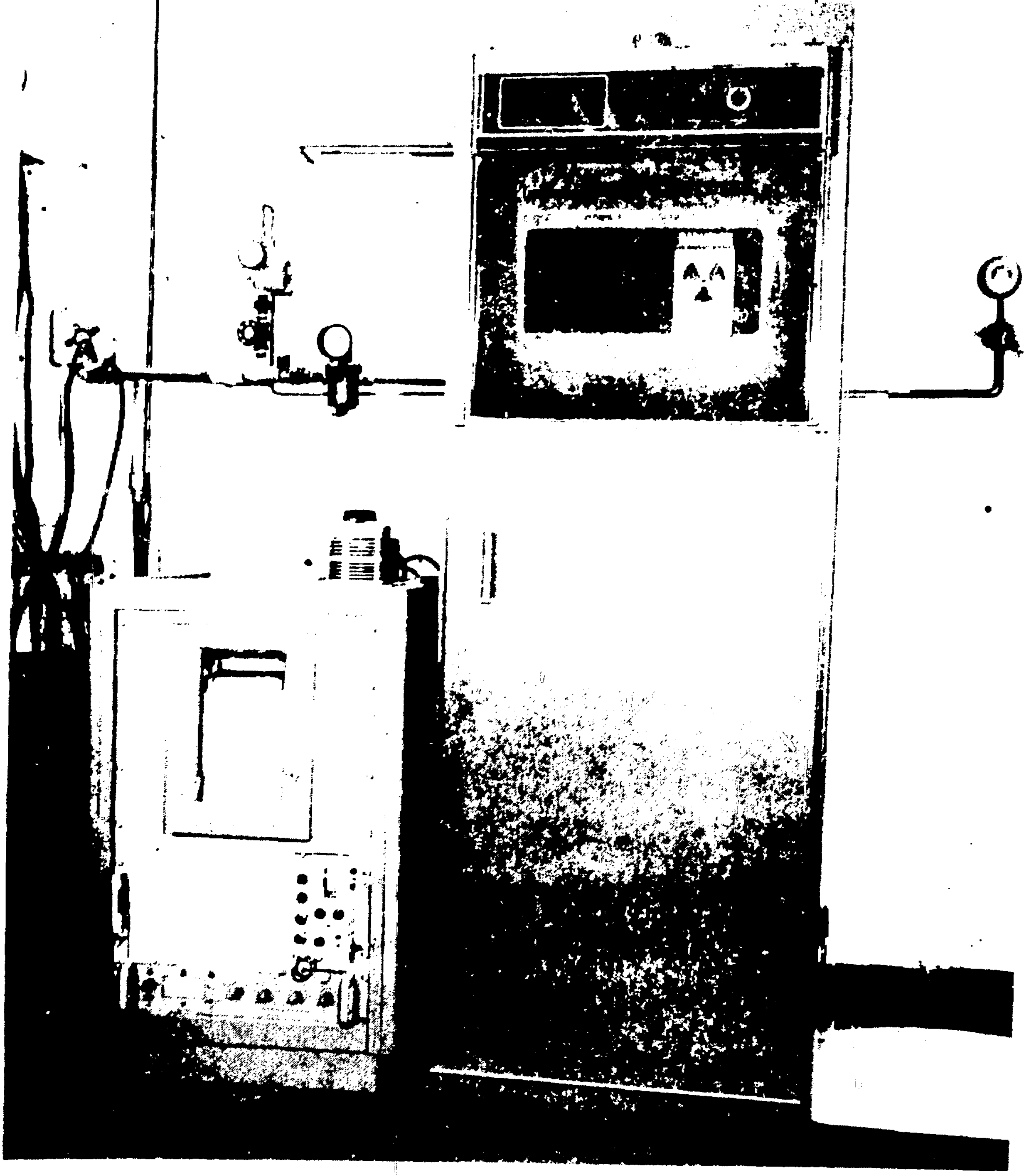
    a. Highly corrosive conditions (>10,000 ppm NO_x)

    b. High temperature (∿200°F)

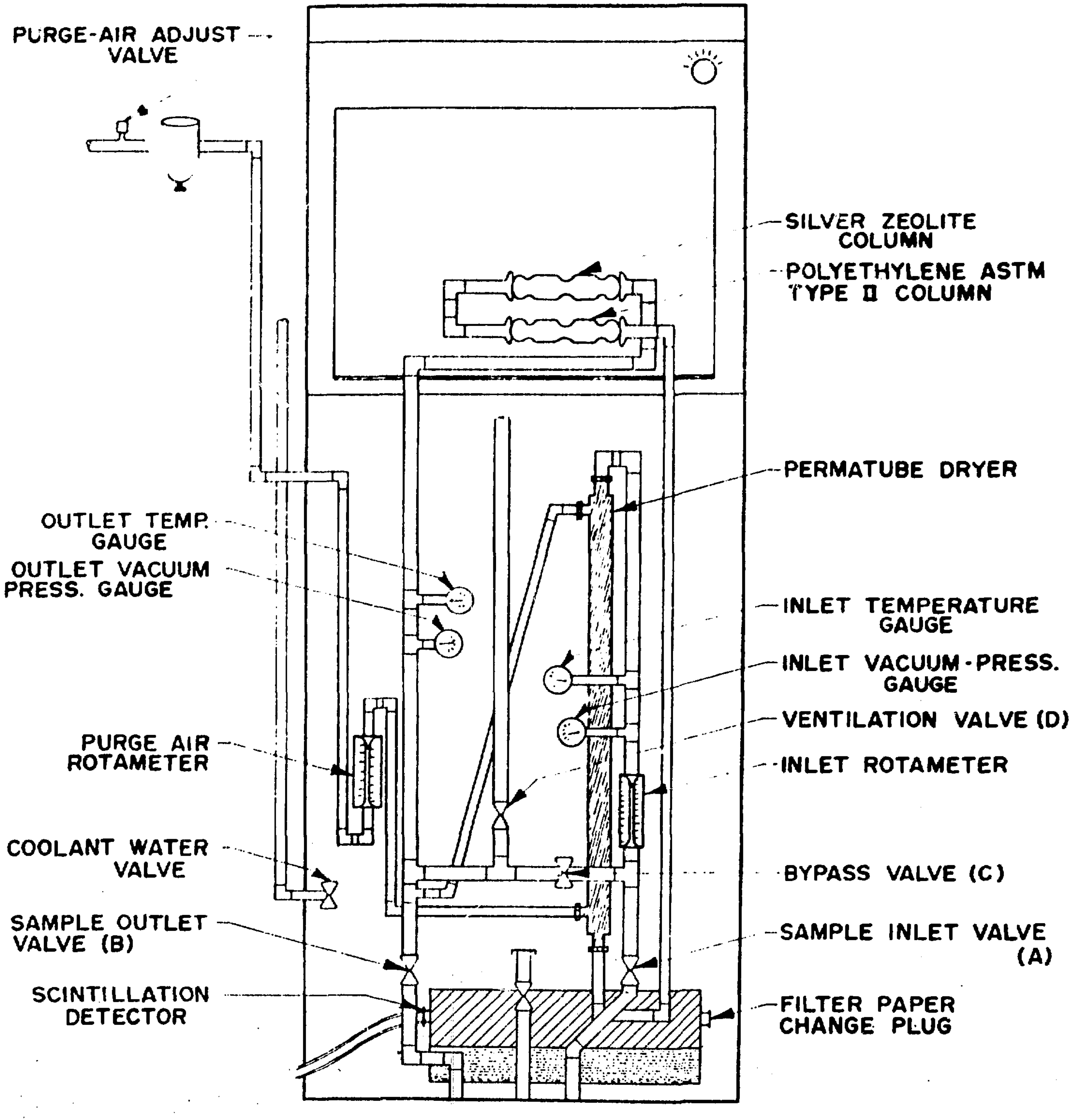
    c. High humidity conditions (dewpoint ∿160°F).

DESCRIPTION OF THE PRESENT MONITOR

Figure 1 is a photograph of the present sampling station. The sampling cabinet is 76 inches high with a standard oven mounted on top which acts as a heat source and containment structure for the polyethylene beads used to collect the gaseous Ru-106 (RuO_4). The electronic cabinet is situated adjacent to the sampling cabinet and houses the high voltage supply to the NaI(Tl) detector, amplifier, single channel analyzer, ratemeter and the chart recorder. Figure 2 shows a diagram of the WCF sampling station. Process off-gas is sampled from the positive pressure side of the off-gas blower and enters through the sample inlet line. This inlet line is heat-taped and maintained at a temperature of approximately 180°F. The sample off-gas is monitored by a rotameter, temperature gage and a pressure-vacuum gage. A 24 inch permatube dryer then



STATION 6 SAMPLER



removes over 99% of the water present in the sample stream without affecting the radioactive particulates or gases present in the stream. The dry sample stream passes through a 47 mm Whatman 40 cellulose filter which is continuously monitored for gross gamma activity by a 2 x 2 NaI(Tl) scintillation detector. After passing through the filter, the sample stream enters a glass column containing approximately 20 g of ASTM Type II polyethylene beads which collect the $^{105}RuO_4$ by absorption(1). The silver zeolite column shown in figure 2 was originally installed to collect I-129, but recent studies show that the polyethylene beads also remove radioiodine, probably in the form of I_2. The polyethylene beads are separately scanned using a Silicon Lithium-Drifted Low-Energy Photon Spectrometer (Si[Li]LEPS) for the 29.8 KeV Xe X-ray produced upon the I-129 decay. The sample gas is then returned to the process off-gas duct on the vacuum side of the blower.

PRINCIPLE OF PERMEATION DISTILLATION

The movement of specific molecules across a physical barrier with the selective exclusion of others is a well known biological phenomenon. Living cells rely upon this principle to maintain their steady-state metabolism with the uptake of a number of nutrients and the elimination of their waste products. Cellulose acetate membranes have been routinely used in reverse osmosis apparatus for the tertiary treatment of waste water in treatment plants throughout the world. It was these observations which led to the use of semi-permeable membranes to remove the water present in the process off-gas stream at the WCF. These membranes are silicon compounds which have a higher permeability for water vapor than for other molecules including most diatomic or noble gases. The basic mechanism of the permeation principle is such that a gas at partial pressure P_1 on one side of the membrane and partial pressure P_2 on the other side of the membrane will permeate through the membrane until P_1 and P_2 become equal. A diagram of the permeable membrane dryer installed in the WCF monitoring station is shown in figure 4.

Wet sample gas enters the silicone polymer tube bundle while water vapor continuously diffuses across the membrane and is removed by the counter-flowing dry purge air. The water vapor (and HTO) in the sample stream permeates through the cylindrical tubes according to the partial vapor pressures in the sample stream and purge stream.

The permeation of the water vapor is governed by the following equation(2):

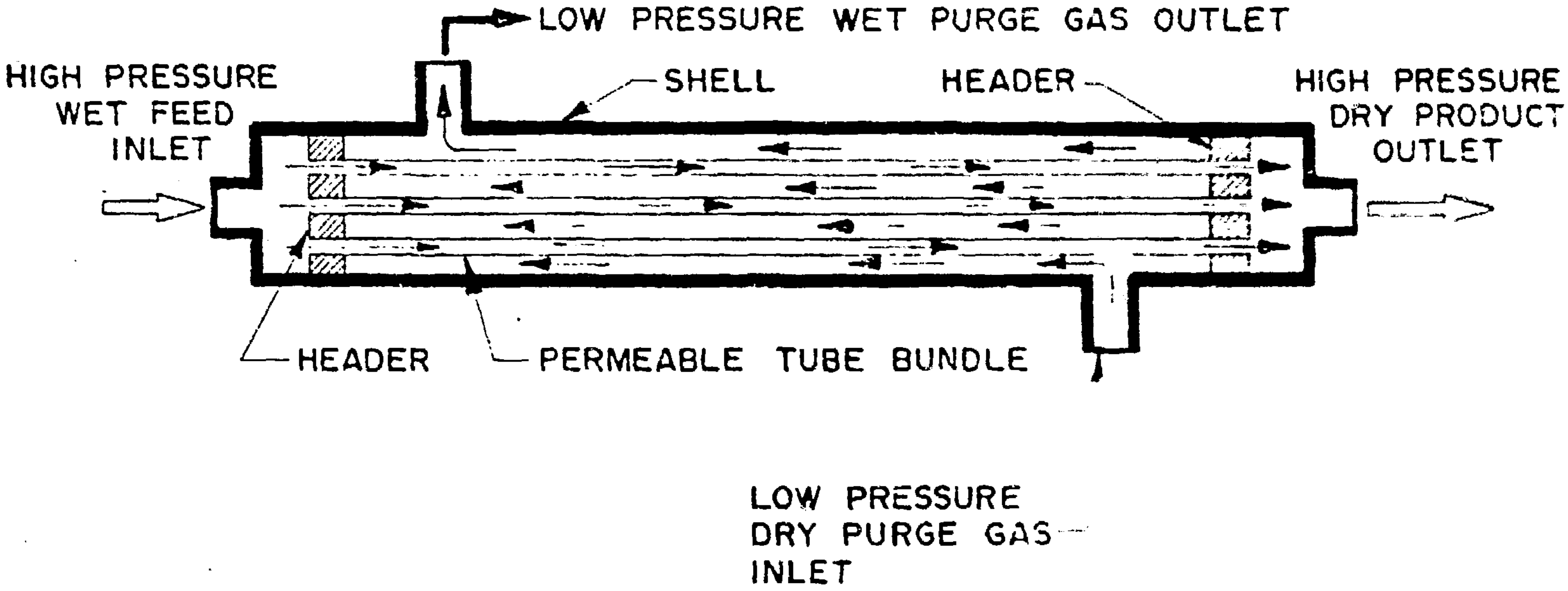
$$P_2 - P_1 = \frac{\phi \ \ln (b/a)}{2\pi \ LP} Y_{CG} \quad .$$

where: P = Permeability of the tube to the diffusing gas (H_2O, HTO).

a, b = inner and outer tube radii.

P_1, P_2 = outer and inner bulk partial pressures of the diffusing gas, cm Hg.

PERMATUBE DRYER



ϕ = carrier gas flowrate, cm^3/sec. (STP).

L = Polymer tube length, cm.

Y_{CG} = Volume fraction of the diffusing gas in the purge stream.

For a fixed dryer design having a constant purge gas flow, all of the variables except Y_{CG}, p_1, and p_2 become constant. Also note that with a large dry purge gas flow, p_1 becomes small compared to p_2. Therefore, $p_2 - p_1$ approaches p_1 with a large purge stream flow. Thus, the entire equation can be reduced to:

$$p_2 = K \cdot Y_{CG}$$

where, $K = \frac{\phi \ln (b/a)}{2\pi \; LP}$

It is seen from this equation that the concentration of water vapor present in the purge gas is directly proportional to that in the sample stream. This illustrates that quantitative removal of water vapor and tritium oxide is possible using permeable membranes to remove moisture from a sample stream. Since p_2 must be large for quantitative results, K must also be large (implying a large ϕ, since L and P are large) when Y_{CG} is small.

It has been shown by Howell et al.$^{(3)}$ that semipermeable membranes can be used to remove HTO over other gases such as HT, Ar, Kr, and Xe. The enhancement of HTO over these other species can be further increased by situating more than one membrane stage across the purge stream. It is clear that there are many applications in the field of Health Physics with which these permeable membranes have many applications such as in area monitors, stack monitors, etc.

OPERATION OF A COMMERCIAL PERMATUBE DRYER

The commercial permatube dryer originally used in the WCF process off-gas station was supplied by Permapure Products, Inc. Box 70 Oceanport, NJ 07757. The particular model was PD-1000-24S which consisted of 200 polymer tubes housed in a stainless steel shell 24 inches long. The stainless steel shell and two tube bundles are shown in figure 3. The tube bundle directly beneath the shell was deliberately failed by heating it to a temperature of approximately 250°F for 48 hours. The tube bundle in the lower portion of the figure has been in the off-gas station for approximately 5 weeks and operated continuously at approximately 180°F. These polymer tubes show no fatigue of deterioration under the very corrosive operating environment of the process off-gas. However, the potting material used to bind the tubes together was attacked by the high NO_x concentrations present and developed leaks after approximately 5 weeks of continuous operation. A new tube has been designed by ACC and installed in the off-gas station which utilizes 5 polymer tubes (Nafion tubing 0.10 i.d. x 0.12 o.d.) mounted in an all stainless steel header. This new dryer is approximately 96% efficient (inlet dp: 150°F, outlet dp: 50°F) compared to the 99% efficiency of the PD-1000-24S permatube. These tubes hold up well under calciner conditions, but the efficiency appears to be too low for proper sampling conditions when the WCF is operated at full capacity. Several new approaches to constructing a durable efficient permatube are being investigated by ACC.

There are several requirements that must be met to ensure satisfactory operation of the commercial permatube.

1. Gaseous samples should be filtered to at least a 10 micron particle size.

2. Condensable products such as oil, acid mists and water droplets must be removed.

3. The purge gas flowrate must be at least 1 1/2 times the actual volume of the sample gas flowrate.

4. Conditioning of the dryer is required for best results. Experiments performed at the ICPP indicate that the commercial PD-1000-24S (200 tube bundle) dryer operates approximately 60% efficient (inlet dp = 150°F, outlet dp = 120°F) without conditioning and with 99% efficiency (inlet dp = 150°F, outlet dp = 20°F) after 6 hours of purge gas flow.

Radioactive particulate plateout in the polymer tubes appears to reach equilibrium after a short period of operation. In fact, from gross radiation readings on the stainless steel tubing and that of the polymer tubes, it appears that stainless steel tubing (type 316) has a greater affinity for particles than the tubes. This is most probably due to the high NO_x concentration in the sample stream which tends to regenerate the tubes (Permapure Products suggest using a 10% HNO_3 solution to regenerate the tubes). A study was performed(8) which indicates that radioactive plateout equilibrium is reached in stainless steel tubing after a period of 1 month ([10 cfm flowrate] equilibrium is approximately 0.25% of the total activity transported through the tubing).

ANALYSIS FOR GASEOUS Ru-106 AND MOLECULAR I-129

The polyethylene beads in the WCF station are removed weekly and gamma-scanned for RuO_4. Efficiency is calculated by the difference in activity in each column section according to the equation(5),

$$E = 1 - \frac{(F_2)}{(F_1)}$$

where,

F_1, F_2 = measured activity in each column section.

In terms of total activity,

$$A = \frac{(A_1)^2}{A_1 - A_2}$$

where,

A_1, A_2 = Activity in each column.

I-129 sampling is also performed using the polythylene beads as the sampling media. The analytical method involves using the Si[Li]LEPS and scanning for the 29.8 KeV X-ray. Counting time ranges from approximately 7 to 10 hours and usually yields a coefficient of variation of about 25%.

ERRORS DUE TO ANISOKINETIC SAMPLING(6)

The sample stream drawn into the WCF off-gas station is a proportional sample of the total process off-gas exhaust. No special effort is taken to ensure isokinetic sampling. This sampling error is briefly discussed in this section. If the sampling rate is higher than the local stream velocity a greater fraction of smaller rather than larger particles will be drawn into the probe. If sampling is much lower than the stream velocity, large particles will be impacted into the collecting probe.

There is a relatively small sampling error when the particles are small as is seen at the WCF off-gas monitoring station.

$\frac{\text{Probe inlet velocity}}{\text{Duct velocity}}$	$\frac{\text{Observed concentration in sample}}{\text{Actual concentration}}$
0.5	1.06
0.7	1.02
0.9	1.00
1.0	1.00
1.2	0.98
1.5	0.96
1.7	0.94
2.0	0.86

(Values for a 4 μm diameter particle.)

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Question (Sidney Porter):

Is there any exchange of halogens into your purge gas?

Answer:(Russel Roberts): To take that of one gas, Krypton 85, as you saw earlier, the permeation constant is unique for various diffusing gases. Krypton 85, which is another gas, has the highest permeability constant outside of Tritium, and that about two orders of magnitude lower than that of water. But as far as halogens go, I have the paper which has done a study at Livermore on it and it seems that halogens have, every chemical has, a permeability constant but it's very, very low compared to the primary ones. So you get the selectivity of permeation across these things. But as far as molecular halogens, I don't believe that we have a problem there. We see Iodine in the form of I_2 in the polyethylene beads and we seem to corrolate that with charcoal beads and we get a fairly good agreement. So I would say that permeation of halogens is quite small across the tubes.

MULTISTATION IODINE-125 CONTINUOUS AIR MONITOR WITH MINICOMPUTER ALARM AND DATA REDUCTION

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ABSTRACT

This paper describes the components, operation, and calibration of a Multistation Continuous Air Monitor for the analysis of Iodine-125 and the functions of the Minicomputer in providing alarm functions and data reduction to units specified by regulation for permanent records. The sensitivity and accuracy, as well as the justification for purchase and comparison of costs with other types of air monitoring systems are also described.

INTRODUCTION

The use of 1 to 100 millicurie quantities of Iodine-125 in chemical production operations leads to the need to monitor the concentration of this highly volatile radionuclide in the working environment and in effluents to uncontrolled areas. A detailed analysis was made as to the purpose of the monitoring and the operational value of the data to be produced. The instrumentation available as stock items to meet the resulting requirements was surveyed, which resulted in the development of the Multistation Continuous Air Monitor described below.

COMPONENTS AND OPERATION

The components of the Air Monitor are assembled to collect and quantify gaseous Iodine-125 on a continuous basis. The Iodine-125 is collected by absorption using a commercially available activated charcoal (North American Carbon, Inc., Type G617) of ASTM Mesh size 8-16. The charcoal is contained in a volume of approximately 5cc in the center of a polypropylene tube 8cm long with 1.27cm I.D., and 1.6cm O.D. The charcoal is held in place using a small plug of glass wool which in turn is held in place by a "00" single hole natural rubber stopper in both ends of the tube, as shown in Figure 1. This volume accommodates 1.6 to 2.0 grams of charcoal. The sample air is drawn through the charcoal absorber

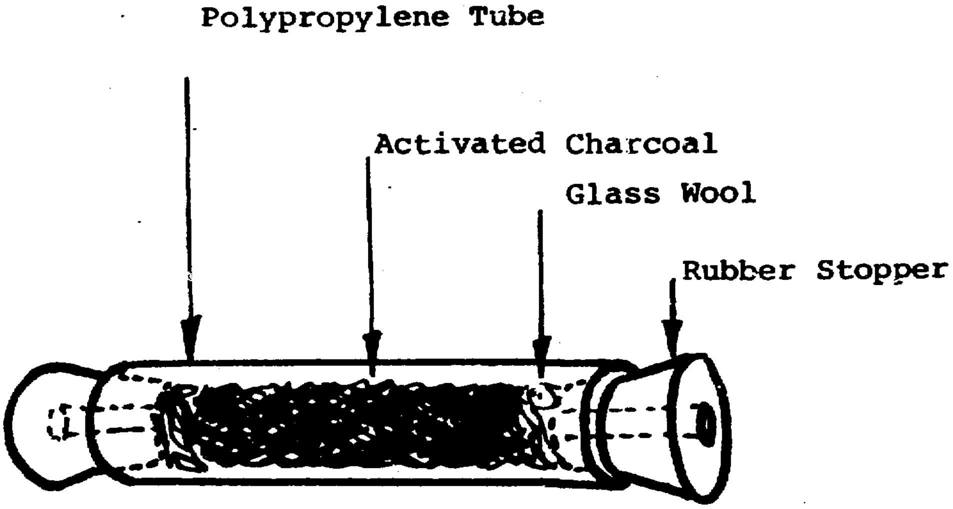


Figure 1. Charcoal Absorber

at a flow rate of 50 to 5000 cubic centimeters per minute allowing a residence time between .061 and 6.1 seconds, which is above the recommended minimum residence time of 0.01 seconds (IAEA73). The absorption efficiency for all species of Iodine of this configuration is greater than 99.9%. The sample air is delivered to the absorber by means of 1/4" I.D. Tygon tubing which is changed on a monthly basis due to contamination on the inside by the sample air. We are currently investigating the use of a teflon and stainless steel tubing to try to eliminate this problem. The flow of air is maintained using a small vacuum pump regulated by a flow meter with a needle valve.

The low-energy radiation from the Iodine-125 is detected with an efficiency of 75-80% by a 7.62 x 7.62 cm NaI (Tl) detector with a 1.75cm diameter well through the side of the crystal, in which the charcoal absorber is placed. The detector and photomultiplier tube (PMT) are shielded by 7.62cm of lead in order to eliminate the effects of transient background. This assembly is housed in a wooden box which protects the components from contamination and weather.

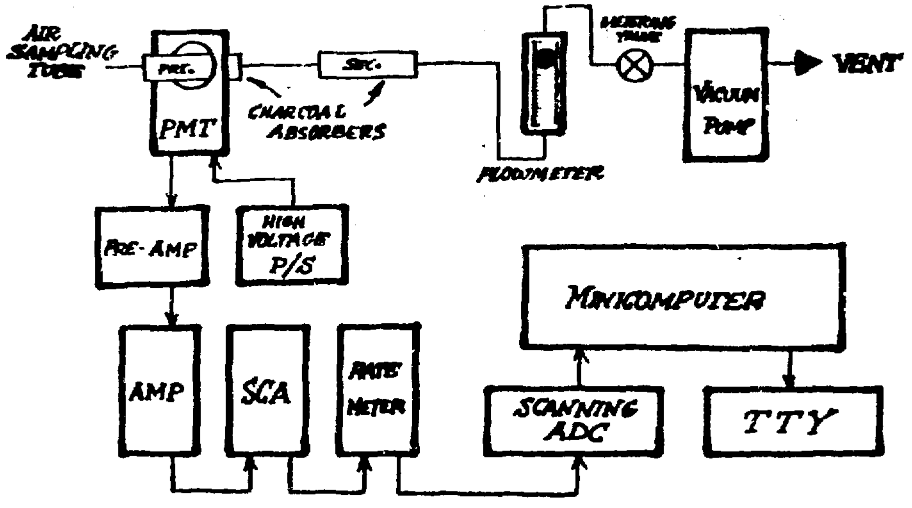


Figure 2. Block Diagram of
Air Monitor Components

The nuclear electronics are assembled from Canberra Industries, Inc. NIM components to form an independent single channel analyzer for each station as diagrammed in Figure 2. The high voltage for the PMT is supplied directly by Model 3002, ±0-3000V DC, 0-10 mA High Voltage Power Supply, or by Model V4 High Voltage Fanout Box, which is supplied by the Model 3002. The high voltage is set to optimize the performance of the PMT in conjunction with the gain of the amplifier. The signal from the PMT is input to a Model 805 Scintillation Preamplifier which prepares the signal for transmission through as much as 200 ft. of cable. The signal from the preamplifier is shaped and amplified by a Model 816 Spectroscopy Amplifier with the gain set so that the output is 51.2 KeV per volt. The amplified signal is then analyzed by the Model 830 Single Channel Analyzer with the baseline and window set to accept only the pulses characteristic of Iodine-125 (2 peaks of 27.5-35 KeV and coincidence at 55-70 KeV). The second generation of this Air Monitor System utilizes a combination Model 2015 Amplifier/Timing Single Channel Analyzer to perform the same function as the Models 816 and 830. The discriminated signal is then integrated by a Model 1480 Linear Ratemeter, or Model 1481L Lin/Log Ratemeter modified to use a time constant of 100 seconds to indicate the average count rate.

The recorder output from the ratemeter of each station is connected to a Kaye Instrument Digistrip Transmitter Model DIST-020M-1TT-117160 Scanning Analog to Digital Converter (ADC) with time of day clock and 14 input channels modified to include a teletype turn-on-off signal. This device scans all of the ratemeters and converts the analog millivolt output of each of the ratemeters to its digital equivalent. The sequential channel scan interval can be selected to be 1 to 77 minutes. The second generation of this system makes use of an Intel SBC711 Analog Input Board which performs the same function, but interfaces directly with the Minicomputer and accepts up to 32 channels and can perform repetitive single channel, sequential channel scan and random channel scan functions at the direction of the Minicomputer, thus providing greater flexibility.

The Minicomputer used in this Air Monitor System is an Intel SBC 80/10 Single Board Computer which incorporates an Intel 8080 Microprocessor and 1000 bytes of random access memory (RAM) and 4000 bytes of programmable read only memory (PROM). In order to increase the capabilities of the Minicomputer, 1000 bytes of RAM and 4000 bytes of PROM were added. Additional capabilities include interface with both teletype (TTY) and video communication' terminals, alarm capabilities for signal lights and telephone alarm system interface, and stand-alone ±5 and ±12V DC power supply. The function of the Minicomputer is to process the automatically entered count rate data with manually entered data (system variables) to produce concentration data in an acceptable format with units in μCi/cc and record it on the TTY, and to compare the data with various limits established and trigger the appropriate alarms when these limits are surpassed. In the second generation system the Minicomputer will provide the time of day clock and will control the scanning function of SBC711 Analog Input Board based on data and alarm conditions. The output device currently being used is a Teletype Model ASR33 with full duplex and motor turn on-off control.

CALIBRATION

The calibration of this Air Monitor System makes use of x-ray-x-ray coincidence method of Iodine-125 standardization discussed by Horrocks(Ho75) and using the insitu charcoal absorber as the standardizing source. This technique uses the following equation to determine the actual activity of the source.

$$S = N_C \left[\frac{1 + 2R}{2R} \right]^2 \qquad \text{(A)}$$

where:

S = Source activity (dpm)

N_C = Counts in coincidence peak

N_S = Counts in singles peak

$R = N_C/N_S$

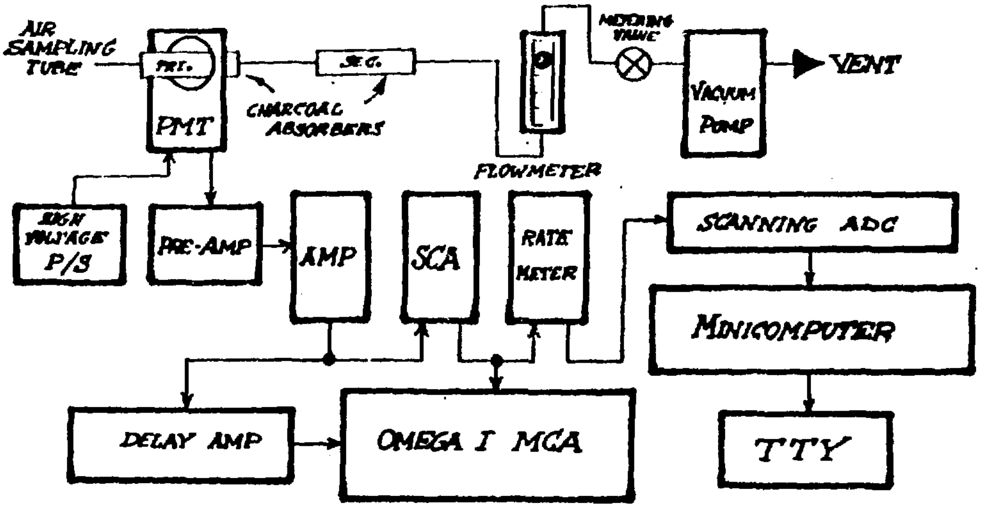


Figure 3. Block Diagram of Air Monitor

Calibration Setup

The components used to determine the source activity are a Canberra Model 1457 Delay Amplifier with the delay set at 1.75 microseconds and the gain set at 1, and a Canberra Omega 1 Multichannel Analyzer (MCA). The setup for calibration requires that the following connections be made. Figure 3 shows the block diagram of the setup.

1. Connect the Amplifier Output to the Delay Amplifier Input.
2. Connect the Delay Amplifier Output to the ADC Input of the MCA.
3. Connect the output of the SCA to the Gate Input of the MCA.

By using the Region of Interest integration capabilities of the MCA the counts in the two peak areas as shown in Figure 4 can be determined and utilized in the calibration equation (A) above to determine the source (absorbed sample) activity. The counting data of the station being calibrated is determined using the setup factors of the ratemeter; i.e., scale factor (CPS full scale) and recorder output setting (mV full scale), and the millivolt reading on the TTY output for the station of interest. The insitu efficiency then simply becomes the ratio of the calculated count rate to the calculated source activity. In the event that the ratemeter is set for logrithmic output an additional calculation is necessary, as will be explained in the Minicomputer Software Section that follows.

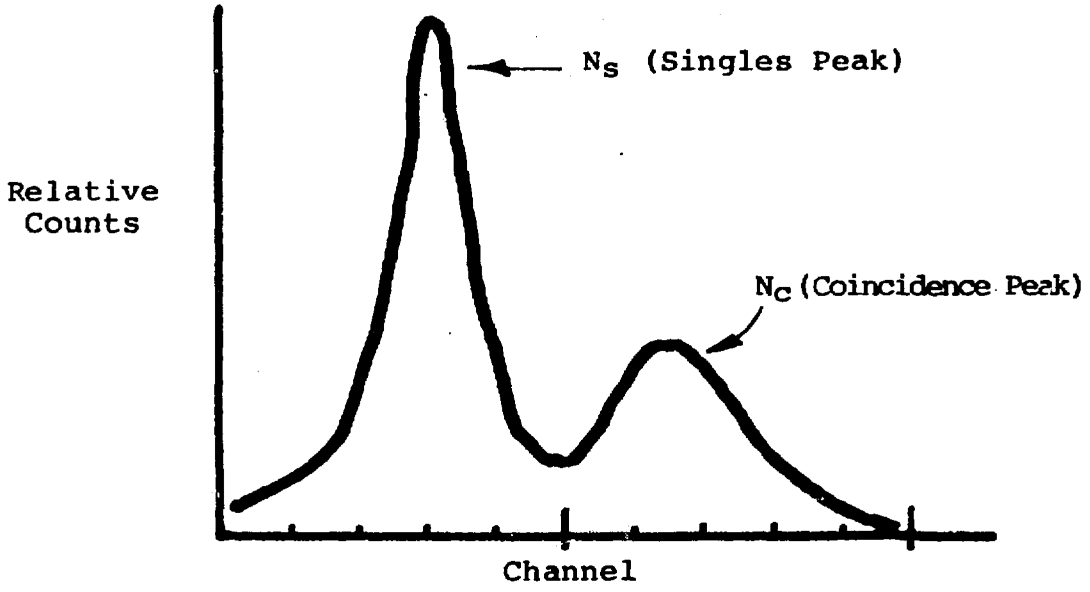


Figure 4. Pulse Height Spectrum of Iodine^{-125} from MCA Showing Singles and Coincidence Peaks

The Air Monitor System is calibrated weekly, or when the charcoal absorber is changed, whichever is most frequent. The calibration procedure can be performed in about 5 minutes for each station without disturbing the operation of the system. A log of the efficiency of each station is maintained to determine any drift or other deterioration of the system.

Other calibrations that need to be performed on this Air Monitor System include the ratemeters, the ADC, and the flow meters. These calibrations are usually performed on an annual basis, except the flow meters which are only on installation or change of the float ball.

Some of the problems which have occurred with the Air Monitor System are the drift of the PMTs with age (we are using used detectors), the geometry of the charcoal absorber, which is dependent on centering the charcoal section of the tube in the detector well, and the absorption-desorption equilibrium on the air sample delivery tube, as discussed previously.

MINICOMPUTER SOFTWARE

The Minicomputer Software has been developed to provide as much flexibility and as little maintenance as possible within the constraints of the hardware. The input which is required is of two types, manual and automatic. The manual input data are those factors which relate to the setup of each station and the alarm levels as follows.

1. Known Dead Station Numbers.
2. High Range Log Station Numbers.
3. Low Range Log Station Numbers.
4. Flow Rate for each Station (cc/min).
5. Upper Scale Limit for each Station (mV).
6. Rate of Rise Limit for each Station for selected scan internal (pCi/cc).
7. Scale Factor for each Station (CPS full scale).
8. Counting Efficiency for each Station ($\frac{\text{cpm}}{\text{dpm}}$).
9. Full scale Recorder Output (mV).

Each time one of these parameters changes, the change must be entered manually into the Minicomputer using the teletype or video input keyboard. The data which is automatically input into the Minicomputer is received from the ADC as a digital millivolt reading corresponding to the output of the ratemeters. In the first generation Air Monitor the time of day is also included as one of the 15 channels of output from the ADC. In the second generation the ADC will provide up to 32 channels of data and the Minicomputer will provide the time of day clock.

The calculations that the Minicomputer performs are as follows:

1. Convert millivolt reading to counts per second (cps).
    a. Linear Ratemeter

$$\text{cps} = \frac{(\text{mV reading})\quad(\text{Scale Factor})}{(\text{Full Scale Recorder Output})} \qquad \text{(B)}$$

b. Logrithmic Ratemeter

$$\text{cps} = \text{LD X } 10 \left[(\text{ND})\ (\text{mV reading})/(\text{Full Scale Recorder Output})\right] \qquad (C)$$

where LD is the lowest decade; i.e., 10 or 100
and ND is the number of decades

2. Calculate the net change in cps, to be used in concentration calculations and alarm functions.

$$\Delta\,\text{cps} = \text{cps2} - \text{cps1} \qquad (D)$$

where cps2 is the value of the previous scan and cps1 is the value of the current scan.

3. Calculate the concentration during the scan interval.

$$\mu\text{Ci/cc} = \frac{(\Delta\text{cps})}{(\ 3.7 \times 10^{4}\ \text{cps}/\mu\text{Ci})\ (\text{eff})\ (\text{flow rate})(\text{Scan interval})} \qquad (E)$$

The output of the Minicomputer is formatted so that the data presented is in the appropriate units and form to be used as a permanent record with no additional manipulation necessary. The current first generation output format includes each of the following items, some of which are identified in Figure 5.

1. Input Error Notation (such an error terminates the operation of the Minicomputer and initiates an alarm).
2. Initial Automatic Raw Data at turn-on (mV).
3. Calculated Concentration (μCi/cc).
4. Scan Interval Raw Data (mV).
5. Manual Data Input Parameters (every 12 hours).
6. Alarm Notations (coded).
7. Time of Day (24 hour clock).
8. Extra Line Feed at each Hour (for format clarity).

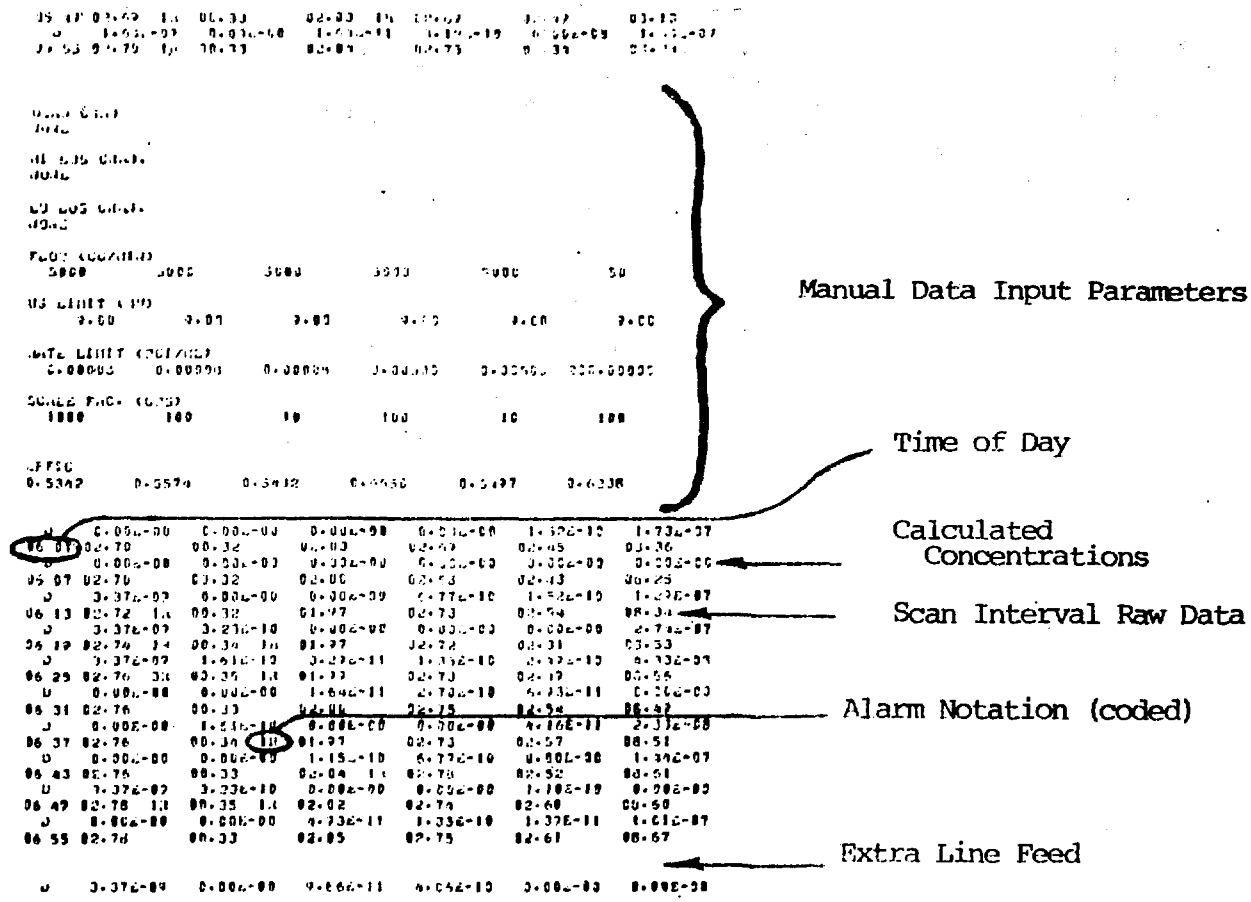


Figure 5. Minicomputer Teletype Output Format (First Generation)

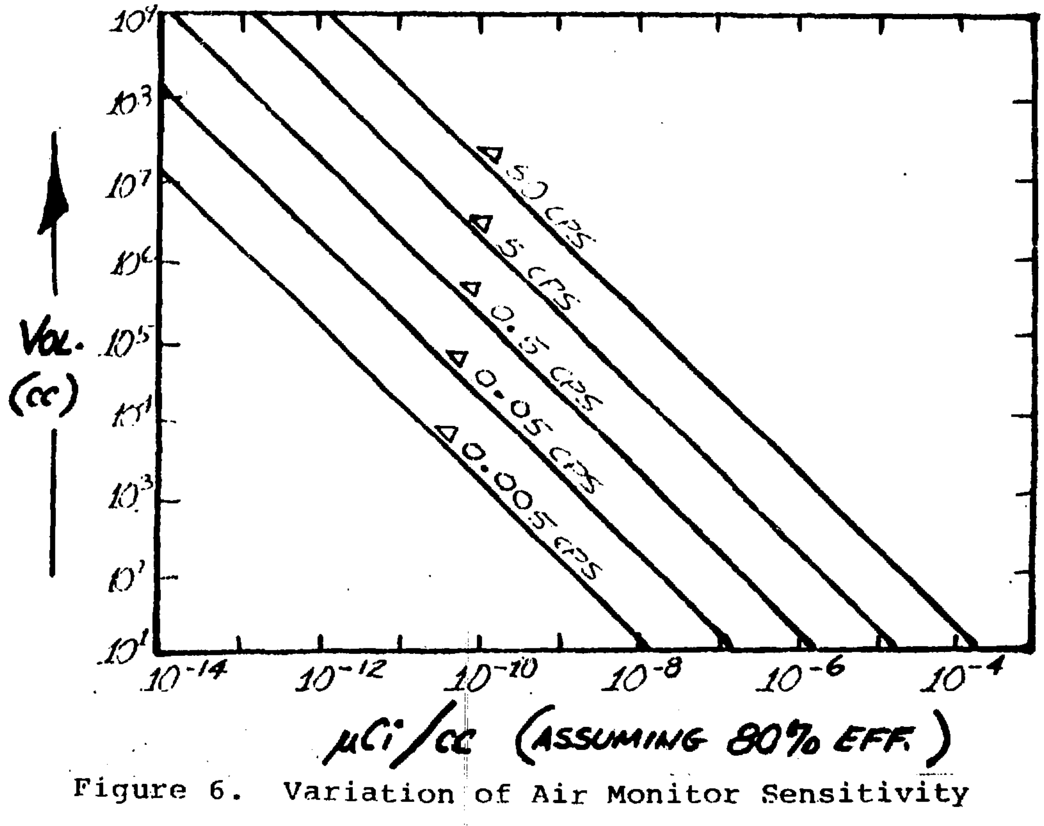


Figure 6. Variation of Air Monitor Sensitivity with Air Sample Volume and Change in Count Rate

The alarm functions of the Minicomputer are designed to compare the current station data (both raw and calculated) with the manual input parameters to provide a live time indication of the operating conditions. The specific alarm functions of this Air Monitor System in the order of lowest to highest priority are as follows:

1. Up-Scale Limit Alarm, Code UL - This alarm limit set in mV serves as a percentage of full scale since the ratemeter is scaled from 0 to 10. This function is useful as an action or working level alarm.

2. Full Scale Alarm, Code FS - This alarm indicates that the ratemeter has reached the end of the scale range. Care must be used in arbitrarily switching to the next highest scale because of changes in sensitivity and accuracy, and the need to adjust the manual input parameters. In some cases this can be used to incidate that it is necessary to change the charcoal absorber.

3. Rate of Rise Alarm - 1st Occurrence, Code 1R - This alarm function compares the current concentration with the previous concentration, and if the change is greater than the manual input value the alarm is initiated. This alarm is useful as a local working level alarm.

4. Rate of Rise Alarm - 3rd Successive, Code 3R - This alarm indicates that the Rate of Rise Alarm -1st Occurrence condition was maintained for two successive scan intervals. This alarm indication is treated as an Emergency Condition Alarm.

5. Dead Channel Alarm, Code DC - This alarm function indicates that there has been a failure of the electronics of a particular station.

The distribution of the alarm indications vary with the facility and operation, but can be categorized by responsibility as follows:

1. Operation Supervisor receives FS, 1R, and 3R alarms.

2. The Facility Health Physics Representative receives all alarms.

3. The Facility Security Office receives 3R and DC alarms for 24-hour notification of personnel on the Health Physics call list.

With the greater capabilities of the second generation interface between the ADC and the Minicomputer it will be possible to adjust the scan interval and sequence in the event of an alarm condition. This will allow for greater flexibility in the alarms and improved incident data for evaluation and reporting requirements.

SENSITIVITY AND ACCURACY

The sensitivity of this Air Monitor System is dependent on several factors which are related to the various components. The first factor is the recorder output factor which is the ratio Scale Factor and the recorder full scale millivolt reading of the ratemeter. The lower the scale factor and the higher the full scale recorder millivolt reading, the more sensitive the system will be.

For example, with a Scale Factor of 10 cps and a full scale millivolt reading of 20 mV, the smallest detectable change in the ADC of 0.01 mV is equivalent to 0.005 cps. The second factor affecting the sensitivity of the system is the overall counting efficiency of each station. Obviously the higher the efficiency the more counts will be recorded per unit time, thus increasing the sensitivity. The third and most influencing factor is the volume of sample air which is drawn through the charcoal absorber per scan interval. This is dependent on the product of the flow rate and the elapsed time of the scan interval. As can be readily observed in equation (E), by increasing either or both of these factors the sensitivity will be increased. Figure 6 demonstrates the effects of some of these factors on the sentitivity.

The accuracy of this Air Monitor System is dependent on both the random error related to radioactive decay and the systematic errors introduced by the components. The random error of each concentration computation is quantified using the standard deviation of the change in the count rate.

$$\sigma_{\Delta cps} = \left(\frac{cps1 + cps2}{RC}\right)^{\frac{1}{2}} \left[2\,\frac{RC}{T}\,(1-\frac{RC}{T} + \frac{RC}{T}\,e^{-T/RC}\right]^{\frac{1}{2}} \qquad \text{(F) Pr64}$$

where $\sigma_{\Delta cps}$ = Standard deviation of the change in count rate

cps, cps_2 = Count rates (cps) of the current and previous scans, respectively

RC = Time constant of the ratemeter (100 seconds for this sytem)

T = Scan interval (sec)

PREVIOUS COUNT RATE (cps)	PERCENT RANDOM ERROR OF CHANGES IN COUNT RATE (cps) (@ 2σ, T=360)				
	+0.005	+0.05	+0.5	+5	+50
0.01	449	75.1	20.5		-
0.1	1286	142	23.8	6.48	
1.	4021	406	44.9	7.52	2.04
10.			129	14.2	2.38
100.				40.6	4.49
1000,				127	12.9
10,000.					40.2
100,000.					127

Figure 7. Percent Random Error at Various Count Rates and Changes in Count Rates.

The table in Figure 7 shows the random errors at various count rates and changes in count rates at the 95.5% (2σ) confidence level.

The systematic error of the components and methods combinded is approximately 2.6% @2σ which is added in quadriture to the random error to produce the overall error of the system.

In Figure 6 the sensitivity of the Air Monitor System is shown for changes in count rates of 0.005 and 0.05 cps, and in Figure 7 we see that the errors for these changes are extremely large. From this the question may arise as to value of measuring count rates at very low levels where these large errors occur with changes in count rates as much as 50%. The value lies in the anticipation of very large increases with very good statistics which may occur in an emergency situation and moderate changes also with good statistics which would result in evaluating the change over a long period of time such as 4 to 24 hours for the purpose of obtaining the average concentration for the period to sensitivity levels of several times less than the maximum permissible concentration for an uncontrolled area (8 x 10^{-11} μCi/cc).

JUSTIFICATION FOR THE PURCHASE OF A MULTISTATION AIR MONITOR SYSTEM

The justification for the purchase of a Mult station Air Monitor System involves a detailed evaluation of regulatory requirements, recommendations of recognized national and international organizations, and the experience gained and data required in the operational use of large quantities of Iodine-125. In making this type of evaluation the regulatory requirements take on the highest degree of importance in that compliance must be maintained. In order to put air monitoring into the proper perspective it is appropriate to consider the regulatory definition of the term survey.

> "Survey means an evaluation of the radiation hazards incident to the production, use, release, disposal, or presence of sources of radiation under a specific set of conditions. When appropriate, such includes a physical survey of the location of materials and equipment, and measurements of dose rates, doses, and quantities and concentrations of radioactive material" Ca73§30258(w)

On the basis of this definition, air monitoring can be considered to be a survey for the determination of compliance with the specific requirements of the regulations. One of these areas is the mandate that all radiation exposures and releases of radioactive material in affluents to uncontrolled areas be as low as is reasonably achievable (ALARA) US78§20.1(c).

In order to determine whether a program is ALARA it becomes necessary to make the appropriate surveys on which comparisons can be made. The surveys can also be used to demonstrate compliance with certain regulatory requirements as follows:

1. Occupational Dose Limit - 7.5 rem per quarter for the thyroid. (Ca74)
2. Exposure of Indidivuals to Concentrations of Radioactive Material in Controlled Areas - 5×10^{-9} µCi/cc Iodine-125 (Ca73§30266).
3. Permissible Levels of Radiation in Uncontrolled Areas.
    a. 2mrem in one hour, or
    b. 100 mrem in 7 consecutive days, or
    c. 0.5 rem in one year (Ca73§30268).

4. Concentrations in Effluents to Uncontrolled Areas - 8 x 10^{-11} μCi/cc Iodine-125 (Ca73δ30269).

5. Personnel Monitoring required at 25% of maximum permissible levels (Ca73δ30276).

6. Incident Reports to meet classification and time requirements (Ca73δ30295 & 30297).

In addition to making surveys to meet regulatory requirements, they also demonstrate the efforts to meet the recommendations of the International Commission on Radiological Protection (ICRP) and the National Council on Radiation Protection and Measurement, particularly in the areas of ALARA (ICRP65 ¶ 52, NCRP71 ¶ 178) and Maximum Permissible Levels (ICRP65 ¶56, ICRP59, NCRP71 ¶ 238), which lead to the regulatory requirements described above. In addition to these recommendations, a number of references state specific programs for making air monitor surveys. (ICRP66¶7, ICRP65¶30, ICRP69 Chapter F & I, ICRP59¶79 & 86, NCRP64 Section 5.8, IAEA66 Sections 6.3.1 & 6.3.2, La67, NCRP71¶224.) The objectives of these programs can be summarized as follows.

1. Monitor the air to determine the possible inhalation exposure of the personnel.

2. Provide an alarm in the event of an unusual or unexpected increase in the level of air contamination.

3. Provide information necessary to plan for appropriate programs for personnel monitoring for internal contamination.

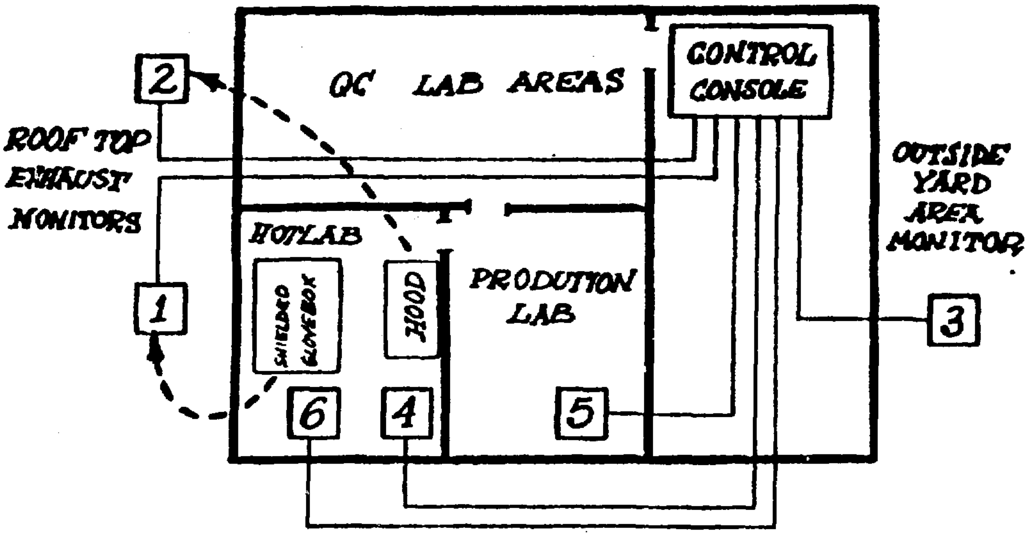
4. Assess the efficacy of the Engineering and Administrative Programs established to control the airborne contaminants.

In addition to meeting regulatory requirements and recognized organization recommendations, the operating experience and requirements lead to the need for air monitoring information for the following purposes.

1. Correlate thyroid doses with actual operations.

2. Verify controls.

3. Verify compliance.



STA #1	Located on Roof - Monitors Filtered Gloveobx Effluent.
STA #2	Located on Roof - Monitors Filtered Hot Lab Hood Effluent.
STA #3	Located Outside - Monitors Yard Area Ambient Air.
STA #4	Located in Hot Lab - Monitors Hot Lab Ambient Air.
STA #5	Located in Production Lab - Monitors Ambient Air.
STA #6	Located in Hot Lab - Monitors Glovebox Interior Air.

Figure 8. Air Monitor Station Locations
(Simplified Facility Diagram)

4. Emergency condition alarms.
5. Maintaining uncontrolled areas.
6. Estimate Thyroid Burdens (Ga75).

On the basis of the regulatory requirements, recommendations, and operating requirements, it was determined that an Air Monitor System to monitor the areas of the facility shown in Figure 8 was justified.

CRITERIA	DESCRIBED SYSTEM	PASSIVE SAMPLE	STAND ALONE
1. Cost of Program	$25,622	$21,200	$39,000
a. Fixed Capital	11,144	20,000	---
b. Variable Capital/ Station	2,413	200	6,500
c. Labor $/Yr @ 14/hr	4,368	26,208	13,104
2. Data			
a. Output	Differential	Average	Integral
b. Reduction	Automatic	Manual	Manual
c. Frequency	Live	Set Intervals	Live
3. Alarms	Yes, 5 modes	No	Yes, 1 mode

Figure 9. Comparisons of Air Monitor System Costs and Functions.

COMPARISONS OF COSTS AND FUNCTIONS

Once the justification for the purchase of an Air Monitor System had been determined, the task at hand was to find a system that would provide the necessary functions required by the regulations and operations that would not require excessive capital or operating expenses. In addition to the system described in this paper, two others were investigated. The first was the use of passive sampling where an air sample is collected on charcoal absorber, dismantled, and counted in an appropriate instrument. The second was stand-alone integrating air monitor systems such as used in nuclear reactor facilities. It soon became clear on the basis of the criteria shown in the cost and function comparison for 6 stations as shown in Figure 9 that the described system provide all of the attributes required and desirable.

CONCLUSION

The conclusion which is reached as a result of the description of the Air Monitor System and the discussion of justification and comparisons is that it is very feasible to obtain an Iodine-125 Air Monitoring System which will provide a wealth of data to meet regulatory and operating requirements and the recommendations of other organizations with reasonable capital expenditure and operating costs.

ACKNOWLEDGEMENTS

I wish to acknowledge the management support of L.A. Micco and R. J. Baumann in the development of this Air Monitor System and the unrelenting dedication of S. D. Enyart for the hardware assembly, test, and installation, and art work, and R. K. Sallee for the development of the Minicomputer software.

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Question (Bill Friedman, California Department of Health):

Doug, I have a few questions; I don't know if you realize it, but you gave a 40-minute paper in 20 minutes. Can this system be adapted for I-131?

Answer (Doug):

I am sure it can.

Question (Bill Friedman):

What will it take to do it?

Answer (Doug):

Just changing the settings on the single channel analyzer.

Question (Bill Friedman):

So this doesn't amount to any cost?

Answer (Doug):

No.

Question (Bill Friedman):

What will be the lower limit of sensitivity for both of those isotopes?

Answer (Doug):

I don't have it for I-131. I do have it for I-125. The sensitivity is dependant upon a number of factors but we can get down, if you sample over a 4-hour period, you can get down as low as 3.52×10^{-13} microcuries per cc. There are some inherent inaccuracies with the system at that level and I have a number of things here; at the lowest detectable change on the instrument are the error runs in the neighborhood of 142% at 1 sigma and the error is reduced very rapidly to reasonable areas of 10% at 2 orders of magnitude higher than that.

Question (Bill Friedman):

What is the response time for a system when you are approaching your present level? First of all, I should ask what spread is there between the present level and the actual accepted limit, for both of those isotypes? I was just talking about I-125.

Answer (Doug):

Well, we have a 100 sec time constant in the rate meters, so that is the determining factor for the scanning time. So, if you're set on a scanning time of one minute, then you have one minute response on the system.

Question (Bill Friedman):

Suppose you have a huge release of volatile iodine?

Answer (Doug):

You know it at the next scanning period.

Question (Bill Friedman):

Would you be able to get the people out in time before they get a lung full?

Answer (Doug):

Yes, I believe you can. The current instruments we are using were described analog to digital converter have some limitations. The new one that we are building will be controlled by the micro processor, so when it sees something go up on one station, he can switch over to a more frequent monitoring scanning period which will give you more accurate and more immediate results.

Question (Bill Friedman):

OK, then, when you are able to adapt the systems from I-125 to I-131, do you reposition the switch and set for the other isotope?

Answer (Doug):

All you have to do is just adjust the windows on the single channel analyzer.

Question (Bill Friedman):

What are you going to do if you are using both of those isotopes in the same area? Say two different labs close to each other. Can you handle that too, or do you have to have two different systems?

Answer (Doug):

You have to have two channnels.

ATOMIC ENERGY OF CANADA LIMITED

MONITORING TRITIATED WATER IN AIR AND WATER EFFLUENTS

by

R.V. Osborne and N.W. Tepley

SUMMARY

Current "on-line" methods of monitoring effluents for tritium (as tritiated water, HTO) measure concentrations in air above 250 nCi/m^3 (∿ 10 kBq/m^3) and in water above 1 μCi/kg (∿ 40 kBq/kg). Some of the problems encountered in such monitoring are the presence of fission and activation products in the effluents and, particularly in water monitoring, the often dirty quality of the sample. In a new design of monitor, HTO is collected directly from air by a flow of liquid scintillator (LS). For water monitoring a flow of air continuously samples the water and transports HTO to the LS.

The key features of the new design are that the high detection efficiency of LS is realizable, that the rate of use of LS is only ∿ 2 mm^3/s, that the controlled evaporation and metering of air provides the low flow of HTO needed for mixing with LS, and that accurate metering of a dirty effluent is not needed. The sensitivities for detecting tritium on-line are improved by at least an order of magnitude.

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R.V. Osborne and N.W. Tepley

INTRODUCTION

Monitoring tritium in effluents from nuclear power plants is required for health physics reasons. Additionally in CANDU nuclear power stations, which use heavy water as moderator and primary coolant, the presence of tritium in effluents or other fluid streams even at radiologically unimportant concentrations has economic significance. Current on-line methods for monitoring tritium (as tritiated water, HTO) are limited to measuring concentrations in air above 250 nCi/m^3 (∿ 10 kBq/m^3)(1) and in water streams above 1 μCi/kg (∿ 40 kBq/kg)(2). More sensitive methods are needed.

The main problems in such measurements are the presence of gamma radiation at the detector and the presence of fission products and activation products in the monitored fluids(3). Concentrations below those cited are usually measured by taking samples manually; directly from a water system, indirectly from air by collecting HTO in a bubbler(4) or on a desiccant for subsequent counting with liquid scintillator(5).

One one-line method of using liquid scintillator for measuring HTO in air is to collect the HTO in a water stream that is subsequently sampled and mixed with a stream of liquid scintillator flowing through a counter(6). A water stream may be sampled directly. The liquid scintillator and water flows are about 1 cm^3/min and 0.1 cm^3/min respectively. However for continuous monitoring the cost of a liquid scintillator at flows the order of 1 cm^3/min is ∿ $5000/a which is prohibitive for routine application of such devices on several effluents at a power station. At lower flow rates metering the water sample is impractical, particularly if the sample is from a dirty water effluent.

One way of solving the economic and metering problems in a monitor on a water effluent is to sample the effluent periodically, transport the sample to an automatic liquid scintillator dispenser, mix the scintillator and sample, and count. An auto-urinalyzer, developed several years ago(7) for self-monitoring for tritium by the operating staff of nuclear power stations, is currently being updated and will be able to accept any aqueous sample that is presented to it. However this discrete sampling method, whether automatic or manual, has the disadvantage that an undesirably long delay can occur before any release of activity is observed. Nevertheless releases in air and water effluents may be conveniently integrated by this method, using a bubbler for air measurements or a collector for water measurements that is automatically sampled and refilled periodically.

An alternative solution using liquid scintillator for measurements of tritium in both air and water is described in this paper. The practical limits to the variables are discussed, the critical ones are identified and experimental values given. The activities of tritium in air and in water effluents that may be detected are estimated.

DESCRIPTION

The method is shown schematically in Figure 1. Liquid scintillator (LS) and air are metered at flows F_s and F_a respectively to the mixer where HTO is transferred to the LS from the air. Air (at a flow approximately $10 \cdot F_s$), LS and HTO are then pumped to the detector cell in the liquid scintillation counter. The small flow of air is included so that the transfer time from mixer to detector remains short.

The in-coming air may be a continuous sample for an air effluent or other air volume, or it may be a sample of air that has been bubbled (flow F_t) through a continuous sample stream of water (flow F_w) at a controlled temperature. The controlled evaporation and metering of the air/HTO vapour mixture are practicable ways of providing the low flow of water that is needed to mix with the desired low flow of scintillator. The alternative method of condensing the water from the air for mixing with scintillator directly (as in the device described in reference 8 for example) still leaves the problem of metering extremely small quantities of water when a liquid scintillator is used.

The sampling method described also avoids the problem of accurately metering a (possibly) dirty effluent since both the air and the water flow rates, F_t and F_w are not critical. A condition is that F_t is greater than the desired sample air flow F_a. The bubbler temperature is controlled below ambient to avoid HTO condensing from the air leaving the bubbler.

THEORY

The rate at which HTO activity is entering the mixer is given by $F_a \cdot C_a$ where the symbols here and subsequently are as given in Table 1. Liquid scintillator enters the mixer at flow F_s of which some is evaporated by the air. The liquid scintillator flow out of the mixer is $(F_s - \sigma \cdot L \cdot F_a)$. Hence the concentration of HTO in LS leaving the mixer is

$$\varepsilon \cdot F_a \cdot C_a / (F_s - \sigma \cdot F_a \cdot L)$$

and the counting rate from the detector is

$$\left[\varepsilon \cdot F_a \cdot C_a / (F_s - \sigma \cdot F_a \cdot L)\right] V \cdot E.$$

If more than a few counts are accumulated in the counting time chosen (t) and the background counting rate is well known from a longer count, then a criterion for the minimum detectable count at the 95% confidence level (5) is $\approx (10\ B.t)^{0.5}$.

The minimum detectable activity in air (MDC_a) that gives this count is then given by

$$MDC_a = \left[\frac{F_s - \sigma \cdot F_a \cdot L}{F_a}\right] \left[\frac{1}{\varepsilon}\right] \left[\frac{1}{V \cdot E}\right] \left[\frac{10B}{t}\right]^{0.5} \qquad (1)$$

If water is monitored by sampling with air then the rate at which activity is entering the mixer is

$$F_a \cdot H \cdot \eta \cdot \beta \cdot C_w$$

and the minimum detectable activity in water is given by

$$MDC_w = \frac{1}{H \cdot \eta \cdot \beta} \cdot MDC_a \qquad (2)$$

VALUES OF THE VARIABLES

Feasible values of the variables in (1) and (2) are as follows:

$\left[H \cdot \eta \cdot \beta\right]^{-1}$; From previous work$^{(4,9)}$ the evaporation efficiencies in many types of bubbler are known to be close to unity and the isotope-effect coefficient β is about 0.8 at ∿ 10°C - 15°C. Hence for controlled evaporation in this temperature range $(H \cdot \eta \cdot \beta)^{-1}$ ∿ 100 m^3/kg.

$\left[V \cdot E\right]^{-1}$; For a liquid scintillator flow of ∿ 2 mm^3/s through the counting cell and a clearance or response time constant of 200 s the cell volume is ∿ 400 mm^3. An attainable counting efficiency with either a dioxane-based or solgel$^{(10)}$ scintillator is 0.25. Hence an attainable value of $(V \cdot E)^{-1}$ is 10^7 $Bq \cdot s/m^3$.

$\left[10\ B/t\right]^{0.5}$; For a detector of the size noted a typical value for B is ∿ 0.03 s^{-1}. With an effective counting time of 600 s the attainable value of $[10\ B/t]^{0.5}$ is ∿ 0.02 s^{-1}. Note that a long counting time may be assumed in this context; the capability to respond more quickly to higher activities is retained by using a ratemeter circuit with an updating period that depends on the accumulated count. One such a circuit was described in reference (11).

$\left[(F_s - \sigma \cdot F_a \cdot L)/F_a\right]$; This quantity is the ratio of the flow of liquid scintillator through the detector to the sample air flow. One limit will be set by evaporation of the scintillator solvent. Control of a system with such a high air flow that a major fraction of the solvent is evaporated is difficult, although the feed to the mixer can be suitably diluted. A reasonable condition is that $F_s \nless 2 \cdot \sigma \cdot F_a \cdot L$. The vapour densities of xylene, toluene, and dioxane, three possible solvents, differ by a factor of ~ 3 at normal room temperatures. For the least volatile, xylene, $L \sim 5 \times 10^{-5}$. Hence, if the evaporation efficiency is close to unity, the limiting value of $(F_s - \sigma \cdot F_a \cdot L)/F_a$ is 5×10^{-5}. A more restrictive limit may occur on the ratio of F_s/F_a because of water collection inefficiency as discussed in the next section.

$\left[\varepsilon\right]^{-1}$; There are two components to this variable which is the fraction of water in air transferred to the liquid scintillator. One is the fraction of the equilibrium distribution of the water between the air and scintillator that is attained. We have shown that for several designs of mixer, close to equilibrium distribution may be attained for air/scintillator flow ratios up to $\sim 10^4$. The other component is the actual equilibrium distribution of water between air and scintillator which is discussed in the next section.

COLLECTION OF HTO BY LIQUID SCINTILLATOR

The vapour pressure of water mixed with a dioxane-based scintillator or as a sol with a toluene- or xylene-based scintillator is greater than would be observed with an ideal mixture following Raoult's law. We have demonstrated that the evaporative loss rates of water from both a toluene-based scintillator (Ready Solv VI) and a dioxane-based scintillator were higher by a factor of 2 at 10% water loading than they would be from an ideal mixture. The factor increased to 8 at 0.1% water loading for the former scintillator.

Figure 2 shows values of the product $\left[\varepsilon^{-1}\cdot(F_s/F_a)\right]$ that have been observed with various scintillators at various ratios of (F_a/F_s). The effect of humidity in the sample air may be seen; also the increase in capture of HTO from the sample air effected by loading the scintillator with water. The latter appears to be a key to applying the method since it reduces the dependence of the HTO capture on the humidity in the incoming air.

Hence, for monitoring with an acceptable dependence upon the humidity of the air, an attainable value for $\varepsilon^{-1}\cdot(F_s/F_a)$ is 5×10^{-4}. This value we take here as a conservative estimate of the value of $\varepsilon^{-1}\left[(F_s-\sigma\cdot F_a\cdot L)/F_a\right]$ since no credit is taken for the effect of solvent evaporation. For water monitoring by this method, where the input humidity is controlled, a higher air to scintillator flow ratio may be used. Also the insensitivity of the value of $\varepsilon^{-1}\cdot(F_s/F_a)$ to the ratio F_a/F_s at the higher values of the ratio allows for less critical control of the flow rates. The attainable value for $\varepsilon^{-1}\cdot\left[(F_s-\sigma\cdot F_a\cdot L)/F_a\right]$ in this case is $\sim 3 \times 10^{-4}$.

With the above values, we have

$$\begin{aligned} MDC_a &= (5 \times 10^{-4})\cdot(10^{7})\cdot(2 \times 10^{-2}) \\ &= 100\ \text{Bq}\cdot\text{m}^{-3}\ (\sim 3\ \text{nCi}\cdot\text{m}^{-3}) \end{aligned}$$

and

$$\begin{aligned} MDC_w &= (10^{2})\ (3 \times 10^{-4})\cdot(10^{7})\cdot(2 \times 10^{-2}) \\ &= 6\ \text{kBq}\cdot\text{kg}^{-1}\ (\sim 0.2\ \mu\text{Ci}\cdot\text{kg}^{-1}). \end{aligned}$$

These values are 1% and 6% of the maximum permissible concentrations in air and in water for members of the general public respectively. The sensitivity of the method as discussed here is compared in Figure 3 with those of other instruments mentioned earlier. Lower concentrations of HTO in both water and air are detectable with the method described here than with the other on-line monitors noted.

ACKNOWLEDGEMENT

We appreciate the cooperation of Dr. R.M. Brown in providing us with samples of various scintillators for our preliminary measurements.

Table 1. Symbols and Units of Quantities Discussed in the Text

B - Counting rate from the detector with an inactive sample (s^{-1}).
C_a - Concentration of HTO in air ($Bq \cdot m^{-3}$).
C_w - Concentration of HTO in water ($Bq \cdot kg^{-1}$).
E - Detection efficiency in the counter ($s^{-1} \cdot Bq^{-1}$).
F_a - Air flow into the mixer ($m^3 \cdot s^{-1}$).
F_s - Liquid scintillator flow into the mixer ($m^3 \cdot s^{-1}$).
F_t - Air flow through the water sampler ($m^3 \cdot s^{-1}$).
F_w - Water flow through the water sampler ($kg \cdot s^{-1}$).
H - Saturated vapour density in air at the temperature of the water sampler ($kg \cdot m^{-3}$).
L - Fractional content of scintillator solvent in air saturated at the temperature of the mixer.
t - Counting time (s).
V - Volume of liquid scintillator and water sol that is in the counter (m^3).
β - Ratio of the specific activity of the water vapour to that of the water in the water sampler.
ε - Ratio of the tritium activity leaving the mixer in the liquid scintillator to that entering in the air.
η - Fraction of vapour saturation attained in the water sampler.
σ - Fraction of saturation vapour density of scintillator solvent in air leaving the mixer.

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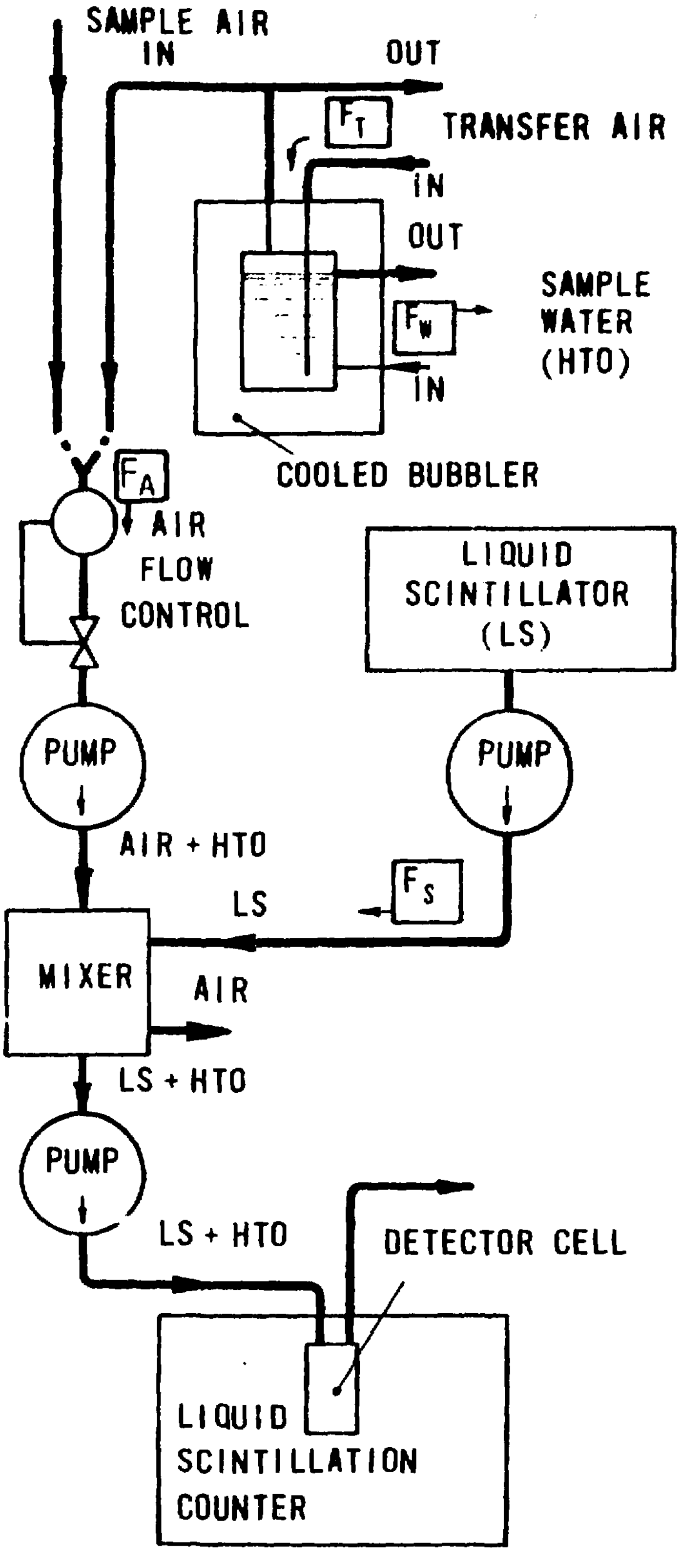
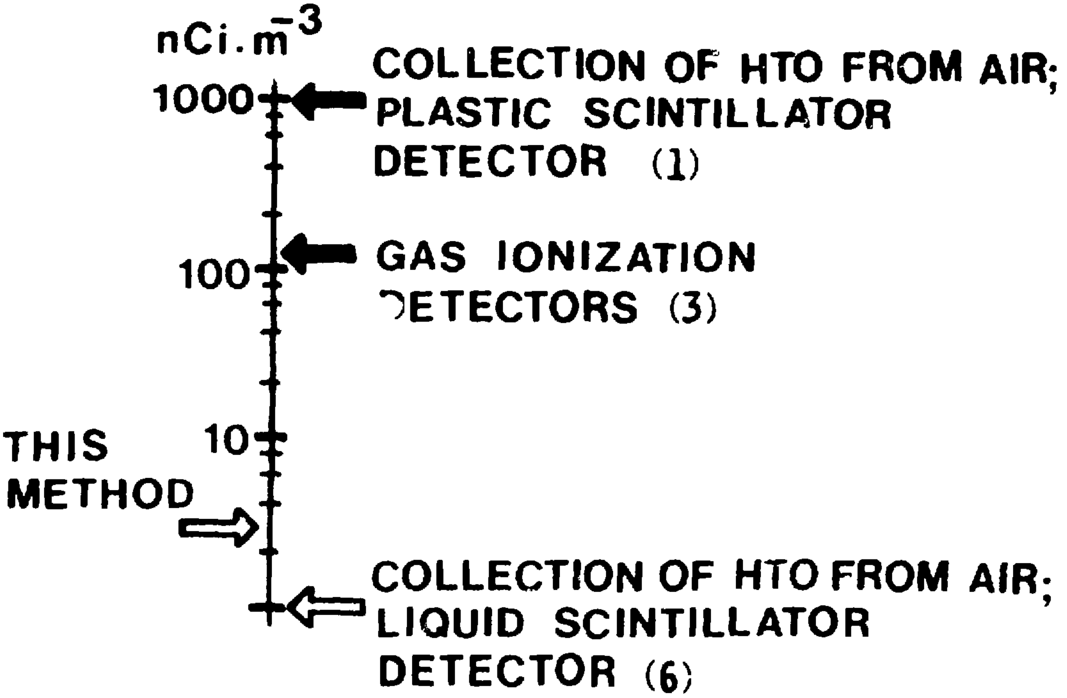


Figure 1. Flow schematic

COMPARISON WITH OTHER METHODS FOR AIR



COMPARISON WITH OTHER METHODS FOR WATER

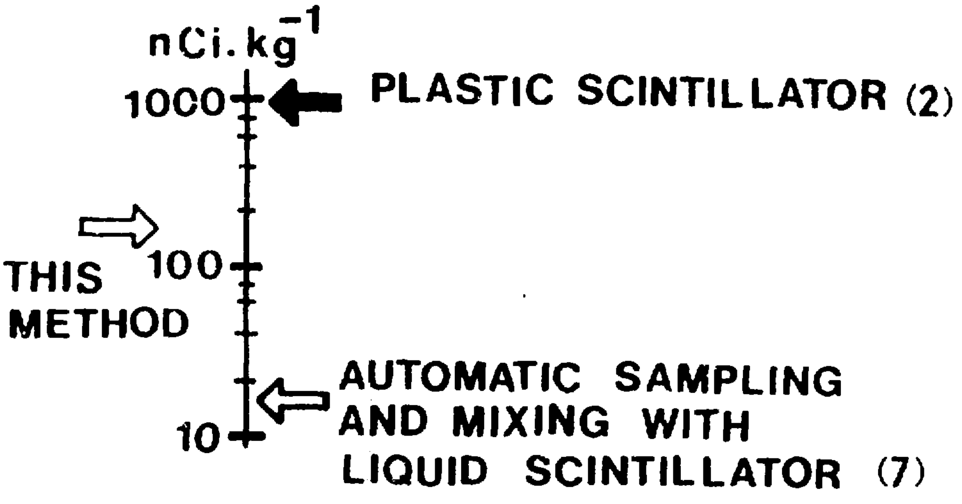


Figure 3. Comparison of the sensitivity of the new method with those of other methods. Black arrows show methods that are currently on-line.

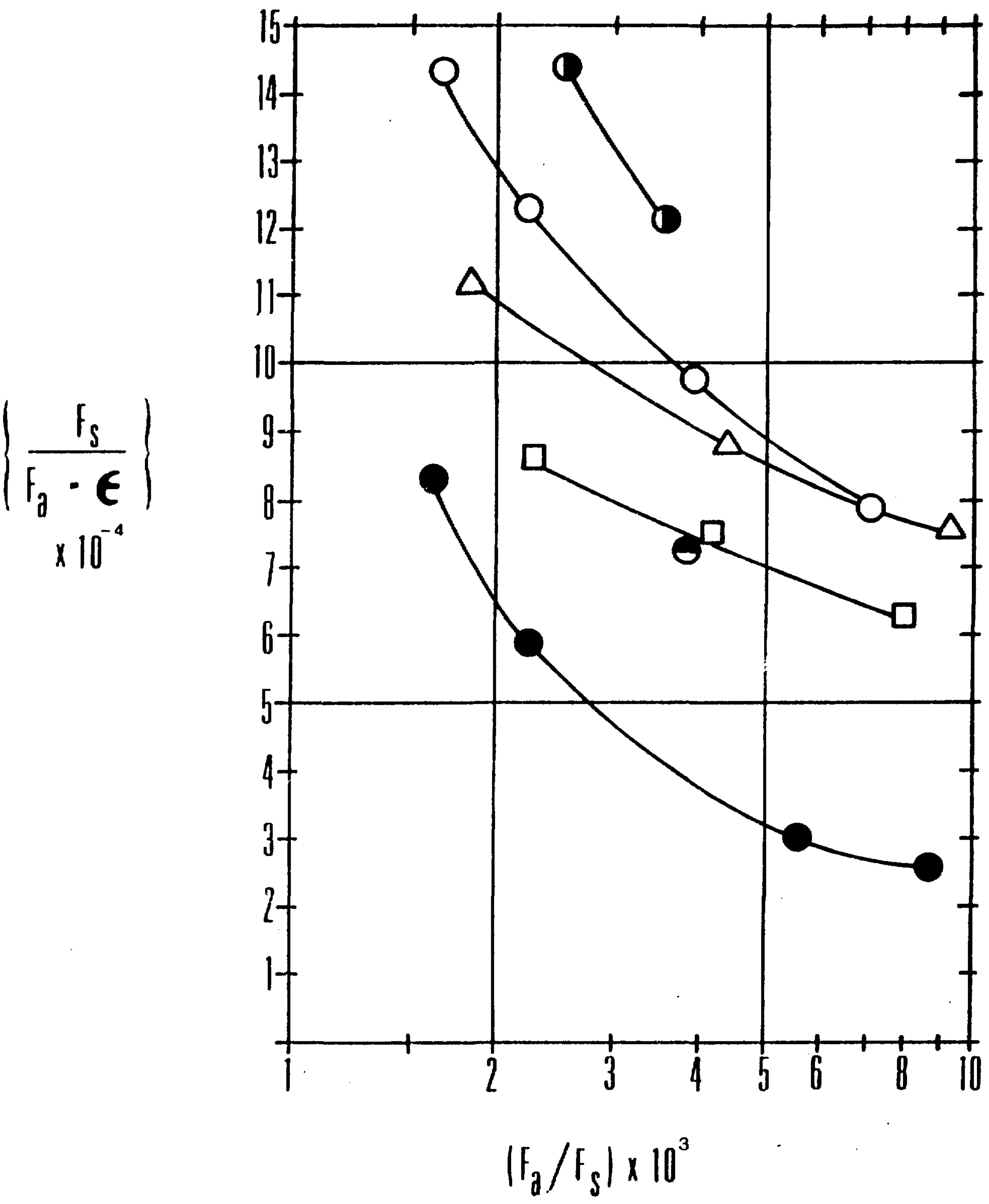


Figure 2. *Transfer of HTO to LS at various values of (F_a/F_s). The ordinate is $[\epsilon \cdot F_a/F_s]^{-1}$; the lower this quantity, the lower is the minimum detectable activity of tritium.*

Key:
- ○ *Ready Solve (Beckman)*
- △ *Aquasol (New England Nuclear)*
- □ *Handifluor (Mallinckrodt)*
- ◐ *Ready Solv VI; 5 g/m³*
- ◒ *Ready Solv VI; 15 g/m³*
- ● *Ready Solv VI + 10% water*

The air humidity was 10 g/m³ except where noted.

Question:

I may have missed it, but what is the response time of this system?

Answer (Dick Osburn):

I didn't say. It depends on whether you are using it as a water monitor and just what the design of the bubbler is. It's got to be on the order of several minutes. From the air, once in the air phase on through to the detector, the rise time would be a few minutes. We are at the early stage of development where bits and pieces are spread along the bench and as we compact it more we can get the response time down to 2-3 minutes.

APPLICATION OF A MICROPROCESSOR SYSTEM TO STREAM MONITORING*

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ABSTRACT - Low-level liquid wastes originating from the Oak Ridge National Laboratory (ORNL) are discharged, after treatment, into White Oak Creek, which is a small tributary of the Clinch River located in east Tennessee. Samples of White Oak Creek discharges are collected at White Oak Dam by a continuous digital proportional water sampler and analyzed weekly for radioactivity. The sampler contains a control system with a microprocessor that has been programmed to solve nonlinear weir equations. This system was designed and installed at ORNL by the Instrumentation and Controls Division and was tested by the Environmental Surveillance and Evaluation Section of the Industrial Safety and Applied Health Physics Division. The control system was designed to measure water flow rates from 0 to 334 ft^3/sec to within 0.1%. Results of our test program and possible applications to other liquid sampling needs are discussed.

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*Research sponsored by the Department of Energy under contract with the Union Carbide Corporation.

BRIEF DESCRIPTION OF WHITE OAK AREA

White Oak Creek (WOC) and Melton Branch tributary surface streams, which flow through the Oak Ridge National Laboratory (ORNL) reservation, receive treated low-level radioactive liquid waste which originates from various Laboratory operations. The creek receives additional low-level liquid waste generated by seepage of radioactive materials from intermediate-level liquid waste holding ponds, hydrofracture sites and solid waste burial grounds. Before converging with Clinch River, both streams flow into White Oak Lake (WOL), a 20-acre (8-ha) impoundment formed in 1943. White Oak Lake serves as the final settling basin for ORNL waste management. The lake was created using a highway fill dam, forming White Oak Dam (WOD), see Fig. 1. The spillway was constructed of a steel sheet piling wall driven in the form of a square around the upstream end of the concrete culvert forming a weir through which WOC flows. A vertical sliding gate is used to control flow water at elevations between 744 and 750 feet above sea level. Water from WOL discharges through the weir at an average flow rate of 15 ft^3/sec (425 liters sec^{-1}) with minimum of ∿ 3 ft^3/sec to a maximum of ∿ 1000 ft^3/sec. The creek meanders for approximately 0.6 miles (1 km) and empties directly into the Clinch River.

The criteria for acceptable amounts of pollution discharged into the environment have been repeatedly lowered. In order to characterize accurately the concentration of pollutants in the environment, a large number of samples are often needed. Logistic and economic problems of sample collection establish practical limitations on the number of samples taken. In order to determine whether the Laboratory's discharges meet federal water-quality standards, monitoring the quality and quantity of the liquid effluent is required. The importance of monitoring WOD, which is the last control point for the liquid discharges from ORNL, require flow-proportional sampling. This sampling is controlled by a microprocessor.

INTRODUCTION TO MICROPROCESSORS

A microprocessor (μP) is the vital part of a microcomputer. It is the central processing unit which performs the calculating and decision functions, and in microelectronics μP's are fabricated on a single silicon substrate of very small size (half a centimeter on a side). These μP's, along with memory and peripheral circuitry, form the complete microcomputers whose complexity falls somewhere between conventional minicomputers and small hand-held calculators. These complete microcomputer systems are assembled on a board whose area does not exceed this page. Microprocessors can lower the cost and increase the flexibility of electronic equipment. When many functions must be performed, μP's can be used economically to replace or upgrade handwired or random-logic designs involving scores of standard digital equipment (To76).

At the present time, about 20 U.S. companies are now manufacturing some 30 different designs of μP chips, ranging in price from $10 to $300. More than 120 companies are incorporating these chips in microcomputer

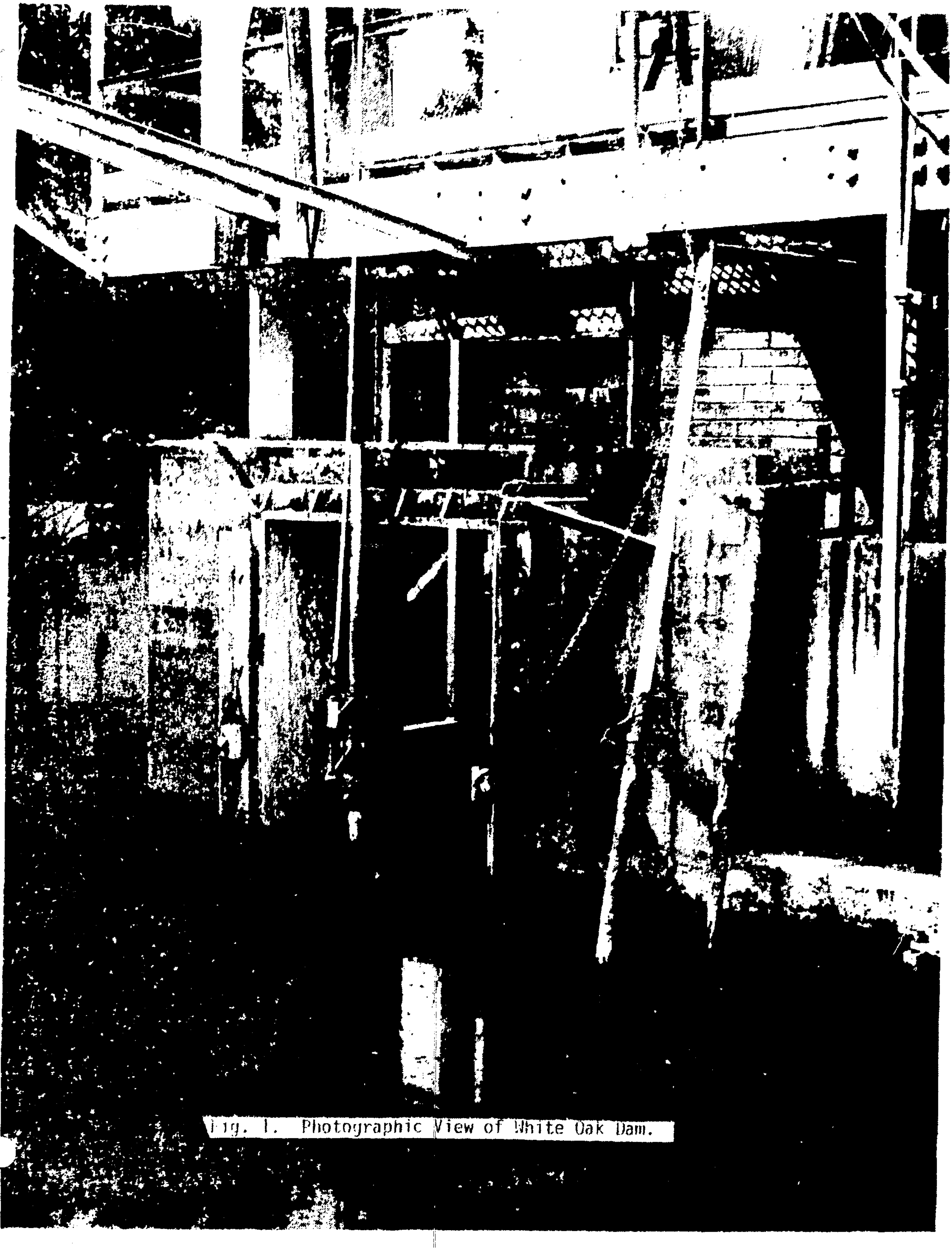

Fig. 1. Photographic View of White Oak Dam.

systems selling for $100 and up, and the number of applications for μP's is proliferating daily in industry, banking, power generation, and consumer products (To77).

DESIGN OF SAMPLING SYSTEM

The microprocessor-controlled sampling system for WOD was designed and installed at ORNL by the Instrumentation and Controls Division (Ro75) and tested by the Environmental Surveillance and Evaluation Section of the Industrial Safety and Applied Health Physics Division. The system was so designed that it takes an equal amount of a water sample for each 10,000 ft^3 of water that flows over the weir. The pre-established requirements for the system were that it should measure 10,000 ft^3 of water within 1% and that it should take repeatable water samples at all anticipated flows. The computation of the volume of discharged water involves two different nonlinear relationships between water height and water flow and requires integration. The microcomputer used for the system was the Intel SIM 8-01, utilizing the 8008-1 microprocessor; this system is equipped with 1K of random-access memory and 2K of programmable read-only memory and is on one printed-circuit board. It is interfaced to an analog input signal, a mechanical water sampler, and three control switches. The control program performs a nonlinear conversion from water height to water flow and numerically integrates flow. The program also controls the mechanical sampler and can test most of the system, including itself, for proper operation (Ro75).

The relationship between height (H), in feet above sea level, and water flow (F), in ft^3/sec is given in Roberts (75). There is a point of discontinuity in the relationship between F and H when the water height is above 750 ft (top of dam) above sea level, because the water actually spills over the dam during heavy rains, thus changing this relationship. Since the F and H relation is nonlinear and in two distinct pieces and the stability and calibration accuracy of analog function generators are not sufficient for this application, digital conversion is more accurate and repeatable than analog conversion (Ro75). Also, typical water-flow rates might be anywhere from < 3 ft^3/sec in dry weather to > 1000 ft^3/sec in spring flooding. This dynamic range of $> 333:1$ is more accurately accommodated by a digital system than an analog one. Further, considering that the flow must be integrated over times as long as an hour between samples, digital integration is preferred to analog, as a digital integrator does not decay or drift over a long period of time (Ro75).

The major computational tasks for this system are the conversion from height to flow and the numerical integration of flow. To accomplish this objective, H is converted to F by a table-lookup and linear interpolation method rather than by a time-consuming floating-point solution to the equations given by Roberts (75). The lookup table of flow values corresponding to 251 equally-spaced height values is stored in one 256 x 8 PROM (programmable read-only memory). This lookup table is presented in Table 1; the flow in ft^3/sec times the time between samples in sec

Table 1

Interrelationships Among Elevation, Flow, and Sampling Time

Elevation		Flow		Time Between Samples (HRS:MIN:SEC)
Scaled	Actual (Ft)	Scaled	Calculated (ft^3/sec)	
0	745.00	0.0	0.00	1340: 56: 39
25	745.13	0.1	0.58	4: 48: 41
50	745.25	0.2	1.61	1: 43: 53
75	745.38	0.3	2.93	0: 56: 53
100	745.50	0.5	4.50	0: 37: 03
125	745.63	0.7	6.27	0: 26: 33
150	745.75	0.9	8.24	0: 20: 13
175	745.88	1.1	10.37	0: 16: 04
200	746.00	1.4	12.66	0: 13: 09
225	746.13	1.6	15.10	0: 11: 02
250	746.25	1.9	17.68	0: 9: 25
275	746.38	2.2	20.39	0: 8: 10
300	746.50	2.5	23.23	0: 7: 10
"				
"				
"				
2000	755.00	200.0	1861.0	0: 0: 53

For $H < 750$ ft: ft^3/sec * Time between Samples $\gtrsim$ 10,000 ft^3

For $H > 750$ ft: ft^3/sec * Time between Samples $\gtrsim$ 100,000 ft^3

will always give a product of 10,000 ft^3 regardless of flow. Further, the stored F and H values are scaled to simplify the digital read-out systems; these scaled values correspond to actual values as seen in Table 1.

The completed μP system is shown in Fig. 2 as a block diagram (Ro75). Since the digital voltmeter accepts voltages in a range from 0 to 2V, the analog voltage, which is proportional to water height, is scaled to this range. A pulse actuates a digital time that controls the mechanical water sampler when the flow data are summed to an integrated water volume of 10,000 ft^3. Water is sampled by diverting a continuous flow of thoroughly-stirred water, and is pumped from the stream into a container for 3.3 sec (Ro75).

Figure 3 shows the panel display of the completed microprocessing system. The scaled water height and flow numbers (Table 1) are displayed on the front of the system. Controls on the front panel (Fig. 3) include a switch to reset the integrated water volume to zero for the purpose of initializing the system, a display inhibit switch for turning the water-height and water-flow displays off and on, a switch which allows an operator to take a water sample at any time, and a test switch to perform several checks of the system hardware and program (Ro75).

A schematic diagram showing the overall sampling and control system is shown in Fig. 4. The height of the water is measured by a float inside a stilling well. This float is connected to a pulley assembly which translates the vertical movement of the water height to the rotation of the pulley shaft. This rotation of the pulley shaft is connected to a chart recorder for the purpose of backup in determining the integrated flow. A potentiometer connected to the pulley shaft also produces an analog voltage proportional to the water level. The analog voltage is then scaled to a range from 0 to 2V for acceptance by the digital voltmeter (Ro75). As stated previously, when the flow data is summed to an integrated volume, a pulse activates the timer for control of the solenoids on the mechanical water sampler. The mechanical funnel directs the water sample into either the low ($H < 750$ ft) or high ($H > 750$ ft) flow-sample container once the solenoid is activated. The water is collected in two 55-gallon stainless-steel drums for a one-week period. One drum receives the sample during low-flow conditions, the other during high-flow conditions.

Several alarm set points are built into the μP system (i.e. pump failure, power failure, etc). These alarm signals are transmitted to the Environmental Monitoring (EM) Control Center using telephone lines.

In addition, the creek activity is monitored on a "real-time" basis using five submerged G-M tubes in a lead-shielded adjustable head pot (see Fig. 4). The signal from these tubes is transmitted over telephone lines to the EM Control Center.

FINISHED WATER QUALITY MONITOR

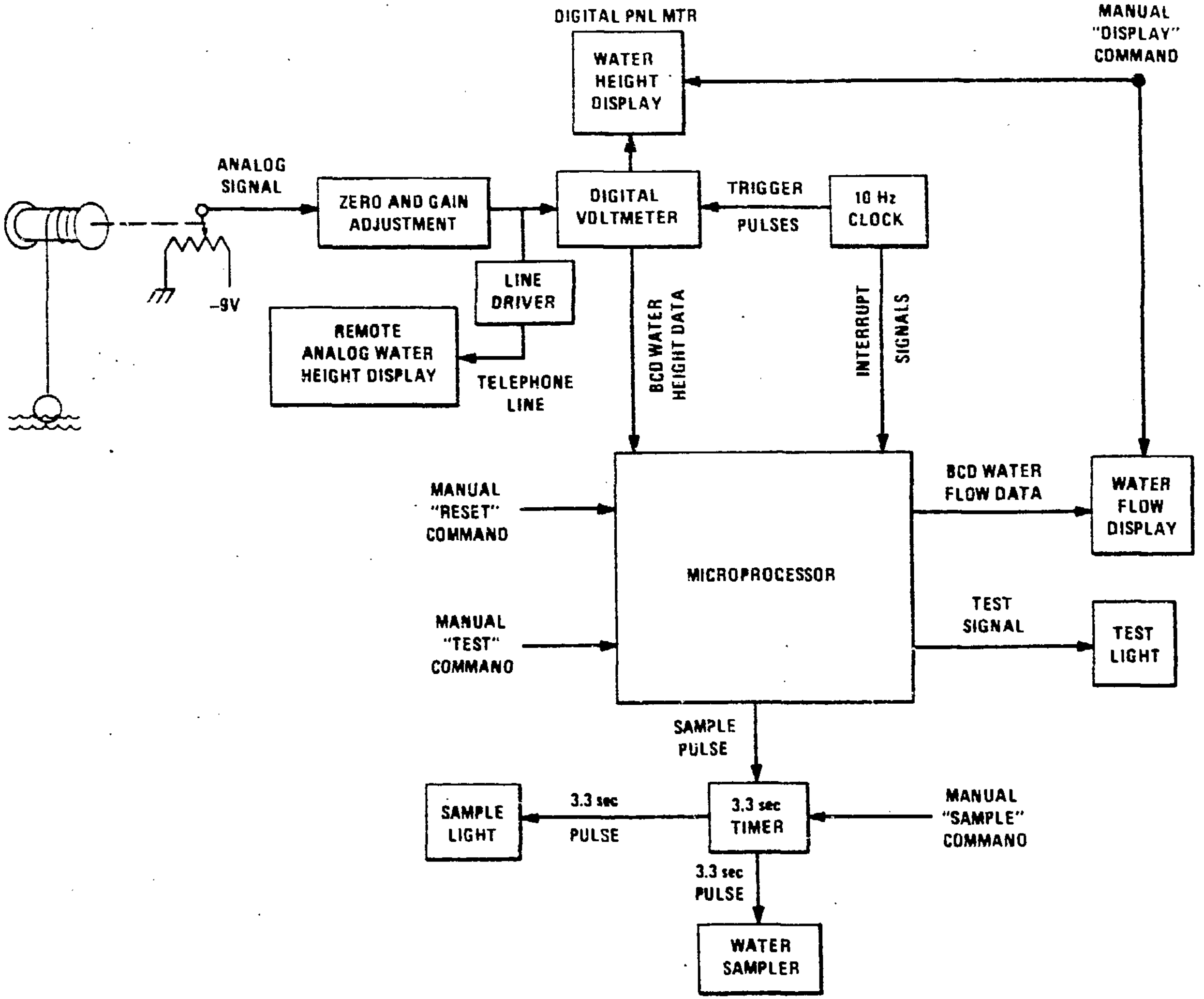

Fig. 2. A Block Diagram of The Finished Microprocessor.

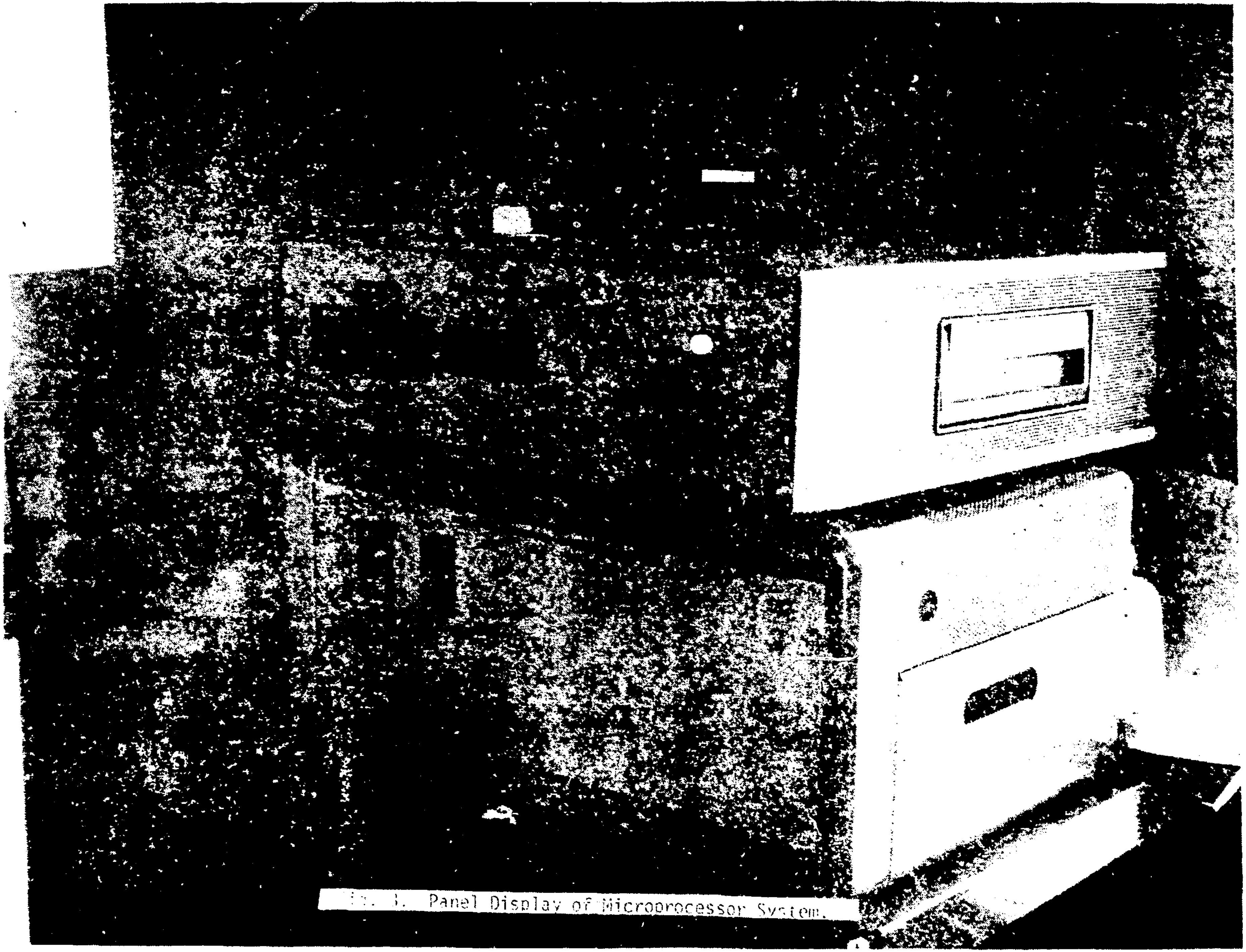

Fig. 1. Panel Display of Microprocessor System.

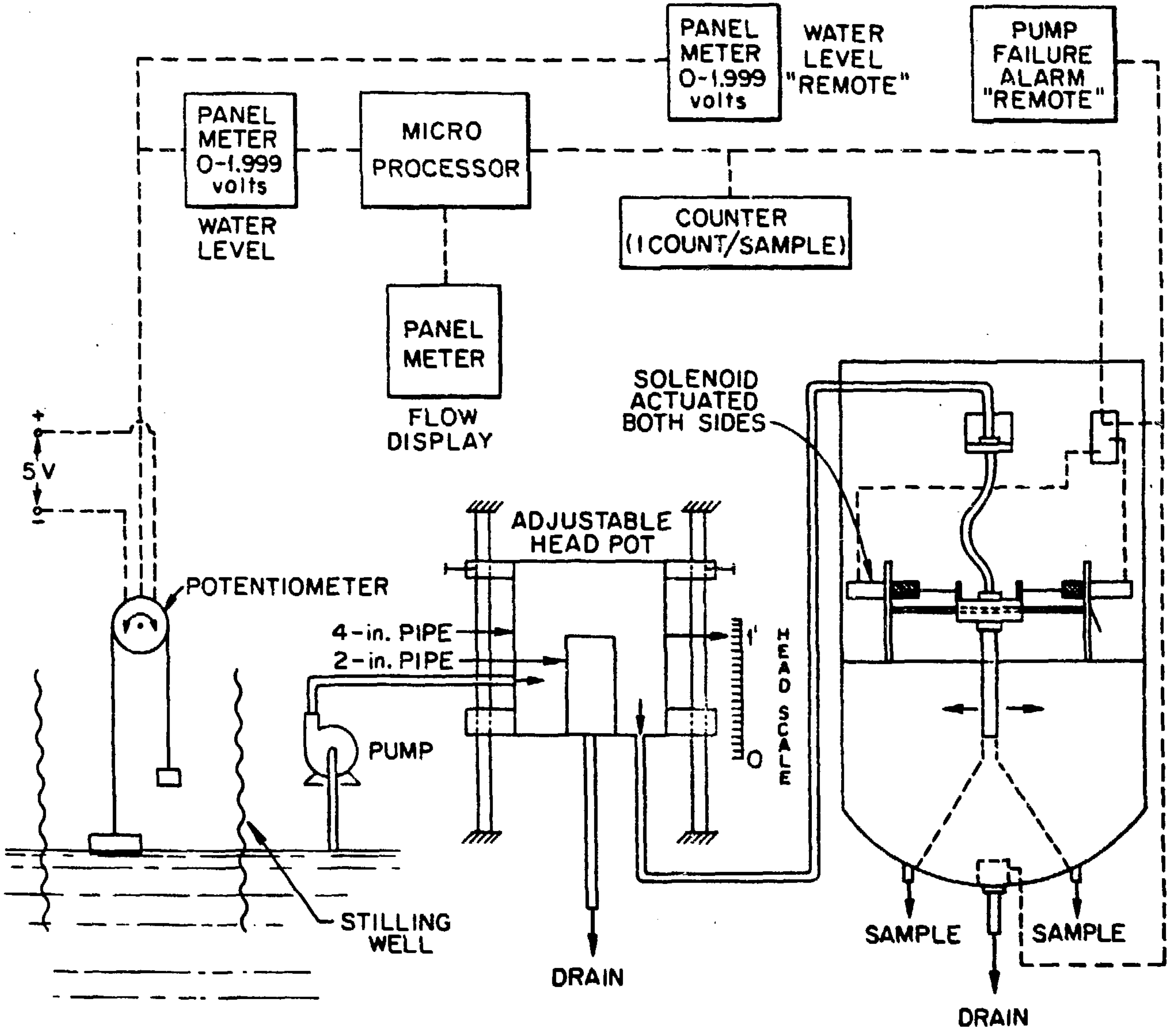

Fig. 4. Schematic of Total Sampling System.

OPERATING EXPERIENCE

The stream monitoring system described has been in operation since September, 1976, as part of the routine water sampling program performed by the Environmental Surveillance and Evaluation Section at ORNL. This water sampler is particularly important because the location of the sampler is at WOD, which is the last control point for the liquid discharges from ORNL. The original system costs were fairly low ($10,000-$20,000) and thus far, the reliability of the instrument has been approximately 90%.

Several tests, using a chart recorder system connected to the pulley shaft for determining the integrated flow, have shown the μP system to be working correctly in the past. The major operational difficulties encountered have been in the areas of voltage changes, telephone transmission, and temperature variations. The latter has become a significant problem recently, and the system is presently being reevaluated. A backup system that takes a sample at regular time intervals is used when the μP system is not operational. Another problem area is accurately determining the relationship between F and H when the water height is above 750 ft; in this case, the water actually spills over the dam. Because the uncertainty greatly increases during these conditions, water samples are collected separately for evaluation.

SUMMARY

A stream monitoring system incorporating an Intel 8008-1 μP has been designed, built, installed, and operated at ORNL. A μP was chosen over hard-wired logic, as a μP has programmable arithmetic capability as well as a lower cost of construction. Further, a μP can be salvaged and reprogrammed for use in another system when it is removed or replaced. The biggest operational difficulties encountered thus far have been in the areas of voltage changes, telephone transmission, and temperature variations. This system, however, has shown itself to be a reliable instrument in the past, and various test programs have shown the μP to be operating as designed. This system has been found to be a remarkable tool for environmental monitoring in obtaining a water sample that is truly proportional to the stream flow. We feel this type of system would be applicable in those cases where flow-proportional or redundant samples are needed, i.e., stack monitoring, air sampling, liquid effluent monitoring, etc.

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ACKNOWLEDGMENTS

Acknowledgment of the design and construction of the microprocessor system by staff members of the Instrumentation and Controls Division of the Oak Ridge National Laboratory is made. Their names are included in Reference Ro75. Mr. E. D. Gupton's initiation of this project is also acknowledged by the authors.

TRITIUM MONITORING INSTRUMENTATION AT ROCKY FLATS

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ABSTRACT

Monitoring for tritium at Rocky Flats is conducted to evaluate health hazards in the work area and assess environmental conditions. A wide variety of monitoring methods are employed including HTO bubblers, ionization chamber gas analyzers, surface smears, silica gel traps, and special direct gas counting. The instrumentation and methods used include both selective and nonselective techniques to evaluate tritium levels in room air, gas effluents, surface contamination, and liquid discharge. Liquid scintillation counting is used for most quantitative measurements.

INTRODUCTION

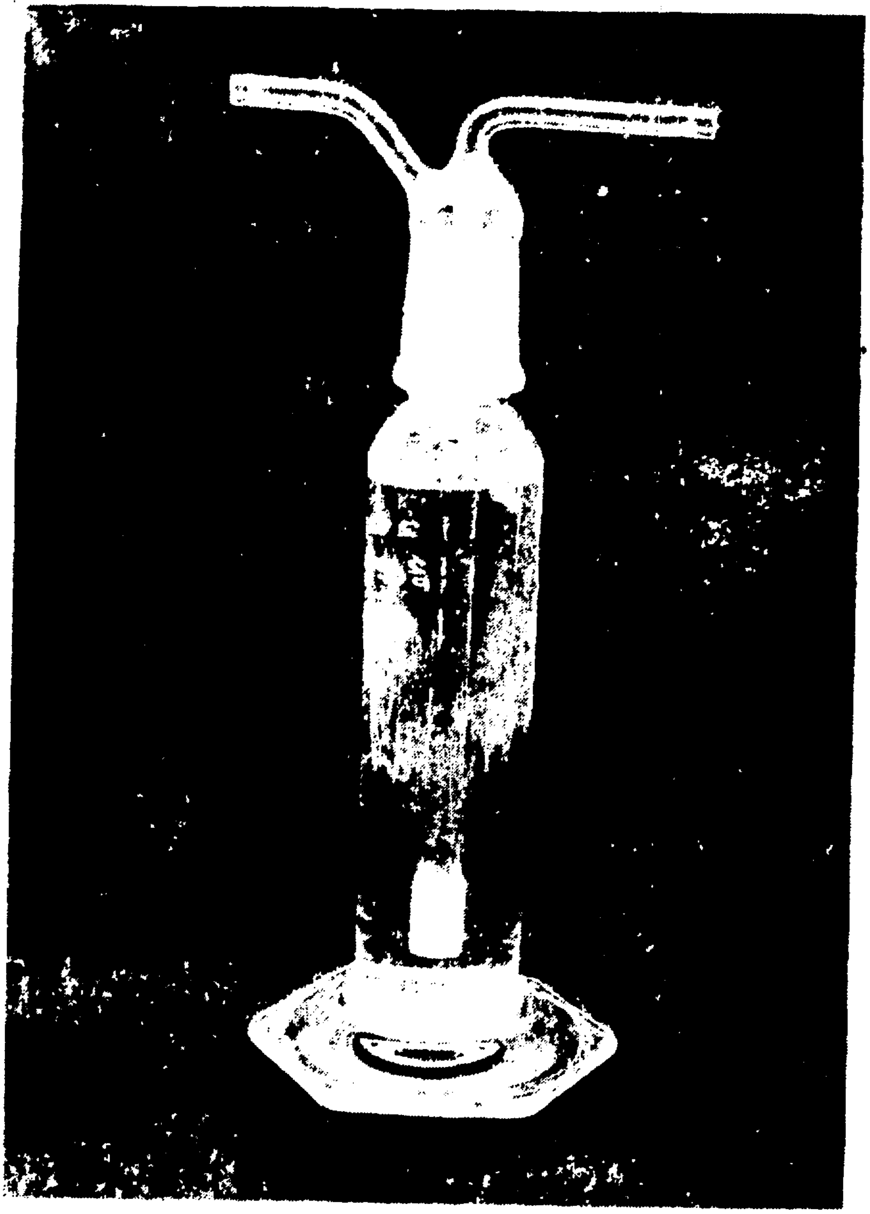
Rocky Flats does not handle tritium as a separate radioisotope in it's production recovery operation, therefore a routine tritium monitoring program was not a standard procedure prior to 1974. Because Rocky Flats received a shipment of tritium contaminated material and an environmental release occurred, a routine tritium monitoring program was instituted. Rocky Flats presently handles special source (SS) and non-SS material which may contain trace quantities of tritium on a case by case basis. Because of the limited tritium handling and, therefore, low tritium background, the monitoring instrumentation and methods employed are designed for low level analysis. Non-integrating gas methods are sensitive to less than 10 microcuries per cubic meter ($\mu Ci/m^3$) while liquid methods are used in the low picocurie per millilitre (pCi/ml) range. Integrating methods (HTO bubblers) that are nonselective are adequate for in-plant monitoring applications when combined with knowledge of the operations being performed.

ROOM AIR AND EFFLUENT AIR MONITORING

Room Air Monitoring for Tritiated Water (HTO)

Activities at Rocky Flats, in which potentially tritium-contaminated material is handled, rely upon an HTO bubbler for routine room air monitoring. These bubblers contain water or ethylene glycol which will trap tritium in the oxide form. These bubblers are used for long-term survey work rather than as an immediate alert to a tritium release. Room air bubblers are located in all areas where tritium contaminated material is handled.

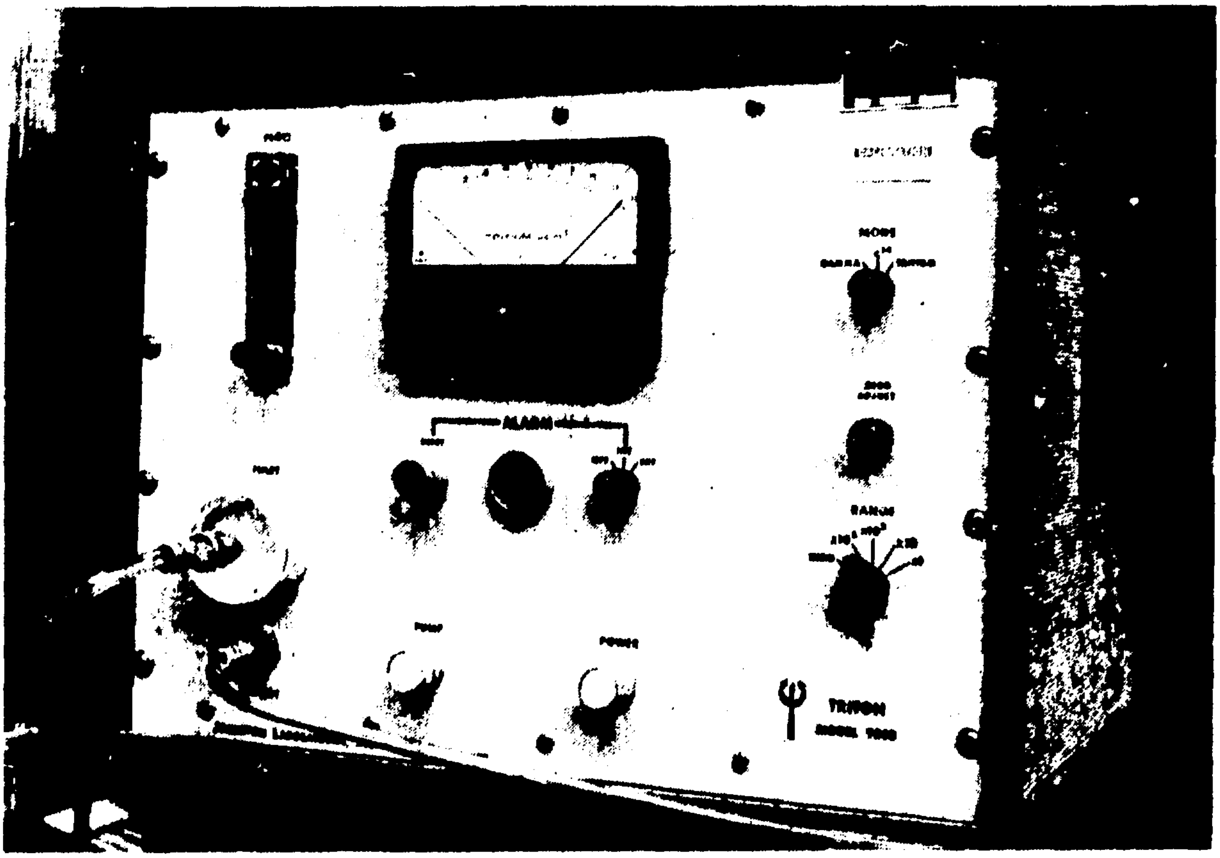
The sampling device is a glass wash bottle or flask with a fritted glass dispersion tip on the end of an inlet tube. (See Photograph I.) Air at a rate of two litres per minute is bubbled through approximately 250 ml of water or ethylene glycol for a two to three day period. Trapping efficiency for all HTO bubblers is greater than 90%. A 40 ml aliquot from the bubbler is taken and placed in a 2-oz wide-mouth bottle for transportation to an analytical laboratory. A 5 ml aliquot is taken and added to 17 ml of liquid scintillation cocktail (Amersham/Searle PCS) which is then analyzed for tritium via liquid scintillation counting techniques. Room air concentrations are expressed in pCi/m^3. Normal "background" levels run 150 pCi/m^3, compared to the maximum permissible concentration in a controlled area of 5 $\mu Ci/m^3$.



PHOTOGRAPH I.
HTO BUBBLER

Room Air Monitoring for Tritium Gas (T_2) and Vapor (HTO)

Operational monitoring involving tritium in the gaseous phase relies upon a Johnston Laboratories Triton 955B Analyzer for routine room air monitoring. (See Photograph II.) The instrument utilizes a positive displacement pump to draw air through a submicron filter and an electrostatic precipitator, thereby removing interfering particles, ions, and smoke. Ionization due to radioactivity within the ionization chamber is detected and amplified by an electrometer. The output current for the electrometer is connected to a panel meter and a recorder jack. An indicating meter scale is calibrated from direct readings in $\mu Ci/m^3$.



PHOTOGRAPH II.
TRITON 955B ANALYZER

Effluent Air Monitoring

Stack exhausts are continuously monitored for tritium oxide using the bubbler system described for room air monitoring. These bubblers are located on stacks leading from all buildings with a potential for tritium work. Typical background levels are 150 pCi/m^3 with fluctuations from 100 pCi/m^3 to 300 pCi/m^3. All bubbler samples are counted on a refrigerated liquid scintillation counting system to lower the background from the cocktail solution. The minimum detectable amount (MDA) of 20 pCi/m^3 is a function of flow rate, volume, and instrument background. The variable energy windows on the counter are set to accept the maximum tritium energy spread. This can be done because normal Rocky Flats samples lack crossover from other beta-emitting materials.

GLOVEBOX MONITORING

In-Line Tritium Gas (T_2) Monitoring

The Johnston Laboratory Triton 955B Analyzer is also used to monitor in-line operations involving possible tritium-contaminated materials for determining background fluctuations only. This analysis is used to alert area personnel of an increase which could lead to glove permeation and personnel exposure. Care is taken to use the analyzer in an air-atmosphere box. Analyzers of this type can be used in atmospheres containing inert gases if they are properly calibrated. This analyzer may be used in areas with gamma-producing radioactive materials by shielding the analyzer with 1/4 inch lead and taking care to filter any particulate material prior to the gas entering the unit. Typical Triton units are set to alarm at 10 $\mu Ci/m^3$. Normal lower detection limit is 2 $\mu Ci/m^3$.

Surface Contamination - - In-Line and Out

In-line and out-of-line surface contamination monitoring is qualitative and has traditionally been accomplished at Rocky Flats by smearing approximately 100 cm^2 area with Reeve Angel® 5.5 cm filter paper. In-line smears are coated with glycerin prior to use. Smears are placed in a 2-oz narrow-mouth bottle with 40 ml of water. The water is shaken with the smear and allowed to equilibrate for two hours. The water is then distilled to remove alpha radioactivity. A 5 ml aliquot is taken from the distillate and added to 17 ml of liquid scintillation cocktail. The sample is then analyzed for tritium via liquid scintillation counting techniques.

Out-of-line smears with alpha radioactivity greater than 500 disintegrations per minute (dpm) per 100 cm^2 area are placed in a 2-oz narrow-mouth bottle with 40 ml of water. The water is shaken with the smear and allowed to equilibrate for two hours. The water is then distilled to remove alpha radioactivity. A 5 ml aliquot is taken from the distillate and added to 17 ml of liquid scintillation cocktail. The sample is then analyzed for tritium via liquid scintillation counting techniques.

Smears with alpha radioactivity of less than 500 dpm are placed in a 2-oz narrow-mouth bottle with 10 ml of water. The water is shaken and allowed to equilibrate for two hours. A 5 ml aliquot is taken and added to 17 ml of liquid scintillation cocktail. The sample is then analyzed for tritium by liquid scintillation counting techniques. Smear results are expressed in pCi/100 cm^2 area.

ENVIRONMENTAL SAMPLING

Liquid Effluents

Process liquid wastes are sent to the waste treatment facility through process waste drains. The liquid effluent is not sampled specifically for tritium prior to the waste treatment facility because of the extremely low levels of tritium present. If a specific operation generates liquid waste with possible tritium contamination, this liquid is held, sampled and analyzed prior to introduction into the normal process waste system.

Storage tanks and the solar evaporation ponds are continuously monitored for tritium using grab-sampling techniques. Analysis is done by liquid scintillation counting. Results are reported in pCi/ml.

Ambient Air Sampling for Tritiated Water (HTO)

Environmental monitoring for HTO in ambient air utilizes silica gel as a collective medium.(See Photograph III.) The sampling device consists of a 1 litre per minute air pump which operates continuously. The pump is powered by a 12-volt DC regulated power supply, converted from 115-volt AC, or in case of power failure, from a backup battery pack. The tritium sample tube is constructed of Pyrex, filled with silica gel to collect moisture from ambient air. The silica gel container is maintained between 40 and 90°F using fans and heaters. This protects against loss of water vapor from overheating or lack of absorption because of extreme cold.

Samples are collected on a weekly basis. The Pyrex tube is removed from the sampler, and the silica gel is removed and heated by heating mantles to drive off absorbed water. The water is collected and analyzed for tritium using liquid scintillation counting techniques. Results are reported in pCi/m^3.

PHOTOGRAPH III.
SILICA GEL SAMPLER

SPECIAL SAMPLING

Any container received at Rocky Flats whose contents have the potential for containing tritium is gas sampled and analyzed for tritium before the container is unpacked, processed, or otherwise handled. This sampling and analysis is done in a tritium surveillance laboratory.

Materials (both SS and non-SS) are packaged in a variety of ways. These include drums from 10 to 80 gallon sizes, pressure cookers, multisize produce cans, bottles, vials, etc. Each type of container requires a specific sampling method. The method must insure that a representative gas sample can be taken safely, without doing damage to the material being shipped. Since material may contain plutonium, this adds another dimension to the sampling method.

Each sampling method allows a gas sample to be taken while the remaining head-space gas is sealed in the original shipping container. This assures that no material will be released to the atmosphere during the sampling or analysis. Absolute filters (0.3 to 2.0 micron) are used to stop any particulate radioactive material from contaminating the gas sample.

Examples of the sampling hardware used include a produce can punch system, drum punch system, and a tube-punching fixture. The produce can punch (See Photograph IV.) uses a neoprene gasket to seal an O-ring fitting to the can. A 1/8-inch needle is sealed in the O-ring fitting and the complete system is evacuated. **Some** head space sampling procedures place the entire produce can in a sealed container and evacuate the outer container or flush with inert gas (argon). Evacuation of the outer container can cause the produce can to collapse or become distorted. The inert gas flushing method dilutes the sample of interest. To sample the can, the pump is closed off and the needle is driven to a preset depth into the can. A gas sample can then be taken for tritium or general gas analysis. The can and manifold remain sealed until the analysis is complete. The same general method is used for drums. (See Photograph V.) A drum sampler uses an O-ring seal on the drum and a solid punch is used to penetrate the drum lid.

A tube punch fixture (See Photograph VI.) is used to extract a sample from metal tubing. The tube is sealed in the fixture with an O-ring and a positive drive punch opens the tube.

Gas to be tritium analyzed is expanded to a known pressure into a previously evacuated liquid-scintillation counting vial fitted with a rubber septum. Gas pressure must be below 200 torr. The vial is then removed from the sampling manifold, and liquid scintillation cocktail (Amersham/Searle PCS) is added using a hypodermic syringe with the plunger removed.

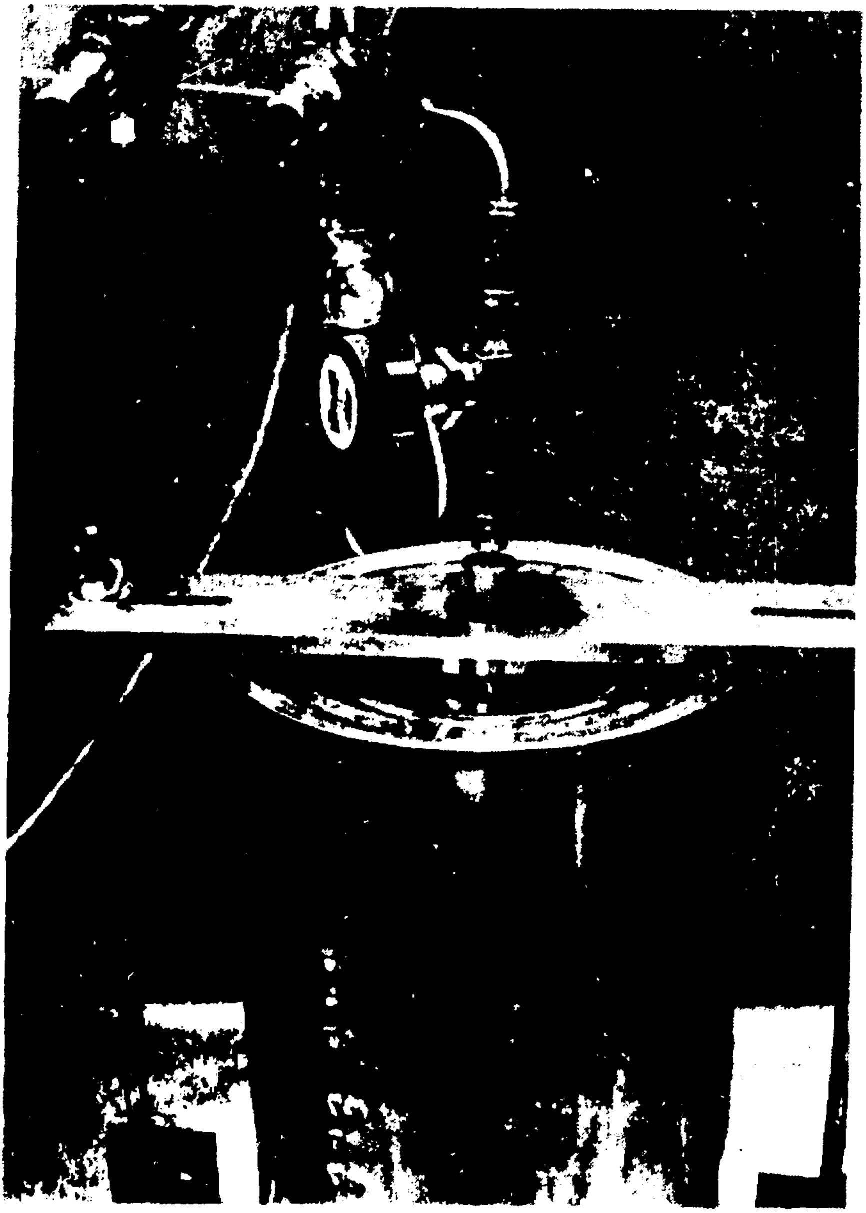
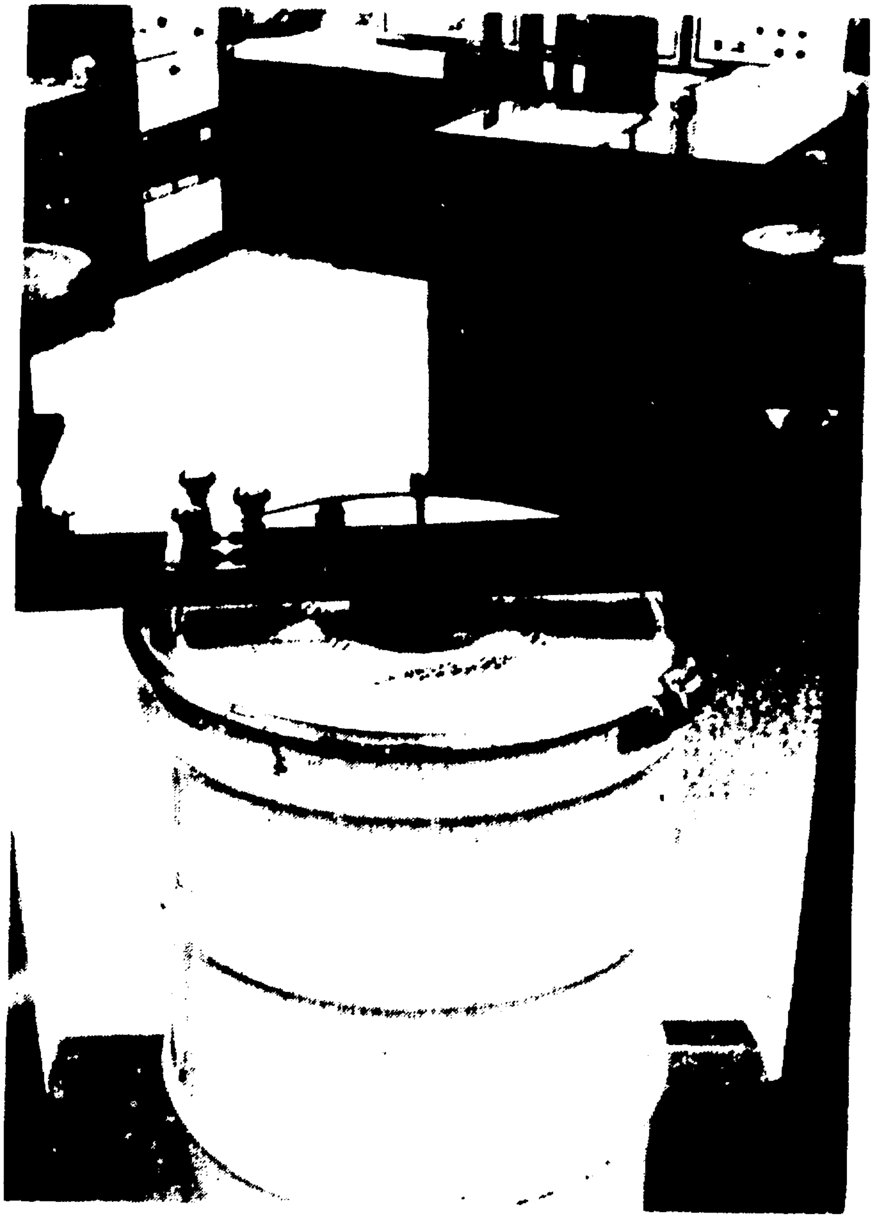


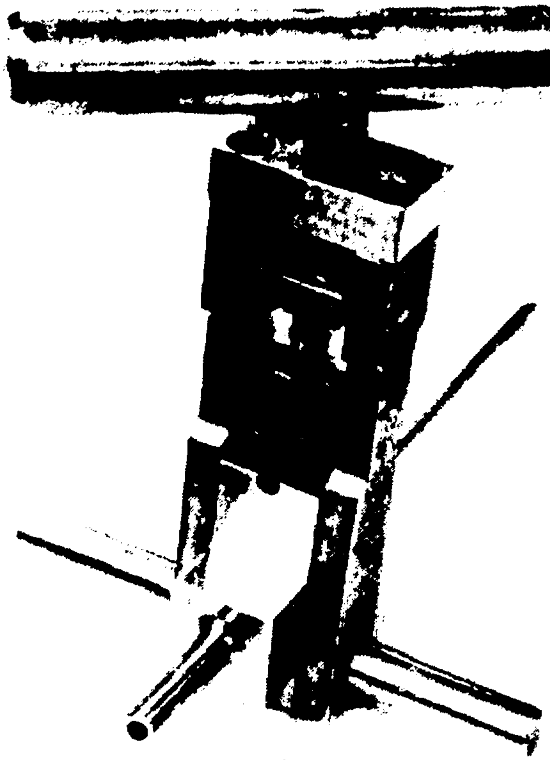
FIGURE [illegible]

[illegible] THE PUMP [illegible]



PHOTOGRAPH V.

DRUM PUNCH FIXTURE

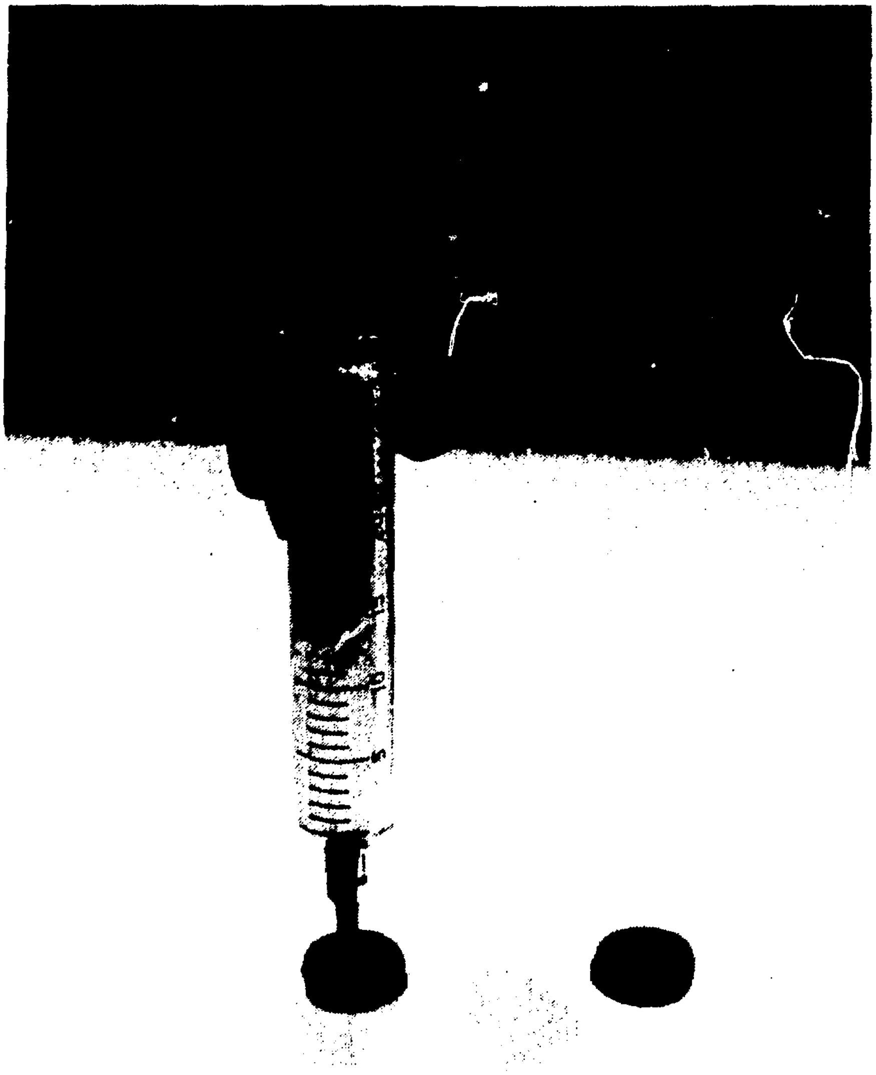


PHOTOGRAPH VI.
TUBE PUNCH FIXTURE

(See Photograph VII.) Cocktail is added until the pressure in the vial equilibrates with atmospheric pressure. The vial is then placed in a liquid scintillation counter for counting. Counting the sample in a normal liquid scintillation counter completes the analysis. A second gas sample may be taken for complete gas analysis by mass spectrometry or gas chromatography if needed.

Standard liquid scintillation counting methods are used in correcting counts per minute (cpm) to disintegrations per minute (dpm). Efficiency curves are generated using a quenched standard set. The dpm value is then converted to tritium activity in curies or microcuries. The gas volume associated with this activity is corrected to standard temperature and pressure (0°C and 760 Torr). The final results are reported as $\mu Ci/m^3$ at STP in the gas phase.

This method determines elemental tritium (T_2, HT, or DT), oxides of Tritium (HTO, T_2O) and organotritium ($CxHyT_2$). The makeup of the matrix gas (helium, argon, air, etc) does not interfere with the analysis. The lower detection limit is 2 $\mu Ci/m^3$. The method is standardized against known tritium mixes from Matheson Gas Products and found to give greater than 80% recovery for sample concentrations of up to 0.01 Ci/m^3.



PHOTOGRAPH VII.
SCINTILLATION COUNTING VIALS

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TECHNICAL SESSION F

RADIATION DETECTORS II

Chair: William McCarthy

A 16-DETECTOR ALPHA SPECTROMETER USING 1 MULTICHANNEL ANALYZER*

by

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ABSTRACT

An alpha spectrometer containing 16 independent detectors and utilizing one 4096-channel multichannel analyzer (MCA) was constructed from commercially available modules. The spectrometer was designed specifically for the counting of low levels of radioactivity in environmental samples. Gated analog routing allows spectral data acquisition into 256 channel regions of the MCA memory as if each region were an independent 256-channel MCA. External live-time clocks and 50-Mhz analog-to-digital converters control timing and acquisition on each unit of eight detectors. Spectral data output is to magnetic tape in units of 256 channels each with a unique tagword. These tapes are then read and processed, and final reports are generated, by a large Control Data 6000 series computer.

INTRODUCTION

A system has been designed which allows simultaneous data collection from 16 solid-state alpha detectors by means of routing all detector signals into one large (4096-channel) multichannel analyzer (MCA). This system was designed

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* For presentation at the 11th Midyear Topical Symposium on Radiation Instrumentation, Health Physics Society, San Diego, California, January 17-19, 1978.

as a replacement for an outdated set of eight 400-channel MCA's. These older MCA's were configured to accept data from only two detectors each and all data output was to paper punch tape or a slow printer. The new data system was designed to utilize the existing detectors, power supplies, and amplifiers while upgrading the data acquisition and output hardware. The major system design criteria were as follows:

1. Magnetic tape must serve as the primary output device allowing the capability of producing computer-readable spectral data. Each spectrum on tape must contain a tagword and at least 200 channels (20 bits per channel) of spectral data. Magnetic tape should also provide a medium for the primary historical file of raw spectral data.

2. The system must contain a high-speed parallel printer for writing out spectral data.

3. A system must be provided which has the capability of acquiring data from 16 detectors simultaneously (with the option of live time gating on each) storing data in regions of at least 200-channel resolution.

4. The system cost must remain below $30,000.

Many commercial system configurations were considered and reviewed before we focused on the present special configuration. The first system considered was one that contained eight 512-channel multichannel analyzers. Each analyzer could be purchased with a high frequency analog-to-digital converter (ADC), and high-speed gating module. In this way data would be routed into memory as two 256-channel spectra per analyzer (MCA). One problem with this system was that a common data bus would be needed to interface each MCA to a tagword generator, magnetic tape formatter, and transport.

Another problem was strictly monetary. A system of this sort would cost three times our $30,000 budget limit.

The second system evaluated was minicomputer based. The system would have provided the most flexibility and future expansion capability of any considered. However, it was also more expensive than our budget would permit. Depending upon the hardware design chosen, a minicomputer system could have cost between $40,000 and $100,000.

The third system evaluated contained four 1024-channel MCA's each handling four inputs (detectors) and interfaced through one common data bus to a high-speed printer, magnetic tape formatter, and transport. This system's major limitation was again monetary and would have cost approximately $40,000 at the time.

Since it appeared that no commercially available systems known could provide for our needs, we began to design a system which could be fabricated by a commercial vendor as a special system. The main idea was to somehow construct a system from one MCA which would be equivalent to 16 MCA's, each capable of producing a 256-channel by 20-bit spectrum with output to magnetic tape, and a high-speed paper printer. The system would also have to meet the complete capability needed by the purchase and utilization of only one 4096-channel by 24-bit* MCA.

THE SPECTROSCOPY SYSTEM

The system designed and now used in routine production counting is depicted in Figure 1. The detectors, bias supplies, biased amplifiers, and linear amplifiers for each input are not described in the figure independently, but are an integral part of each of the numbered blocks at the top. Each

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*There are 20 bits available for data in each channel. Four more bits per channel are used for area intensification, cursor location, region of interest, etc., and are not used for counting-data storage.

component is of high quality commercial grade and serves the function of producing pulses of 0 to 10 volts which are linearly proportional to the energies of alpha particles detected within the range 3.5 to 6.7 MeV.

The next components in the line are Gated Analog Routers (GAR's). There are four GAR's which serve the purpose of routing spectral information through the analog-to-digital converters (ADC's) and into the correct locations of memory in the 4096-channel MCA. The function of the GAR's will be described later.

Receiving GAR output are two independent 50 MHz ADC's. The ADC's digitize analog pulses which have traveled through one of the two connecting GAR's. This digital code then goes to a clock gate which is either open (during live-time acquisition) or closed (timed out) and on to a digiplexing module (DX2). The DX2 readies the digitized pulse for servicing by the MCA. This is achieved by adding an information bit for final routing addressing. The MCA services the pulse by treating it as an address between 0 and 4096. Upon receipt of that address, the MCA goes to the appropriate point in memory, retrieves the numerical value stored at that point, increments it by one, and stores the new number back in the same address. In this way histograms of nuclear events are generated. To recapitulate, the main objective is the production of 16 independent 256-channel by 20-bit alpha spectra. The energy calibration of each spectrum ranges from 3.5 MeV (channel zero) to 6.7 Mev (channel 256) yielding 12.5 KeV per channel, and each spectrum occupies a given unique 256-channel region in the MCA. The output from detector No. 1 is routed to the first 256 channels, detector No. 2 to the second 256 channels (or channels 257 through 512), detector No. 3 to the third 256 channels (or channels 513 through 768) and so on.

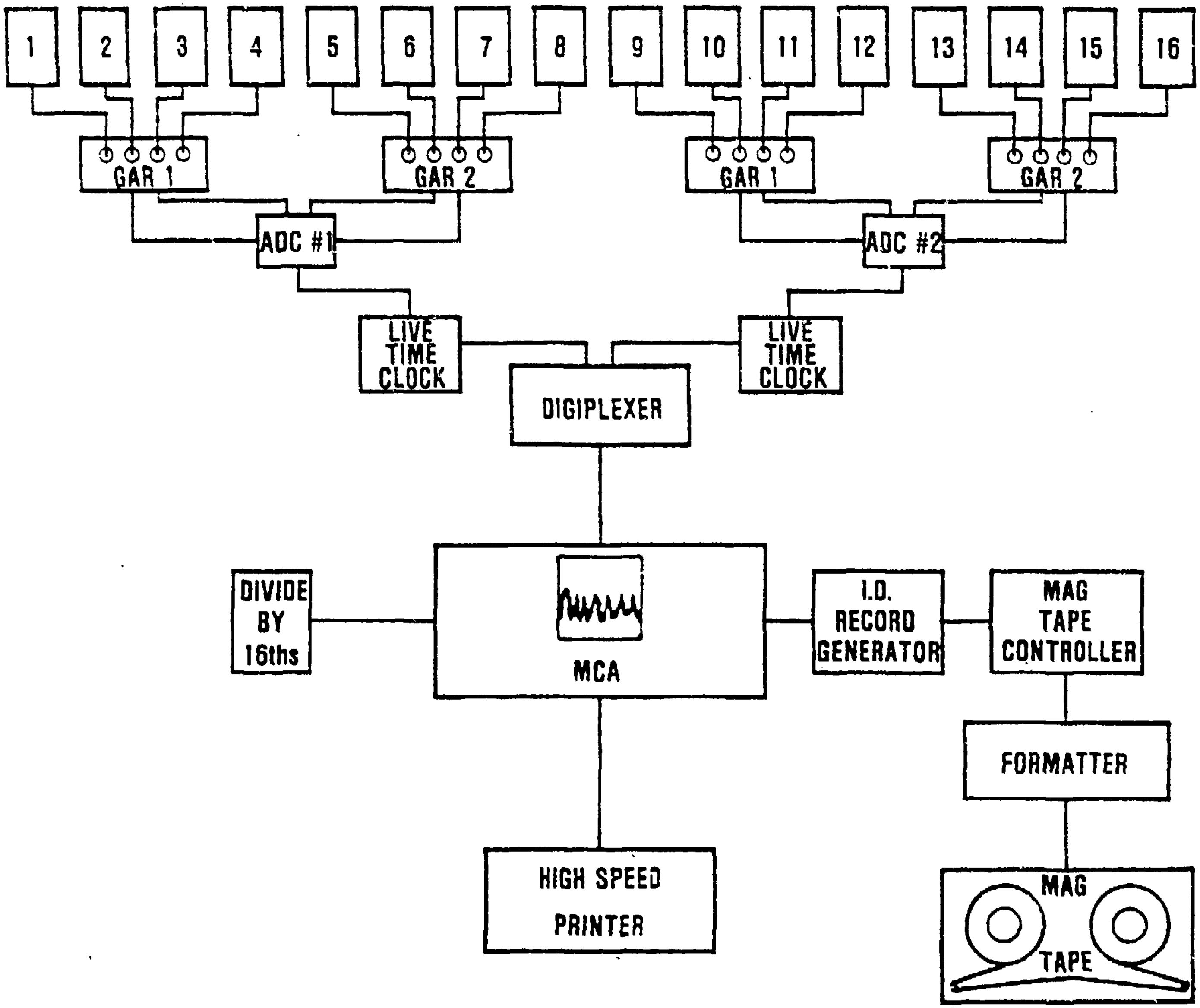

FIG. 1

THE 16-DETECTOR SPECTROMETER

The function of the GAR's and the routing procedure is now described in detail. As an example, we will follow through the system a 6-MeV pulse produced by detector No. 2 (Figure 2). A 6-MeV pulse produces a 10-volt pulse from the final stage of amplification. This 10-volt pulse passes unaffected through the GAR, tripping a gate, and on to the ADC for digitization. A 10-volt pulse is digitized as 200. However, since the second gate of the GAR was tripped, the number 256 is automatically added to the digitized pulse of 200. The result is then 456, which passes on through the clock gate to the digiplexer and awaits servicing by the MCA. When serviced, the contents of memory location 456 are incremented by one. This has the effect of storing an event of 6 MeV in the 200th channel of the second 256-channel spectrum in memory. A digiplexing module serves the function of splitting the addressing into two halves, one for each ADC. All pulses passing through ADC No. 1 and on to the digiplexer are basically unaffected. Those pulses passing through ADC No. 2 all have the number 2048 (2^{11}) added to them by the digiplexer. In this way each set of two GAR's only serves as a router for memory size 2048, and the final addressing is performed by the digiplexer.*

Two four-input GAR's, one ADC, and one clock control the first half of the MCA (2048 channels) and eight detectors. The other eight detectors and second half of the MCA memory are controlled by a symmetrical set of GAR's, an ADC, and a clock. In this way independent counting times may be selected on each unit of eight detectors.

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*Only the most significant bits of the pulse are affected by the routing modules. The GAR's set the 8th, 9th, and 10th bits and the digiplexer sets the 11th bit, forming an 11-bit address.

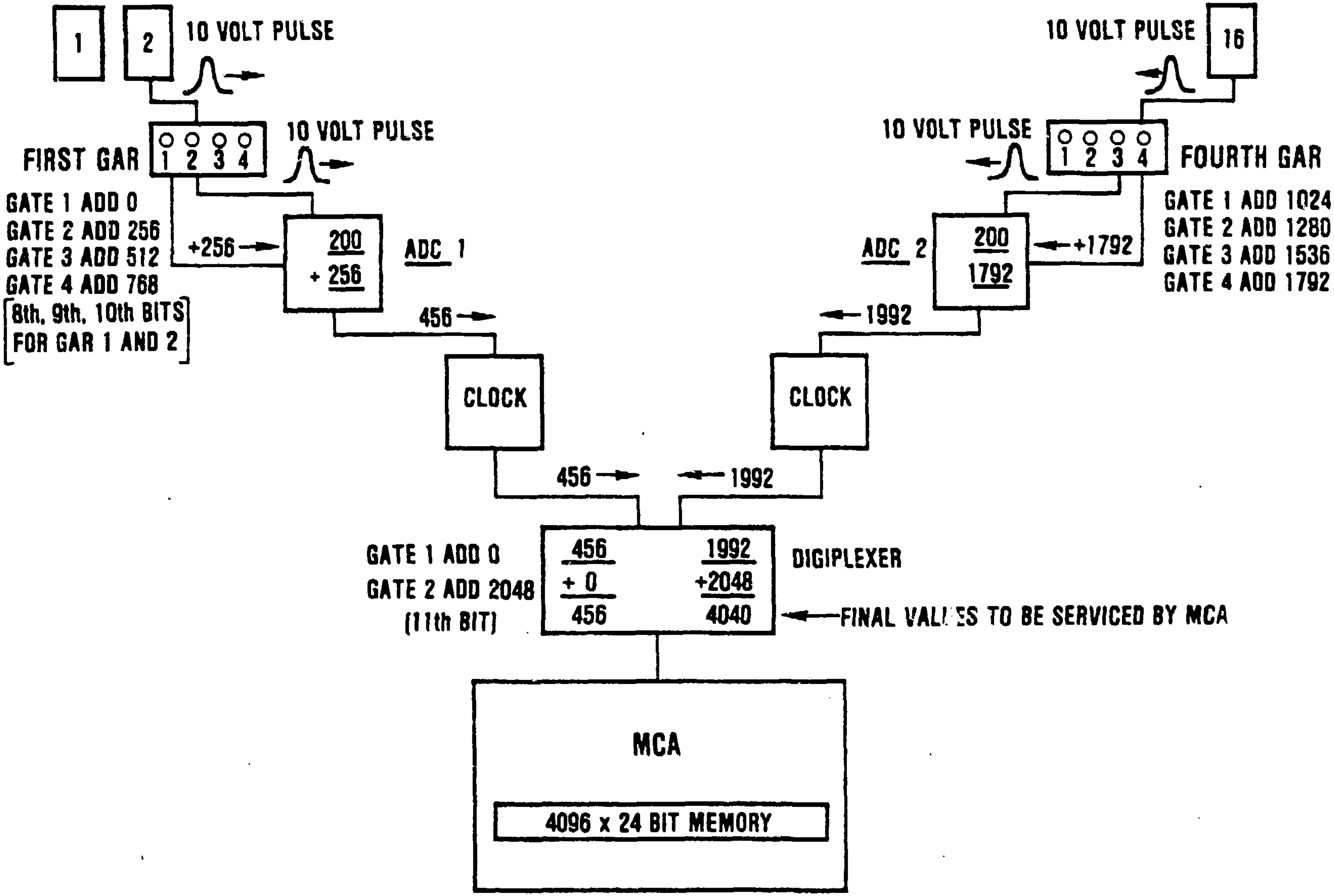

FIG. 2
TWO EXAMPLES OF SIGNAL ROUTING

As a second example of routing procedures, assume a 6-MeV pulse travels through the 4th input of the fourth GAR (Figure 2). This pulse is digitized to 200 by the second ADC, but the number 1792 (address bits, 8,9, and 10 are set to 1) is added to this digit by the GAR, signifying that the 4th gate of the second GAR connected to ADC No. 2 was tripped. As the pulse travels on, the second gate of the digiplexer is tripped, signifying that the pulse originated from ADC No. 2. This allows the number 2048 (2^{11}) to be added to the pulse. The net result of the routing, digitizing, and digiplexing is then the production of the address 4040. This address corresponds to the 200th channel of the 16th spectrum. At this time the location 4040 is incremented by one in the MCA, which signifies that a 6-MeV pulse was detected by detector No. 16.

When spectra have been generated, they are written to magnetic tape and/or printed. Each spectrum written to magnetic tape contains a time-of-day stamp and a 6-digit tagword. The regions of memory to be outputted are selected manually by a division-by-16 module which allows an operator the option of choosing any 256-channel unit of the 4096-channel memory. Each spectrum appears on tape as a unit of 256 contiguous numbers (representing 256 contiguous channels) preceded by the appropriate tagword information. In this way, confusing channel numbers of various spectra are not included and all spectra appear to be of identical origin to the processing computer.

DATA HANDLING ADVANTAGES OF THE SYSTEM

Since all counting data are written to magnetic tape which is compatible with processing by our large computer, data manipulation is possible through an interactive mode terminal. In general, spectral peak shapes produced by a given isotope may differ vastly between counts of environmental level samples. Peak shapes are a function of many variables, some of which are not easily controlled. It is for this reason that when mathematically reducing low-level spectra, batch-run computer codes are difficult to produce and run properly.

We have found interactive graphics to be more adapted to our needs than batch processing. An interactive program has been developed which maps spectra on a graphics cathode-ray tube (CRT), thus allowing an operator to locate peak regions of interest, automatically sum those regions, subtract a previously stored background for the region, and print a net area for further use. In each analysis, many peaks may be summed in this fashion and stored in an on-line array. The type of analysis to be performed may then be specified and other pertinent parameters entered through a conversational mode with the computer operator. The data and parameters may then be routed for processing and final report generation.

Conversational mode operates by the computer asking questions of the operator. These questions may seek information such as the amount of tracer activity added to a sample, the initial sample weight, the operator's name, etc. An operator in this way cannot forget to enter all pertinent information necessary for the particular analysis being performed.

There are at least two advantages of computerizing the data handling and reduction. Prior to computerization, all raw data were checked and peaks were summed by hand. These sums were entered on a key-punch coding sheet with other pertinent parameters, sent to the key-punch operators, punched, verified, and finally submitted to a batch analysis run. Automating this routine has led to the freeing of an operator from the previous method of peak integration by hand. Also the visual observation of peaks on a CRT adds to the ability of correct peak region definition and reduces the tedium presented by working with raw numbers. This technique necessarily reduces human error, and greatly reduces data throughput time.

The interactive system has benefitted the entire operation from other points of view. Reports are generated and edited from a terminal in the laboratory at the instrument site. Should sample comparisons be necessary with historical spectral data, the accessing of that data is faster and simpler from the terminal. Both the hardware and software of the upgraded system have provided more data integrity and less vulnerability to human error than the previous system.

APPLICATIONS OF COMMERCIAL LIQUID SCINTILLATION COUNTERS TO RADON-222 AND RADIUM-226 ANALYSES

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ABSTRACT

The ubiquitous commercial liquid scintillation counter offers automatic sample processing, automatic data recording and the prospect of multiple users. With these features in mind we have explored a number of applications of liquid scintillation counters to environmental and health physics problems. One application, the analysis of radon in water has been described elsewhere and is only briefly reviewed. A method for measuring radon in air, two methods for measuring radium in water, and a technique for leak testing radium needles have also been investigated. An ordinary glass scintillation vial is readily converted into a minature scintillation flask by coating the inside surface with a thin layer in ZnS:Ag phosphor. The lower limit detection is high, about 2 pCi/l for a 1 hour count, but these flasks have proved to be useful in situations where a larger number of samples must be taken in environments with relatively high levels of radon. One technique for the detection of radium in water uses liquid-liquid extraction to concentrate radon into an organic scintillation fluid, the other involves passing the water sample through an ion exchange resin and then sealing the resin and scintillation fluid in a vial. Both techniques offer the prospect of easy and inexpensive analyses with limites of detection at or below 0.5 pCi/l. Radium needles can be leak tested by placing them in vials containing toluene for a few minutes, adding fluor to the toluene and counting. Preliminary data regarding these several methods are given.

Automatic liquid scintillation spectrometers are widely available in many institutions, especially those conducting biomedical research. In most cases, if a liquid scintillation spectrometer is not available for the exclusive use of the health physicist, he can obtain time on an instrument residing in another department. These instruments offer automatic sample processing and automatic data recording, often in computer compatible form, and are used by health physicists for wipe tests and urinalyses (Br 77). We report here preliminary investigations of several additional health physics applications of liquid scintillation spectrometers including methods for radon in water, radium in water, radon in air and leak testing of radium-226 needles. While the ability of liquid scintillation methods to detect radioactive noble gases is documented (Ho 73), we present several practical approaches using standard vials and materials.

Radon-222* in Water

This method has been published elsewhere (Pr-in press) and is only reviewed here. Although several previous methods for measuring radon in water have been described and used, all have the drawback of requiring an appreciable amount of repetitious labor. A widely used technique employes a radon bubbler and a ZnS:Ag phosphor-coated scintillation cell (Lu 57, Lu 64). A carrier gas is passed through the water in the bubbler and conveys the radon, either directly or after a concentration step, into the scintillation cell, which is counted on a photomultiplier tube. A considerable amount of time is required for sample preparation and the counting and sample changing is typically done manually. We took advantage of the large differential solubility (Sc 20) of radon in toluene (about 50 times greater than in water) to develop a method which utilizes a commercial liquid scintillation counter. With this method, sample collection and preparation takes about one minute, the counting and sample changing are done automatically and the data are generated in computer compatible form. A lower limit of detection of 10 pCi/l is achievable with 40 minute counting of fresh samples.

Ordinary glass liquid scintillation (LS) vials are pre-loaded with 5 cm^3 of toluene-based LS fluid. At the sample site water is collected with a medical syringe and 15 cm^3 is injected into the LS vial under the toluene.

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* Henceforth simply radon

The vials are vigorously shaken and allowed to stand for a minimum of three hours to allow for buildup of radon daughters, and then are counted. Calibration is achieved by splitting samples between the LS method and a ZnS:Ag scintillation cell (Lucas) method which is NBS traceable.

A severe field test of the LS method in which 75 pairs of samples were taken, transported by automobile about 1600 km and counted a week after collection indicated that the method is quite reliable. The errors were typically only slightly greater than that expected due to counting statistics. About four percent of the vials exhibited unacceptable leakage. While one vial was found to have a defective cap, most of the leakage was probably due to inadequate tightening of the caps. Where high individual sample precision and accuracy are required, it is recommended that duplicate or even triplicate samples be taken of each source. Employing the easily defensible hypothesis that radon is much more likely to leak out of a vial than into a vial, one can discard anomalously low results.

Radium-226* in Water

The success of the LS method for radon in water led quite naturally to an attempt to develop a method for radium in water based upon measurement of the associated radon. While most ground water supplies have radon concentrations in the range of 100- 10,000 pCi/l (mostly unsupported) (Du-76) radium typically exists at concentrations ranging from a few hundreths to a few pCi/l (Ei 73). Thus the problem becomes one of concentration, either of the radium or the radon, prior to counting. One approach to the concentration involves liquid-liquid extraction, taking advantage of the high (50:1) solubility of radon in toluene reltative to water (Sc 20).

Approximately one liter of water to be analyzed for ^{226}Ra is boiled in a flask to drive off any radon present. Twenty ml of toluene-based liquid scintillation solution is then added to the flask along with sufficient distilled water to bring the miniscus up to the level of the stopper. The flask is then sealed and set aside for at least one half-life of radon to allow for radon in growth. The flask is then agitated for at least two hours in such a way that the toluene fraction is broken into several small bubbles distributed through the aqueous phase. Agitation is stopped and the toluene withdrawn by syringe as soon as a single layer has reformed. The toluene

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* Henceforth simply radium

solution is then injected through a rubber septum into a vial previously evacuated and counted in a liquid scintillation counter. Recovery tests were performed and found to agree well with the recovery predicted by the partition coefficient.

The method was tested by preparing four solutions with known concentrations of radium and analyzing them by the liquid-liquid extraction method while the primary stock from which the solutions were prepared was analyzed by the NBS traceable bubbler and scintillation flask technique. The results of the test are given in Table 1. An advantage of this method is that it is quite sensitive. The net count-rate for solution C, which is near the EPA standard for total radium in drinking water, was 19 counts per minute, while the background count rate was 4 cpm. Radium concentrations an order of magnitude below the standard are therefore readily detected after an hour's counting time, as is shown by the fair agreement seen with solutions A and B. Another advantage inherent in the method is the relative insensitivity to the presence of contaminants in the solution to be tested. This was demonstrated in a radon extraction experiment in which no significant difference was noted between the results in de-ionized water and in sea water.

It is apparent that the radium may also be concentrated. While alternatives exist we chose to test ion exchange resin concentration followed by liquid scintillation counting. Ion exchange resin concentration followed by gamma ray spectroscopy has been successfully employed previously (Ha 72).

TABLE 1

Radium concentrations measured by the liquid-liquid extraction method

Solution	Known ^{226}Ra Activity (pCi/l)	^{226}Ra Activity by LS Method (pCi/l)
A	0.64	0.60
B	0.61	0.71
C	6.10	5.63
D	37.5	35.2

The water to be tested is passed through a column containing from 5 to 10 ml of an appropriate cation exchange resin. The resin is then placed in a 20 ml liquid scintillation vial with enough distilled water so that the resin is covered by a few mm of water. Five to ten ml of liquid scintillation solution is then added and the vial is sealed and counted to check for unsupported radon. The vial is then set aside to allow the ingrowth of radon from radium trapped in the resin. Prior to the next count, the vial is vigorously shaken for a few seconds to insure that the radon generated in the aqueous phase has an opportunity to become dissolved in the organic phase. The vial is then kept undisturbed for at least three but not more than 24 hours prior to counting for the determination of radon activity due to the ingrowth from radium. The radium itself is trapped in the resin and does not contribute to the count rate.

In an initial test of recovery, a liter of water containing 50 pCi/l of radium was passed at a rate of 4 l/hr through three successive columns containing 5 ml each of Amberlite IR 120+ cation resin. The resin from each column was placed in a scintillation vial with 5 ml of LS solution and enough distilled water to bring the vials up to volume. All the vials were sealed and set aside for over seven half-lives of ^{222}Rn. When the vials were counted, it was seen that virtually all of the activity was found in the first column. The gross count rates for the three columns was 436, 28, and 27 cpm, as compared with a background of 25 cpm. The total net count rate for the three columns was 416 cpm, compared to a maximum count rate of 475 cpm which is computed on the basis of 100% retention of radium on the resin and full solution equilibrium in the scintillation vial.

The practicality of the technique is dependent on determining the best resin types, flow rates, and competing ion limitations in order to develop a routine procedure which can then be tested for reliability field conditions. While this technique would not be useful for sea water or other waters with high mineral concentrations, it could prove to be a sensitive and convenient technique for analyzing radium in potable water. A minimal amount of sample preparation and equipment is required, and sensitivity is limited primarily by the amount of water that can be passed through a 10 ml resin bed before the resin becomes saturated.

Radon in Air

We required a simple, inexpensive method for taking and evaluating numerous radon-in-air samples in connection with survey work in caverns.

While the well-accepted Lucas Cells (Lu 57) were suitable, their high unit cost and the requirement for manual sample changing prohibited their exclusive use in our case. As an alternative we constructed scintillation flasks from ordinary 25 ml liquid scintillation vials. After some experimentation with materials and methods, we settled on glass vials with aluminum foil lined caps. A light coating of ordinary silicone vacuum grease is applied to the interior surface of the vials followed by an application of ZnS(Ag) scintillator in powder form. Calibration is effected by opening the vials in a glove box containing a known radon concentration. A rubber squeeze bulb is used to assume complete interchange of glove box air with the air previously contained the vials. After filling, the vials are set aside for a minimum of three hours for equilibrium to occur and then counted. Due to individual variations in construction, each cell has a unique calibration factor. Table 2 gives summary statistics for a recent batch of 30 cells as well as parameters for a typical Lucas Cell.

TABLE 2

Performance Summary of Thirty ZnS Scintillation Vials

	ZnS Scintillation vials		Typical Lucas Cell	
Radon concentration in sampled air	281 ± 8*	pCi/l		
Mean Cell background	0.12 ± .19	c/m	0.1	c/m
Absolute efficiency	4.0 ± 0.5	(c/m)/pCi	5.3	(c/m)/pCi
Relative efficiency	0.60 ± .08	C/alpha particle	0.8	c/alpha particle
Calibration factor	0.100 ± .013	(c/m)/(pCi/l)	0.5	(c/m)/(pCi/l)

* all errors are ± 1.96σ

For the cavern surveys, at least 2 ZnS cells are used at each sample point and time with at least one point being sampled with a Lucas Cell to determine the agreement of the methods for field use. Table 3 lists the comparison of ZnS cells with Lucas Cells for typical Cavern samples. As can be seen there generally is good agreement between the two methods. By taking two or more samples, the means of the ZnS cells are in close agreement with the Lucas Cells.

TABLE 3

Comparison of ZnS and Lucas Cell Field Test Results (pCi/l)

Lucas (± 1σ)*	ZnS Cell (± 1σ)*	ZnS Cell Means
5.1 ± .4	4.9 ± 1.8	--
32.5 ± .8	31.3 ± 3.7	--
101.4 ± 1.4	92.1 ± 6.0	
	104.0 ± 6.4	98.1
5.6 ± .5	6.7 ± 2.0	
	4.4 ± 1.8	5.6
175.6 ± 6.8	185.5 ± 6.4	
	176.3 ± 6.2	
	174.3 ± 6.1	178.7
72.7 ± .7	70.0 ± 5.4	
	63.3 ± 5.2	66.7
19.5 ± .9	16.0 ± 2.8	
	12.3 ± 2.5	14.2
84.4 ± 1.8	80.4 ± 5.5	
	82.0 ± 5.5	81.2

* counting statistics only

The possibility of leakage due to air pressure changes and heat changes during transport were considered. Two ZnS cells were loaded with a high concentration of radon (1-2 nCi/l) and their count rates were determined. They were carried on two sampling trips, one from Houston to San Antonio and one from Houston to Austin, distances of 320 and 260 km respectively, without being opened. During the transport they were subjected to 100°F temperatures and altitude changes of several hundred meters. The decrease in count rates from that expected from radioactive decay alone was about 4%. It is not known if this small loss was due to leakage, radon diffusion through the caps, or statistics. The time interval from the first to the last count was 9 days. Thus far only 4 vials have been observed to leak noticeably, from a population of over 300 field and laboratory samples.

The results of this study indicate that these ZnS cells can be successfully used for environmental sampling of radon at levels on the order of 2 or more pCi/l. ZnS cells are inexpensive and simple to construct. They permit taking many samples with automatic processing to obtain the results. Using 2 or more cells at each sample point improves their agreement with Lucas Cells. ZnS cells should prove to be useful in obtaining profiles of radon-222 levels in Uranium mines, natural caverns and similar environmental conditions.

Leak Testing Radium Sources

Leak testing sealed sources is usually required by law. Leak-testing implantable medical sources is especially important because of direct contact with patients. Radium sources are usually leak tested by adsorbing radon-222 (a radium-226 daughter that is present if the source is leaking) in activated carbon filters or cotton and then using an alpha detector to detect radon and its daughters. Another method is to wipe the surface of the source or its containment vessel and count the wipe. The radon adsorption method requires a long sampling time, perhaps leaving the source(s) unsecured for some time, and the wipe method has questionable sensitivity.

A simpler, shorter, and apparently more sensitive method of leak-testing radium sources is suggested here: the use of liquid scintillation techniques to detect radon and its daughters. Since liquid scintillation is available at many facilities were radium is used, it should be possible to develop a standard leak-testing procedure acceptable to regulatory authorities and useful to radium users.

As mentioned, leaking radium sources also leak radon, in fact, the leakage of radon normally precedes any leakage of radium provided that the encapsulation has not been seriously damaged. The suggested procedure is to immerse the radium source(s) in a liquid scintillation vial containing a toluene-based cocktail. After immersing the sources for a sufficient length of time, the vial(s) can be evaluated for radon content by common liquid scintillation equipment and methods. The vials indicating leakage can also be held for decay of the radon and their possible content of radium determined.

To test the method, cocktails of toluene with PPO (5g /l) and POPOP (0.5g /l) fluors were used. The first test was to determine if the radon would diffuse from the cocktail at a rate that would make the method impractical. Twenty milliliters of cocktail were placed in each of 4 vials and a quantity of radon (in air as a carrier) was bubbled into three of the vials with a syringe, the fourth vial being used as a control. The four vials were then counted by liquid scintillation instrumentation set to detect radon in this particular cocktail. The four vials were then uncapped and left standing in a fume hood. After 30 minutes, 1 vial was recapped and the other vials recapped after subsequent 30 minute intervals. After a suitable delay for ingrowth of the radon daughters, the vials were recounted and their loss by diffusion calculated. It was determined that the radon had dissipated at the rate of about 15% of the original concentration per half hour. The loss rate is acceptable, which is important since recapping the vials during immersion might increase hand exposure to the point that the procedure would be impractical.

A second step was to leak-test a radium needle that had passed successfully previous leak-tests involving the activated carbon and alpha detector method. The needle was immersed for 60 minutes in a vial with 20 ml of the toluene cocktail, with the cap off. The vial was then counted and found not to differ from the normal background. It had been speculated that radium sources may always have some contaminant on the exterior surfaces, fixed there during manufacture, and that the method may be so sensitive as to falsely indicate that sources are leaking. Testing a large number of sources should provide more information than this one test.

The next step was to test sources that were known to be leaking. Two sources that were known to be leaking the greatest and the least at a large hospital and were still in the inventory awaiting disposal, were used.

The two sources were tested in the same manner as the non-leaking needle above and counted with the same liquid scintillation procedure. One was immersed for 35 minutes and the other for 36 minutes. Count rates of 11,000 and 1500 c/m were observed. At the particular window used in this experiment, the background was 25 c/m.

The liquid scintillation method of leak-testing radium sources appears to be useful. Nevertheless, since implantable radium sources come in contact with human tissues, and since there exists the possibility that the POP and POPOP fluors may be toxic, a modification is suggested. Instead of using the toluene cocktail containing the fluor, pure toluene could be used, adding the fluor after removing the source. This would obtain the desired cocktail without contaminating the source with the fluor. Toluene has a high vapor pressure and should be eliminated during the normal autoclaving to which the sources are usually subjected before use.

The successful test results obtained for this evaluation indicate that the leak-testing of radium-226 sealed sources and detection by liquid scintillation can be incorporated into a working system. It has the advantages of simplicity, sensitivity, automatic sample processing, and possible reduced human exposure. Since the purpose of the procedure is not to determine the amount of leakage, but to determine if a source is leaking at all, quantitation is not a requirement. We plan a large scale test of the method in the near future.

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Speaker (Tom Gesell):

I answered all your questions last night, Alan.

Question (Alan Moghissi):

You didn't. I was very impressed with a zinc sulfide coated glass. The literature indicates that zinc sulfide basically is opaque to light. So I am wondering what kind of trick have you used that you are coating the glass, and I presume you haven't altered the instrument at all. Is that correct?

Answer (Gesell):

I presume I am using the same trick that the instrument manufacturers do and they make scintillation-type alpha probes. The zinc sulfide turns out to be adequately transparent to the light; you can view it from the back.

Question (Moghissi):

All right, do you count in coincidence or the additive mode?

Answer (Gesell):

No, we count in normal liquid scintillation counter mode.

Question (Moghissi):

No change in high voltage?

Answer (Gesell):

No.

Question (Moghissi)::

On what kind of discriminator?

Answer (Gesell):

Well, we just simply optimize by using our scintillation counter, ours uses a two-channel unit, and uses simply a two-channel analyzer, and we just adjust the channel size to get a reasonable background . . .

Question (Moghissi):

But a Lucas call has a background of a couple of counts per hour?

Answer (Gesell):

Well, we found our cells have a background of something on the order of 6 counts per hour.

Question (Moghissi):

And you get the same thing with liquid scintillation?

Answer (Gesell):

Yes.

Question (Moghissi):

Son of a gun.

Answer (Gesell):

The point is that we're using the same phosphorous, zinc sulfide, actually less of it, and we can essentially, by running up the bottom energy discrimination, we can discriminate against the low light events.

Question (Moghissi):

You did not reduce your high voltage?

Answer (Gesell):

No.

Question (Moghissi):

You are a miracle-I don't know how you do it.

Question (Bill Wadman, Irvine):

With regard to your toluene collection of radon, what size samples can you process with that method--how large a quantity?

Answer (Gesell):

We have been using a liter. There's no reason--see there is a partition coefficient of 20°C of 50, and it turns a reasonably independent temperature there, one goes to I think 52 at 30°C, anyway, the more water you use, of course, the more sensitive the method becomes until eventually you plateau out.

Question (Wadman):

Have any of the earthquake predictors been around to talk to you about that method?

Answer (Gesell):

No, as a matter of fact I wrote to the Geological Survey about the method and ask them if they had any interest in it, and of course they didn't. I think the reason is that for the earthquake prediction studies continuous readout, and I think what most of them are doing is flowing water around an anulus of the sodium iodide scintillation crystal so they get real time or continuous monitoring. This is certainly a grab sampling method.

Question (Carroll Trent, Berkeley):

For my curiosity, in your zinc sulfide air vial sample system, did you ever try just putting some zinc sulfide coated mylar into the vial?

Answer (Gesell):

We've been meaning to try this and I think it might solve some of our problems of reproducibility from one cell to the next, and we would like to try this. It's just been a matter of some lethargy as far as getting some of this stuff ordered.

Trent: It's a mylar with zinc sulfide.

Answer:

Right, I was just wondering-we might not even have to glue it out of the wall-we thought about that--we might be able just to slip it in there and have it spring out against the wall.

Question (Bussick, from SLAC):

What method of coating did you wind up using?

Answer:

We used the same method that Andy George used at HASL--it's not very elegant. One puts some silicon vacuum grease on the inside of the vial and then shakes zinc sulfide in and pours out the excess.

ENERGY RESPONSE REQUIREMENTS FOR EXPOSURE METERS USED IN DIAGNOSTIC RADIOLOGY

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Abstract

When measurements of exposure and half-value layer are required for primary medical x-ray beams, the energy response of the measuring instrument is an important consideration. Using a computer the energy response to polyenergetic x-rays was simulated for several commercially available exposure meters. The magnitude of errors due to energy response, when measuring exposure and half-value layer, will be discussed for x-ray beams in the diagnostic energy range (including mammography). Energy response requirements for desired accuracy will also be presented.

Introduction

Energy dependence of the ion chamber is one of a number of possible sources of error when measuring primary diagnostic x-ray beams. In order to study the effect of energy response alone on measurements of exposure and half-value layer, a method of isolating this source of error from all others was required. It was decided that a computer simulation of ion chamber energy response to polyenergetic x-rays would allow manipulation of the x-ray beam parameters without introducing extraneous sources of error.

Method

A digital computer was used to simulate the measurement of exposure and half-value layer for several commercially available ion chambers and primary diagnostic x-ray spectra. X-ray spectra were corrected for air attenuation for a target-to-detector distance of 100 cm (1). Spectra for varying amounts of total aluminum filtration were calculated for 45, 70, and 90 kVp, single-phase full-wave-rectified tube potentials, by using the total mass attenuation coefficients of aluminum to exponentially attenuate the spectra. To simplify the computer programming, the total attenuation coefficients of aluminum were calculated over the 10-90 keV range by using the following empirical equation (1):

$$(\mu/\rho) = e^{Y} \qquad (1)$$

where $Y = 6.5951 - 0.47029\,X + 0.010655\,X^{2} - 0.0001466\,X^{3} + 1.0515 \times 10^{-6}\,X^{4} - 3.0436 \times 10^{-9}\,X^{5}$

and X is the photon energy (keV). Figures 1-3 show the calculated spectrum at two extremes of filtration for 45, 70, and 90 kVp, respectively.

Energy responses of four typical ion chambers, which were designed to be used to measure primary diagnostic x-ray beams, were estimated from commercial literature over the energy range of 10 to 90 keV. Figure 4 illustrates the energy responses of the chambers used in this study.

Relative exposure was calculated (2) by using the following relationship:

$$\text{Exposure} \propto \sum_{i} S_{i}\, R_{i}\, E_{i}\, (\mu_{en}/\rho)_{i} \qquad (2)$$

where i is the energy interval, S_i is the relative number of photons in the spectrum, R_i is the ion chamber relative response, E_i is the photon energy (keV), and $(\mu_{en}/\rho)_i$ is the total mass energy absorption coefficient of air. Total energy absorption coefficients of air were also calculated over the 10-90 keV range by using an empirical equation (1):

$$(\mu/\rho)_{en} = e^{Y} \qquad (3)$$

where $Y = 5.0655 - 0.44471\,X + 0.010354\,X^{2} - 0.00013696\,X^{3} + 9.7845 \times 10^{-7}\,X^{4} - 2.878 \times 10^{-9}\,X^{5}$

and X is the photon energy (keV). Equation 3 was found to yield results in good agreement with published data (3).

Half-value layer was determined by calculating the exposure for varying amounts of added aluminum filtration and determining the thickness of aluminum required to halve the exposure. Table I gives the calculated half-value layers and corresponding effective energies for various amounts of total filtration for the 45, 70, and 90 kVp spectra assuming an ideal energy response.

The commercial literature usually presents ion chamber energy response as a function of effective energy. Effective energy of a polyenergetic photon beam is the energy of monoenergetic photons which have a total attenuation coefficient equal to the effective total attenuation coefficient for the polyenergetic photon beam in the same medium (4). For the

Table I

Total Filtration	45 kVp HVL	45 kVp Eff. E	70 kVp HVL	70 kVp Eff. E	90 kVp HVL	90 kVp Eff. E
0.0 mm	0.67 mm	19 keV	-	-	-	-
0.5	0.92	22	1.43	25 keV	-	-
1.0	1.14	23	1.71	27	2.11 mm	29.5 keV
1.5	1.32	24.5	1.97	28.5	2.45	31
2.0	1.46	25.5	2.23	30	2.78	32.5
2.5	1.61	26.5	2.45	31	3.07	34
3.0	1.74	27	2.67	32	3.35	35
3.5	-	-	2.86	33	3.60	36
4.0	-	-	3.03	34	3.84	37
4.5	-	-	3.20	34.5	4.04	38
5.0	-	-	-	-	4.24	39
5.5	-	-	-	-	4.42	39.5

purpose of this study, the energy response was used as if a function of monoenergetic photon energy. The validity of this assumption was checked by calculating the relative response at each effective energy given in Table I and comparing the results with the responses of the original curve at the same effective energy. Agreement was found to be within ±1% for all but a few points.

Exposure Measurement Error

In order to determine the magnitude of the error due to energy response when measuring exposure, the exposures for the tube potentials and filtrations listed in Table I were calculated for each of the four ion chambers. Table II shows the percent deviation from the actual exposure (exposure calculated with ideal energy response) for the various techniques.

As was expected, the chamber results differ from the actual exposure as would be predicted from the energy response curves. These errors can be corrected by applying a correction factor calculated from the energy response at the applicable effective energy.

Half-Value Layer Measurement Error

In order to determine the magnitude of the error when measuring half-value layer due to energy response, the half-

Table II
EXPOSURE ERROR AT 45 kVp

Total Filtration	Manf. A	Manf. B	Manf. C	Manf. D
0.0 mm Al	-1.7%	-5.5%	-13.7%	-0.9%
0.5	+0.8%	-4.4%	-10.8%	+0.3%
1.0	+2.0%	-3.7%	- 9.8%	+0.9%
1.5	+2.7%	-3.2%	- 9.4%	+1.2%
2.0	+3.3%	-2.8%	- 9.2%	+1.3%
2.5	+3.6%	-2.4%	- 9.1%	+1.4%
3.0	+3.9%	-2.1%	- 9.0%	+1.4%

EXPOSURE ERROR AT 70 kVp

Total Filtration	Manf. A	Manf. B	Manf. C	Manf. D
0.5 mm Al	+2.6%	-2.4%	-9.1%	+0.7%
1.0	+3.2%	-1.8%	-8.8%	+0.8%
1.5	+3.8%	-1.2%	-8.5%	+0.9%
2.0	+4.1%	-0.7%	-8.3%	+0.9%
2.5	+4.3%	-0.2%	-8.2%	+0.8%
3.0	+4.5%	+0.1%	-8.1%	+0.8%
3.5	+4.6%	+0.5%	-7.9%	+0.7%
4.0	+4.7%	+0.7%	-7.8%	+0.7%
4.5	+4.8%	+1.0%	-7.7%	+0.6%

EXPOSURE ERROR AT 90 kVp

Total Filtration	Manf. A	Manf. B	Manf. C	Manf. D
1.0 mm Al	+3.5%	-0.8%	-7.8%	+0.7%
1.5	+3.8%	-0.2%	-7.7%	+0.6%
2.0	+4.1%	+0.3%	-7.4%	+0.6%
2.5	+4.3%	+0.7%	-7.2%	+0.5%
3.0	+4.4%	+1.1%	-7.0%	+0.4%
3.5	+4.5%	+1.4%	-6.8%	+0.4%
4.0	+4.5%	+1.7%	-6.7%	+0.3%
4.5	+4.6%	+1.9%	-6.5%	+0.2%
5.0	+4.6%	+2.1%	-6.4%	+0.1%
5.5	+4.6%	+2.3%	-6.2%	+0.1%

Table III
HVL ERROR AT 45 kVp

Total Filtration	Manf. A	Manf. B	Manf. C	Manf. D
0.0 mm Al	+0.04 mm	+0.02 mm	+0.05 mm	+0.02 mm
0.5	+0.03	+0.02	+0.02	+0.01
1.0	+0.02	+0.02	+0.01	0.0
1.5	+0.02	+0.02	+0.01	0.0
2.0	+0.02	+0.02	+0.01	+0.01
2.5	+0.02	+0.03	+0.01	0.0
3.0	+0.02	+0.03	+0.01	0.0

HVL ERROR AT 70 kVp

Total Filtration	Manf. A	Manf. B	Manf. C	Manf. D
0.5 mm Al	+0.03 mm	+0.04 mm	+0.01 mm	0.0 mm
1.0	+0.04	+0.06	+0.03	0.0
1.5	+0.03	+0.06	+0.02	0.0
2.0	+0.03	+0.06	+0.03	-0.01
2.5	+0.02	+0.06	+0.03	-0.01
3.0	+0.02	+0.06	+0.02	-0.02
3.5	+0.01	+0.06	+0.03	-0.02
4.0	+0.01	+0.06	+0.03	-0.02
4.5	+0.01	+0.06	+0.03	-0.02

HVL ERROR AT 90 kVp

Total Filtration	Manf. A	Manf. B	Manf. C	Manf. D
1.0 mm Al	+0.03 mm	+0.08 mm	+0.04 mm	-0.01 mm
1.5	+0.02	+0.08	+0.04	-0.02
2.0	+0.02	+0.08	+0.05	-0.02
2.5	+0.02	+0.09	+0.07	-0.02
3.0	+0.01	+0.09	+0.07	-0.02
3.5	+0.01	+0.09	+0.07	-0.03
4.0	0.0	+0.07	+0.07	-0.03
4.5	0.0	+0.08	+0.08	-0.03
5.0	0.0	+0.08	+0.08	-0.03
5.5	-0.01	+0.07	-	-0.03

value layers "measured" by each of the four chambers were calculated using the method previously described. Table III shows the deviation in millimeters of aluminum from the actual half-value layer as calculated with ideal energy response for 45, 70, and 90 kVp.

Surprisingly, even for chamber C which exhibited the greatest response variation of the four chambers in the 10-90 keV range, the half-value layer measurement error due to energy response was less than 0.1 mm. The error would be expected to increase for measurements in the 25-30 kVp mammography range using chambers having rapidly decreasing response below 20 keV. However, over the range of 45 to 90 kVp with wide variation of filtration used in this study, there appears to be little half-value layer measurement error due to energy response.

Energy Response Requirements

The energy range of 10-90 keV was divided into three regions of interest which roughly correspond to the different energy response characteristics found within this range for a typical ion chamber. Figure 5 illustrates the regions: region I includes 10 keV up to 20 keV, region II includes 20 keV up to 40 keV, and region III includes 40 keV and greater. In order to determine the effect energy response in each of these regions has on exposure measurement, the contribution to the total exposure of photons in each region was calculated. Table IV presents the percent of the total exposure due to each region.

Table IV

Technique	Contribution to Exposure Region I	 Region II	 Region III
45 kVp, 0.0 mm	47.7%	51.6%	0.7%
45 kVp, 0.5 mm	30.4%	68.5%	1.1%
45 kVp, 1.0 mm	19.6%	79.0%	1.4%
45 kVp, 3.0 mm	3.2%	93.4%	3.4%
70 kVp, 0.5 mm	17.1%	69.9%	13.0%
70 kVp, 2.5 mm	2.4%	72.0%	25.6%
70 kVp, 4.5 mm	0.3%	63.4%	36.3%
90 kVp, 1.0 mm	9.2%	63.9%	26.9%
90 kVp, 2.5 mm	2.3%	58.7%	39.0%
90 kVp, 5.5 mm	0.1%	43.4%	56.5%

Data presented in Table IV may be used to determine the energy response necessary to achieve a desired level of ion chamber performance when measuring an x-ray beam of given tube potential and filtration. For example, at 45 kVp with 0.5 mm aluminum, regions I, II, and III contribute 30.4%, 68.5%, and 1.1% to the total exposure, respectively. Thus, a 10% decrease from the ideal energy response in each region would result in exposure measurement errors of approximately 3%, 7%, and 0.1%, respectively. As can be seen, the effect of energy response in region III is relatively insignificant. If the tube potential remains the same and the filtration is increased to 3.0 mm, a 10% decrease in response will result in errors of approximately 0.3%, 9%, and 0.3%. Response in region II is now the only significant contribution. In this manner the energy response requirements to provide a specific uncorrected error can be determined over the tube potential and filtration ranges of interest.

Since, in most cases, energy response does not significantly affect half-value layer measurement, energy response tailored to exposure error limitations will provide sufficient half-value layer measurement accuracy. Measurement of half-value layer will only be significantly affected by energy response if the filtration added during the measurement produces effective energies for which the energy response changes rapidly. For example, if chamber C were used to measure the half-value layer of an x-ray beam whose effective energy ranged from 10 keV to 18 keV as filters were added, the resulting error would be significant due to the response change from 0.5 to 0.9.

Conclusion

As expected, exposure measurement error due to energy response is significant. However, over the range of this study, energy response appears to have little affect on half-value layer measurements. In order to adequately correct for chamber energy response, the response or correction factors must be known over the entire effective energy range for which the chamber is to be used. The number of known points should be adequate to define the response curve, i.e. fewer points in regions of slowly changing response and more points in regions of rapid change.

Acknowledgement

Computing assistance was obtained from the Health Sciences Computing Facility, University of California at Los Angeles, supported by NIH research resources grant RR-3.

FIGURE 1

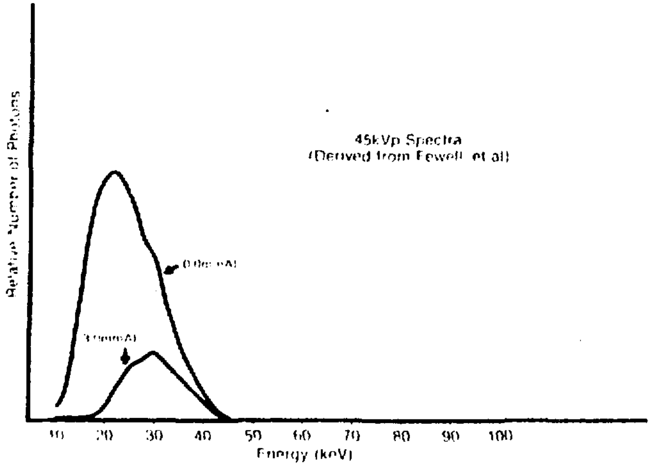


FIGURE 2

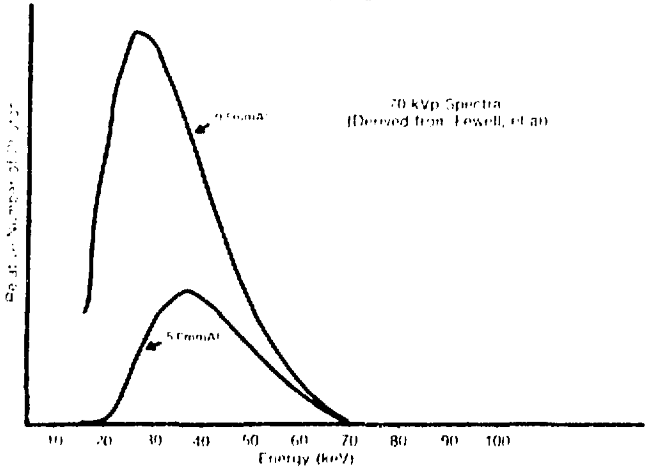


FIGURE 3

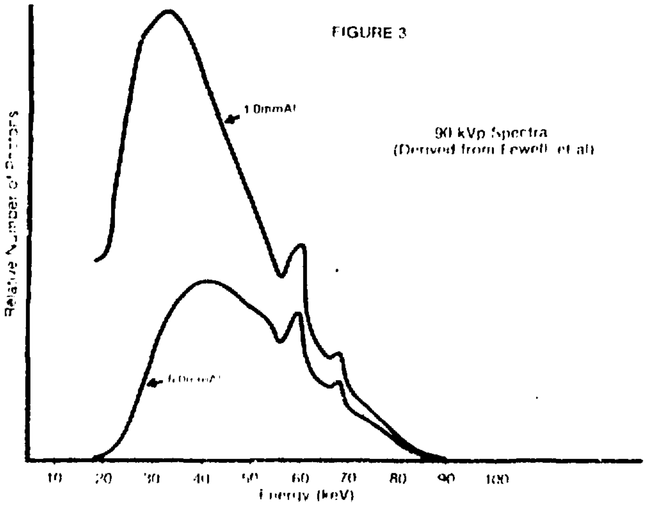


FIGURE 4

TYPICAL ION CHAMBER ENERGY RESPONSES

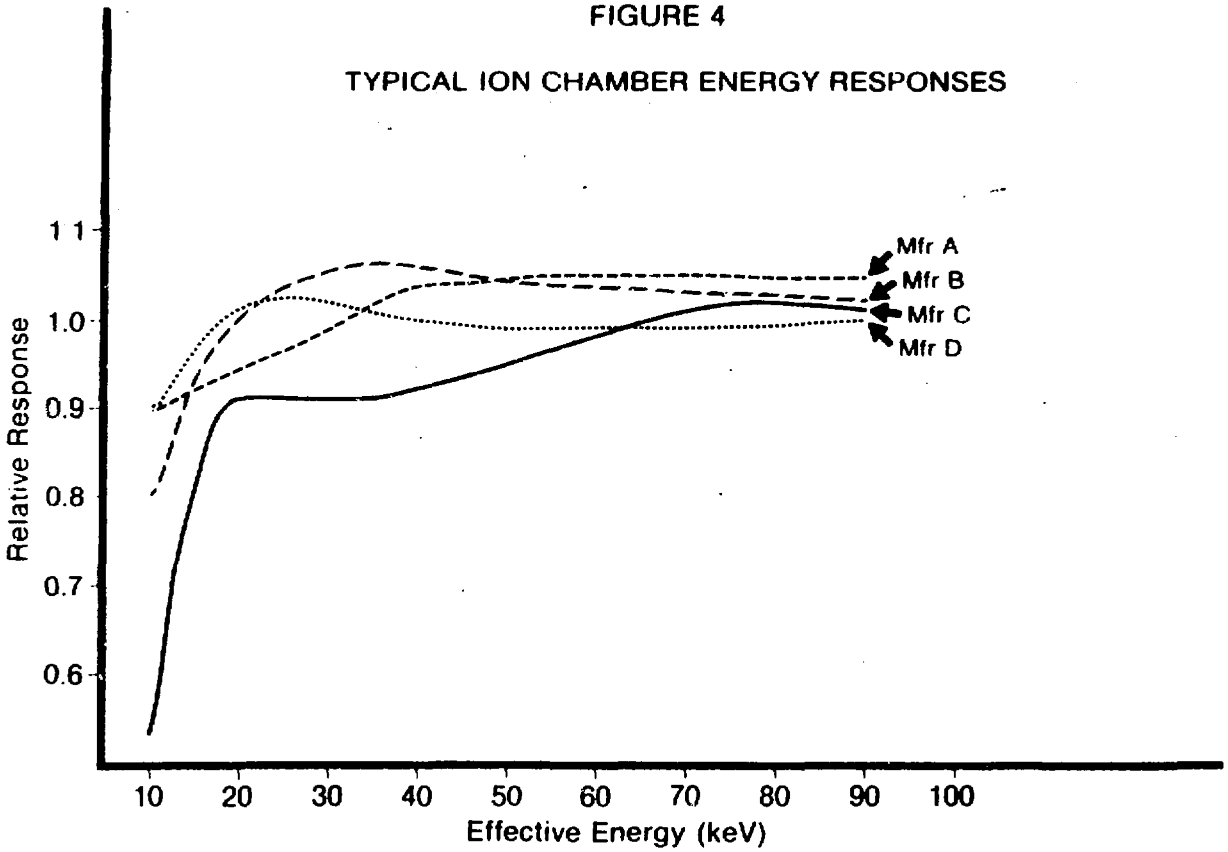
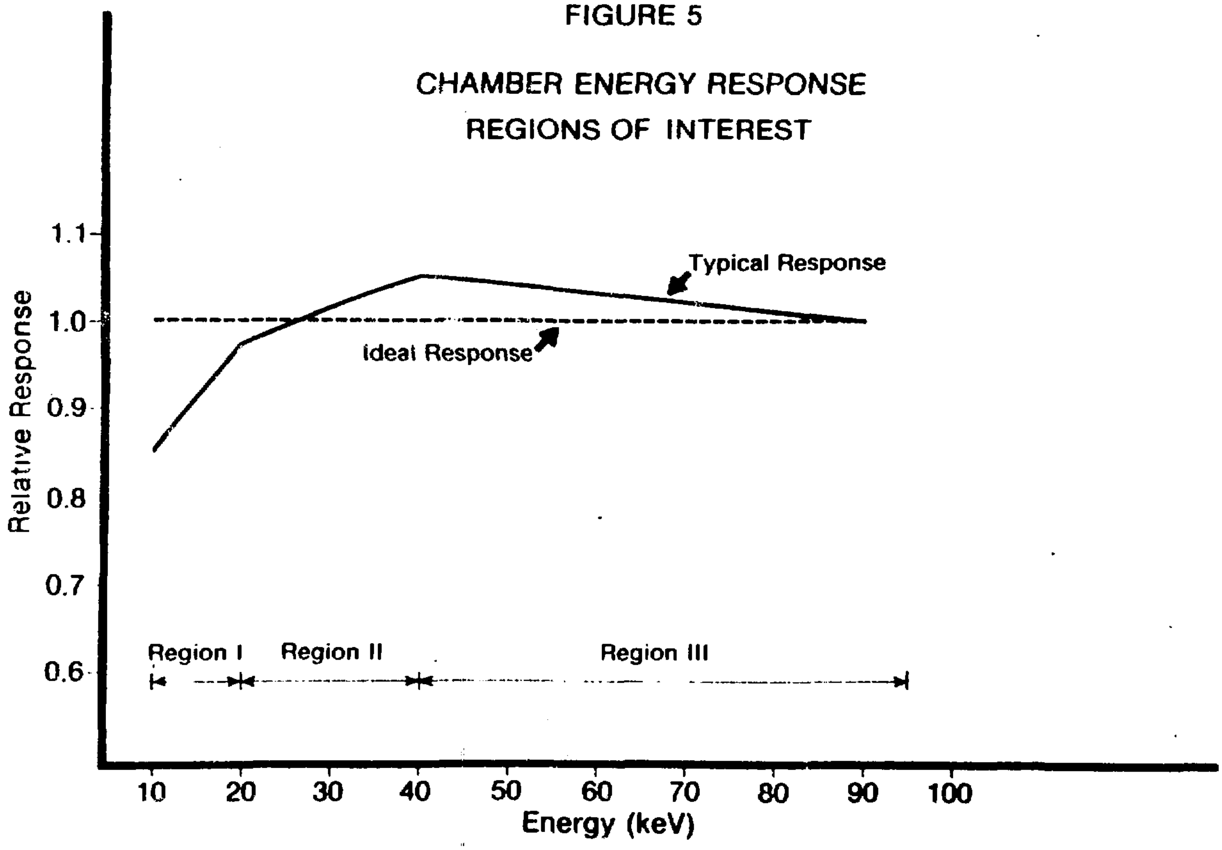


FIGURE 5

CHAMBER ENERGY RESPONSE
REGIONS OF INTEREST



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Question (Dwayne Spear, Battelle Northwest):

I am interested if you found that the medical profession has shown any interest in this type of data.

Answer (Bruce Horn):

By medical professions, you mean physicians?

Question (Dwayne):

Yes, and people using the X-Ray equipment.

Answer (Bruce Horn):

I'd probably have to say no; I'm a physicist who works for a medical group, and I guess it's my responsibility to make such measurements and therefore I worry about the problems involved.

Question (Tom Morgan, Bureau of Radiological Health):

Couple of things I'd like to ask. One, in your program, your modeling, you did not make a scattered component is that correct?

Answer (Bruce Horn):

That's true. The half-value-layer were merely simulated by, since we had exposure measurements every half millimeter determining when the exposure happened; it was not meant to show the errors that are possible in H.V.L. measurements, which scarrer is a significant contributor, but merely to show the effect the energy response.

Comment (Tom Morgan):

My point being I was pleased that, at the lower energy some empirical data that we had taken in investigating this agreed very closely with yours, but the higher energy where we were actually making measurements under a number of geometries. We found errors that would range up to 25-40%, which is a combination of the effect of the energy response as it is affected by the scatter component of the beam. The only point I would like to make here is that it's not totally independent as it would appear with this, if you do factor in the geometry and the scatter components. If I might, just one other comment with respect to the previous question: I think the medical community, and especially the Bureau of Radiological Health, is interested in providing this kind of information to the medical community, and they seem to be receptive and looking for this kind of guidance.

A COINCIDENCE COUNTING SYSTEM FOR DETECTION OF LOW LEVEL ACTIVITIES

by

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There is increasing use of isotopes that decay by electron capture. In a number of these isotopes a gamma ray is emitted by the nucleus following the capture of the orbital electron with a known half-life. If the half-life of the excited state is small compared to the resolving time of the coincidence circuit then the absolute activity of the source can be determined.

We have devised a system based on coincidence counting for measuring the absolute activity of such isotopes and have routinely used such a system for measuring the activity of ^{123}I.

The system consists of two NaI(Th) crystals, a high voltage supply, 2 preamplifiers, 2 amplifiers, two single channel analyzers, one coincidence unit and three scalers. A block diagram of the counting system is shown in *Figure 1. Figure 2 shows the actual set up of the equipment.* The crystals used were 3 x 1/4 and 3 x 1 inches, for the x-ray and gamma-ray respectively. The measurement of the absolute activity of an isotope by the method of coincidence counting is based on the following equations:

$$N_x = N_D \cdot E_x (f_{x,\ \mathrm{total}}), \qquad (1)$$

$$N_\gamma = N_D \cdot E_\gamma \cdot f_\gamma , \qquad (2)$$

and $$N_c = N_D \cdot E_x \cdot E_\gamma (f_{x\,\mathrm{capture}}) \cdot f_\gamma . \qquad (3)$$

Substitiuting Egns (1) and (2) into Egn (3):

$$N_D = \frac{N_x \cdot N_\gamma}{N_c} \left(\frac{f_{x,\ \mathrm{capture}}}{f_{x,\ \mathrm{total}}} \right) \qquad (4)$$

Where:

N_D= absolute activity of source in disintegrations per minute (dpm)

N_x= Net count rate of the x-ray detector in counts per minute (cpm)

N_γ= Net count rate of the γ-ray detector (cpm)

N_c= Net count rate of the coincidence circuit (cpm)

E_x= Counting efficiency of the x-ray detector

E_γ= Counting efficiency of the gamma-ray detector

$f_{x,total}$ = The total number of 28 KeV x-rays per disintegration (electron capture and internal conversion)

$f_{x, capture}$ = The fractional number per disintegration, of 28 KeV x-rays resulting from electron capture alone.

The absolute activity of the source can therefore be determined solely on the basis of observed counting rates and factors derived from the decay scheme of the isotope. Figures 3 and 4 show the details of the energetics of the ^{123}I decay.

This method of measuring the absolute activity of a source has been shown to be independent of the source-crystal geometry and source volume. It has also been shown that this measurement is relatively independent of the setting of the windows of the single channel analyzers. (1,2)

In using this system for ^{123}I, the derived factors required can be found in the literature (3,4). Then Egn (4) becomes for ^{123}I:

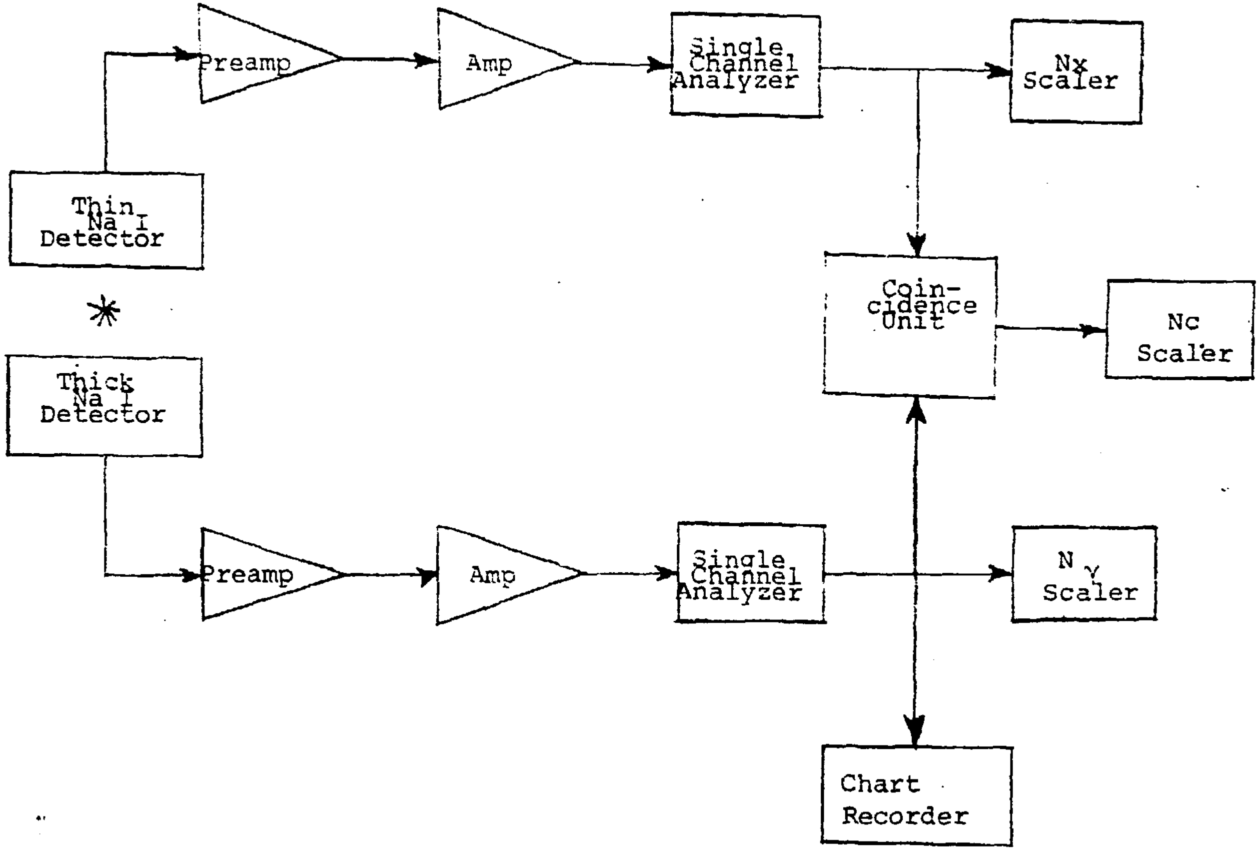
$$N_D = 0.86 \frac{N_x N_\gamma}{N_c} \qquad (J)$$

It should be noted that this counting system is completely unshielded and was routinely used in a nuclear medicine department of a large hospital, where the background is expected to be relatively high. Background levels were not found to be a problem in this counting system.

We have used this system for performing ^{123}I thyroid uptake examinations as well as measuring the amount of ^{123}I in urine. Table I shows the results of the urine measurements. For counting times ranging between 1 and 10 minutes the activity was measured with a fractional standard deviation of between 1.1 and 7.4 percent. The range of activities found in a 10 ml sample was between 10^{-3} and 10^{-1} μCi.

Besides measuring x-gamma coincidences, this type of system with the oppropriate size crystals can be used to measure the absolute activity of isotopes in which gamma-gamma coincidences occur. Isotopes which can be possible candidates for x-gamma coincidence includes: ^{54} Mn, ^{57}Co, ^{58}co, ^{63}Zn, ^{73}As, ^{85}Sr, and ^{87}Y.

Figure I. Block diagram of the circuit used in the coincidence counting



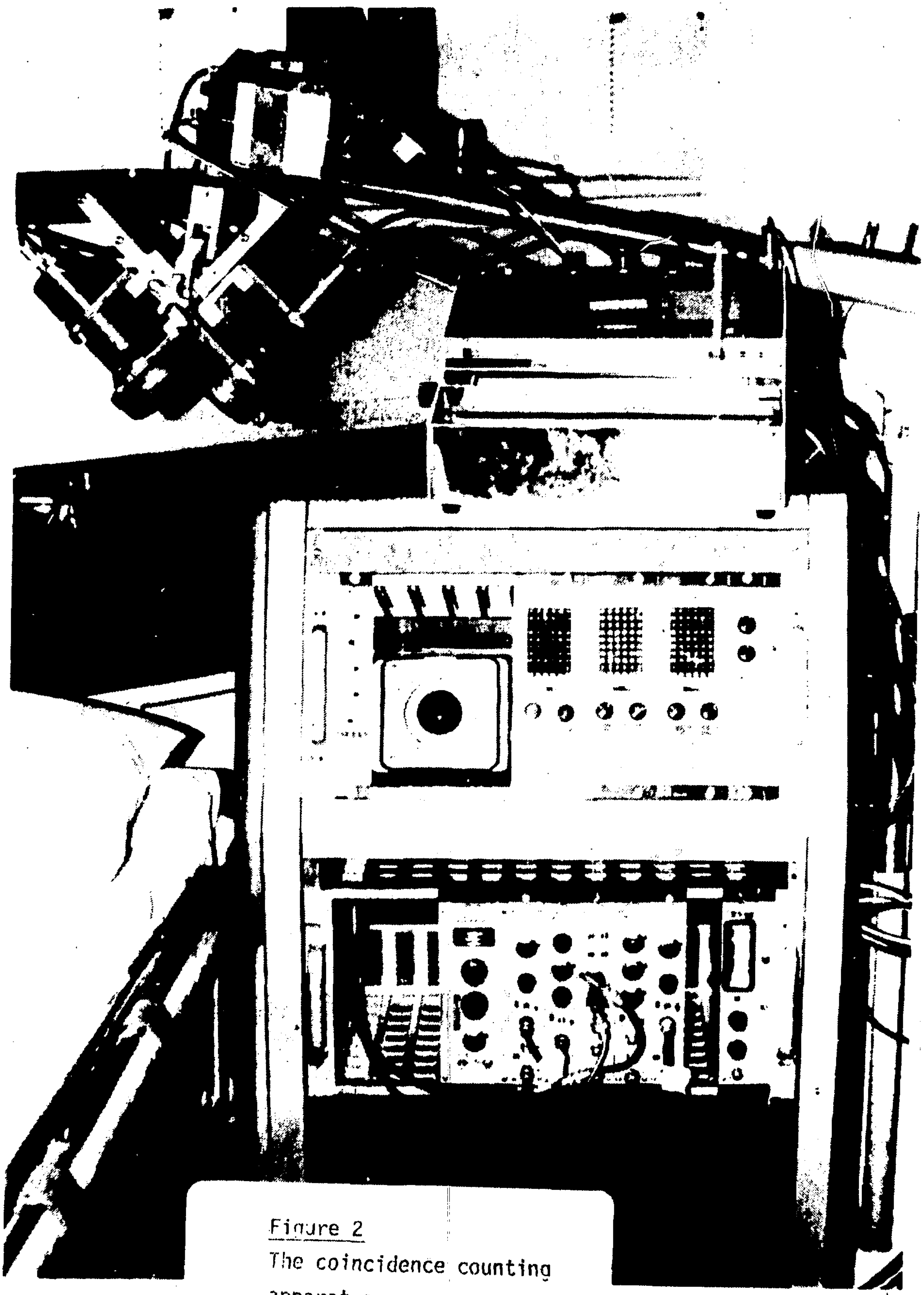


Figure 2

The coincidence counting apparatus.

Figure 3. Iodine-123 Decay Scheme (Simplified).

Half Life=13.0 hrs

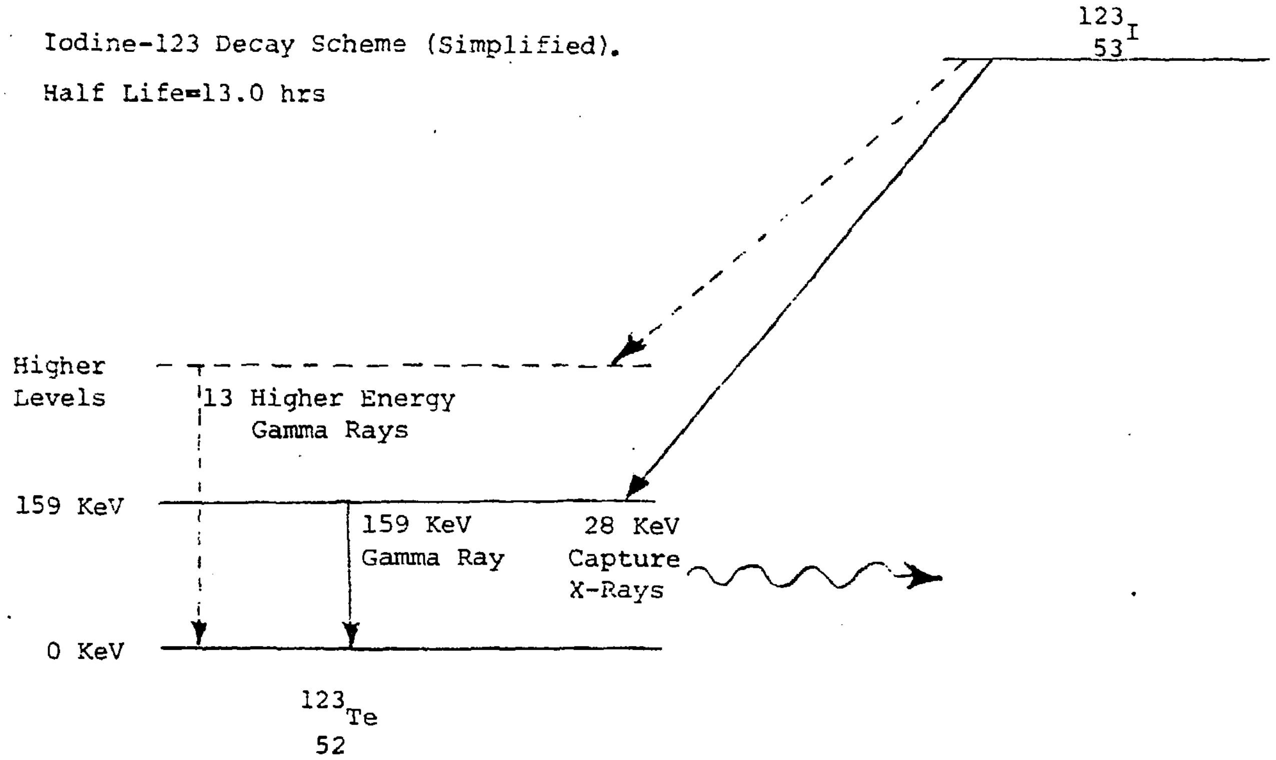


Figure 4. Iodine-123 photon emissions

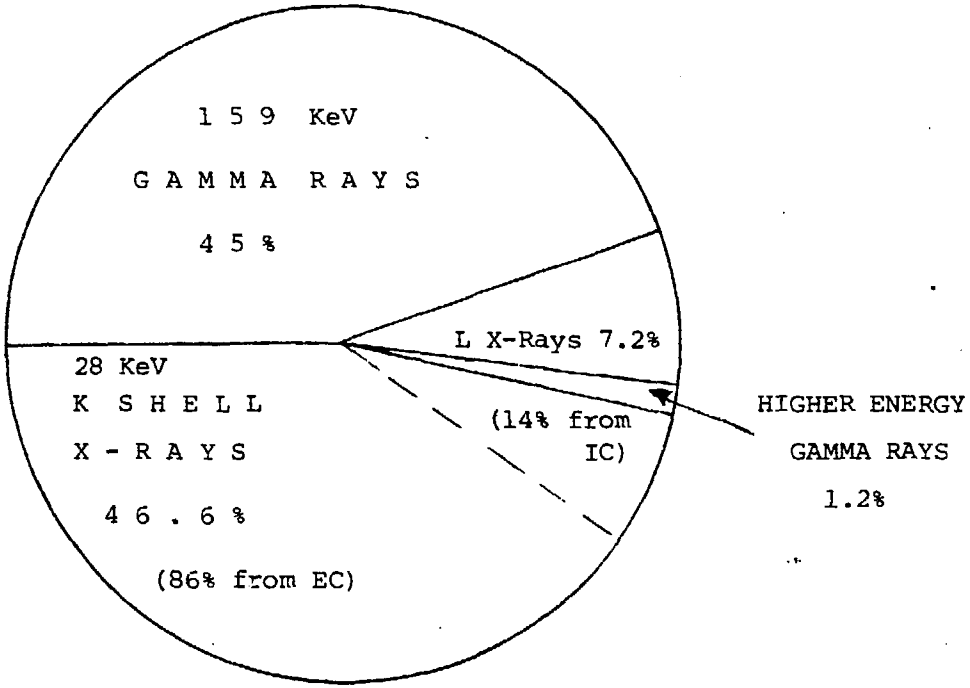


Table I. Measurement of ^{123}I activity in urine specimens.

Urine volume (cc)	time (min)	N_c (count)	Activity µCi	% Uncertainty (± %)
10.8	2	[illegible]	0.086	2.2
10.0	4	76[illegible]	0.030	1.1
10.0	1	2892	0.040	1.9
10.0	8	1133	0.006	3.0
10.0	2	2605	0.102	2.0
10.0	1	369	0.010	5.2
10.0	4	8745	0.034	1.1
10.0	2	1075	0.041	3.0
10.0	10	236	0.001	6.5
10.0	2	1081	0.051	3.0
10.0	2	2331	0.02	2.1
10.0	1	3802	0.06	1.6
10.0	4	938	0.006	3.2
20.0	4	4388	0.036	1.5
10.0	2	2177	0.022	2.1
10.0	10	181	0.004	7.4

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TECHNICAL SESSION G

COMPUTOR CONTROL OF RADIATION INSTRUMENTATION

Chair: Frank Gallagher

Frank Gallagher III, Chair:

I guess my first introduction to the computer was back a number of years ago, perhaps a dozen or so, when I took my first class in programming, and about my only vivid recollection of that is the long trudge over to the computer center in terrible weather--snow, sleet, whatever it was, and then to find out that my program had not run for some reason, and I had to do it all over again and come back in three or four hours. Well, I gradually reached the point where I could be a chauvinist and send a secretary, or I could send a technician, or something like that, but we still lots of IBM cards and problems like that. As an example of the improvement in computer technology, our new engineering building at the university was completed about the middle 1960's, about 1965 or 66, and in there all-knowing design they built a computer system into the building--in other words, it could not be removed through the door, or a window or anything like that. So while my industrial hygiene and safety colleagues were on vacation last summer I had to go consult on the problem of removing the computer from the building. Suddenly it was obsolete, the new computer was going to occupy a very small corner of the room, but they had to start getting their cutting torches out, and remove all of this steel and vacuum tubes and everything else that the old-fashioned computer would need in order to operate.

We've seen a tremendous development of mini-computers and micro-processors over the last couple of years, and I guess the first thing that developed out of that would be the hand-held calculators and some of the very fancy watches that are on the market now. Now some of the large portable instruments can even be purchased with the rather fancy mini-computer capabilities. Some of the liquid scintillation counters are this way. We need to convert some of our existing equipment however, that we have spent so much money on already and won't be able to replace for maybe ten or fifteen years, so this is a problem of finding interface, and specially designed equipment to handle the calculations that we need. Also, we need things like new electronics for survey meters. It would be very nice to get disintegrations per minute rather than counts per minute, digital printouts and things like that. In the next few years we're probably going to have to handle the conversion to S.I. units or meters will have to read out in both dpn, beuarels, Rads, grays, and all that nonsense. So computer technology is going to have to help us out in many of these areas.

This afternoon we're going to hear about computer interface with area monitors, air monitor systems, various types of access controls, and interlocks to high radiation areas, which are computer controlled. I'll say one thing about the speaker. It suddenly occurred to me while we were listening that the 1969 symposium that was sponsored by the Southern California chapter, was held in Los Angeles, and during that meeting we had some pretty good rains also, and 1969 was the last time I could not get to my house because of the floods. There's something about all those "laws" that are very true.

INTEGRATED MINICOMPUTER ALPHA ANALYSIS SYSTEM

by

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ABSTRACT

Approximately 1,000 stack and occupation air samples from plutonium and uranium facilities at LASL are analyzed daily. The concentrations of radio-nuclides in air are determined by measuring absolute alpha activities of particulates collected on air sample filter media.

The LASL-designed and built Integrated Minicomputer Pulse system (IMPULSE) is an interface between many detectors of extremely simple design and a Digital Equipment Corporation (DEC) PDP-11/04 minicomputer. The detectors are photomultiplier tubes faced with zinc sulfide (ZnS). The average detector background is approximately 0.07 cpm.

The IMPULSE system includes two mainframes, each of which can hold up to 64 detectors. The current hardware configuration includes 64 detectors in one mainframe and 40 detectors in the other. Each mainframe contains a minicomputer with 28K words of Random Access Memory. One minicomputer controls the detectors in both mainframes. A second computer was added for fail-safe redundancy and to support other laboratory computer requirements. The main minicomputer includes a dual floppy disk system and a dual DEC 'RK05' disk system for mass storage. The RK05 facilitates report generation and trend analysis.

The IMPULSE hardware provides for passage of data from the detectors to the computer, and for passage of status and control information from the computer to the detector stations. Essentially all data collection is accomplished in the computer proper. Batch loading and processing of samples, as well as individual sample analysis, is possible. Air sample filter location codes are stored on disk so that they can be assigned by the operator to a specific set of samples which have been loaded into

IMPULSE in a predetermined order. For a 10-minute sample analysis time, the lower limit of detectability is about 0.2 dpm.

The computer prints sample locations, radioisotopes of interest, sampling and analysis dates and times, gross counts, cpm, dpm, lower-limits-of-detectability, and dpm per m^3 (or dpm per ml). Also printed are appropriate MPC's, %MPC's released, μCi/ml, and μCi's released. Errors (1σ) are also calculated and printed.

INTRODUCTION

Approximately 1,000 building-stack and occupational air samples from plutonium and uranium facilities at LASL are analyzed daily. The concentrations of plutonium and uranium in air are determined by measuring the absolute alpha activity of particulates collected on filter media. In the past, commercial count-analysis systems were used to count large numbers of air samples. The available systems employed a single proportional counter and a mechanical sample changer to sequentially count many samples. The principal shortcomings of these systems have been excessive turnaround times, low sample throughput rates, and high maintenance costs associated with mechanical changers and specialized electronics. Also, the gas-flow proportional counter systems generate only raw count data. It is very difficult and costly to interface a computer to such systems to convert raw count data into a desired form.

The IMPULSE system was conceived at LASL, with the key design objective directed to overcoming the shortcomings associated with the gas-flow proportional counter systems. IMPULSE employs a large number of logically independent counting stations (channels), which are coordinated by a single system controller that interfaces to a minicomputer. The computer memory and I/O structure substitutes for external counters, scalers, and timers. The raw count data, acquired under computer control, are immediately available for analysis and mass storage to archival media. Single digit LED displays at each detector allow an operator to visually scan the functional status of each individual detector. An audible alarm is available for pre-selected fault conditions. Almost all data collection is accomplished in the computer proper. The only exception is a first-in/first-out (FIFO) memory buffer in the IMPULSE hardware. This is to provide pulse pair resolution in the system at much higher rates (up to 10^6 cps) than is possible at computer-interrupt service times (10^4 cps).

A primary design objective was system reliability. This objective was achieved through selective redundancy and through the use of proven design concepts and system components. The system hardware is simple in construction and concept.

The minicomputer is programmed so that backgrounds, efficiencies, and lower limits of detectability can be automatically acquired and stored on disk files for each detector. The software calculates and stores errors (1σ) associated with these parameters. The computer notifies the operator when re-calibrations are required. The software automatically re-determines backgrounds during the night-time. All old calibration data is replaced in file by the new data. If updated calibration data falls outside of pre-specified tolerance limits, then the new data is rejected and the detector is disabled until the operator investigates the problem. All calibration results are printed out on local printers.

Raw count data are acquired under computer control. When operating in the batch mode, an operator is required to provide minimal input information to the computer. First, he must identify the batch name which identifies a segment of a disk file that contains filter information that is of a permanent nature (all files can be updated to account for changes in sample parameters). The information stored for each sample in a batch includes the filter name, isotope of interest, sampling rate and period, filter analysis time, and effluent rates for stacks.

An operator must also identify the detector where the first sample of a batch will be located. Samples are sequentially loaded into available detector stations. Samples can be selectively omitted from a batch.

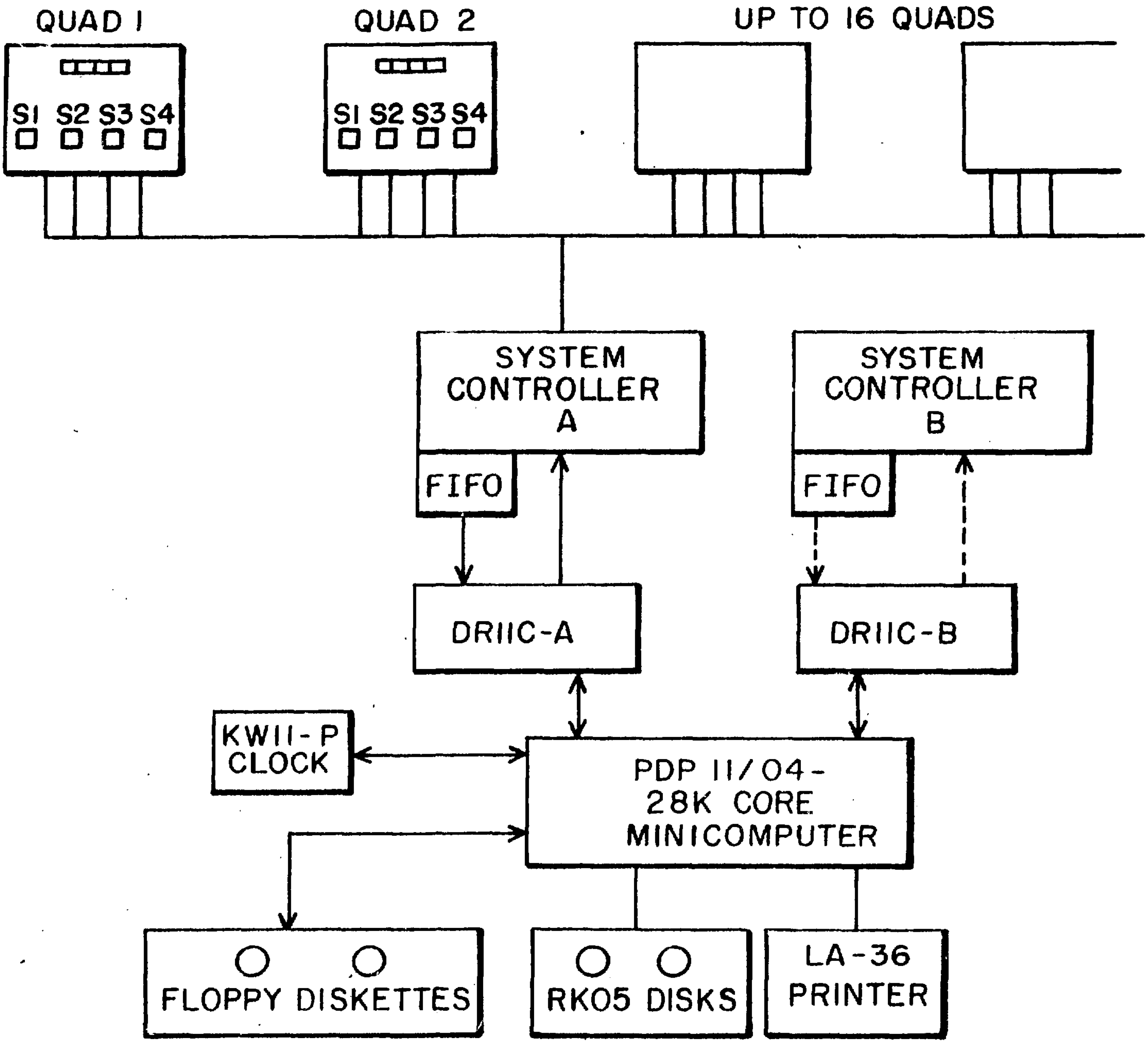
As part of the batch input command string, an operator can over-ride analysis times or sampling times already pre-specified in the batch file.

In the individual sample(s) mode, the input command string is similar to the batch input command string. For both the individual sample(s) and the batch configuration, the operator can send information to mass storage or prevent information from being stored. In either case, results are always printed as hard copy. All data in mass storage can easily be updated by an operator if required. Trend analysis or report generation can be conducted by an operator for any period of interest.

Hard copy results of data fall into two categories. First, all samples analyzed are tabulated in the order in which they are loaded into the detectors. Second, after all results are tabulated, another sheet is printed which identifies only those samples which have exceeded appropriate warning detection levels. This information is provided to the appropriate operational health physicist for necessary action.

SYSTEM CONFIGURATION

A block diagram for a single IMPULSE mainframe is shown in Figure 1. The system controller was designed to handle 64 counting stations. Each computer is equipped with two general-purpose digital I/O interfaces such that a single computer can control both mainframes concurrently. The system software (ALPHA) developed for IMPULSE allocates memory, resources, and logic such that a two-system-per-computer configuration appears to an operator as a contiguous 128-channel system. Only 104 detectors are presently being used with IMPULSE. As required, more alpha counting stations may be added. The addition of beta counting stations is also under consideration.



IMPULSE

INTEGRATED MINICOMPUTER
ALPHA ANALYSIS SYSTEM

All the components in each mainframe are housed in a group of five standard cabinet racks. The only external component is the keyboard/printer terminal associated with the minicomputer. The mass storage capability is located only on one mainframe.

The system is modular to the maximum practical extent. Two identical failures are required to cause the system to be non-operational, although minor recabling may be required in some cases of single failures. Failure of one system controller reduces the system to 64 detector stations.

In addition to the redundancy concept, the system hardware has demonstrated a high degree of reliability. This is attributed to conservative design.

COMPUTER HARDWARE

The computer hardware configuration consists of the following DEC equipment:

(a) PDP 11/04 with 28-K word memory: 16-K words of core and 12-K words of MOS.

(b) KW11-P programmable clock.

(c) LA-36 Decwriter (keyboard/printer).

(d) Two DR11C general-purpose digital (programmed) I/O interfaces.

(e) An RX11 diskette (floppy) controller with two drives providing 256-K words of mass storage.

(f) RT-11 single-job operating system software.

One computer has been equipped with an RK05 disk controller with two drives (2.5-M words) for archival data storage requirements and retrieval capabilities.

OPERATIONAL CONCEPT

The provision of two PDP 11/04 minicomputers in the system, although increasing the overall system reliability, is primarily in support of the operational concept. Each mainframe is supported by a minicomputer during test, development, and maintenance operations. This allows two parallel, independent operations to be performed. Operationally, it is routine for any necessary maintenance to be performed on one mainframe, while the other mainframe is used for routine analysis purposes.

During normal operations, when no development or maintenance is being performed, both mainframes are supported by one minicomputer. This minicomputer has the dual RK05 mass storage device. The other minicomputer may then be used for trend analysis, report preparation, and other laboratory purposes. Each minicomputer is equipped with a dual "floppy" diskette device for the operating system software, the system analysis software, and limited mass-storage capability.

DETECTOR STATIONS

The counting stations of IMPULSE are organized into "quads" of four detectors each. The quad is a standard rack-mountable chassis 8-3/4" high. There are 26 quads per mainframe. The quad houses all components necessary to hold the samples, detect the particles, and produce well-conditioned output pulses. A count station consists of a ZnS-faced PM tube and a light-baffled sliding drawer. Filter media to be counted are inserted into recesses in the drawers, and the drawers are closed to expose the ZnS-faced tubes to alpha radiation from particulates collected on the samples. The recesses hold the standard 2-1/8" diameter HV-70 filter papers used extensively at LASL for air sampling.

A 2-inch photomultiplier tube, RCA 6655A, was chosen. The tube is shielded with two layers of electromagnetic shielding, foil-wrapped around the glass portion. An over-wrap of two layers of black tape assists in light shielding. The PM tube is mounted vertically with the face down. The tube is held in this position by an aluminum housing with two internal O-rings. The O-rings position and hold the tube, as well as providing a light seal. The detection mechanism is a layer of silver-activated ZnS, which is applied to the face of the PM tube by dusting on the surface of double-sided transparent adhesive tape. This method of screen application is simple, reliable, and repeatable.

A small metal box on the base (top) of the tube houses all components directly associated with the operation of the tube. Connectors on this box provide high voltage (1000 volts), diagnostic and pulse signal outputs, as well as the tube socket. A potentiometer is used to drop the system high voltage to the voltage selected for an individual tube. A unique hybrid circuit, mounted within the tube base, is used to reduce the number of connections required within this assembly. The hybrid contains the entire dynode divider string, stabilizing capacitors, and a 1000:1 voltage divider for diagnostic purposes. In addition, an opto-isolator is mounted on the hybrid substrate. The opto-isolator is used for gross over-current detection. Use of this manufacturing technique has resulted in a 60% reduction in required connections, with attendant savings in assembly cost. The hybrid assembly is laser-trimmed, and thus offers increased precision when compared to discrete components.

The pulse signal from the PM tube is passed through a short length of coaxial cable to a plug-in card which contains the signal processing functions for a single tube. The design of this card provides acceptable signal processing with minimum electronics. A single operational amplifier, followed by a comparator, provides signal conditioning and threshold discrimination. Each pulse is then converted to a digital signal of known width. The signal processor card is also used to monitor two associated error conditions (tube over-current and sample drawer open), to provide this information to the external system and to generate a pulse whenever an error condition changes.

A 2-1/8-inch diameter sample is held in a drawer assembly beneath a tube, within a shallow cylindrical cavity. The drawer assembly is Teflon-coated for lubrication and ease of decontamination, if required. A thin vane on the rear of the drawer interfaces with a slotted optical switch in the main chassis to provide a "drawer open" indication.

The entire mechanical assembly for holding the tube and sample is manufactured from commercial stock to relatively loose tolerances, and very little machining is required. Of particular interest is the fact that there are no unusual measures for light sealing, such as felt pads, aluminized Mylar, external housings, etc. A simple 2-bounce light baffle at the drawer front, modest spring-loading of the drawer, two O-rings, and the tape wrapping on the PM tube provide sufficient light seal.

Four detector assemblies are mounted in the main quad chassis. Also included are two modular low-voltage power supplies, a potted high-voltage distribution assembly, and a printed circuit board for interconnections. Each of the two IMPULSE mainframes contains 16 quads. Each mainframe has a 1000-volt power supply which can supply the power needs of the detector stations of both IMPULSE mainframes.

Each count station has a single digit LED display associated with it. A 4-bit code, selecting one of 16 characters, can be sent to any LED. The LEDs are used to display the logical or physical status of a count station to the operator. The information is under direct control of the minicomputer and may be arbitrarily used for any purpose under software control.

SYSTEM CONTROLLER

The system controller consists of four "data modules", each servicing four quads, a master controller section, a 64-word (16-bits each) FIFO, and logic required to interface to a DEC, DR11C general-purpose digital (programmed) I/O port to the computer. Logic in the controller includes a data latch and 3 data lines for each of 64 possible stations in a system. Two data lines are read-only and present status for drawer open-closed and PM tube excess-current conditions. The third line is read-write and provides on-off control of data transfer from the controller to the computer.

The controller provides decoding-processing of five operational commands as output from computer-software via the DR11C output register:

(a) READ-DATA: A general system clear command that clears the FIFO, FIFO-full flag, and data latches.

(b) READ-DISABLE: Logically disables a specified channel from sending "hit" data to the computer.

(c) READ-ENABLE: Logically enables "hit" data transmission for a channel.

(d) READ-STATUS: Encodes, for a specified channel, a 16-bit status word containing channel number, states of the three data lines, and a flag marking it as status data. Channels need not be read-enabled to read status. The encoded status word is stored in the FIFO, which subsequently is read by the program via the DR11C input register.

(e) WRITE-LED: Extracts a 4-bit character code from the command word and writes it into a LED for a specified channel. This character is latched by the display so no refresh cycles are required.

A data "hit" (sufficient current flow in a PM) is sensed by the PM signal processor and sets the data latch associated with the hit channel. If that channel is read-enabled, the data module controlling that channel "signals" the master which then encodes the three data lines as status, six bits as channel address, and one bit as a hit flag into a 16-bit word, which it stores in the FIFO. Pulsed event data is accepted asynchronously, and may be <u>any</u> event data not requiring precision timing.

All information flow from the system controller to the computer is directed through the FIFO buffer. The data-word processing rate of the controller is approximately 750 kHz. The computer software hit-request handling rate is approximately 10^4 events per second. The 64-word FIFO provides buffering external to the computer to handle instantaneous bursts of data input. The controller senses for a FIFO-full condition after each data-word entry to the FIFO. If the FIFO becomes full, an internal FIFO-full flag is set and the contents of the FIFO, when read by software, are tagged as having contributed to the FIFO-full condition. The controller will terminate writes to the FIFO until software clears the full condition via the READ-DATA command. Data input from the controller is handled on an interrupt basis as described below. Occurrences of a FIFO-full condition are audibly reported and are logged by the software as a time-tagged message on the console printer.

Each controller is equipped with a low voltage power supply. The low voltage supplies are each capable of supplying power to both system controllers.

MAINTENANCE AND REPAIR

Aside from routine cleaning of sample drawers, the minicomputers are the only system components requiring periodic maintenance.

Periodic background counts and calibrations under software control can uncover trends indicating required adjustments. A special set of software routines has been provided to assist in fault isolation and repair.

Maintenance and repair of major items normally consists of replacement of a module or other plug-in component.

SYSTEM SOFTWARE

The ALPHA program, which supports the IMPULSE system, is written about 95% in FORTRAN, with the remainder in assembly language. The latter segment is restricted to interrupt handling of the clock and I/O between the computer and system controller; this software needs to be as efficient as possible and is not expected to change much from its current form.

The program is configured to handle two system controllers. Sufficient self-timing and checking features are built into the program to ensure that one computer can handle input from two systems for the expected aggregate count rates.

ALPHA is structured into five segments:

1. Main Program. The main program consists of a rather large DO LOOP, which serves as a system coordinator. It continuously scans all data channels (count stations) checking for logical consistency between hardware and software status of each channel. The loop monitors elapsed time-out or maximum integer-count of channels assigned to counting. The results of a time-count or count-out are further processed, depending on the type of count assigned to a channel. The main loop also monitors for completion of command input by the operator via the terminal. Upon sensing "command input done", the main program branches to a command preprocessor, which validates and then executes legal commands. Completed command processors return to the beginning of the main program loop.

A secondary loop, operating once each second, issues a read-status command for all channels, updates the LED digit display for each channel, and does other low-rate support functions.

2. Command Processors. A total of 28 commands are available to the operator for control and interrogation of the system. Some commands have subfunction operands such that the total discrete command set numbers about 50.

Character-wise input is handled on an interrupt basis, using RT-11 system facilities but with "home brand" editing and echo print. Commands are not executed until the "return" key is struck such that command input is handled concurrent with counting and main loop processing.

A limited subset of the commands requires further operator input. Such input is "timed-out" by the system so the operator cannot "hang" the system for longer than 30 s.

Commands are checked for validity and structure; command arguments are inspected for "reasonableness" to the maximum extent possible. Some critical commands, which affect operation of the system in a gross sense, are password protected to minimize the possibility of unintentional usage or "intentional" operator misuse.

Considerable effort was expended in designing a simple but adequate command set. Processing 1000 samples per day could prove to be very complicated after several days or weeks of operation -- a messy command set compounds the probability of operational error.

Commands are available to assign channels to counting either singly or on a mass-load basis. Attempts to use channels not logically available are rejected.

Each sample count entered to the system must be identified with an alpha-numeric name, filter installation date, sampling-time, air-flow rate, and the time between sample removal and analysis. Since filters are normally collected and counted in batches, with 10 to 90 filters per batch, the program provides for loading in a batch mode. Batch names and filter names within batches are prestored on a disk file that is read for automatic name and count-time assignment. Another mass-load mode allows designation of a six-character keyword name for a set of N samples. The first sample in this mode is assigned the keyword name, and the remaining N-1 names are program-generated as the keyword plus two digits of sequence number 01 through 99.

Name entries are cross-checked before acceptance. Attempts to use names reserved for system use or names already being counted by the system are rejected.

3. Interrupt Processors. This segment consists of three interrupt handlers written in assembly language.

The KW11-P clock handler operates at a 1-ms interrupt rate and maintains two timing services to the main program:

(a) A double precision 1-s resolution time used to time-out count operations and time label channel calibration data.

(b) A six-decade calendar clock (seconds through year) used to date and time-tag significant events and counted results. This clock service furnishes day, month, and year update (including leap years) - facilities not available under our current version of the RT-11 operation system.

Input from the FIFO buffer is handled by either of two interrupt processors, depending on whether input is "hit" or status data. A single hit logged into a nonfull FIFO generates an "A-request" interrupt via the DR11C. Status word input generates a "B-request" interrupt. A FIFO-full condition with all "hit" or a mix of "hit" and status data generates a B-request. The interrupt processing from a two-system configuration is partially merged, such that the A and B interrupt-processor procedures handle both systems.

Interrupt processing consists of first extracting the 6-bit channel address from the data word which serves as an index into either of two arrays, and then inspecting the word for data type. "Hit" data results in incrementing an indexed 16-bit word within an integer count array (0 to 65,000 counts per channel). Status data input results in storage of the new status in an integer hardware-status array and setting a new status flag subsequently processed by the main loop program.

"Hit" data also contains status bits that may indicate error conditions at time of the hit; an excess-current or drawer-open condition results in a software read-disable of the offending channel.

4. Calibration. Each physical channel in the IMPULSE system is handled asynchronously with respect to other channels. The program maintains a software status array which defines one of 12 functional states for a channel:

(a) Sample count.

(b) Background count.

(c) Calibration-standard count.

(d) Idle but needs background.

(e) Idle but needs calibration.

(f) Idle and needs both background and calibration.

(g) Idle, a proper count has been assigned but drawer is open.

(h) Idle, a sample or calibration count is done but the station drawer has not been opened..

(i) Idle by virtue of a persistent excess-current condition in the PM.

(j) Idle by manual command (channel out of service).

(k) Idle and available for counting.

It is this state that is periodically sent to LEDs corresponding to channels. States s and h blink every second to attract the operator's attention.

When the ALPHA program is first loaded from disk, all channels that respond properly to a read-status command are assigned to logical state f. Those not responding properly are assigned state j.

To use a channel for sample counting, the operator is <u>forced</u> to first do a background count and then a calibration-standard count. The results of these two sequential counts are conditionally accepted, based on percentage change from prior background or standard counts. Acceptable background or calibration counts are logged into memory along with a time-tag as to when the count was taken. These calibration data remain valid for a manually determined period of time (days), after which the calibration data become "outdated" and the software status reverts to state d, e, or f. To use the channel again, the operator is forced to recalibrate. Calibration data, or extension of time thereof, may be manually entered. On idle channels whose background time has expired, the program automatically initiates background counts. This feature can be locked out or restricted to hours when the system runs unattended.

5. Data Storage and Retrieval. Most counted sample results are written to disk archival storage. Information written includes sample name, date and time of filter installation to exposure, air flow rate, elapsed count time, raw count acquired, and calibration data for system channel used.

These data are retrievable from storage via operator command. Selective retrieval using one or more of the information attributes is provided.

IMPULSE SAMPLE ANALYSIS METHODS AND PROCEDURES

All plutonium and uranium facilities at LASL are carefully monitored for the purposes of public and worker safety. The concentrations of these radionuclides in air are determined by measuring absolute alpha activities of the air samples.

IMPULSE CALIBRATIONS AND STATISTICAL CONSIDERATIONS

EFFICIENCY

The mean efficiency for the 104 detectors of the IMPULSE system is approximately 35 ± 1 percent (1σ). The range is from 32 to 38 percent. For any given detector, repeated efficiency measurements taken over a period of many months have been found to differ from the mean for that detector by no more than ±.01 (or approximately 3% of the mean).

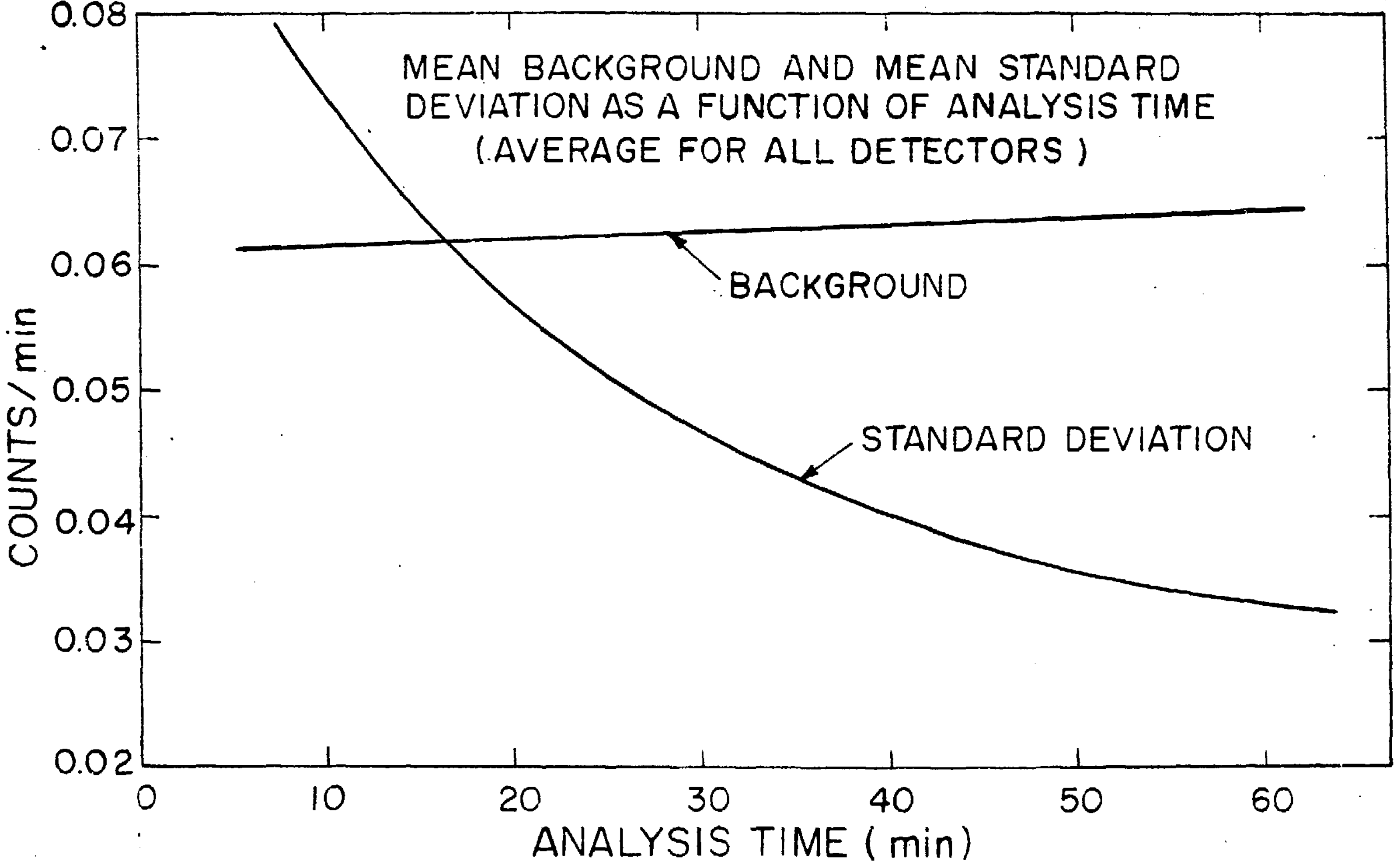
BACKGROUND

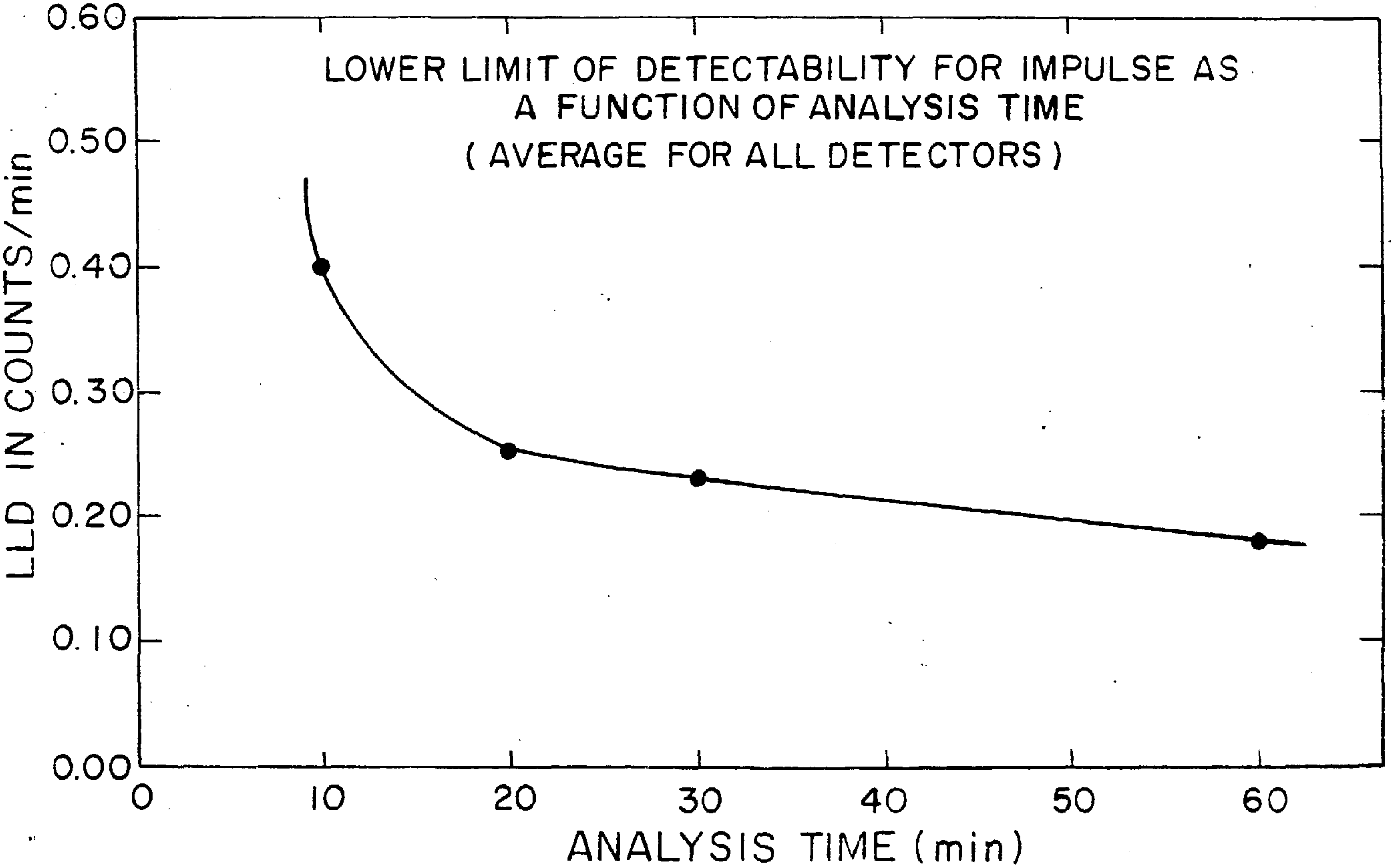
The IMPULSE system determines a background by calculating the mean and the standard deviation of a pre-specified number of repeat counts for the analysis time of interest. After the means and standard deviations (1σ) are generated for each detector, this information is stored as a permanent file on RK05 disk. Backgrounds and standard deviations are stored as cpm values. This file can be updated as subsequent recalibrations are conducted. With this information, the software will also be able to generate LLD's for each detector when sample analyses are to be conducted.

The mean background for all IMPULSE detectors for all times in the range of 10 to 60 minutes is $6.2 \times 10^{-2} \pm 3.2 \times 10^{-2}$ cpm. In Figure 2, we show the mean background (cpm) for all IMPULSE detectors as a function of analysis times in the range considered. The background is observed to increase slightly with increasing analysis time. This increase is statistically insignificant because the background values for all analysis times in the range considered are well within one standard deviation of each other. There is, however, a significant decrease in the mean standard deviation with increasing analysis times. This decrease is also shown in Figure 2.

Background distributions for the range of analysis times considered are Poisson in nature. The lower-limit-of-detectability (LLD) is defined as the 99.8 percent cumulative value for a given Poisson distribution. The 99.8 percent cumulative value is determined once the mean background is known. A simple algorithm is used to calculate counts corresponding to the 99.8 percentile. There is a decrease in the LLD with increasing analysis time, as shown in Figure 3.

Physically, if a measured count falls below the 99.8 percent cumulative value, we consider it to be background. The cumulative value chosen to describe an LLD is quite arbitrary. For our purposes, the LLD is so low with respect to the MPC values that we are trying to detect, that it really doesn't matter whether one chooses 99.8, 90, 80, etc. The 99.8 value was chosen as a reasonable way to accept or reject very low count data. Thus, if a filter count is less than or equal to the LLD, it is


MEAN BACKGROUND AND MEAN STANDARD DEVIATION AS A FUNCTION OF ANALYSIS TIME (AVERAGE FOR ALL DETECTORS)
BACKGROUND
STANDARD DEVIATION
COUNTS/min
0.08
0.07
0.06
0.05
0.04
0.03
0.02
0
10
20
30
40
50
60
ANALYSIS TIME (min)


LOWER LIMIT OF DETECTABILITY FOR IMPULSE AS
A FUNCTION OF ANALYSIS TIME
(AVERAGE FOR ALL DETECTORS)
LLD IN COUNTS/min
0.60
0.50
0.40
0.30
0.20
0.10
0.00
0
10
20
30
40
50
60
ANALYSIS TIME (min)

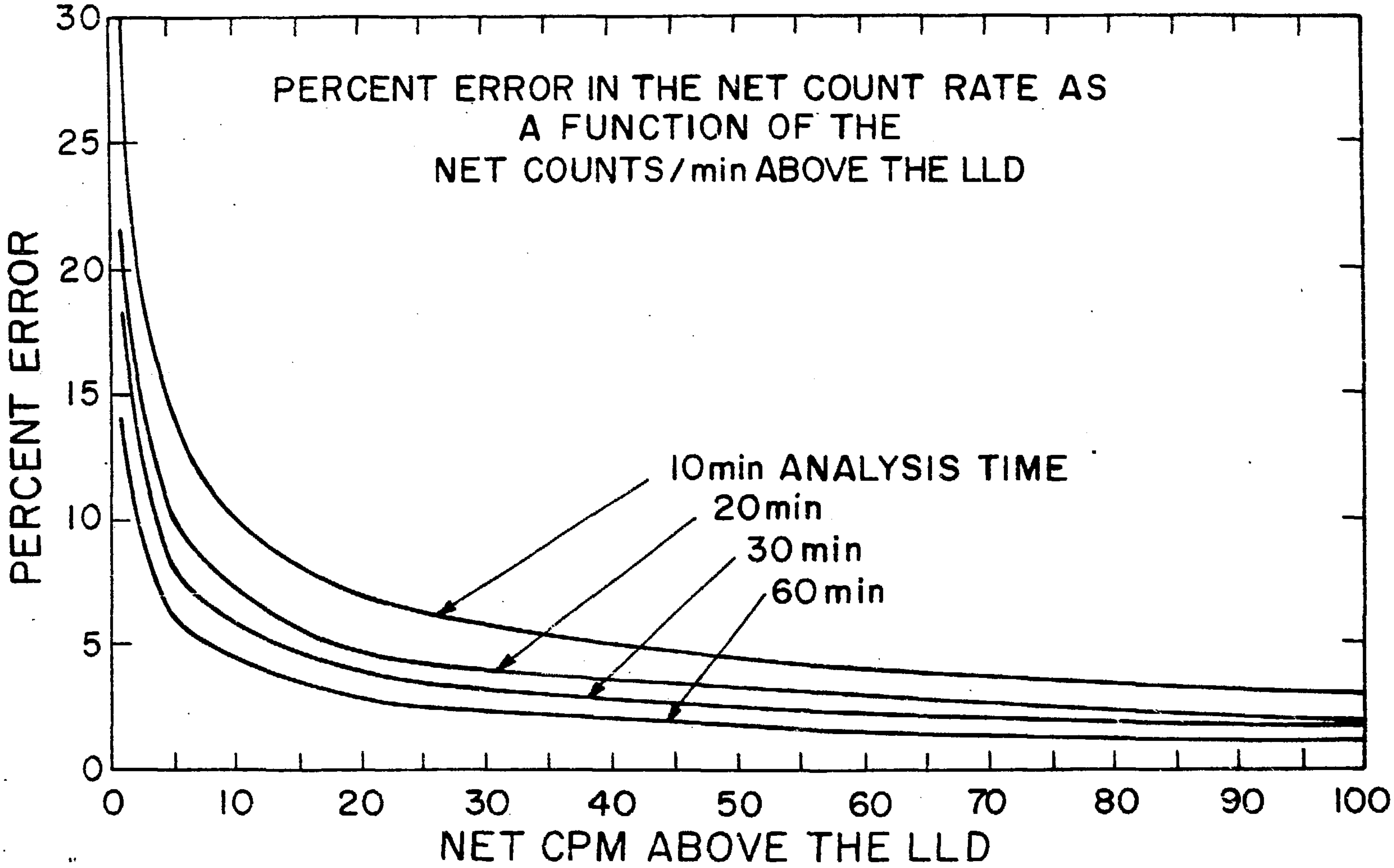
considered to be a background. If a count is greater than the LLD, the net count is determined by subtracting the value of the mean background for the given detector.

The percent error in the net count rate (1σ) as a function of the number of counts per minute above the LLD is presented in Figure 4. This relationship is shown for analysis times of 10, 20, 30 and 60 minutes.

STACK EFFLUENT RELEASE DETERMINATIONS

At the LASL there are 92 stacks from process facilities which are continuously monitored in order to determine gaseous effluent releases of plutonium or uranium. All stacks are sampled for 168 hours with 2-1/8 inch diameter HV-70 filter paper. The sampling rate is 2 cfm. All stack filters are initially screened to identify possible problem areas. The documentable release values are determined by counting all samples 168 hours after removal from the sampling heads. The radon/thoron contribution to the measured activity is then insignificant. We have found this approach to be more acceptable than the use of Koval's equation, which requires two measurements separated in time by many hours to correct the long-lived filter activity for radon/thoron contributions. Koval's calculation introduces unacceptable errors into the determination of long-lived, low-level alpha activity of a sample.

Prior to 1977, all gross alpha counting was accomplished at LASL with commercial systems which employed a single, thin-window, gas-flow, proportional counter and a 100-station, mechanical, sample changer to sequentially count many samples. The detector efficiency has been about 32 percent for ^{239}Pu alpha particles. The average detector efficiency for IMPULSE is 35 percent. The gas-flow proportional counter background has been approximately 0.2 cpm for a 10 minute analysis time. Work-load and


PERCENT ERROR IN THE NET COUNT RATE AS
A FUNCTION OF THE
NET COUNTS/min ABOVE THE LLD
PERCENT ERROR
30
25
20
15
10
5
0
0
10
20
30
40
50
60
70
80
90
100
NET CPM ABOVE THE LLD
10min ANALYSIS TIME
20min
30 min
60 min

statistical considerations have constrained us to 10-minute analysis times with the commercial systems. The average IMPULSE detector background is 0.07 cpm for a 10-minute analysis time. The LLD for the commercial systems for 10-minute analysis times has been about 0.025 MPC for soluble ^{239}Pu in air (MPC is 6.0×10^{-14} μCi/ml). For a 10-minute IMPULSE analysis time, the LLD is about 0.016 MPC.

The total μCi released from a stack is represented by:

$$\frac{(X-B)Y(4.50\times10^{-7})}{t\varepsilon S}\ \mu\text{Ci}$$

where,

X = sample counts in analysis time t

t = sample analysis time (minutes)

B = background counts in t minutes

Y = stack effluent rate in cfm

ε = detector fractional efficiency

S = sampling flow rate in cfm (cubic feet per minute)

Except for the sample counts, all of this information is in the software, ready to be used for the analysis of a batch of filters. Since the stack effluent rates are also stored on disk, it is an easy matter to generate the μCi released per week from a given stack. The error (1σ) associated with a determination of the total μCi released from a stack is represented by:

$$\frac{(X-B)Y(4.50\times10^{-7})}{t\varepsilon S}\left\{\frac{(X+\sigma_B{}^2)}{(X-B)^2}+0.0406\right\}^{\frac{1}{2}}\ \mu\text{Ci}$$

where, σ_B = 1σ standard deviation of mean background in counts per t minutes

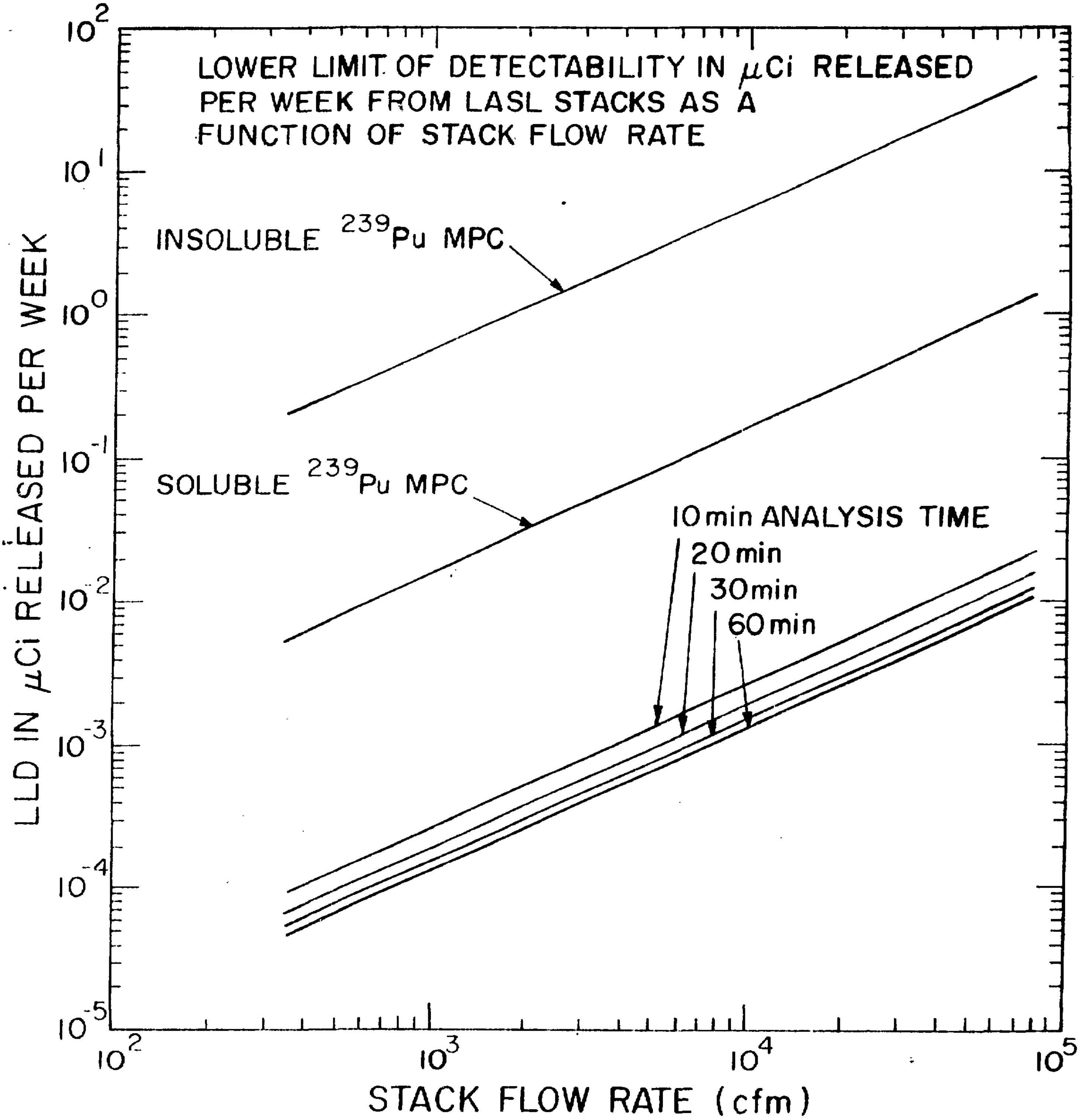
For these equations, the error (1σ) associated with the efficiency was assumed to be 5%. The error associated with the sampling rate was 12.5%

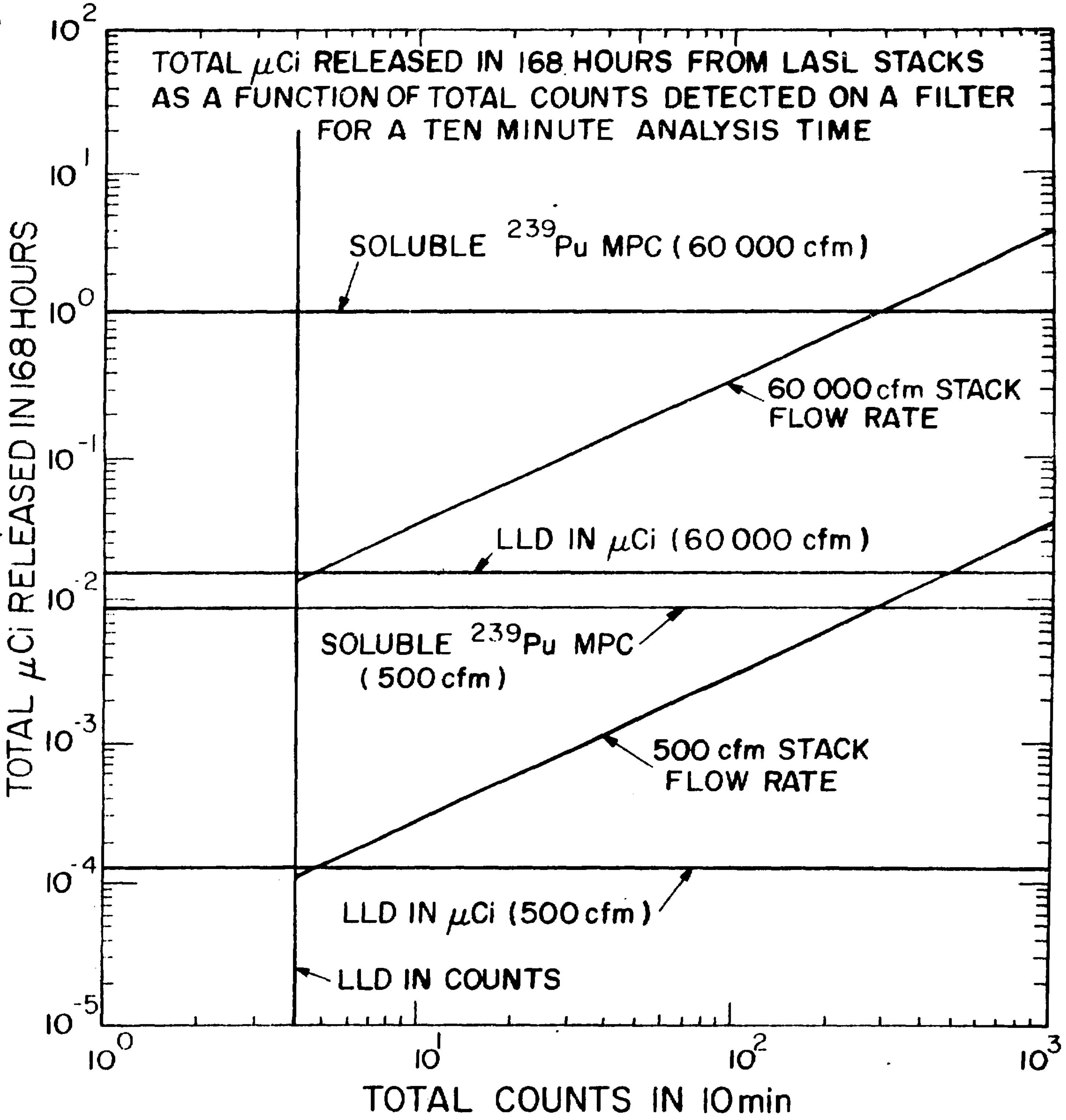
and that of the effluent rate was 15%, based on LASL experience. The errors associated with time are assumed to be insignificant.

In Figure 5, the LLD, expressed in terms of μCi released per week from a stack (the minimum detectable release), is plotted as a function of the stack flow rate in cfm. At LASL, the minimum stack flow rate is about 500 cfm. The maximum flow rate is about 60,000 cfm. A curve is shown for each of 10, 20, 30 and 60 minute sample analysis times. These curves represent the determination of airborne plutonium levels in HEPA filtered stack effluent air streams where the sampling rate is 2 cfm. A value of 37.6 ± 2.8 is used to represent the percent alpha radiation loss by the effect of the deposition of airborne contamination within the filter medium for HEPA filtered plutonium stacks. Also shown in Figure 5 are the curves representing the μCi equivalents of the soluble and insoluble non-occupational MPC's for ^{239}Pu expressed as a function of the stack flow rate. From this figure one can get an idea of what an increase in analysis time buys in terms of the minimum detectable release for LASL stacks.

In Figure 6, we show the total μCi released from a stack in 168 hours as a function of the total counts from a filter for a 10 minute analysis time. Curves are shown for stack flow rates of 500 cfm and 60,000 cfm. Also shown is the LLD in counts and the LLD expressed in μCi for the 500 cfm and 60,000 cfm stacks. Only the soluble MPC equivalents in μCi for ^{239}Pu are shown for the two stack flow rates.

After an analysis is conducted, data is sent to the mass storage device (RK05) and is printed out as hard copy from the terminal. This is true for samples that are analyzed in the individual or batch modes. Two types of hard copy are generated by the software. The software first


LOWER LIMIT OF DETECTABILITY IN μCi RELEASED PER WEEK FROM LASL STACKS AS A FUNCTION OF STACK FLOW RATE
INSOLUBLE ^{239}Pu MPC
SOLUBLE ^{239}Pu MPC
10 min ANALYSIS TIME
20 min
30 min
60 min
LLD IN μCi RELEASED PER WEEK
10^{2}
10^{1}
10^{0}
10^{-1}
10^{-2}
10^{-3}
10^{-4}
10^{-5}
10^{2}
10^{3}
10^{4}
10^{5}
STACK FLOW RATE (cfm)


TOTAL μCi RELEASED IN 168 HOURS FROM LASL STACKS AS A FUNCTION OF TOTAL COUNTS DETECTED ON A FILTER FOR A TEN MINUTE ANALYSIS TIME
SOLUBLE 239Pu MPC (60 000 cfm)
60 000 cfm STACK FLOW RATE
LLD IN μCi (60 000 cfm)
SOLUBLE 239Pu MPC (500 cfm)
500 cfm STACK FLOW RATE
LLD IN μCi (500 cfm)
LLD IN COUNTS
TOTAL μCi RELEASED IN 168 HOURS
10^2
10^1
10^0
10^-1
10^-2
10^-3
10^-4
10^-5
10^0
10^1
10^2
10^3
TOTAL COUNTS IN 10 min

generates a summary report for all samples. The software then generates a report for those samples which exceed certain warning levels which an operator can specify in the ALPHA program.

OCCUPATIONAL AIR SAMPLE EVALUATIONS

The analysis of occupational air sampling filters is important both for the safety of the worker and for the health physics control of operations which may release uranium or plutonium. At LASL, occupational air samples are removed from their sampling heads daily or weekly, depending on the particular area of concern. Filter sampling rates are either 1.5 or 2.0 cfm.

Filters are allowed to decay either over-night or over a weekend to allow a substantial amount of the radon/thoron to decay away. To date, samples have been counted for 10 minutes on the proportional-counter systems to determine all dpm/m^3 values. If air concentrations are normal and below the appropriate MPC values, no further analysis work is done. If the dmp/m^3 values measured are equal to, or greater than, the appropriate MPC values, the samples are counted 168 hours after removal from the sampling head in order to allow the radon/thoron to decay to insignificant levels. Interim analysis and evaluations are conducted at the specific request of the health physicist responsible for responding to these "above warning level" results.

The concentration of radioactivity in air is represented by:

$$\frac{(X-B)(5.89\text{x}10^{-1})}{t\varepsilon Sq}\ \frac{\text{dpm}}{\text{m}^3}$$

where all of the parameters have been defined previously except for q, which is the filter sampling time in hours.

The error (1σ) associated with this concentration is equal to:

$$\frac{(X-B)(5.89\times10^{-1})}{t\varepsilon Sq}\left\{\frac{(X+\sigma_B{}^2)}{(X-B)^2}+0.018\right\}^{1/2}\frac{\text{dpm}}{\text{m}^3}\ .$$

One has the same software capabilities for the occupational air filters as one does for the stacks. Summary reports similar to the stacks are generated for the occupational air samples at the request of the operator for either individually loaded samples or for air filter batches. An example of a summary report and a warning level report is shown in Table 1 for occupational air samples. Radioactive sources were used to generate the high percent MPC results.

DATA RETRIEVAL

All counted sample results are retrievable from the mass storage device (RK05) via operator command. With the software, an operator can generate a variety of outputs. First, an operator can selectively identify any individual stack or occupational air filter(s) and request a tabulation of the air concentrations of the activity for any period of time. Second, an operator can request a plot of the percent MPC as a function of time for any period of interest for any sample location. The plot is linear in time and logarithmic in the percent MPC. Decade changes result in printed character changes. Third, an operator can generate individual or batch reports for occupational air samples for any period of time. Time dependent data and summary data are provided. Finally, annual stack reports can be generated. Standard formats are stored on file and printed with summary data for the period of interest.

SUMMARY

IMPULSE was designed to be a cost effective system characterized by simplicity, reliability and enhanced performance over commercially available

TABLE 1

EXAMPLE SUMMARY AND WARNING LEVEL REPORTS FOR OCCUPATIONAL AIR SAMPLES

IMPULSE SUMMARY REPORT

Area: CMR-FHS   Type: Plutonium-Rooms   Sampling Period: 28-NOV-77@9 thru 5-DEC-77@16

Analysis Time: 5.0 min   Analysis Date: 5-DEC-77@16
Sampling Rate: 2.0 CFM   Sampling Time: 168. Hours

Filter No.	Filter Ident.	Primary Isotope	Gross Counts	LLD	Bkgd CPM	Bkgd +- Error	Effic	Spec. Activity DPM/M3	Spec. Activity +- Error	%MPC	Page 1
1	S-012	PU-239	2152.	8	0.133	0.061	0.367	2.054	0.038	46.2	
2	2023	PU-239	8.	6	0.066	0.055	0.356	0.008	0.001	0.2	
3	2113	PU-239	37.	5	0.050	0.050	0.343	0.038	0.002	0.8	
4	2119	PU-239	0.	7	0.105	0.104	0.371				
5	2123	PU-239	1.	6	0.077	0.083	0.354				
6	2124	PU-239	3958.	5	0.050	0.050	0.366	3.788	0.070	85.3	
7	2137	PU-239	28.	7	0.094	0.072	0.370	0.026	0.001	0.6	
8	3023	PU-239	3.	6	0.066	0.050	0.382				
9	3110	PU-239	9.	5	0.044	0.052	0.365	0.008	0.001	0.2	
10	3111	PU-239	9.	4	0.027	0.026	0.361	0.009	0.001	0.2	
11	3115	PU-239	2.	5	0.055	0.046	0.372				
12	3118	PU-239	24.	6	0.077	0.056	0.341	0.024	0.001	0.5	
13	3124	PU-239	3817.	8	0.122	0.056	0.357	3.745	0.069	84.3	
14	3127	PU-239	4018.	6	0.083	0.049	0.367	3.835	0.070	86.4	
15	3130	PU-239	1.	5	0.055	0.039	0.361				
16	3131	PU-239	4144.	5	0.044	0.052	0.378	3.841	0.070	86.5	
17	3136	PU-239	12.	5	0.055	0.046	0.348	0.012	0.001	0.3	
18	4023	PU-239	2.	5	0.055	0.088	0.359				
19	4063	PU-239	4075.	6	0.072	0.071	0.356	4.010	0.074	90.3	
20	4064	PU-239	10.	6	0.072	0.036	0.367	0.009	0.001	0.2	
21	4069	PU-239	4306.	5	0.050	0.050	0.359	4.202	0.077	94.6	
22	4110	PU-239	4542.	6	0.083	0.055	0.379	4.198	0.077	94.5	
23	4112	PU-239	3.	6	0.072	0.071	0.380				
24	4113	PU-239	6.	8	0.111	0.096	0.381				
25	4118	PU-239	3.	4	0.027	0.026	0.352				
26	4122	PU-239	4619.	6	0.066	0.061	0.359	4.507	0.083	101.5	Warning
27	4124	PU-239	3.	4	0.033	0.043	0.350				
28	5023	PU-239	14.	7	0.105	0.095	0.394	0.012	0.001	0.3	
29	7016	PU-239	2.	6	0.072	0.075	0.370				
30	7023-1	PU-239	47.	5	0.044	0.039	0.362	0.045	0.002	1.0	
31	7123-2	PU-239	10.	6	0.072	0.071	0.366	0.009	0.001	0.2	
32	7034-A	PU-239	4.	5	0.044	0.039	0.372				
33	7051	PU-239	2.	4	0.033	0.049	0.362				
34	7055	PU-239	0.	9	0.150	0.079	0.375				
35	7095-B	PU-239	5.	6	0.077	0.071	0.374				
36	7112	PU-239	2.	9	0.161	0.082	0.375				
37	7118	PU-239	1.	6	0.077	0.066	0.359				
38	7120	PU-239	1.	9	0.161	0.119	0.373				
39	7122	PU-239	3.	6	0.065	0.030	0.362				
40	7124	PU-239	3.	5	0.044	0.039	0.373				

TABLE 1A

IMPULSE WARNING LEVEL REPORT

Area: CMR-FHS   Type: Plutonium-Rooms   Sampling Period: 28-NOV-77@9 thru 5-DEC-77@16

Analysis Time: 5.0 min   Analysis Date: 5-DEC-77@16
Sampling Rate: 2.0 CFM   Sampling Time: 168. Hours

Filter		Primary	Spec. Activity		
No.	Ident.	Isotope	DPM/M3	+- Error	%MPC Page 1
26	4122	PU-239	4.507	0.083	101.5 Warning

TABLE 1

EXAMPLE SUMMARY AND WARNING LEVEL REPORTS
FOR OCCUPATIONAL AIR SAMPLES

IMPULSE SUMMARY REPORT

Area: CMR-FHS Type: Plutonium-Rooms Sampling Period: 28-NOV-77@9 thru 5-DEC-77@16

Analysis Time: 5.0 min Analysis Date: 5-DEC-77@16
Sampling Rate: 2.0 CFM Sampling Time: 168. Hours

Filter		Primary	Gross		Bkgd			Spec. Activity		
No.	Ident.	Isotope	Counts	LLD	CPM	+- Error	Effic	DPM/M3	+- Error	%MPC Page 1
1	S-012	PU-239	2152.	8	0.133	0.061	0.367	2.054	0.038	46.2
2	2023	PU-239	8.	6	0.066	0.055	0.356	0.008	0.001	0.2
3	2113	PU-239	37.	5	0.050	0.050	0.343	0.038	0.002	0.8
4	2119	PU-239	0.	7	0.105	0.104	0.371			
5	2123	PU-239	1.	6	0.077	0.083	0.354			
6	2124	PU-239	3958.	5	0.050	0.050	0.366	3.788	0.070	85.3
7	2137	PU-239	28.	7	0.094	0.072	0.370	0.026	0.001	0.6
8	3023	PU-239	3.	6	0.066	0.050	0.382			
9	3110	PU-239	9.	5	0.044	0.052	0.365	0.008	0.001	0.2
10	3111	PU-239	9.	4	0.027	0.026	0.361	0.009	0.001	0.2
11	3115	PU-239	2.	5	0.055	0.046	0.372			
12	3118	PU-239	24.	6	0.077	0.056	0.341	0.024	0.001	0.5
13	3124	PU-239	3817.	8	0.122	0.056	0.357	3.745	0.069	84.3
14	3127	PU-239	4018.	6	0.083	0.049	0.367	3.835	0.070	86.4
15	3130	PU-239	1.	5	0.055	0.039	0.361			
16	3131	PU-239	4144.	5	0.044	0.052	0.378	3.841	0.070	86.5
17	3136	PU-239	12.	5	0.055	0.046	0.348	0.012	0.001	0.3
18	4023	PU-239	2.	5	0.055	0.088	0.359			
19	4063	PU-239	4075.	6	0.072	0.071	0.356	4.010	0.074	90.3
20	4064	PU-239	10.	6	0.072	0.036	0.367	0.009	0.001	0.2
21	4069	PU-239	4306.	5	0.050	0.050	0.359	4.202	0.077	94.6
22	4110	PU-239	4542.	6	0.083	0.055	0.379	4.198	0.077	94.5
23	4112	PU-239	3.	6	0.072	0.071	0.380			
24	4113	PU-239	6.	8	0.111	0.096	0.381			
25	4118	PU-239	3.	4	0.027	0.026	0.352			
26	4122	PU-239	4619	6	0.066	0.061	0.359	4.507	0.083	101.5 Warning
27	4124	PU-239	3.	4	0.033	0.043	0.350			
28	5023	PU-239	14.	7	0.105	0.095	0.394	0.012	0.001	0.3
29	7016	PU-239	2.	6	0.072	0.075	0.370			
30	7023-1	PU-239	47.	5	0.044	0.039	0.362	0.045	0.002	1.0
31	7123-2	PU-239	10.	6	0.072	0.071	0.366	0.009	0.001	0.2
32	7034-A	PU-239	4.	5	0.044	0.039	0.372			
33	7051	PU-239	2.	4	0.033	0.049	0.362			
34	7055	PU-239	0.	9	0.150	0.079	0.375			
35	7095-B	PU-239	5.	6	0.077	0.071	0.374			
36	7112	PU-239	2.	9	0.161	0.082	0.375			
37	7118	PU-239	1.	6	0.077	0.066	0.359			
38	7120	PU-239	1.	9	0.161	0.119	0.373			
39	7122	PU-239	3.	6	0.065	0.030	0.362			
40	7124	PU-239	3.	5	0.044	0.039	0.373			

IMPULSE WARNING LEVEL REPORT

Area: CMR-FHS Type: Plutonium-Rooms Sampling Period: 28-NOV-77@9 thru 5-DEC-77@16

Analysis Time: 5.0 min Analysis Date: 5-DEC-77@16
Sampling Rate: 2.0 CFM Sampling Time: 168. Hours

Filter		Primary	Spec. Activity		
No.	Ident.	Isotope	DPM/M3	+- Error	%MPC Page 1
26	4122	PU-239	4.507	0.083	101.5 Warning

systems. The hardware is very simple in concept. Although the software is powerful, the system is easy to operate and to use productively in support of operational health physics requirements. System reliability is improved by component redundancy. Long-term calibration studies have shown that the system is very stable. The system is flexible, easy to use, and easy to maintain. Finally, the use of minicomputers for control inherently provides the data in a suitable format for computer processing and provides the capability for processing the data and preparing required reports most expediently.

ACKNOWLEDGEMENTS

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THE MONITORING SYSTEM OF THE TRITIUM RESEARCH LABORATORY
SANDIA LABORATORIES, LIVERMORE, CALIFORNIA

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ABSTRACT

Computerized tritium monitoring is now in use at the Tritium Research Laboratory (TRL). Betatec 100 tritium monitors, along with several Sandia designed accessories, have been combined with a PDP 11/40 computer to provide maximum personnel and environmental protection.

Each individual monitoring system, in addition to a local display in the area of interest, has a visual/audible display in the control room. Each system is then channeled into the PDP 11/40 computer, providing immediate assessment of the status of the entire laboratory from a central location.

Measurement capability ranges from $\mu Ci/m^3$ levels for room air monitoring to KCi/m^3 levels for glove box and process system monitoring.

The overall monitoring system and its capabilities will be presented.

INTRODUCTION

The Tritium Research Laboratory (TRL) is used for experimental work with isotopes of hydrogen and their compounds. It was designed to conform to the Department of Energy philosophy, which is, to limit the release of any radioactive material to a level "as low as reasonably achievable". In order to effectively meet this goal, the philosophy of operation at the TRL is "containment and cleanup".

To contain any accidental release, experimental work with tritium takes place in sealed glove boxes. These glove boxes are connected to two cleanup systems: the Gas Purification System (GPS), and the Vacuum Effluent Recovery System (VERS).

The monitoring system of the TRL was designed to insure the safe and effective use of these systems, and to provide personnel and environmental protection. The system includes a monitoring capability from KCi/m^3 levels for glove box and cleanup system monitoring, to $\mu Ci/m^3$ levels for room air and stack effluent monitoring. Each tritium monitor is interfaced to the TRL control computer, which continuously scans each of the tritium monitors, assesses the hazards based upon the monitor readings, and takes the appropriate action according to the hazard involved.

The system operates automatically. Action taken by the computer is determined by the location of the monitor, and the monitoring system involved (Figure 1). The basic monitoring systems are: (1) Room Air Monitoring, (2) Glove Box Monitoring, (3) GPS Monitoring, (4) VERS Monitoring, and (5) Stack Effluent Monitoring.

MONITORING SYSTEM GENERAL DESCRIPTION

Room Air Monitoring

In order to protect personnel working in the TRL, the room air of each room is continuously monitored for tritium. In a work environment, tritium monitors must be capable of measuring the maximum permissible concentration (MPC) of tritium in its two predominate forms. For a 40-hour week, the MPC for tritium gas is 2 mCi/m^3. The MPC for tritiated water vapor, for a 40-hour week, is 5 $\mu Ci/m^3$. Room air tritium monitors are capable of reading from 1 $\mu Ci/m^3$ to 20 Ci/m^3.

Each room has been provided with a selectable, multi-inlet system which helps reduce the need for portable monitors. The diameter of the inlet lines has been kept small, (nominally 1/4"), to reduce the overall volume of the lines. In addition, these lines are kept as short as possible, and the entire detection system has been mounted in the room of interest, in order to reduce the response time of the monitor.

Each room has a readout device (a visual/audible display) mounted, in the room, and at the entrance to the room, providing the user with the current status of their laboratory.

Each room air monitor, as well as all other tritium monitors, has a visual/audible display located in the control room, which provides the TRL operations staff with the capability of immediate assessment of the status of the entire laboratory.

Glove Box Monitoring

As previously stated, all experimental work with tritium takes place in sealed glove boxes. These glove boxes are 4-feet wide, 8-feet high and 8-feet long. Each has been equipped with viewing windows, glove ports, access doors, a pass-through port, heat exchanger, vacuum pumps, pressure controls, a fire alarm, and a tritium monitor.

The atmosphere of each glove box is continuously monitored, and in the event the tritium concentration exceeds a pre-determined limit, valves connecting the glove box to the GPS will be opened and the glove box atmosphere will be processed by the GPS.

The range of the tritium monitor used for glove box monitoring was selected based upon: the potential for gram amounts of tritium being involved in a given experiment, and the allowable leak rates of experimental equipment within the box. Glove box tritium monitors are capable of reading from 1 mCi/m^3 to 20 KCi/m^3.

Gas Purification System Monitoring

The GPS is capable of reducing the tritium concentration in a glove box to less than the part per billion level--less than 2.5 mCi/m^3. As the atmosphere of the glove box is circulated through the GPS, the tritium is oxidized on a catalyst bed. This tritiated water is then collected on a series of molecular sieve dryers.

Range selection of the monitors used on the GPS was based on tritium concentrations which may be seen at different points throughout the system.

The influent comes from a glove box; therefore, the GPS inlet monitor is identical to a glove box monitor, with a range from 1 mCi/m^3 to 20 KCi/m^3. This monitor serves a dual purpose: first, it provides an input to a totalizer which records the total amount of tritium processed by the GPS; second, it serves as a cross-check of the appropriate glove box monitor.

There are four other monitors associated with the GPS. These are: the dryer 2 and 3, and the GPS exhaust monitors with a range from 1 μCi/m^3 to 20 Ci/m^3; the dryer 1 and 2 monitor, with a range from 10 μCi/m^3 to 200 Ci/m^3; and the dryer regeneration system monitor, with a range from 1 mCi/m^3 to 20 KCi/m^3.

Vacuum Effluent Recovery System Monitoring

The VERS collects the effluent from all vacuum systems in the TRL. These gases are collected in a monitoring tank, and an analysis is performed. Activity below the cleanup capability is exhausted to the stack. Activity above the cleanup capability is processed through the VERS, which removes tritium by the same process, and to the same level, as the GPS.

There are a total of four tritium monitors associated with the VERS: the 10 ft^3 hold tank, the non-contaminated hold tank, and the VERS exhaust monitors each have a range from 1 μCi/m^3 to 20 Ci/m^3. The contaminated hold tank monitor has a range from 10 μCi/m^3 to 200 Ci/m^3.

Stack Effluent Monitoring

The tritium concentration in the stack effluent is monitored using two monitors with overlapping ranges. Each of these monitors provide an input to a totalizer which records the total amount of tritium exhausted up the stack. The range of the low level stack effluent monitor is from 1 $\mu Ci/m^3$ to 20 Ci/m^3. The range of the high level stack effluent monitor is from 100 $\mu Ci/m^3$ to 2 KCi/m^3.

COMPONENT DESCRIPTION

Before an appreciable understanding of the monitoring system can be realized, a description of each of its major components is necessary. These have been divided into two categories: electronic, and mechanical components. It should be emphasized that all of the electronic and the majority of the mechanical components are interchangeable, requiring only minor modifications. Along with each component, the appropriate modifications will be described.

Electronic Components

Basic Tritium Monitor

The tritium monitors are Overhoff and Associates, Betatec 100 tritium monitors, design and built to Sandia specifications. Each is an auto-ranging instrument, capable of reading from 1 to 1000 units, on each of five ranges, (overall range: 1 to 1.999×10^7 units per instrument). Each monitor consists of an ion chamber with a vibrating reed electrometer, and a readout device--the meter module (Figure 2).

Ion Chamber

The ion chamber provides a known volume in which to measure the tritium concentration. The three ion chamber configurations in use in the TRL are

(Figure 3):

1) Dual two litre ion chambers - Low level monitoring, including room air and low level stack effluent. Measurable concentration: 1 $\mu Ci/m^3$ to 20 Ci/m^3.

2) Single two litre ion chambers - Intermediate level monitoring at two levels, including high level stack effluent, and portions of the GPS and VERS. Measurable concentrations: 10 $\mu Ci/m^3$ to 200 Ci/m^3, and 100 $\mu Ci/m^3$ to 2 KCi/m^3.

3) Dual 20 cc ion chambers - High level monitoring, including portions of the GPS and all glove boxes. Measurable concentration: 1 mCi/m^3 to 20 KCi/m^3.

All ion chambers, regardless of configuration, must meet a leak rate specification of 10^{-6} std cc/sec. This allows any ion chamber to be used on any tritium monitoring system in the TRL.

Electrometers

A vibrating reed electrometer is mounted on each ion chamber. The electrometer provides a usable voltage to the meter module, as a function of the tritium concentration in the ion chamber. (The current required in a dual two litre ion chamber for a reading of 1 $\mu Ci/m^3$ is equal to 2×10^{-15} amperes.) All electrometers are interchangeable, except for their feedback resistors. These resistors provide the desired range selection, and are matched to the ion chamber configuration.

Meter Modules

The information from the ion chamber and electrometer is supplied to the meter module, (Figure 4). The meter module provides visual display information, electrical information for additional accessories, and power supply voltages for the ion chamber, electrometer, and some accessories. The meter modules are interchangeable electrically, but the units/multiplier information must be changed to provide the desired range indication for each monitor. This is done by changing the stick-on decals to the left and right of the range indicator lamps.

The front panel of the meter module is divided into two sections: the display section, and the alarm section. The display section provides a direct readout of the tritium concentration in the appropriate units. The units/multiplier information, located to the left and right of the illuminated range lamp, is applied directly to the reading displayed on the digital panel meter.

The alarm section provides each tritium monitor with a low and high alarm. These alarms can be set at any level throughout the entire range of the instrument, and are completely independent of each other.

Remote Displays

The remote display (Figure 5) is a Sandia-designed accessory to the basic tritium monitor, and is used only with the room air tritium monitoring system. It is normally located at the entrance to the room of interest, providing the user of the room the capability of assessing potential hazards, prior to entering their laboratory. The remote display contains the necessary visual and audible components of the meter module, each in the same relative position of its counterpart.

Console Displays

The console display (Figure 6) is another Sandia-designed accessory to the tritium monitor, and is the method of visual/functional display in the control room. In addition, it is the primary method of interfacing between the basic tritium monitor and the TRL control computer.

Like the remote display, the console display also contains the necessary visual and audible components in the same relative position as its counterpart on the meter module. Unlike the remote displays, console displays are used on most of the tritium monitoring systems in the TRL. Interchangeability requires only the changing of the stick-on decals for the appropriate units/multiplier information, and the appropriate switch module configuration.

A five unit switch module is located in the lower, center portion of the console display front panel. For room air monitors, four switches control the four solenoid valves on the room air sample manifold (see Mechanical Components). The fifth switch controls the use of a backup room air tritium monitor, located on the room air calibration manifold.

For glove box monitors, only one switch is used. This switch controls the use of a backup glove box monitor, located on the glove box calibration manifold.

GPS and VERS console displays have no switches installed. A blank switch plate is installed in all unused portions of the switch modules.

Totalizers

There are two totalizer units used in the monitoring system of the TRL. These Sandia-designed accessories work as a system (Figure 7), to record the total amount of tritium: 1) released in the stack effluent, and 2) processed by the GPS. A digital clock, a scanner, and a printer complete this totalizer system.

The average flow rate of the appropriate monitoring system, and the integration interval are dialed into each totalized by thumbwheel switches.

Tritium totals on each totalizer are provided by the integrator and accumulator readouts. The former is the total tritium released or processed during the current integration interval. The latter is the total tritium released or processed since the system was last reset to zero.

At the end of each integration interval, each readout and the time of day is scanned, and these readings are printed.

Each totalizer has been provided with a low and high level alarm, which will be activated if the tritium totals exceed the respective pre-set level during the integration interval.

Hallway Manifold System Flush Control

The hallway manifold system (HMS) flush control unit (Figure 8), is a Sandia-designed accessory to the room air and stack effluent monitoring systems. It is located in the control room and provides the capability of purging the room air exhaust, and room air calibration manifolds (see Mechanical Components) with compressed air or nitrogen. In addition, the HMS flush control unit provides an alarm input to the guard alarm interface unit in the event of low air flow in either the room air exhaust, room air calibration, or stack effluent manifolds.

Glove Box Flush Control

The glove box flush valve control unit (Figure 8) is a Sandia designed accessory to the glove box monitoring system and is located in the control room. Its function is to provide: 1) automatic purging of the glove box air sample lines and calibration manifold whenever the glove box calibration monitor is not sampling a glove box; 2) an alarm input to the guard alarm interface in the event of low air flow in the glove box calibration manifold; and 3) automatic re-routing of the glove box calibration monitor sample to the VERS in the event the tritium concentration exceeds a pre-determined level.

There are two additional electronic components of the overall monitoring system: the TRL control computer, and the guard alarm interface. Each will be discussed under the computer section of this report.

Mechanical Components

Central Pumping Station

Air samples are drawn into the various manifolds by the central pumping station (Figure 9). Half of the central pumping station consists of dual sample pumps which draw samples through the room air exhaust, the room air calibration, and the stack effluent manifolds. There is a shut-off valve, a flowmeter, and a flow switch in each of these lines.

portions of the HMS flush control unit. The sample pumps have a rated capacity of 4.5 SCFM each, and are exhausted to the stack.

The other half of the central pumping station consists of a dual sample pumps, which draw their samples into the glove box calibration manifold. This manifold is also equipped with shut-off valves, a flowmeter, and a flow switch. This flow switch provides the low flow alarm input to the glove box flush valve control unit. The sample pumps have a secondarily contained head, and have a rated capacity of 0.4 SCFM. They are normally exhausted to the stack. Solenoid valves in the exhaust line controlled by the glove box flush valve control unit divert the exhaust into the VERS whenever the tritium concentration exceeds a pre-set limit.

Room Air Sample Manifold

Each room in the TRL is equipped with a room air sample manifold. This sample manifold is a selectable, four-way inlet system for the ion chamber. At the inlet of each line, there is a hose barb, and an in-line particulate filter. At each inlet to the sample manifold, there is a normally open solenoid valve. These valves can be selectively closed by activating the switches on the appropriate console display, located in the control room.

Under normal conditions, all four valves are open, allowing the monitor to measure a more representative sample of the room air tritium concentration. During potentially hazardous conditions, valves may be closed to select a portion of the room to be monitored. Should even more localized monitoring be desired, a piece of flexible, plastic tubing attached to a hose barb, with the other three valves closed, will allow the room air tritium monitor to be used as a "sniffer".

Room Air Exhaust and Calibration Manifolds

Room air is drawn into the sample manifold and through the ion chamber. The exhaust from the ion chamber is drawn through the hallway manifold selector (a three-way solenoid valve), which is normally open to the room air exhaust manifold. The air samples from all room air tritium monitors are drawn through this manifold, by the central pumping station.

Changing the hallway manifold selector, which is accomplished by operating one of the switches in the switch module on the appropriate console display, re-routes the sample into the room air calibration manifold. The room air calibration tritium monitor is mounted on this manifold, allowing a cross-check of each room air tritium monitor reading against a single unit. It also provides a method of backing up any of the room air monitors, whenever maintenance on a monitor is required. Samples are drawn into this manifold by the central pumping station.

Glove Box Calibration Manifold

Provisions have been made to draw a sample from a glove box and measure the tritium concentration in another backup monitor. At the glove box, there is a normally closed solenoid valve in an additional sample line. This valve can be opened by activating the switch on the appropriate glove box console display. The sample is then drawn from the glove box into the glove box calibration manifold and tritium monitor by the central pumping station.

All three of the above manifolds are located in the main hallway and run the length of the TRL. Due to their length and their potential for the buildup of tritium contamination, provisions have been made to flush these lines. Solenoid valves have been installed in the room air exhaust and calibration manifolds. These valves are controlled by the HMS flush control unit, located in the control room. By activating the appropriate switches, the room air exhaust or the room air calibration manifold will be continuously flushed with nitrogen or compressed air.

A similar system has been installed for the glove box calibration manifold. Solenoid valves in this line, however, are activated by the glove box flush valve control unit, also located in the control room.

Stack Effluent Sample Manifold

The stack effluent tritium monitors draw their sample from a sample array, (Figure 10), located in the TRL building exhaust duct. Since the sample point is upstream of the building exhaust fans, this sample array was designed to provide a representative sample of the TRL exhaust air by using a multi-point sample system, located just downstream of a laminar flow unit. The sample is drawn directly from the exhaust duct by the central pumping station, and is returned to the exhaust duct downstream of the sample array.

Pump Modules

For glove box, GPS, and VERS monitoring, the sample is drawn by a pump module, (Figure 11). A pump module consists of an ion chamber, pump, flowmeter, inlet and outlet shut-off valves, and interconnecting tubing. Any ion chamber can be mounted in a pump module, however, the interconnecting tubing may have to be re-routed, due to differences in the two litre and dual 20 cc ion chambers.

The pump itself, which is identical to the pumps used on the glove box portion of the central pumping station, has a secondarily-contained head. This is to minimize the potential for a tritium release into a room in the event of a pump head failure.

Pump modules are mounted as close to the desired sample point as possible. For glove box monitoring, the pump draws a sample through a hose barg and in-line particulate filter, located just inside the glove box. Flow is then through the pump module and returned to the glove box. For GPS and VERS monitoring, the pump draws a sample directly from the line of interest, and returns it to the line downstream of the sample point.

MONITORING SYSTEM DESCRIPTION

Each of the above components has been combined in various configurations to form the individual monitoring systems. When the appropriate individual monitoring systems are combined, they make up one of the basic monitoring systems. The following are the basic monitoring systems and their sub-systems, with a component listing for each.

Room Air Monitoring System

A typical room air monitoring system (Figure 12) consists of: 1) a room air sampler, a dual two litre ion chamber with electrometer, and a meter module, located in the room of interest; 2) a remote display and a hallway manifold selector, located just outside the room of interest; and 3) a console display with the appropriate switch module configuration, located in the control room. The console display provides the input to the computer.

In addition, the room air monitoring system includes the room air exhaust manifold, the central pumping station, and the room air portion of the hallway manifold system flush control unit. The room air calibration monitoring sub-system consists of: 1) the room air calibration manifold, and central pumping station; 2) a dual two litre ion chamber with electrometer, located on the room air calibration manifold; and 3) the meter module and room air calibration manifold portion of the hallway manifold system flush valve controller, both located in the control room. The computer input from this sub-system is provided directly from the meter module.

Glove Box Monitoring System

A typical glove box monitoring system (Figure 13) consists of: 1) a pump module with dual 20 cc ion chamber, electrometer, and a meter module, mounted in the glove box equipment housing; and 2) a console display, located in the control room. The console display provides the input to the computer.

The glove box calibration monitoring sub-system consists of: 1) the solenoid valve at the glove box of interest; 2) the glove box calibration manifold; 3) a single two litre ion chamber, located on the glove box calibration manifold; and 4) the meter module and the glove box flush valve control unit, located in the control room. The computer input is provided directly from the meter module.

GPS and VERS Monitoring System

A typical GPS or VERS monitoring system consists of: 1) a pump module with the appropriate ion chamber and electrometer located as close to the desired sample point as possible; 2) a meter module, located in an electronic equipment rack at the entrance to the cleanup systems equipment room; and 3) a console display, located in the control room. For GPS and VERS monitors, the console display provides the input to the computer.

The GPS inlet tritium monitor, provides an additional input to the totalizer system.

Stack Effluent Monitoring System

The stack effluent monitoring system (Figure 14) consists of: 1) the multi-point sample system, located inside the ventilation exhaust duct; 2) a dual two litre and a single two litre ion chamber with electrometer, located on the stack effluent sample manifold; 3) the central pumping station; and 4) meter modules for both monitors, located in the control room.

Computer inputs are provided directly from each meter module. Both meter modules provide an addtional input into the totalizer system.

TRL COMPUTER SYSTEM

TRL Control Computer

The control computer is a Digital Equipment Corporation PDP 11/40 computer. The computer and its peripherals, which include two disc drives, a keyboard printer, a video terminal, and an external real-time clock, are located in the control room.

The computer continuously scans each of the tritium monitors and VERS pressure readings, and records the data. It also scans each of the glove boxes for pressure alarms, fire alarms and the correct GPS inlet and outlet valve positions. Whenever a high glove box tritium level exists, the computer starts the GPS, selects the correct clean-up gases for the glove box, selects the appropriate GPS dryers, connects the glove box to the GPS, and runs the GPS until the tritium level is sufficiently reduced.

Whenever the computer finds an alarm condition, it prints a message on the keyboard printer indicating the alarm and its time of occurrence. It also sends an alarm to the guard alarm interface unit.

Guard Alarm Interface

The guard alarm interface unit (Figure 15) is a Sandia-designed accessory to the overall TRL monitoring system. Its function is to provide an alarm in the control room whenever an abnormal condition occurs in any of the several building systems. In addition, the interface will relay the alarm to Security Headquarters after normal working hours.

The interface is divided into seven alarm sections, the first four of which are presently active. In order of highest priority, they are: emergency, danger, hazard, and operational alarms. Each section has a series of lamps indicating the cause of the alarm. Additional lamps indicate the condition in which the interface is functioning.

When in the "send alarm" condition, all alarm occurrences will be sent to Security Headquarters. When in the "do not send alarm" condition, only the emergency alarms and one hazard alarm will be sent.

An alarm may be generated by the computer or any of several TRL systems such as the GPS, VERS, or portions of the tritium monitoring system. An alarm may also be generated by any of several building systems, such as the ventilation, chilled water, or electrical power distribution system. Whenever the alarm is generated, the interface unit will illuminate the appropriate lamp, and sound a local alarm in the control room.

If the alarm occurs during normal working hours, the control room operator will notify the appropriate personnel. If the alarm occurs after normal working hours, the guard alarm interface unit relays the alarm to Security Headquarters, and Security will notify the appropriate personnel.

ACKNOWLEDGEMENT

The authors wish to acknowledge the advise and comments made by W. R. Wall. He was responsible for the design and operation of the electrical systems at the TRL and his help on this paper has proved invaluable.

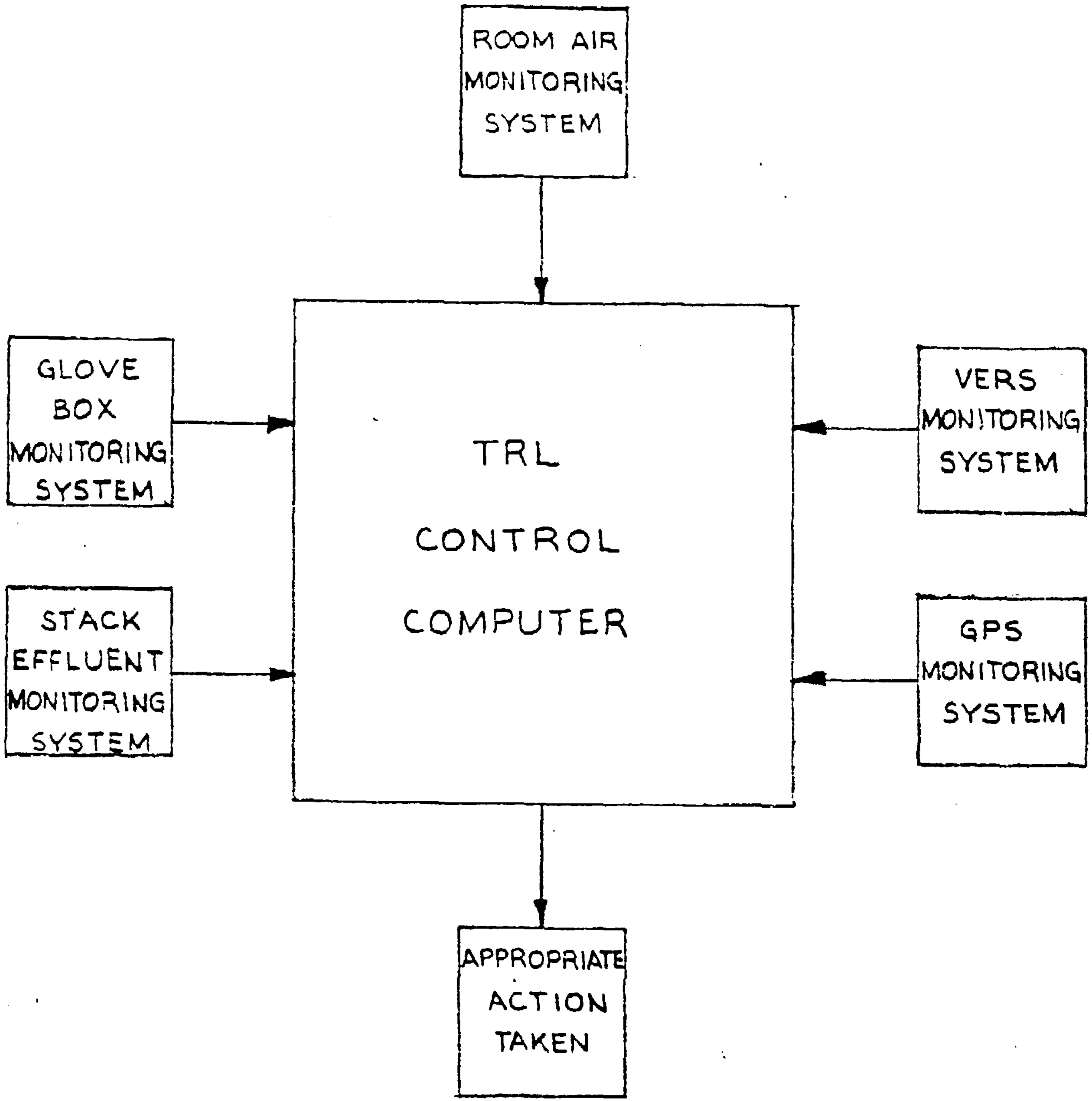


Figure 1: Tritium Monitoring System

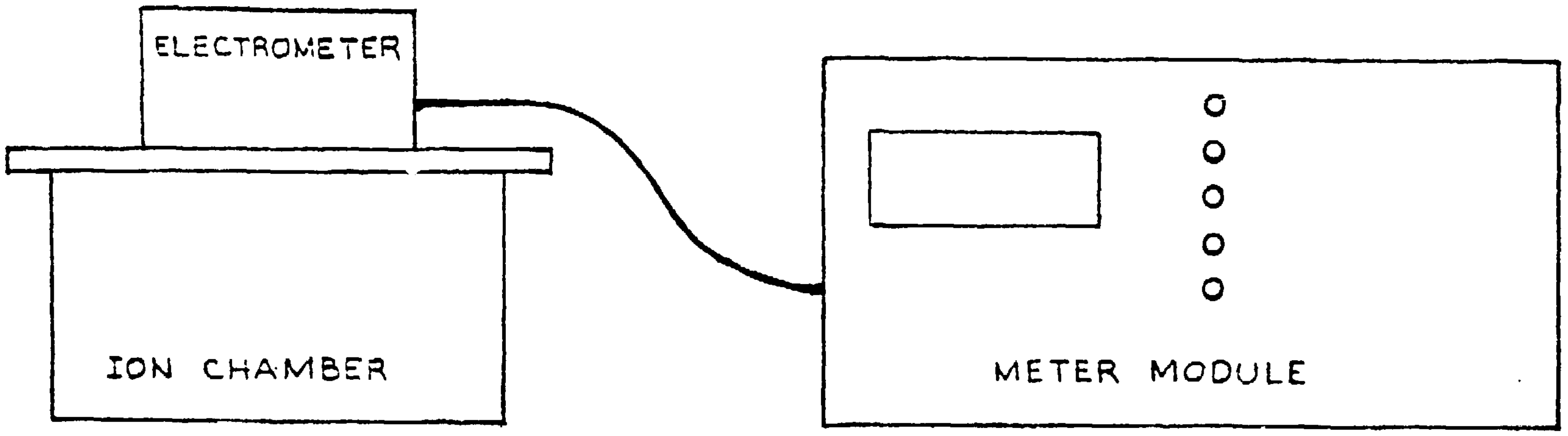


Figure 2: Basic Tritium Monitor

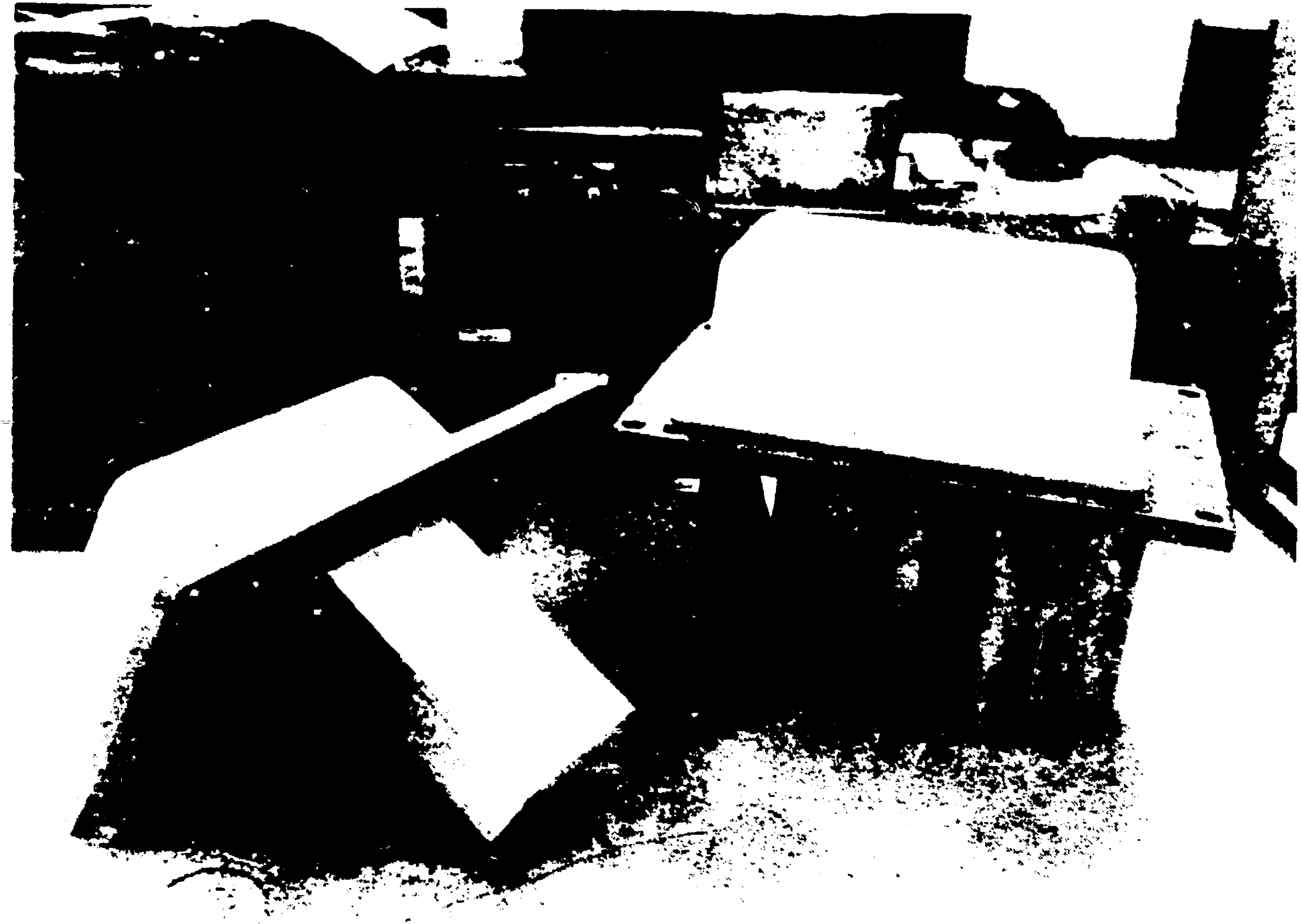


Fig. 3: Dual Two Litre and Dual 20 cc
Ion Chambers.

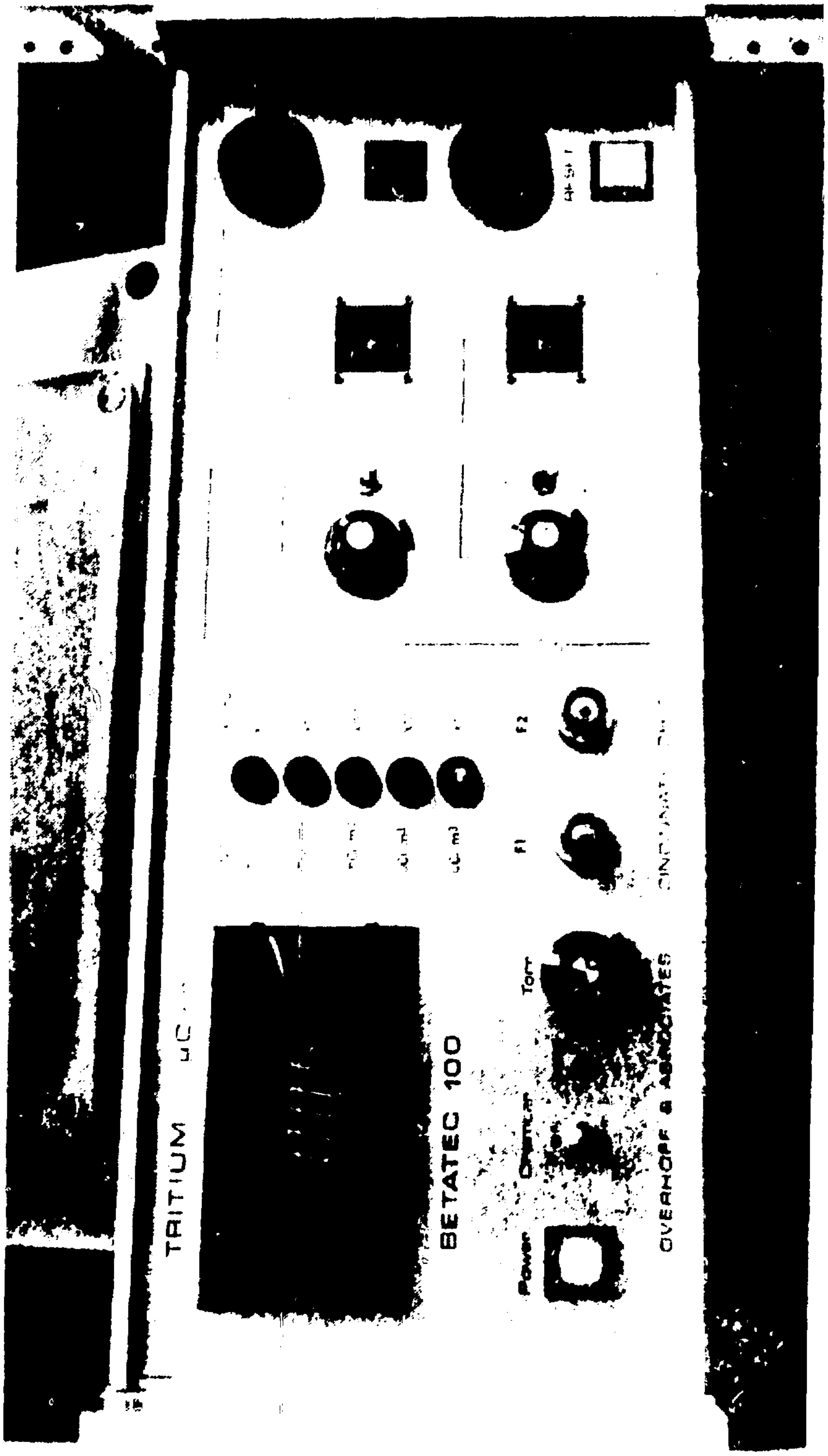


Fig. 4: Meter Module

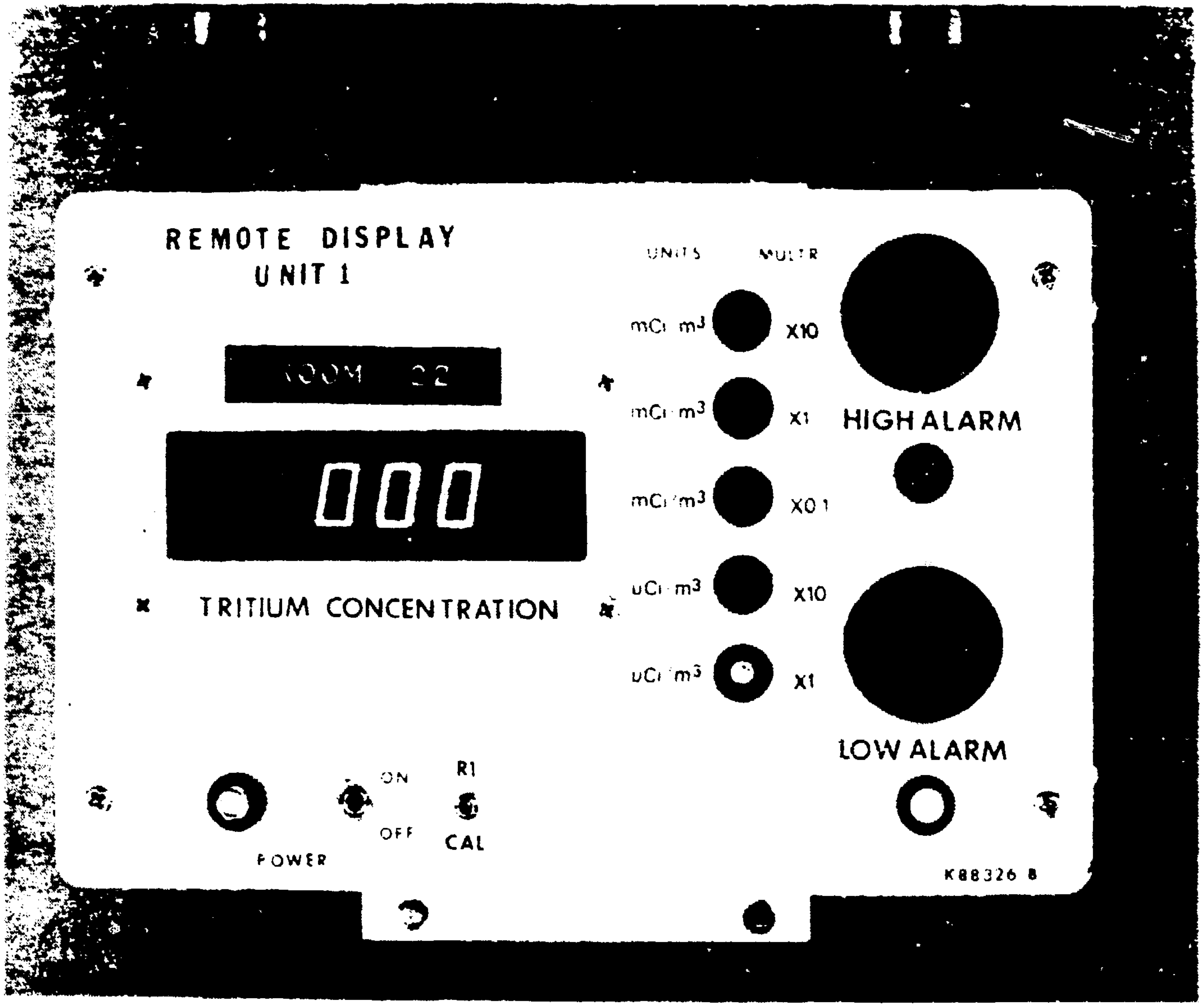


Fig. 5: Remote Display

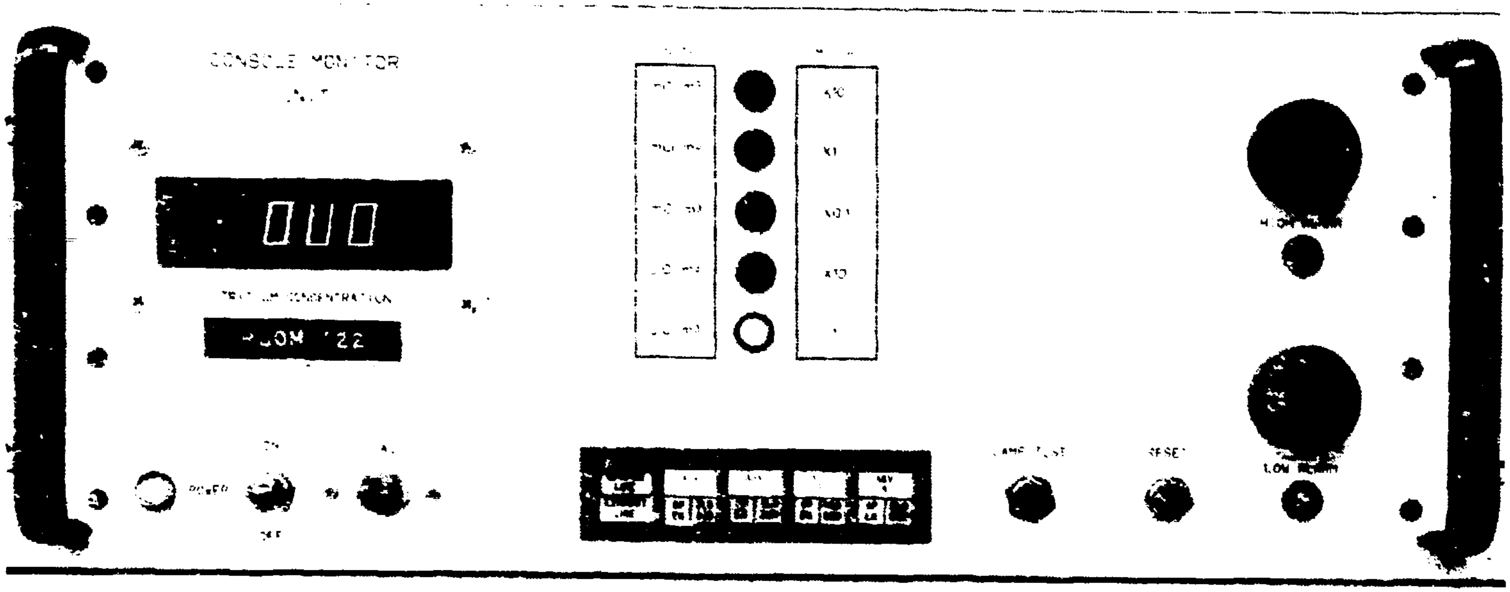


Fig. 6: Console Display

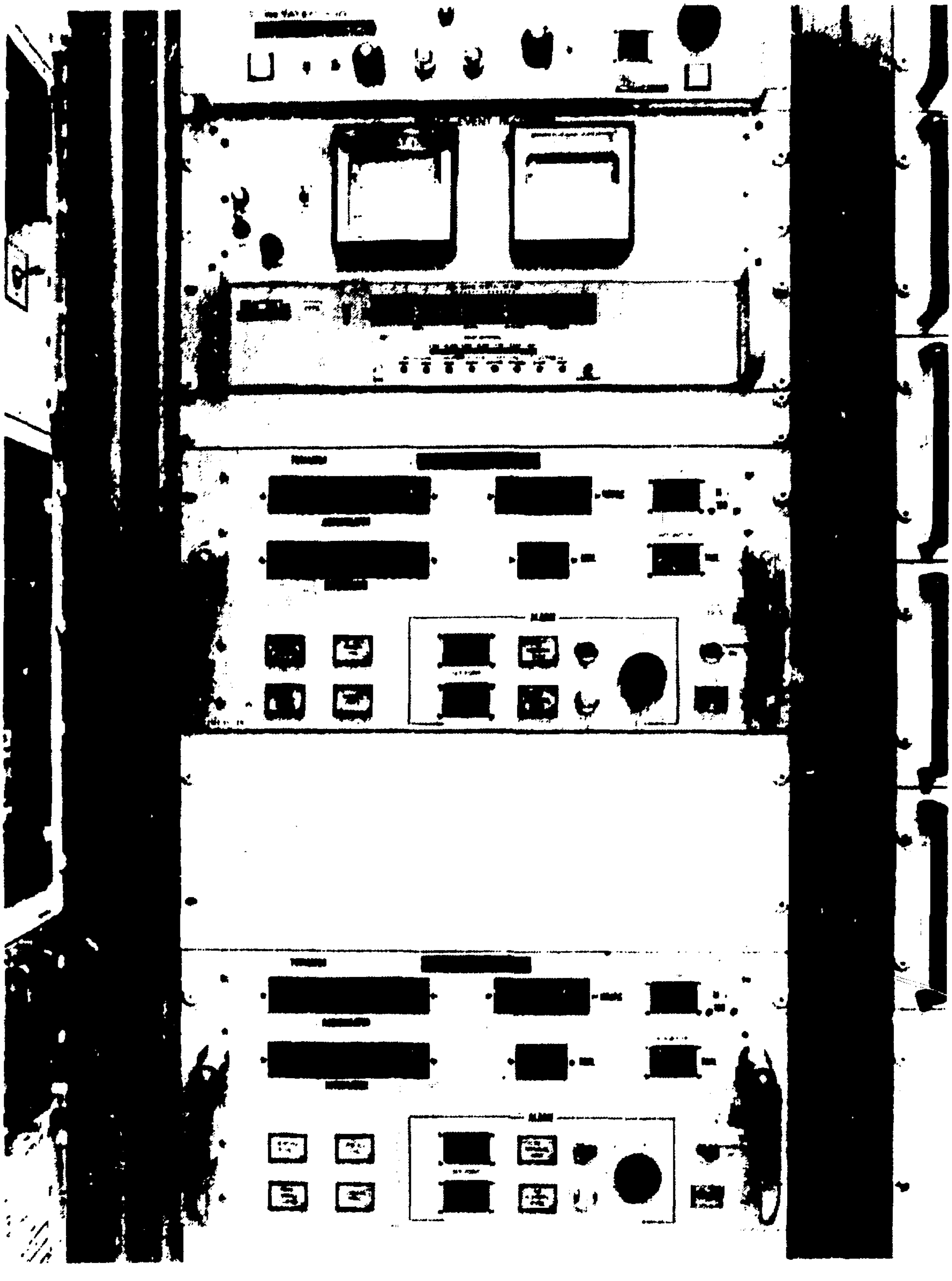


Fig. 7: Totalizer System

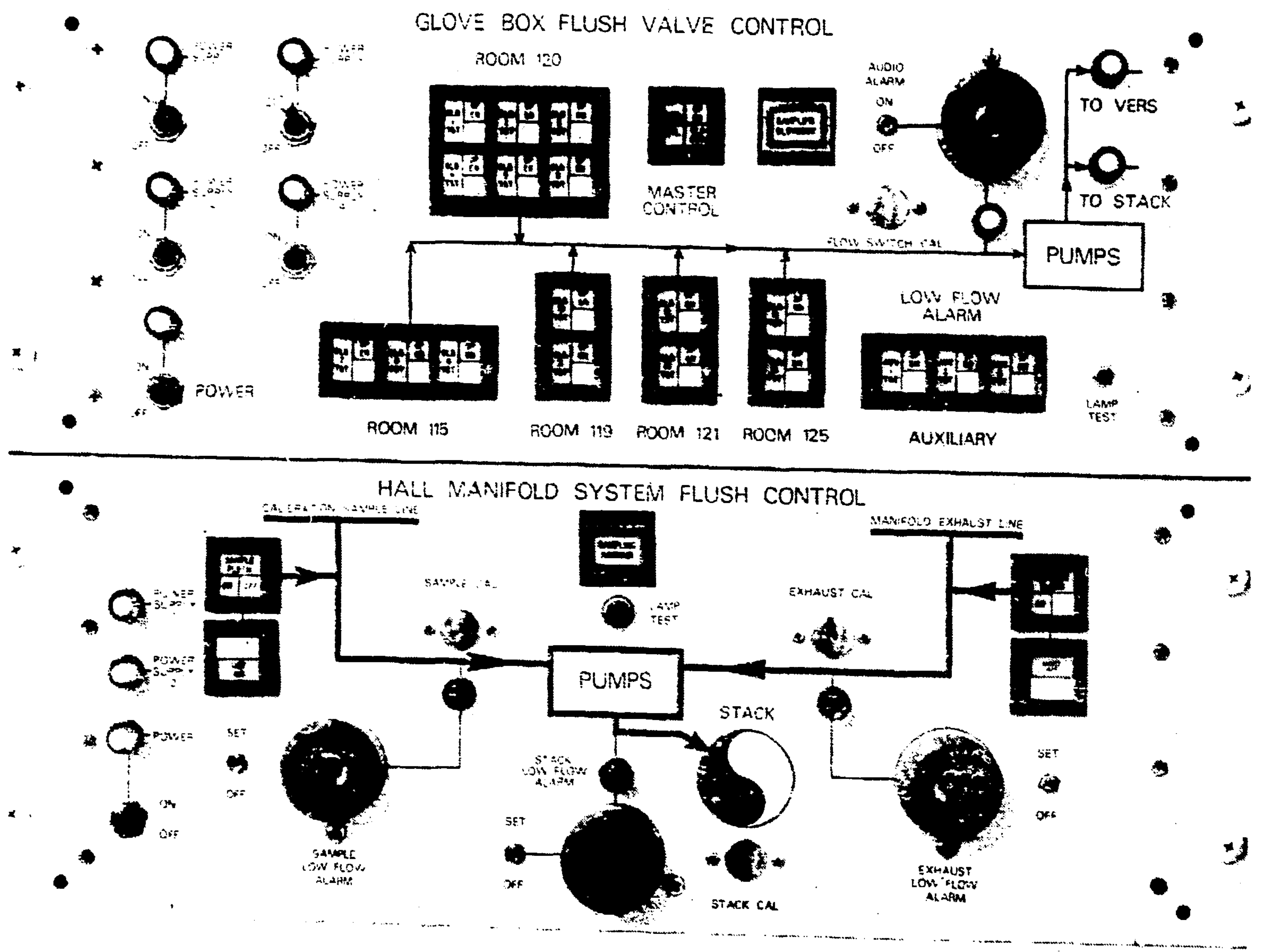
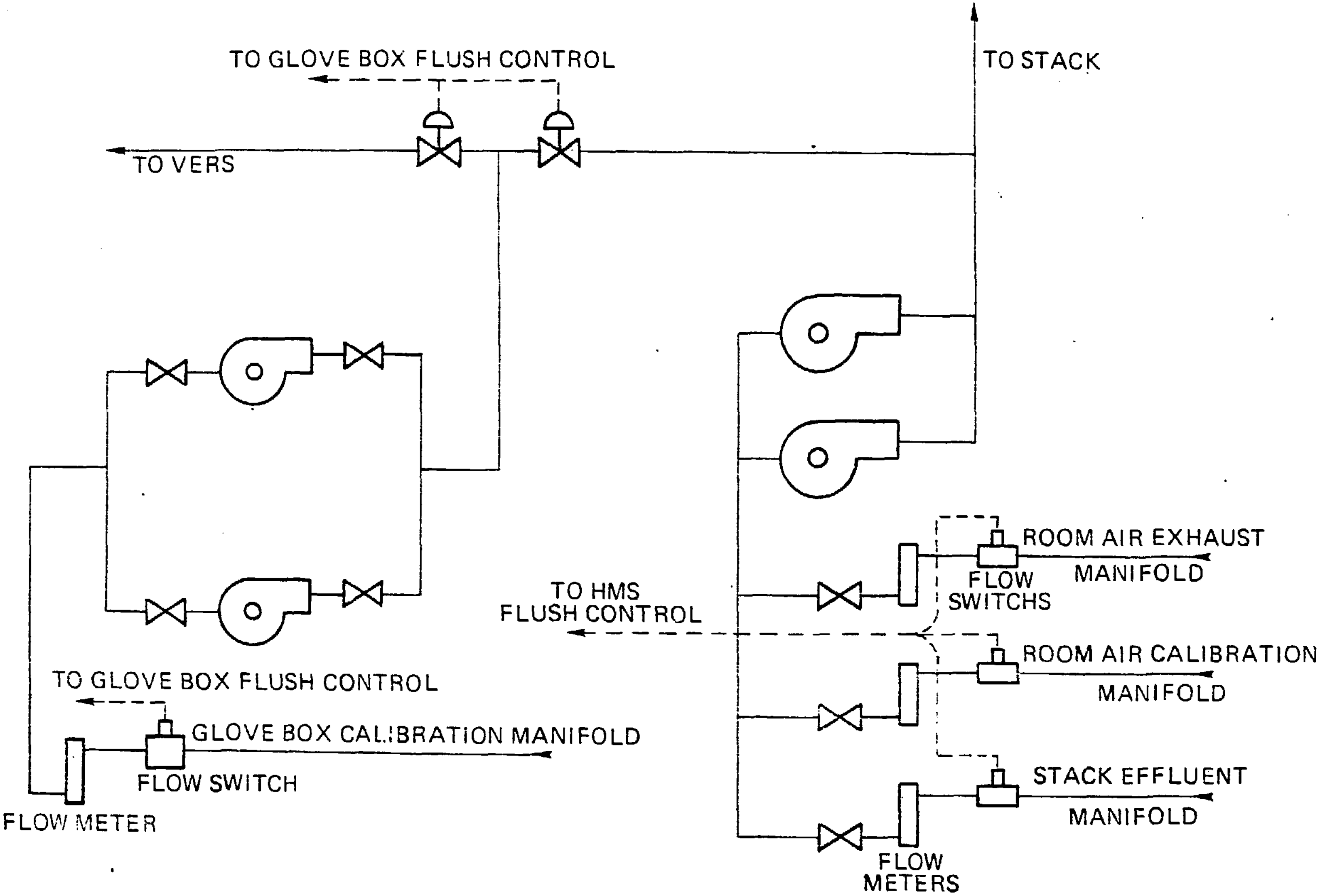


Fig. 3: Hall Manifold System Flush Control
Glove Box Flush Valve Control



CENTRAL PUMPING STATION

Fig. 9

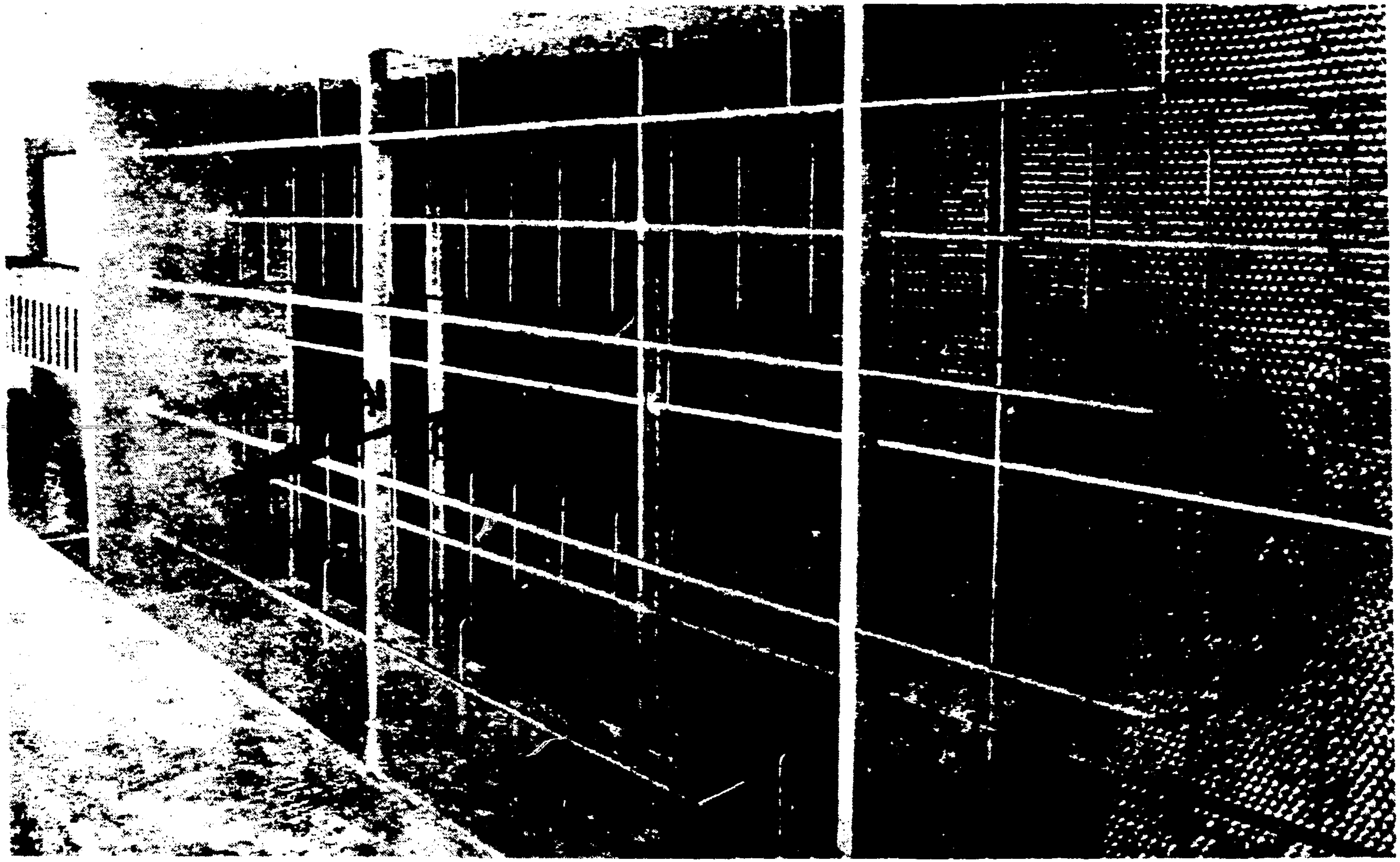


Figure 10: Stack Effluent Sample Array.

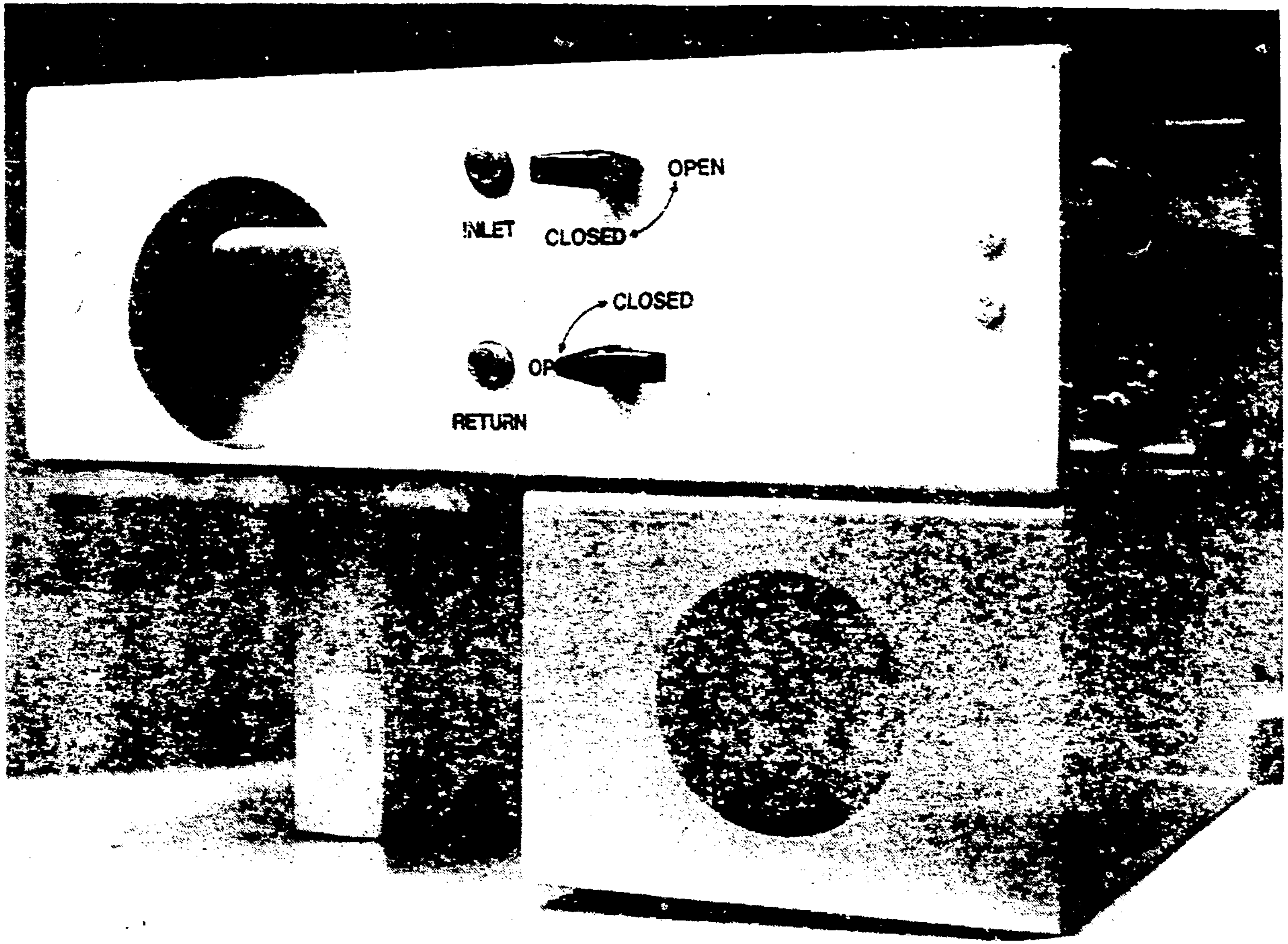


Fig. 11: Pump Module with Dual 20 cc Ion Chambers

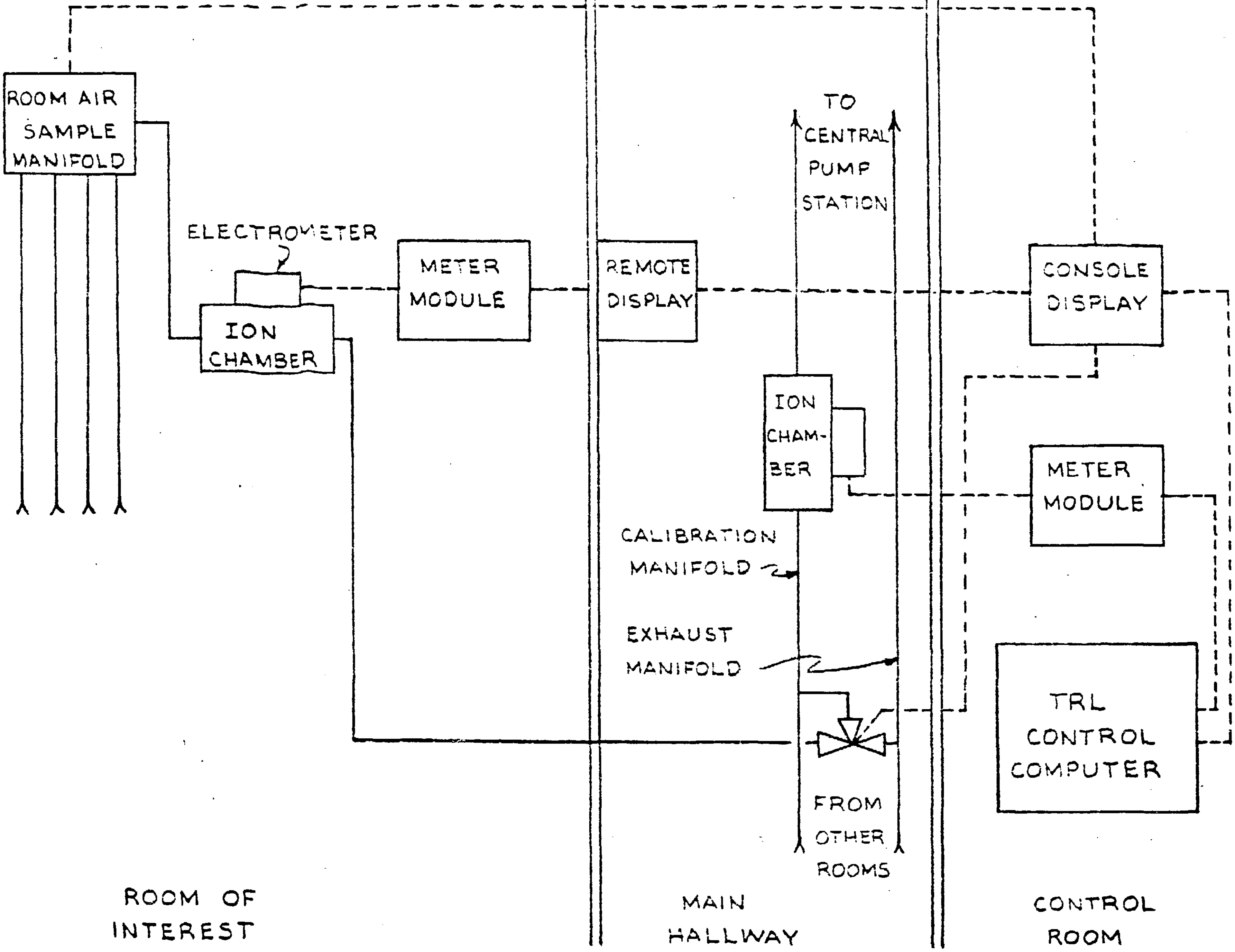


Figure 12: Room Air Monitoring System.

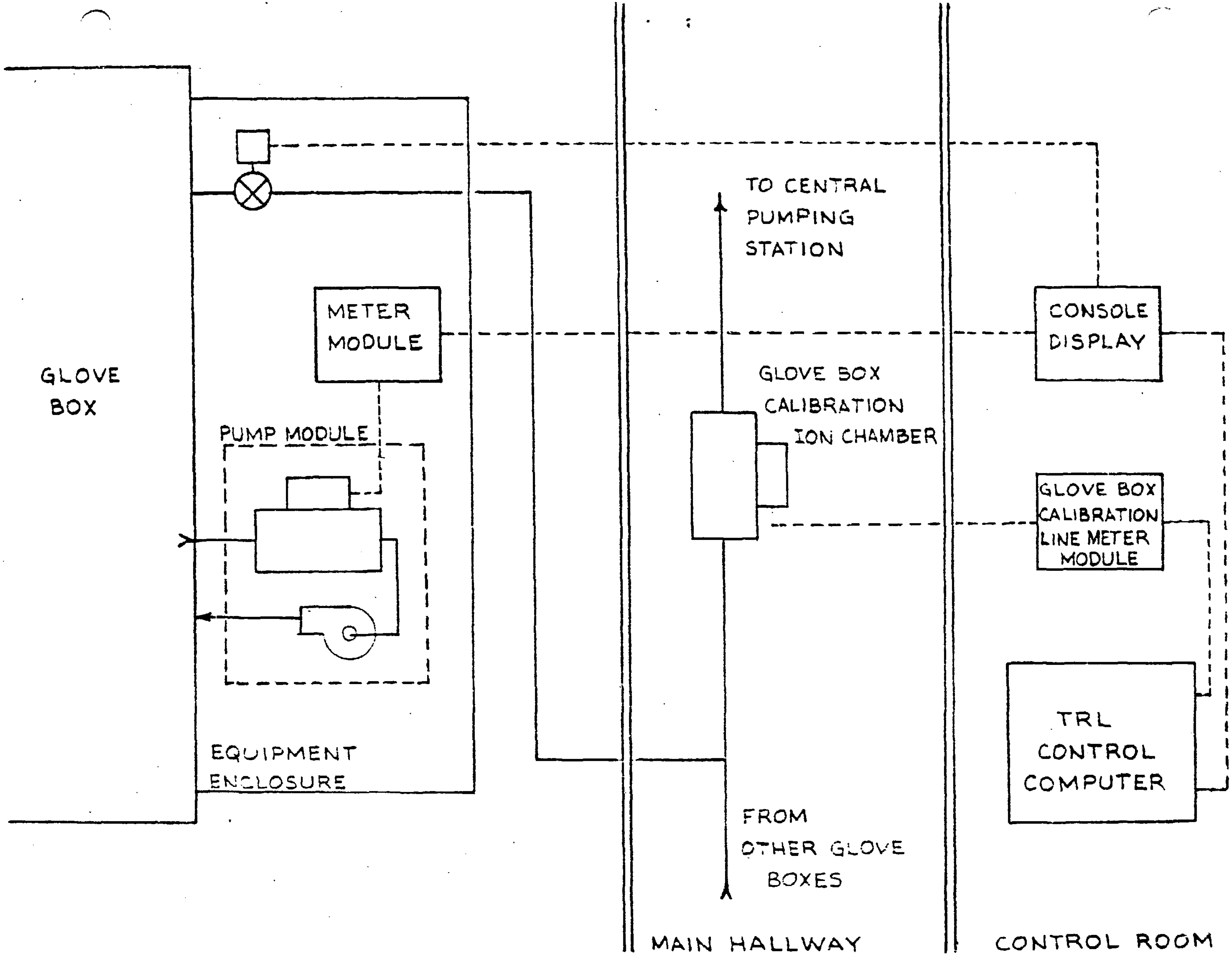


Figure 13: Glove Box Monitoring System

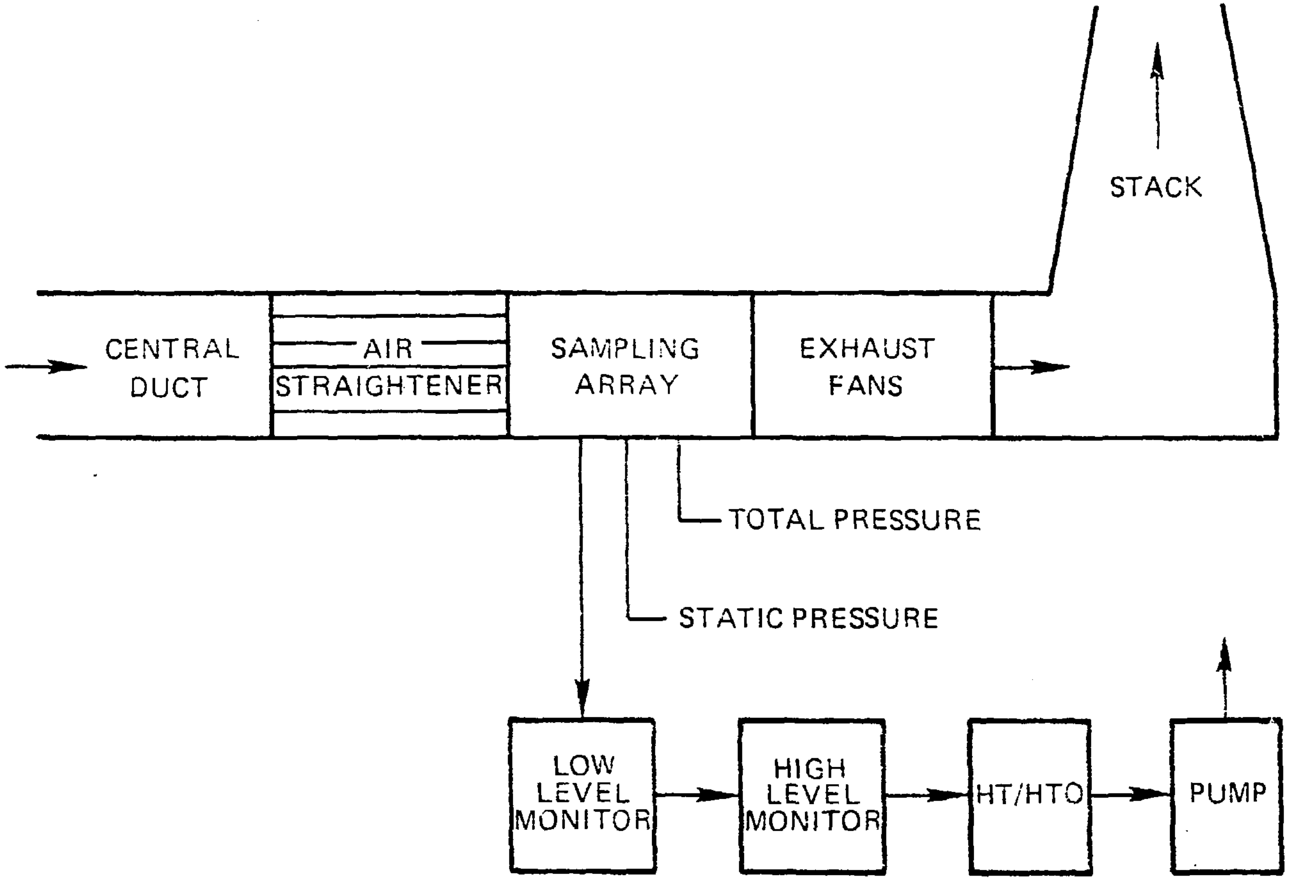


Figure 14: Stack Effluent Monitoring System.

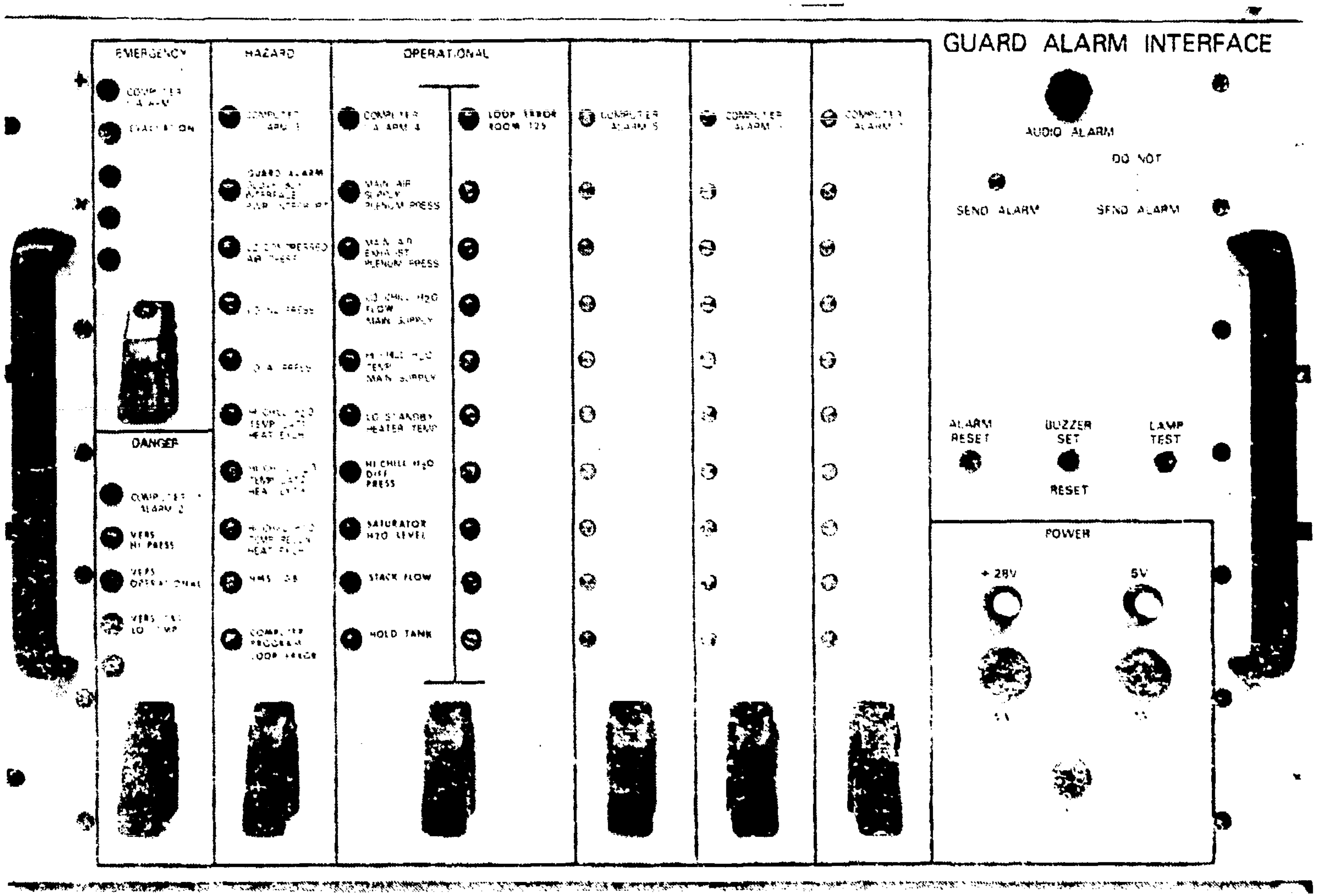


Fig. 15: Guard Alarm Interface

BACKFITTING OF EXISTING NUCLEAR POWER PLANTS WITH PARTICULATE, IODINE AND NOBLE GAS MONITORS

by Merton R. Marley and Eric L. Geiger
Eberline Instrument Corporation

Wall Street Journal reported "Utility Discloses Release of Radioactive Gases." During a 24-hour period, a nuclear power station released radioactive iodine at more than twice the rate permitted for the site. A November 1977 news release states: "NRC staff to begin a new systematic safety evaluation of eleven operating nuclear power facilities." Many existing work area air monitoring and effluent monitoring systems are not capable of continuously monitoring for ^{131}I in the presence of larger concentrations of noble gas. These systems must depend on noble gas monitors to provide warning when limits are approached or exceeded. In some cases, this is not adequate and events as reported in the **Wall Street Journal** happen, adding "fuel to the fire" for the anti-nuclear forces. Analog (ratemeter) radiation monitoring systems currently in use at most operating nuclear plants are inadequate to measure concentrations of airborne radioactive particulate, iodine and noble gases at the concentration dictated by ALARA (as low as reasonably achievable). The presence of variable "background" sources and obsolete electronics limits the usefulness of these existing ratemeter systems.

Existing nuclear power plants need a better way to continuously monitor for ^{131}I in the presence of much larger concentrations of noble gases; a more precise and accurate way to document airborne concentrations of particulate, iodine and noble gases in work areas and in effluents; an easier way to obtain history information; a way to measure, evaluate and alarm when trends as well as high concentrations indicate a potential for exceeding limits specified in the regulations or technical specifications; an easy way to document compliance with regulations and technical specifications and for reporting to NRC. These needs can be met with the Eberline stand-alone digital microcomputer-based system for monitoring radioactive particulate, iodine and noble gas (PING-2). This system may be used for backfitting in existing nuclear power plants. Advantages of the new system include:

1. Good statistical precision from integrated counts over various time periods.
2. Fast response of alarm features when concentrations are elevated.
3. Continuously updated history files for the last twenty-four each 10-minute, 1-hour and 24-hour intervals.
4. Background subtraction features for particulate, iodine and noble gas measurements that correct for variations in background sources.
5. Data are easily retrieved in a usable form, including average concentrations for various time periods, rate of change, release rate and other features.

PING-2 Components

(See Figure 1: PING-2 Overall View.) The Eberline Model PING-2 is a very accurate and sophisticated digital system which is simple to operate and maintain.

(See Figure 2: PING-2 Components.) The major components are the printer, keyboards, microcomputer, sampler assembly with detectors, pump, digital displays with status indicators, area monitor and audible alarm. The first three components are located in a sliding drawer within the top of the PING-2. The remainder are located on the front panel or underneath the metal enclosure

(See Figure 3: Simplified Block Diagram of the Sampler Assembly.) Eberline Sampler Model SA-6 is one compact, lead shielded assembly, which holds detectors for particulate, iodine and noble gas. This sampler has three inches of lead shielding with one inch of lead between the detectors. Motor driven check source assemblies are provided for each detector. The air sample goes through a filter paper on which particulate is deposited, then through the charcoal cartridge which traps the iodines, then into the gas chamber. The particulate filter is monitored by a beta scintillation detector on one side (Eberline Model RDA-3A) and a solid-state alpha detector on the other side (Eberline Model RDS-1). The beta detector measures the beta-emitting radionuclides on the filter. The alpha detector measures radon/thoron daughter products on the filter and is used to estimate the contribution of these daughter products to the beta particulate levels. A proportional amount of the alpha measurement is subtracted from the beta measurement. This technique tends to nullify the effects of fluctuating radon/thoron levels and is one important background subtraction feature of the PING-2.

The charcoal cartridge is monitored by a 2 x 2 inch NaI (Tl) gamma scintillation detector (Eberline Model RDA-2A) which is gain-stabilized to eliminate the effects of drift caused by fluctuations in temperature and/or aging. This gain stabilized feature is very important for ^{131}I measurements. The measurement is accomplished using a single channel analyzer (SCA) with its window calibrated to the 364 keV energy of ^{131}I. An additional SCA is provided with its window calibrated to an energy just above the ^{131}I energy to provide a measure of the background in the iodine window. The effects of a fluctuating background and the contribution from noble gases held up in the charcoal can be nullified by this background-subtraction feature which provides a much more reliable real-time measurement of ^{131}I trapped by the charcoal cartridge.

The gas chamber is monitored by a beta scintillation detector (Eberline Model RDA-3A). Background correction for this channel is derived from the area monitor (Eberline Model DAI-1). Since the external gamma radiation has a measurable effect on the beta measurements, the area monitor is used as a source of background subtraction for measurements of particulate and noble gas. This is the third background-subtraction feature of the PING-2. A fixed background also may be subtracted from each channel, this being the fourth background subtraction feature of the PING-2.

The microcomputer is the heart of the system. A microcomputer is, by definition, a computer that is based on a microprocessor as its central processing unit (CPU). The Eberline microcomputer is Intel 8080A-based, designed and optimized in hardware and software for the task associated with the control and data acquisition/display of radiation monitors. (See Figure 4: PING-2 Microcomputer Block Diagram.)

CPU Board

The CPU board utilizes the 8080A microcomputer as the CPU element. The data word (bus) is 8 bits parallel. The address bus is 16 bits parallel, providing capability of 16,535 memory locations of data words.

Supporting the CPU are the clock generator and the system controller. All logical operations are performed in the CPU. Instructions are fetched (addressed) from memory and executed. A series of instructions then becomes a routine. A series of routines becomes a program. A program is simply a logical sequence of instructions for performing a task. Also residing on the CPU board is 7 level priority-interrupt logic, the system's real time clock logic and input/output ports.

Interrupt is a method of accomplishing a task that is not logical within the normal program flow, for instance, the counting of random events of radiation. After that task is performed, the CPU will resume the task it was performing when interrupted. The real time clock is maintained in 10 millisecond intervals via software (program).

Two input ports of 8 bits each and one output port of 7 bits are provided. One input port is normally used as the method to assign an address (identifier) to the microcomputer by setting bits low using jumpers. Another port is an input for system type information. The last port is an output port for controlling system type equipment, e.g., sample pump and flushing logic.

PROM Board

The programmable read-only memory (PROM) board provides capacity for 8192 8-bit bytes (words) of program and/or constants. This memory is nonvolatile. The program is fixed and not reprogrammable while in use. The memory chips normally supplied are UV-light erasable and reprogrammable at the Eberline factory. Instructions stored in these PROMs are addressed by the CPU for execution.

RAM Board

The RAM board has capacity for 2096 bytes of random access memory. The initializing routine checks RAM, sets input/output (I/O) ports according to their functions and initializes all channel parameters to beginning values. These values allow for an orderly running condition at start-up, and are subsequently changed by the operator in the calibration process. Note the initialization program is executed only when the instrument is first turned on. Data contained in RAM are volatile; that is, the data contained in RAM will be lost when power is removed.

Parallel Input/Output Board

The parallel I/O board provides 24 bits of input, 24 bits of output and 24 bits that may be either input or output, depending upon the application.

Detector Input/Output Board

The detector I/O board provides the logic and input/output capability to support two radiation detectors. A maximum of three of these boards is permissible.

Display

The Eberline Display II is a LED display module used to display data output by the microcomputer. The format of the data display is three significant mantissa digits with a 1½-digit exponent ($\pm 10^{19}$ maximum). An array of 6 lights displays the current status of the channel associated with the data. Dual inputs are provided and the input to be displayed is selected by a front panel switch. Three displays are incorporated in the PING-2. Status display outputs provide a means to actuate external circuitry such as alarm lights and/or audible alarms. Maximum rating of the outputs is 15 VDC and 30 mA.

Printer

The Eberline printer assembly is a 5 x 7 dot matrix per character line printer. Its simplicity insures long life and ease of maintenance. Unlike many printers, it has no clutches, timing discs or reversing mechanisms to control the transport of the printing head. All drive electronics are located on a single printed circuit board attached to the front of the printer assembly. All timing and control are accomplished in the computer.

Keyboard

The keyboard is the communication point from man to machine and has two subgroups. One group is the instruction pad and the second group is the data pad. All commands begin with a single entry via the instruction pad, followed by an entry or entries via the data pad and terminated with the ENTER key. Numerical data (calibration constants, etc.) are entered in fixed decimal point scientific notations. The instruction and data pads are color-keyed according to functions.

Theory of Operation

The PING-2 is a closed loop, fixed tasking operating system with interrupt-driven data acquisition and real time clock. Changes in operating conditions are printed within seconds of occurrence, data presented only if significant. A history file is automatically maintained in an interpretable, orderly manner. A few entries on the keyboard can recall any data, status or parameters of any detector. System security is provided by two key-operated switches which enable the keyboard and/or parameter file modification. For purposes of clarity and ease of understanding, four subsections are presented.

1. Operating System: When the PING-2 is first turned on, it begins running. (See Figure 5: Day and Time.) The initializing routine checks RAM, sets input/output (I/O) ports according to their function(s) and initializes all channel parameters to beginning values. (See Figure 6: Initializing Detector Parameters, Figure 7: Channel Parameters, and Figure 8: Channel Parameter File After Edit.) The program then drops into the closed loop, which is the operating system. The operational status of each detector is conditionally determined on each pass of the program.
   a. Status. Status is the operational condition of the instrument. Since the validity of the data is directly affected by the operational status of the instrument when the data were acquired, the status is always carried along with the data and appropriate light lit on display channel.
   b. (See Figure 9: Status Lights.) The display status lights function as follows:

   Green light lit is normal operating condition.

   White light marked M is lit when in calibrate mode or maintenance mode. When lit, the printer prints out the mode of operation

   Yellow light marked F is lit to indicate failure mode. You may have three different failure modes and the printer prints out the mode: FAIL LO is no counts from the detector within 10 minutes. FAIL HI is excessive count rate (1.2 x 10^6 cpm). FAIL EX is external fail such as pump flow stop.

   Yellow light marked T is lit for trend alarm. The alarm is determined in % per minute increase; the data on the printout are in units of activity.

   Yellow light marked A is lit for alert alarm. This alarm is displayed when the calculated value exceeds the value input as alert alarm setting.

   Red light marked H is lit for high alarm. When this alarm is lit the audible alarm is on. There is an acknowledge switch for the audible alarm. The high alarm is displayed when the calculated value exceeds the value input as high alarm setting. Data are printed at each 10-minute interval during a high alarm.

2. Significant Count: The basic counting mode of the PING-2 is a preset count. The preset value of 512 counts at the detector determines the significance of the data used for display and alarm determination. The precision of any individual count rate determination is therefore ±8.8% (2 sigma). Time to acquire the 512 counts is measured, then divided into the counts acquired, resulting in counts per minute (cpm).

   Any background subtraction sources specified are then calculated and subtracted from the count rate. This results in a net cpm which is proportional to activity at the detector.

   The net cpm is multiplied by the calibration constant entered in the channel parameter file. The result is displayed in units as entered in the parameter file. The data are compared to high and alert alarm settings for alarm determinations.

   Low count rates (less than 51 cpm) would never reach a significant count within the 10-minute history management interval. This condition causes the counting mode to revert to a 10-minute preset time.

   High count rates could cause significant count determinations with insignificant time. Therefore, a period of 5 seconds is the minimum time between significant count calculations.

3. History File Management: History files are maintained on each channel (detector) in three time details. These files are: (1) 23 each 10-minute intervals, (2) 24 each 1-hour intervals and (3) 24 each 1-day intervals. The data presented for these intervals are the averages of the data accumulated during each interval. (See Figure 10: Beta Particulate, 24-hour Interval Log.) Any data presented with other than a NORMAL status word indicate that the instrument was abnormal for the entire interval. Any data accumulated within an interval with a normal or alarm status condition are preserved and presented as normal. When a channel parameter file or a history file is commanded, the file is copied from its operating location in memory to a working buffer. While in the working buffer, the channel parameter file may be modified, and calculations may be performed on the history file to display the data in different formats.

   When a history file is commanded, the file is first printed as it is maintained; that is, the data flagged with abnormal status are the data accumulated with that status. (See Figure 11: Beta Gas, 10-minute Averages.) After the printing, all abnormal data are discarded in the working buffer so that only data under normal or alarm conditions are retained. At this time, remaining data points are interpolated to fill in for any abnormal data points that were deleted from the working buffer. Printouts of these data will maintain the status flag in the left margin to flag the data as calculated data. The interpolation is limited to abnormal status data points that occur between normal status data points. (See Figure 12: Interpolation of Beta Gas, 10-minute Averages.)

   For integrating type detectors, the differential for consecutive data points is calculated. Sample flow rate and time are blended into the calculation. The resultant data multiplied by the calibration constant are in units of concentration (radioactivity per cubic centimeter).

   (See Figure 13: Keyboard Assembly.) The use of the PRINT. FILE command is conjunction with the HISTORY FORMAT switch will print the file(s) as follows:

   a. RE-DISPLAY: The resultant printout is the time detail of concentration or radioactivity at the detector.
   b. RATE OF CHANGE: The resultant printout is the time detail of the change (differential) of the radioactivity. This printout has value in the study of changes in concentration, long-term trend studies, etc.
   c. AVERAGE: The resultant printout is a single value representing the average concentration or radiation rate over the time period of the file.
   d. RELEASE RATE: Since this calculation is to determine the total release out of the stack or duct being monitored, the flow in the stack or duct must be known. This value must be entered (Units = cc/min) when the printout requests. The resultant printout is then a time detail of the radioactivity released per minute.
   e. INTEGRATE: The resultant printout is the time detail of the total radioactivity at the detector over the time period of the file. For the area monitor, this would be the total accumulated exposure over the time of the file.

   Note that all abnormal status data points that cannot be interpolated from normal data have a "?" printed in place of data.

4. Automatic Data Logging (See Figure 14: Data of File.)
   a. All data/conditions logged are preceded by a printout of day-of-year and time-of-day. The day-of-year is a 365-day calendar. The time-of-day is a 24-hour clock. Midnight is 0000, 11:00 p.m. is 2300. The clock and calendar are set by the operator.
   b. All changes of status are logged when they occur. Data are included when change of status is to or from an alarm condition.
   c. Data, status and channel number are printed every ten minutes when status is high alarm or calibrate.
   d. The 1-hour interval file from all channels is printed every 24 hours. The time of day preselected for printout of this log is input by the operator.

      A 4-hour log interval is also operator-selectable. When enabled, the 10-minute interval file from all channels will be printed every 4.0 hours. The time of printout is keyed to the 24-hour log time selected.

NOTE

> All printouts of history data are ordered from the oldest data to the most recent data. The last data printed are for the last *period completed, e.g., last ten minutes, last hour, yesterday.*

OPERATING THE INSTRUMENT

1. Keyboard (See Figure 13. Keyboard Assembly.) The keyboard is the *communication point from* man to machine. The printer is the counterpart from machine to man. To operate the machine properly, the operator needs a good understanding of both keyboard input and printer output.

   The keyboard has two subgroups. The group closest to the status lights is the instruction pad, while the group closest to the key switches is the data pad. An entry of instruction and data is then a command. All commands begin with a single entry via the instruction pad, followed by an entry or entries via the data pad and terminated with the ENTER key. Numerical data (calibration constants, etc.) must be entered in fixed decimal point scientific notation. Each key depressed is displayed on the single-digit display.

   The instruction pad is color-keyed according to function. The blue keys are for inquiry of data or parameters. A command beginning with a blue key is followed by the channel number and ENTER.

   a. FILE: This command sequence will cause the parameter file of the channel number specified by the second key pressed to be printed. If the second key is 0, the system parameter file is presented. This allows the setting of the calendar, clock and log times. If the second key entered is for a channel file (1-6), the file for that channel will be printed.
   b. DATA: A command sequence of DATA, channel number (0-6), ENTER will cause the current data and status of the specified channel to be typed.
   c. EDIT: The black key is used to command changes of parameters in the channel files. The use of this key is restricted by the EDIT key-lock switch.
   d. HISTORY Keys: These three keys were described previously.

   The green keys are for control functions. A command beginning with a green key is followed by ON (or OFF), except for CK SRC, which is followed by channel number.

   a. BIT: This command provides an additional control bit on each detector's wiring terminal board. The command sequence is BIT, ON (or OFF), channel number (1-6), terminated by ENTER.
   b. FLUSH: An output bit is provided to control a flushing mechanism for the sampler.
   c. PUMP: The command sequence of PUMP, ON (or OFF), ENTER turns the sample pump on (off).
   d. ALM: A command sequence of ALM, OFF, channel number (1-6), ENTER will disable the alarming capability of the specified channel. Replacing OFF with ON will re-enable the alarming function of the specified channel. The instrument is initialized with alarms enabled.
   e. CK SRC (See Figure 15: Check Source.): A command sequence of CK SRC, channel number (0-6), ENTER will cause the check source to expose to the detector on the specified channel. If the specified channel is 0, the check source will actuate on all channels which have a check source

connected. Once commanded, the check source routine is automatically terminated after significant data are accumulated.

The red key (PRINT) is used to print the status of all channels and to calculate on history data in conjunction with the HISTORY FORMAT switch. A command beginning with the PRINT key is always followed by the ALL STAT key or the -/FILE key and the ENTER key on the data pad. The data pad is also color-coded to work with the instruction pad. Other keys are:

a. RUB OUT: An entry error may be corrected by depressing the RUB OUT key. Each time the RUB OUT key is depressed, one entry is erased. The display does not present the rubbed-out entry.
b. -/OFF: The -/OFF key is also used to signify "no."
c. +/ON The +/ON key is also used to signify "yes."

This paper has primarily discussed the Eberline PING-2, a self-contained microcomputer-based radiation measuring system. The rapid development of microcomputers has opened the door to unlimited uses for radiation measuring systems. Additional sampler-detector systems available today include moving filter particulate air samplers, iodine samplers, gas samplers, water samplers and area monitors.

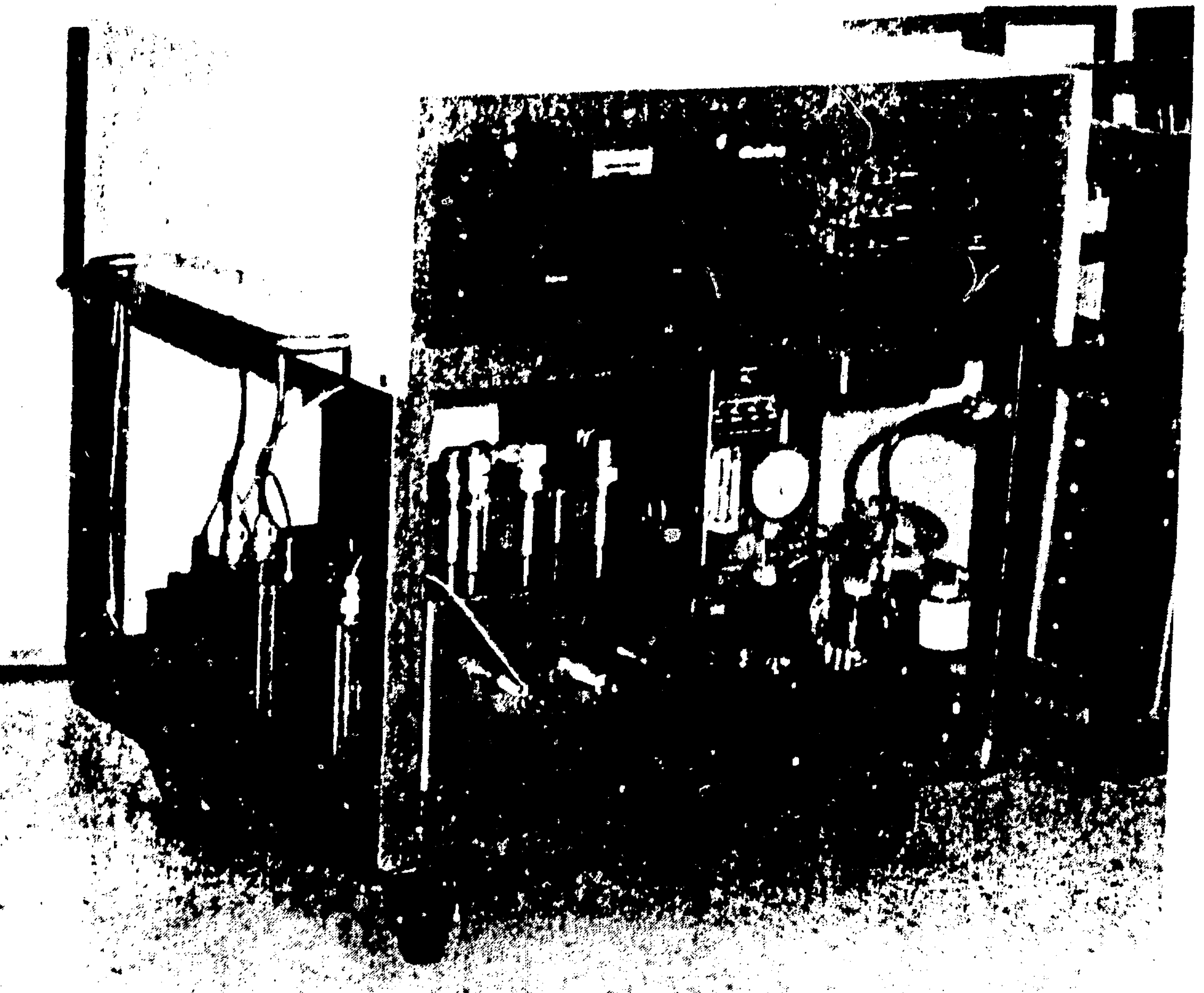
(See Figure 16: Microcomputer Console.) All of the sampler-detector systems mentioned previously can be used with the Eberline microcomputer console. Each microcomputer console is capable of handling data from 192 sampler-detector combinations. The console may output data to operate CRT's for visual display of data and status, may write data into a magnetic tape or communicate with another in-house computer.

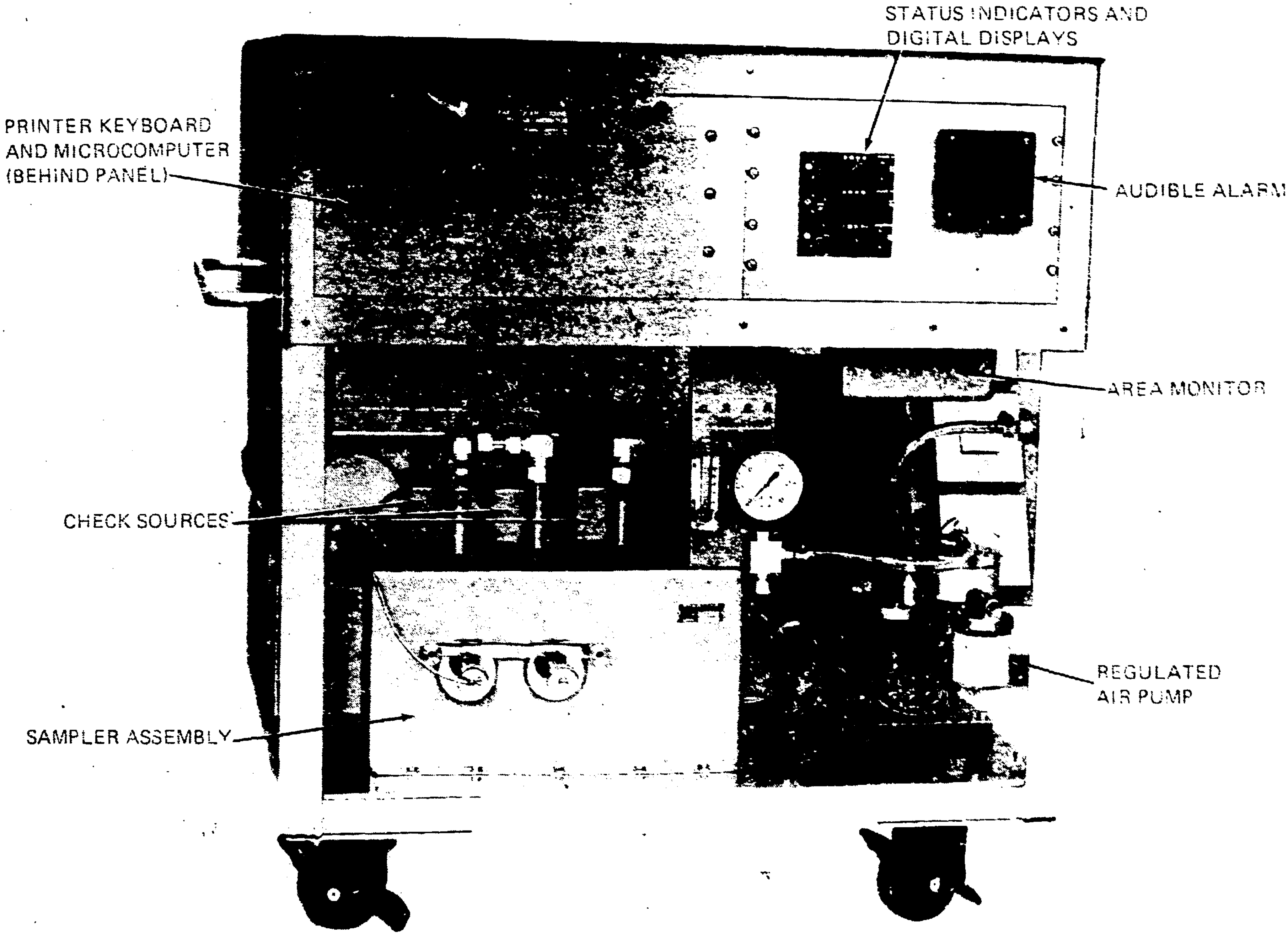
(See Figure 17: Microcomputer Console Panel.) This unit has its own printer, keyboards and status lights and operates similarly to the PING-2. Its functions can be precisely defined (as in the case of the PING-2) or can be provided with extensive flexibility to handle many different types of detector systems.

(See Figures 18 and 19: Field Microcomputer.) Each field microcomputer can accept inputs from six different sampler-detector units. Up to 32 field microcomputers may be connected in parallel via two twisted pair cables back to the microcomputer console.

Conclusion

A stand-alone microcomputer system that comes complete with hardware and software to measure airborne particulate, iodine and noble gases has been described. The system described was designed to meet a need at existing nuclear power plants but also is ideal for work area and effluent monitoring at new nuclear power plants. A microcomputer console and field microcomputer system has also been described. This latter system can be used with many combinations of sampler-detector systems and will accommodate up to 192 channels of detector input. One of the major benefits envisioned by Eberline for this equipment is the upgrading of instrumentation at existing nuclear power plants.




STATUS INDICATORS AND DIGITAL DISPLAYS
PRINTER KEYBOARD AND MICROCOMPUTER (BEHIND PANEL)
AUDIBLE ALARM
AREA MONITOR
CHECK SOURCES
REGULATED AIR PUMP
SAMPLER ASSEMBLY

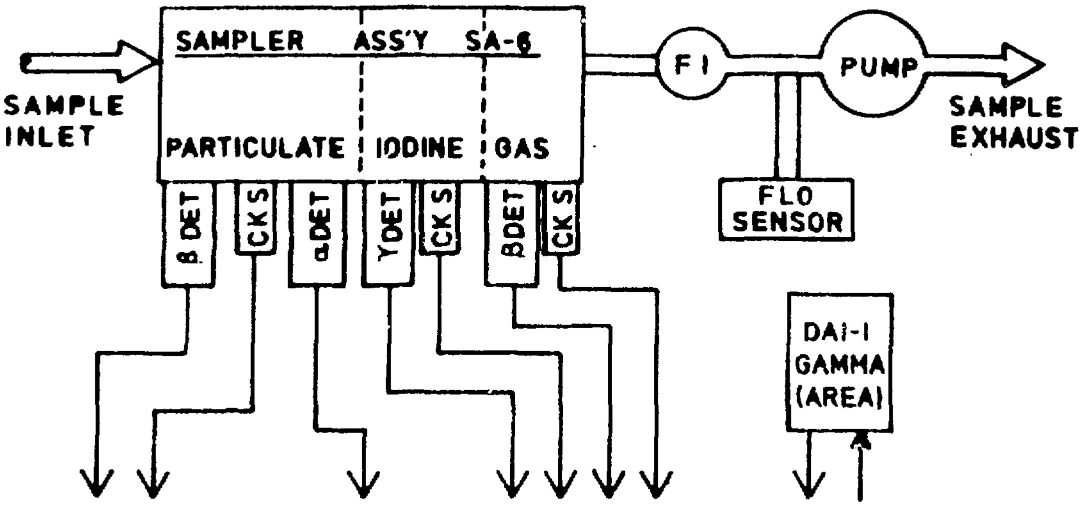


Figure 3. Simplified Block Diagram of the Sampler Assembly

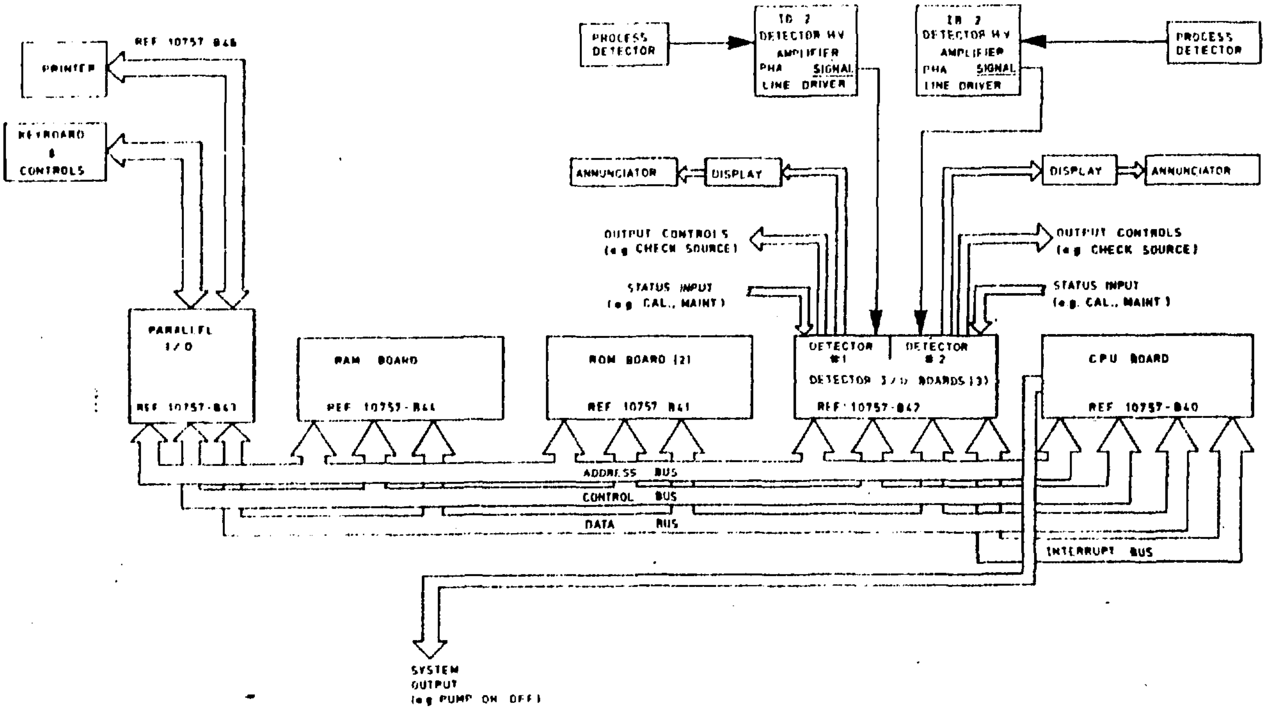
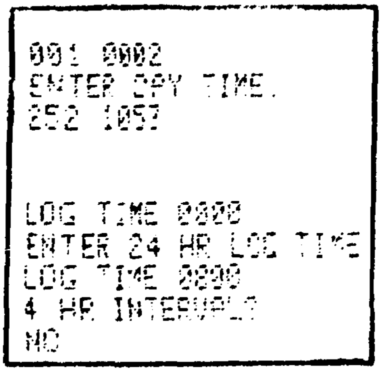
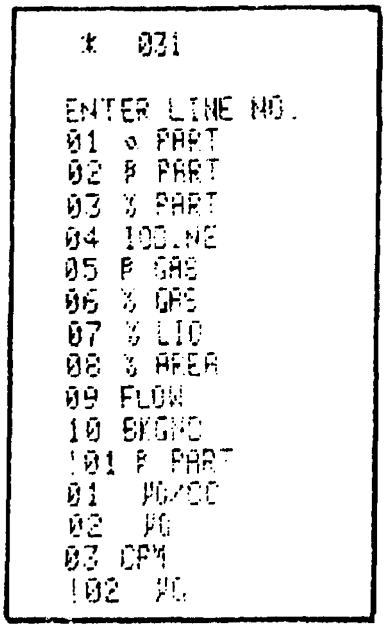


Figure 4. PING-2 Microcomputer Block Diagram



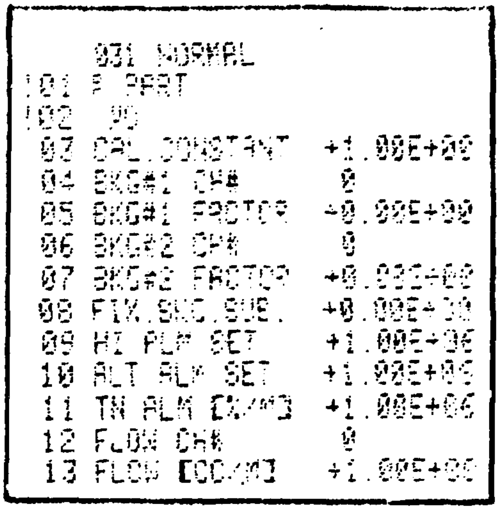
```
001 0002
ENTER DAY TIME.
252 1057


LOG TIME 0800
ENTER 24 HR LOG TIME
LOG TIME 0800
4 HR INTERUPT?
NO
```

Figure 5. Day and Time



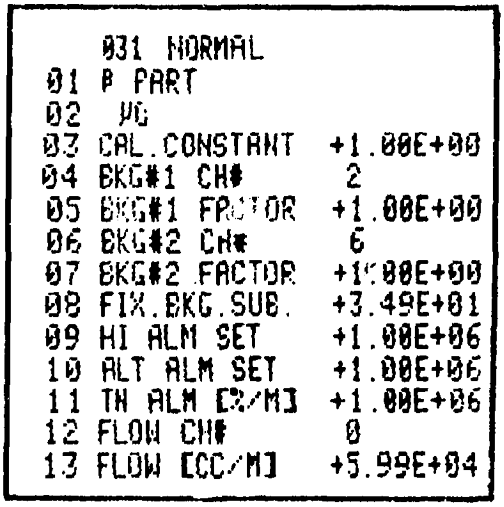
```
  *   031

 ENTER LINE NO.
 01 α PART
 02 β PART
 03 γ PART
 04 IOD.NE
 05 β GAS
 06 γ GAS
 07 γ LIQ
 08 γ AREA
 09 FLOW
 10 BKGND
 !01 β PART
 01   µG/CC
 02   µG
 03 CPM
 !02  µG
```

Figure 6. Initializing Detector Parameters



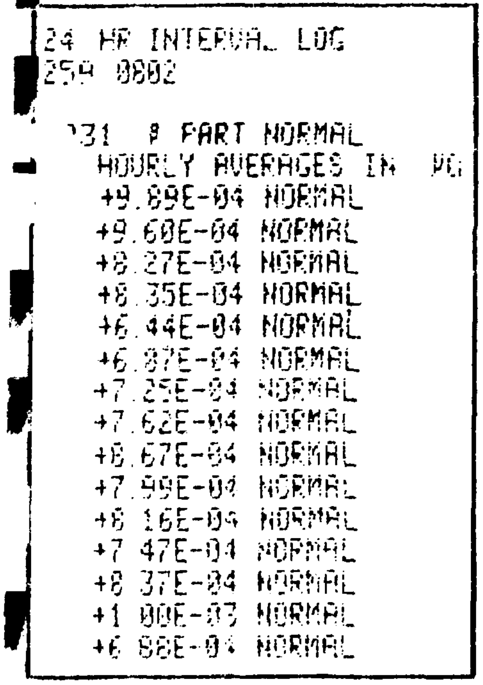
```
     031 NORMAL
!01 β PART
!02  µG
 03 CAL.CONSTANT   +1.00E+00
 04 BKG#1 CH#        0
 05 BKG#1 FACTOR   +0.00E+00
 06 BKG#2 CH#        0
 07 BKG#2 FACTOR   +0.00E+00
 08 FIX.BKG.SUB.   +0.00E+00
 09 HI ALM SET     +1.00E+06
 10 ALT ALM SET    +1.00E+06
 11 TN ALM [%/M]   +1.00E+06
 12 FLOW CH#         0
 13 FLOW [CC/M]    +1.00E+00
```

Figure 7. Channel Parameters



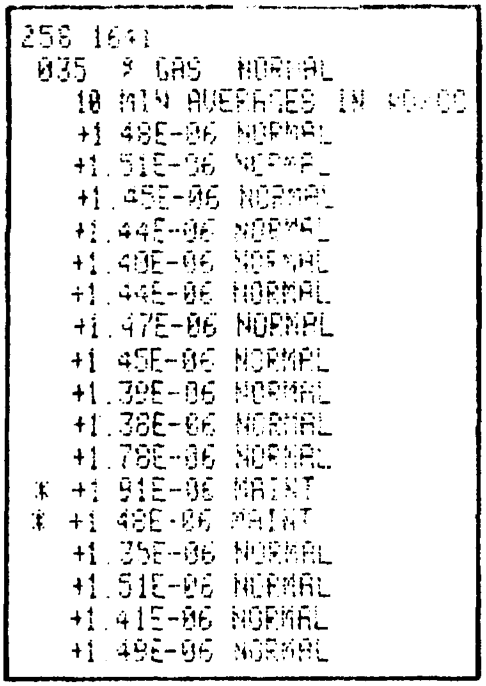
```
     031 NORMAL
 01 β PART
 02  µG
 03 CAL.CONSTANT   +1.00E+00
 04 BKG#1 CH#        2
 05 BKG#1 FACTOR   +1.00E+00
 06 BKG#2 CH#        6
 07 BKG#2 FACTOR   +1.00E+00
 08 FIX.BKG.SUB.   +3.49E+01
 09 HI ALM SET     +1.00E+06
 10 ALT ALM SET    +1.00E+06
 11 TN ALM [%/M]   +1.00E+06
 12 FLOW CH#         0
 13 FLOW [CC/M]    +5.99E+04
```

Figure 8. Channel Parameter File After Edit



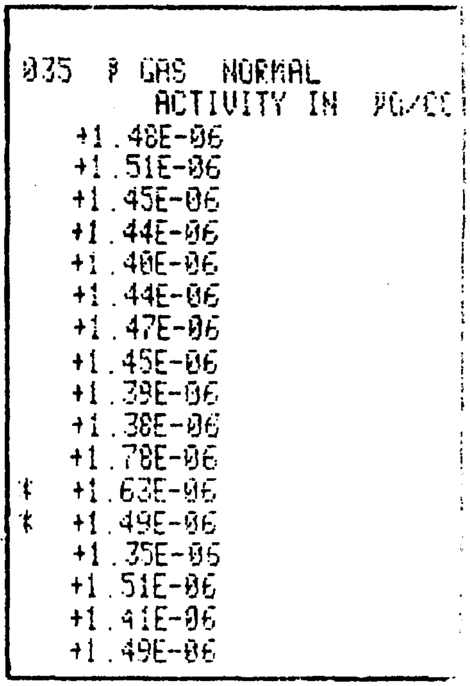
```
24 HR INTERVAL LOG
259 0802

 031  β PART NORMAL
   HOURLY AVERAGES IN  PG
    +9.89E-04 NORMAL
    +9.60E-04 NORMAL
    +8.27E-04 NORMAL
    +8.35E-04 NORMAL
    +6.44E-04 NORMAL
    +6.87E-04 NORMAL
    +7.25E-04 NORMAL
    +7.62E-04 NORMAL
    +8.67E-04 NORMAL
    +7.99E-04 NORMAL
    +8.16E-04 NORMAL
    +7.47E-04 NORMAL
    +8.37E-04 NORMAL
    +1.00E-03 NORMAL
    +6.88E-04 NORMAL
```

Figure 10. Beta Particulate, 24-hour Interval Log



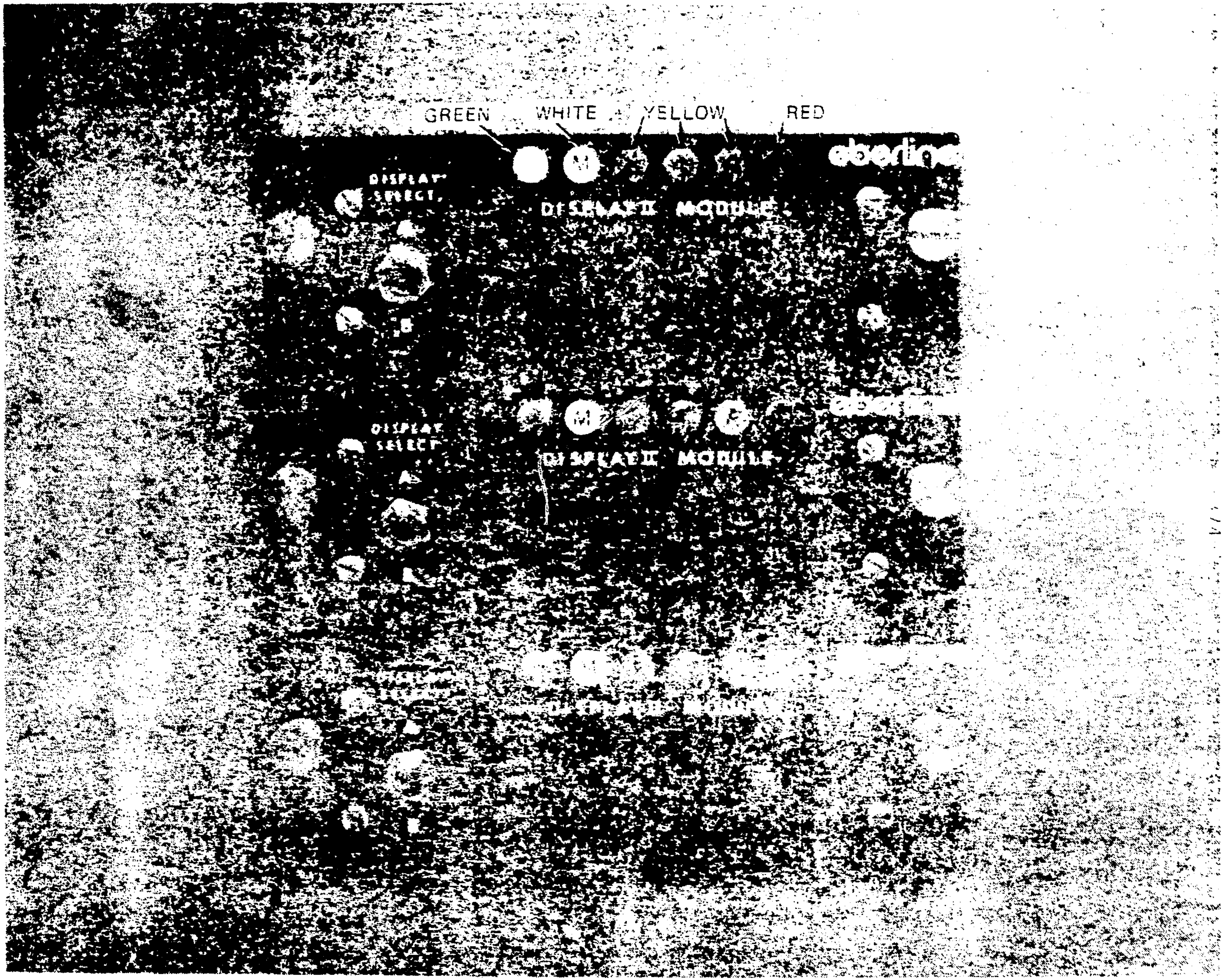
```
256 1641
 035  β GAS  NORMAL
    10 MIN AVERAGES IN PG/CC
     +1.48E-06 NORMAL
     +1.51E-06 NORMAL
     +1.45E-06 NORMAL
     +1.44E-06 NORMAL
     +1.40E-06 NORMAL
     +1.44E-06 NORMAL
     +1.47E-06 NORMAL
     +1.45E-06 NORMAL
     +1.39E-06 NORMAL
     +1.38E-06 NORMAL
     +1.78E-06 NORMAL
  *  +1.91E-06 MAINT
  *  +1.48E-06 MAINT
     +1.35E-06 NORMAL
     +1.51E-06 NORMAL
     +1.41E-06 NORMAL
     +1.49E-06 NORMAL
```

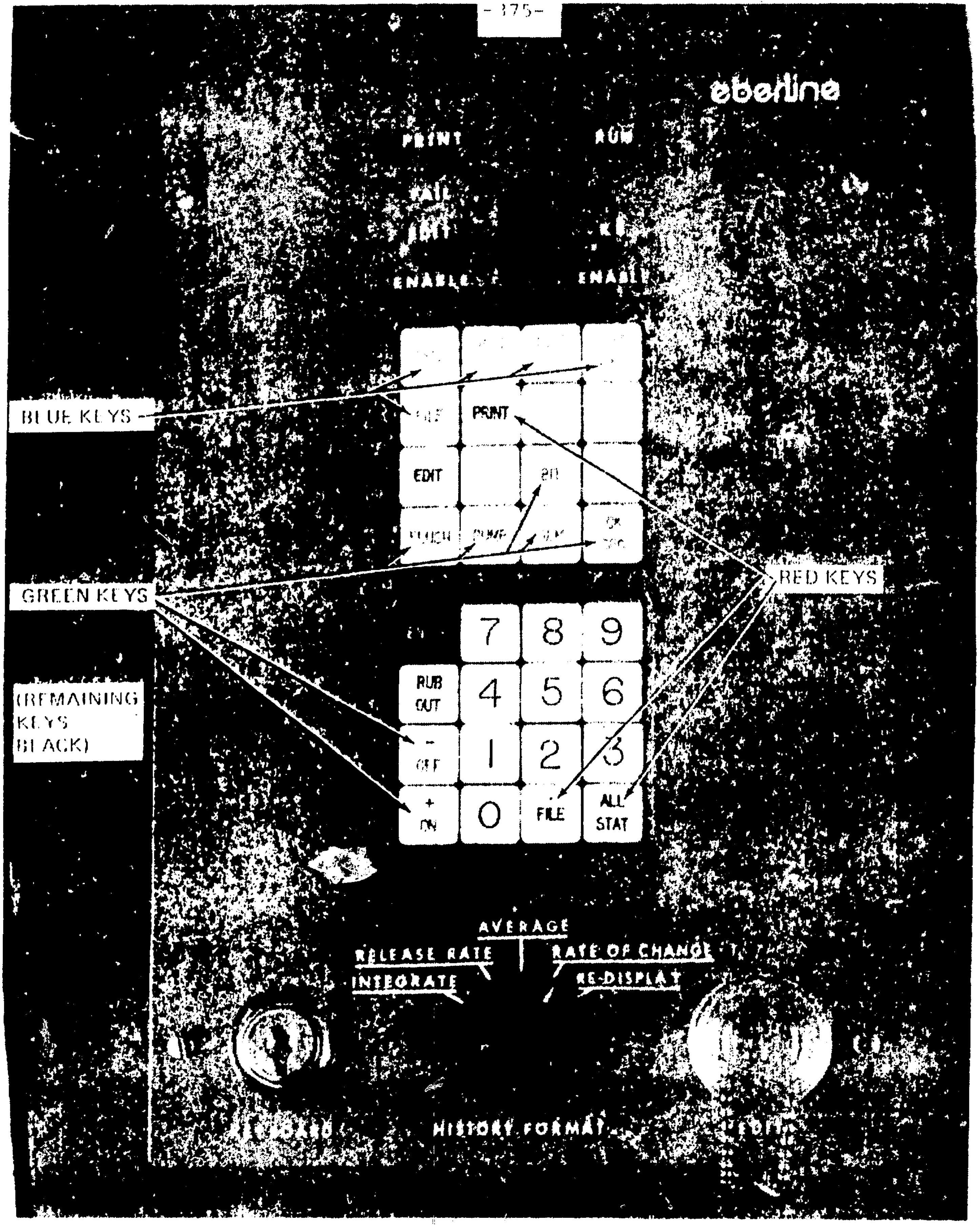
Figure 11. Beta Gas, 10-minute Averages

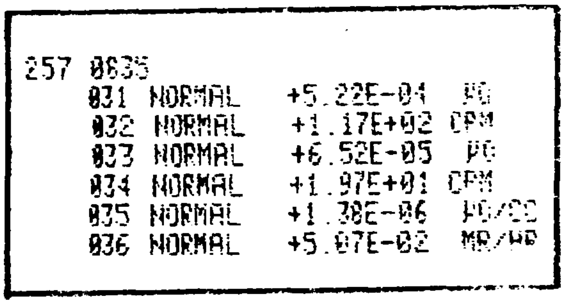


```
035  β GAS  NORMAL
        ACTIVITY IN  PG/CC
    +1.48E-06
    +1.51E-06
    +1.45E-06
    +1.44E-06
    +1.40E-06
    +1.44E-06
    +1.47E-06
    +1.45E-06
    +1.39E-06
    +1.38E-06
    +1.78E-06
 *  +1.63E-06
 *  +1.49E-06
    +1.35E-06
    +1.51E-06
    +1.41E-06
    +1.49E-06
```

Figure 12. Interpolation of Beta Gas, 10-minute Averages

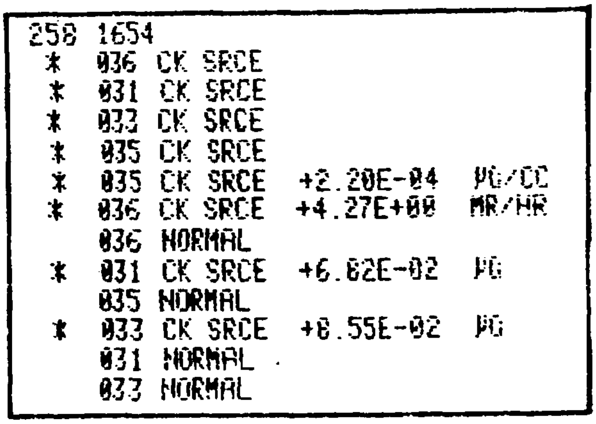

GREEN
WHITE
YELLOW
RED
DISPLAY SELECT
DISPLAY II MODULE
DISPLAY SELECT
DISPLAY II MODULE


eberline
PRINT
RUN
FAIL
EDIT
ENABLE
ENABLE
BLUE KEYS
PRINT
EDIT
GREEN KEYS
RED KEYS
(REMAINING KEYS BLACK)
7
8
9
RUB OUT
4
5
6
- OFF
1
2
3
+ ON
0
FILE
ALL STAT
AVERAGE
RELEASE RATE
RATE OF CHANGE
INTEGRATE
RE-DISPLAY
KEYBOARD
HISTORY FORMAT
EDIT



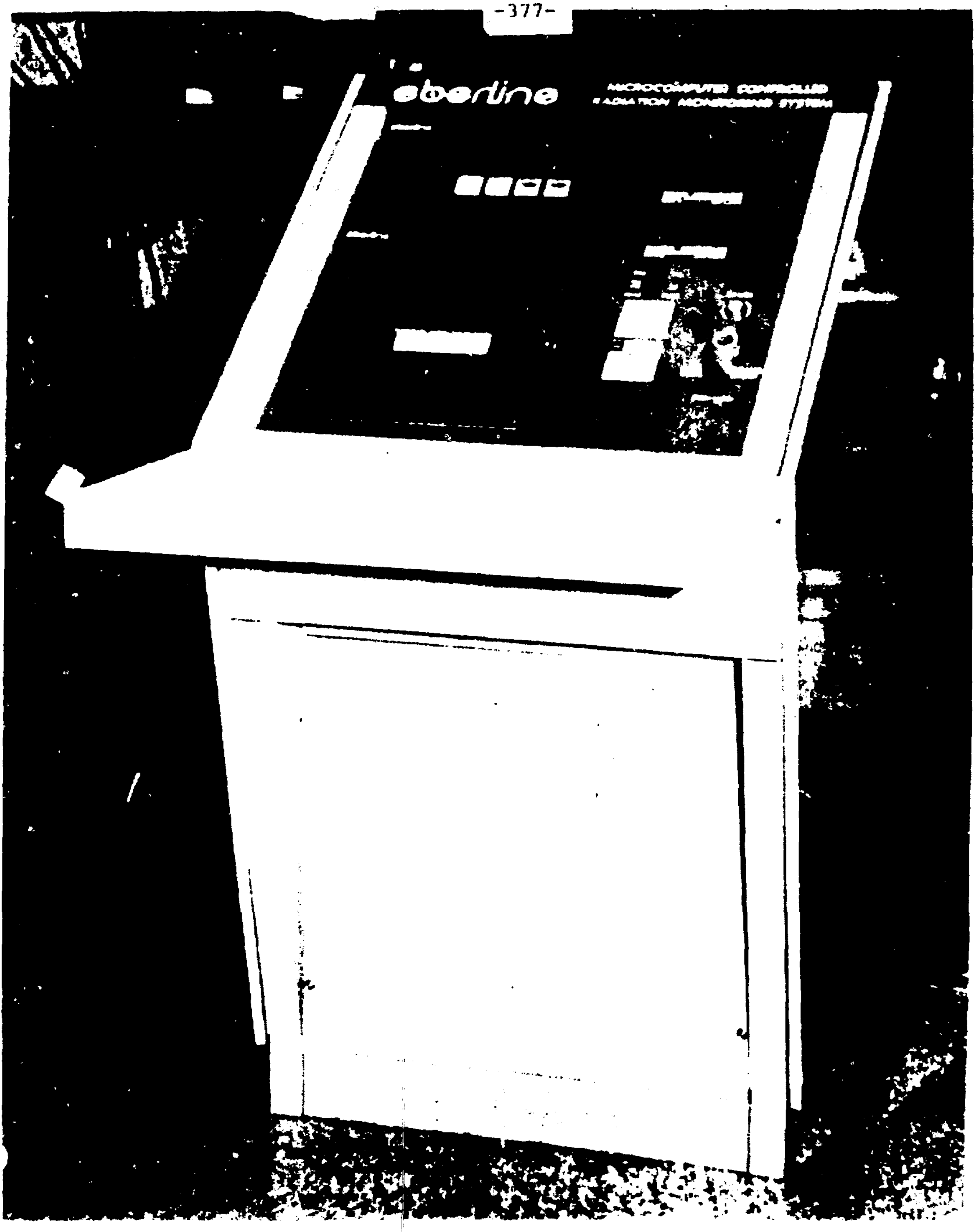
```
257 0635
    031 NORMAL   +5.22E-04  µG
    032 NORMAL   +1.17E+02 CPM
    033 NORMAL   +6.52E-05  µG
    034 NORMAL   +1.97E+01 CPM
    035 NORMAL   +1.38E-06  µG/CC
    036 NORMAL   +5.07E-02  MR/HR
```

Figure 14. Data of File



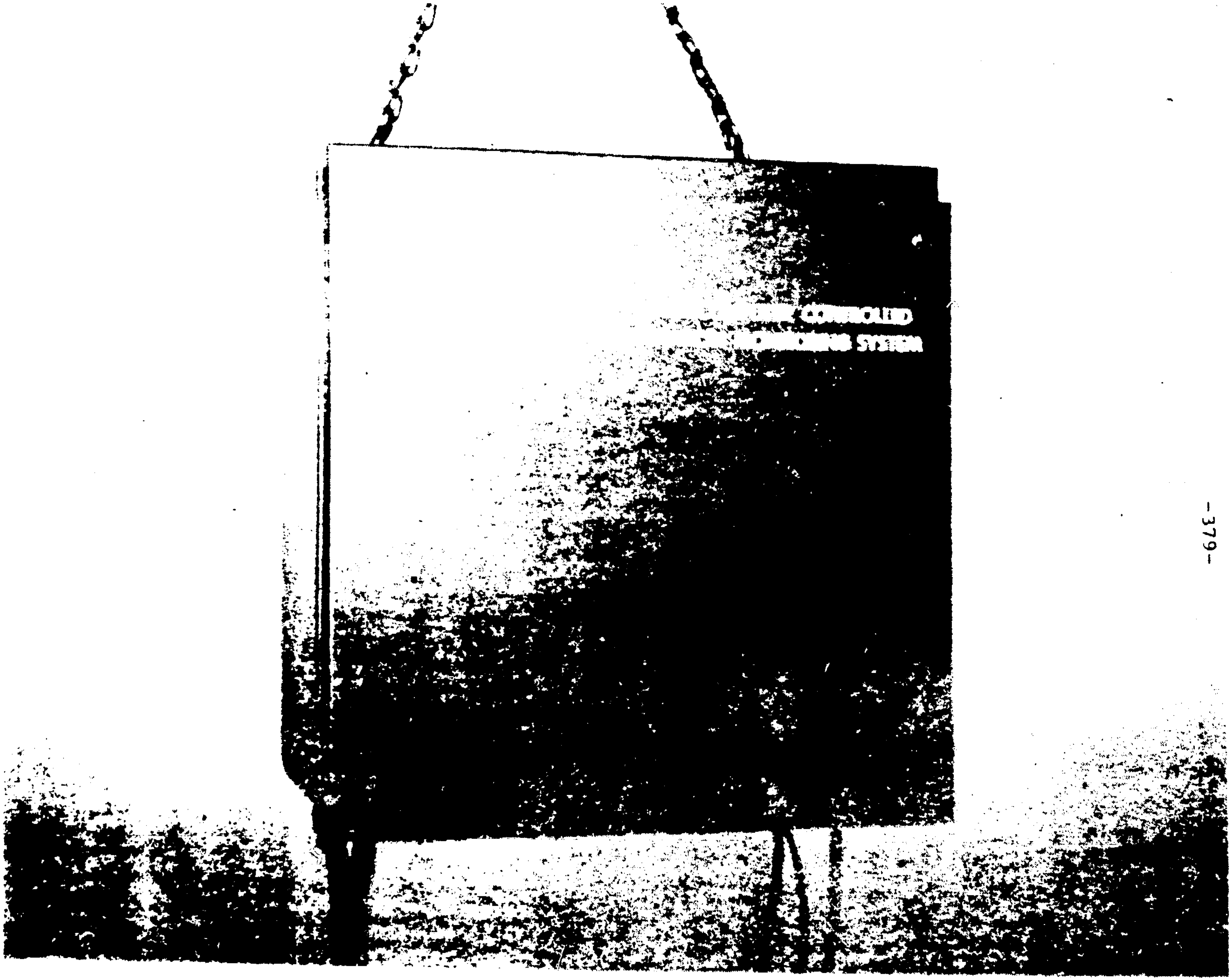
```
258 1654
 *  036 CK SRCE
 *  031 CK SRCE
 *  033 CK SRCE
 *  035 CK SRCE
 *  035 CK SRCE  +2.20E-04  µG/CC
 *  036 CK SRCE  +4.27E+00  MR/HR
    036 NORMAL
 *  031 CK SRCE  +6.62E-02  µG
    035 NORMAL
 *  033 CK SRCE  +8.55E-02  µG
    031 NORMAL
    033 NORMAL
```

Figure 15. Check Source


eberline
MICROCOMPUTER CONTROLLED
RADIATION MONITORING SYSTEM

eberline

MICROCOMPUTER CONTROLLED RADIATION MONITORING SYSTEM


CONTROLLED
SYSTEM

TECHNICAL SESSION H

PORTABLE INSTRUMENTATION

Chair: Ron Kathren

HAND-HELD HIGH RESOLUTION GAMMA-RAY SPECTROMETER

Wayne Garber and Robert F. Hill *

INTRODUCTION

From the beginning of our profession, the health physicist has sought the means to make better measurements of the radiation fields present. Progress has been from the ability to detect radiation up to the ability to identify the energy of the individual particles and photons. With the advent of the germanium detector, gamma-ray spectroscopy has progressed to the point that resolution of a few tenths of a percent are routinely achieved in off-the-shelf instruments. This kind of performance, however, has been achieved only for the large fixed laboratory type of instrument. What this has meant is that the health physicist must collect a sample and take it to the counting laboratory for analysis. Regardless of the efficiency of such a setup, this necessarily results in time delays for receiving results.

In the area of portable instrumentation there are a number of instruments that give gross activity (some based upon Na(Tl) technology which give energy information). With the perfection of the hyperpure germanium detector technology, it has become possible to cycle the germanium detector to room temperature from its liquid nitrogen operating temperature without harm. This has allowed smaller cryogenic containers to be used; hence, more transportable systems could be built. Germanium spectrometers still had not progressed to the point of being truly hand-held and fully portable. Three years ago we undertook to develop a fully portable high resolution hand-held gamma-ray spectrometer that could be used for fluorescence work in gold mines to assay the ore quality[1,2]. Shown in Figure One is the spectra detected from the instrument using a ^{109}Cd source for excitation of the gold.

---

* EG&G/ORTEC, Incorporated
100 Midland Road
Oak Ridge, Tennessee 37830 U.S.A.

[1] Hill, Robert F. and Wayne Garber, 1977, Portable X-ray Analyzer for *In Situ* Gold Qualifications, 23rd Nuclear Science Symposium, to be published in IEEE Transactions on Nuclear Science.

[2] Gedcke, D.A., R. Rolle, and P. J. D. Lloyd, 1977, A Portable Gold Analyzer for *In Situ* Ore Analysis, 26th Annual Conference on Applications of X-ray Analysis, to be published.

INSTRUMENT REQUIREMENTS

When a portable instrument design is contemplated, many new considerations must be considered, in addition to the design of a laboratory instrument. For operator convenience and safety, the following requirements must be met:

1. Won't spill liquid nitrogen on the operator no matter how he holds the dewar.
2. Easy to manipulate and point the detector part of the system.
3. Lightweight so that it is not a burden to the operator.
4. Have sufficient battery power and liquid nitrogen holding time in order to operate for a normal eight hour working shift.

In addition to the above, the system must meet the following reliability and environmental conditions:

1. Detector must continue to operate reliably after repeated thermal cycling from liquid nitrogen to room temperature.
2. System must be able to withstand the numerous shocks that it will receive that portable use implies.
3. Must be able to operate in a wide range of temperature and humidity conditions without failure or false readings.

PROBE

In the fluorescence version of the instrument, the probe assembly contains the radioisotope source with collimator and shutter, detector element, cryostat, preamplifier, display, and operating controls. Figure Two shows a schematic cross section of the probe. The high resolution non-fluorescence version has the radioactive source and shutter removed.

A cooled hyperpure germanium detector is used to obtain high resolution characteristics. Hyperpure germanium is also utilized to allow repeated cycling to room temperature of the device without harm. The entrance window and front contact of the detector has been ion-implanted for ruggedness. The input FET for the preamplifier is placed at the detector so that it is cooled; hence, improving its noise performance.

The cryostat and liquid nitrogen dewar constitute the bulk of the probe since it has sufficient capacity for a working days' operation. The probe weight is under five pounds. The fill tube is arranged so that no liquid nitrogen can spill when the cryostat is in any position. The cryostat is designed to be strong enough to protect the detector from shock, while being lightweight enough to be handled easily. The copper liquid nitrogen tank is supported on thin mylar disks and is completely surrounded by superinsulation.

The preamplifier and high voltage power supply are potted and all leads shielded or current limited to assure safety.

The high voltage power supply was placed upon the probe to eliminate any high voltage cables and hence unsafe wires if it should get severed during operation. On the fluorescence version, a five digit LED numeric readout is provided on the probe to show analyzer status or measured value. While a measurement is being made, the display shows percent of measurement complete. When the measurement is complete, the display shows the value. After the value is stored in memory, the display shows memory location.

CHEST PACK

The chest pack contains the analog pulse processing circuits, digital ratemeter, pulse height analyzers, digital data processing circuits, memory, and batteries as shown in Figure Three. The chest pack is a sealed structural foam box worn on a harness.

The analog pulse processing circuits contain the slow amplifier, fast amplifier, pile-up rejector peak detector, and biased amplifier.

The slow amplifier uses a semi-Gaussian filter with 0.5 μs shaping time constant for good energy resolution, consistent with the high count rates encountered in a fluorescence measurement.

The fast amplifier and pile-up rejector prevent processing and analysis of distorted pulses up to a counting rate of 75,000 counts per second. At the counting rate, the spectrum is broadened by a few electron volts. The biased amplifier is utilized to expand the region of interest in the spectrum.

The digital circuits sequence the analyzer operation, scale the discriminated data, and convert the data to a value of interest. A memory is also provided which minimizes data recording by the operator. An up/down counter is utilized to continuously calculate the peak-to-background ratio of the lines of interest. This raw counting data can then be converted to units of interest before being stored in the memory. The memory section is designed without a power off switch so the results stored in the memory cannot accidentally be lost. Readout of the memory and programming of the arithmetic in the digital section is by a separate support module. The support module has a small electrostatic printer from a hard copy of the data.

The Ni-Cd batteries included have sufficient power to operate the system for twelve hours. Automatic shutoffs are designed to turn the system off when the batteries are too low for proper operation. Even though the system has been shut off, the memory continues to operate for several days at this power level.

All of the electronics are placed into a structural foam box. Operating controls are accessed through a lid that is o-ring sealed. Total weight of the chest pack is under fifteen pounds. Shown in Figure Four is a photograph of the instrument with the electronics module removed from the case.

For those applications where more analyzing power is needed in the elctronics, the probe can be utilized with standard NIM electronics, thus making a transportable system. Shown in Figure Five is the high resolution non-fluorescence mode portable probe connected to standard NIM electronics. This system has a full width half maximum resolution of <570 eV at 122 keV.

INSTRUMENT APPLICATIONS

An early application of the instrument for gold ore assay was discussed earlier. Gold ore analysis is not of general interest to the health physicist. The same technique applies to the interesting heavy elements of plutonium and uranium. We had an incident several months ago that illustrates an interesting application of this instrument. A package wrapped in plain brown paper addressed to the Applications Laboratory arrived in our receiving department. The Applications Laboratory was expecting some uranium ore from one of our Sales Engineers, so they surveyed the package and decided the package was safe to open. Inside was a plastic garbage bag of powdered material they decided was the Colorado uranium ore. To make a further check, this bag of material was placed upon a large Ge(Li) coaxial detector. They obtained a spectrum with many lines in it which did not immediately identify the material. It was also observed that the spectrum from this detector was independent of the sample. At this point, the health physicist was called in to access the danger. The material was identified as the expected ore.

Had we had the instrument that we have been describing, this incident could have been avoided. Using the instrument as shown in Figure Five, the spectrum shown in Figure Six was obtained. There are some lines in this low energy gamma-ray spectra that indicate the presence of uranium.

Even though the system performed well as a gamma-ray spectrometer, the time to make the measurement was too long for a practical field health physics measurement. Next it was decided to see what results would be obtained if the box was fluoresced. Using the system shown in Figure Four, the fluorescence spectra shown in Figure Seven was obtained. We have shown the full spectrum obtained rather than a final calculated peak-to-background ratio value as the instrument gives in field operation.

Cobalt 57 was utilized as the excitation source. In this twenty second run, the presence of uranium is clearly visible. Using this technique, we have been able to detect 400 ppm of uranium. Brodzinski and Wagman state that 100 ppm should be attainable[3].

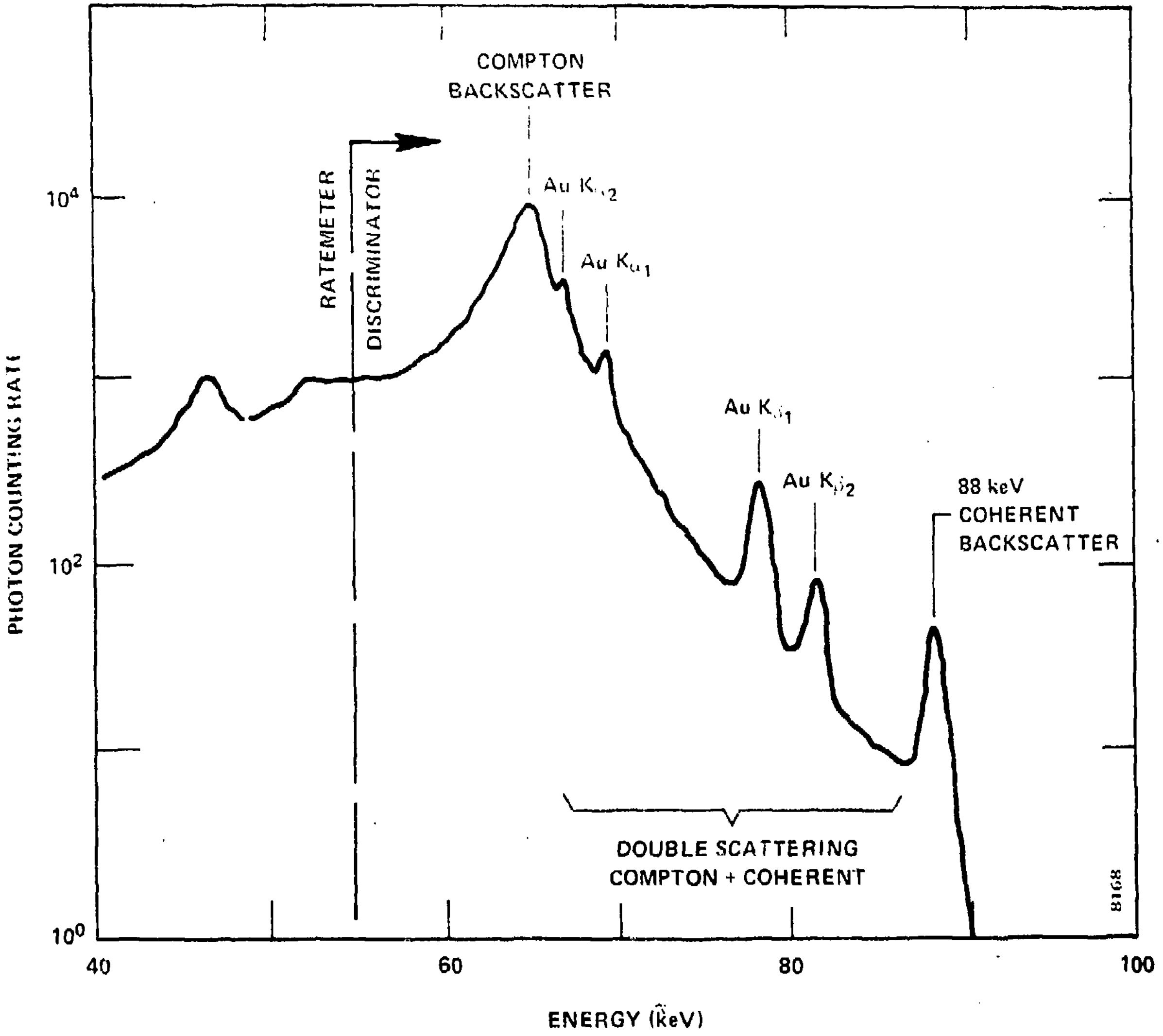
---

[3] Brodzinski and N. A. Wagman, <u>Californium 52 *In Situ* Activation and Photon Deposit Evaluation</u>, BNWL-SA-5561.

SUMMARY AND CONCLUSIONS

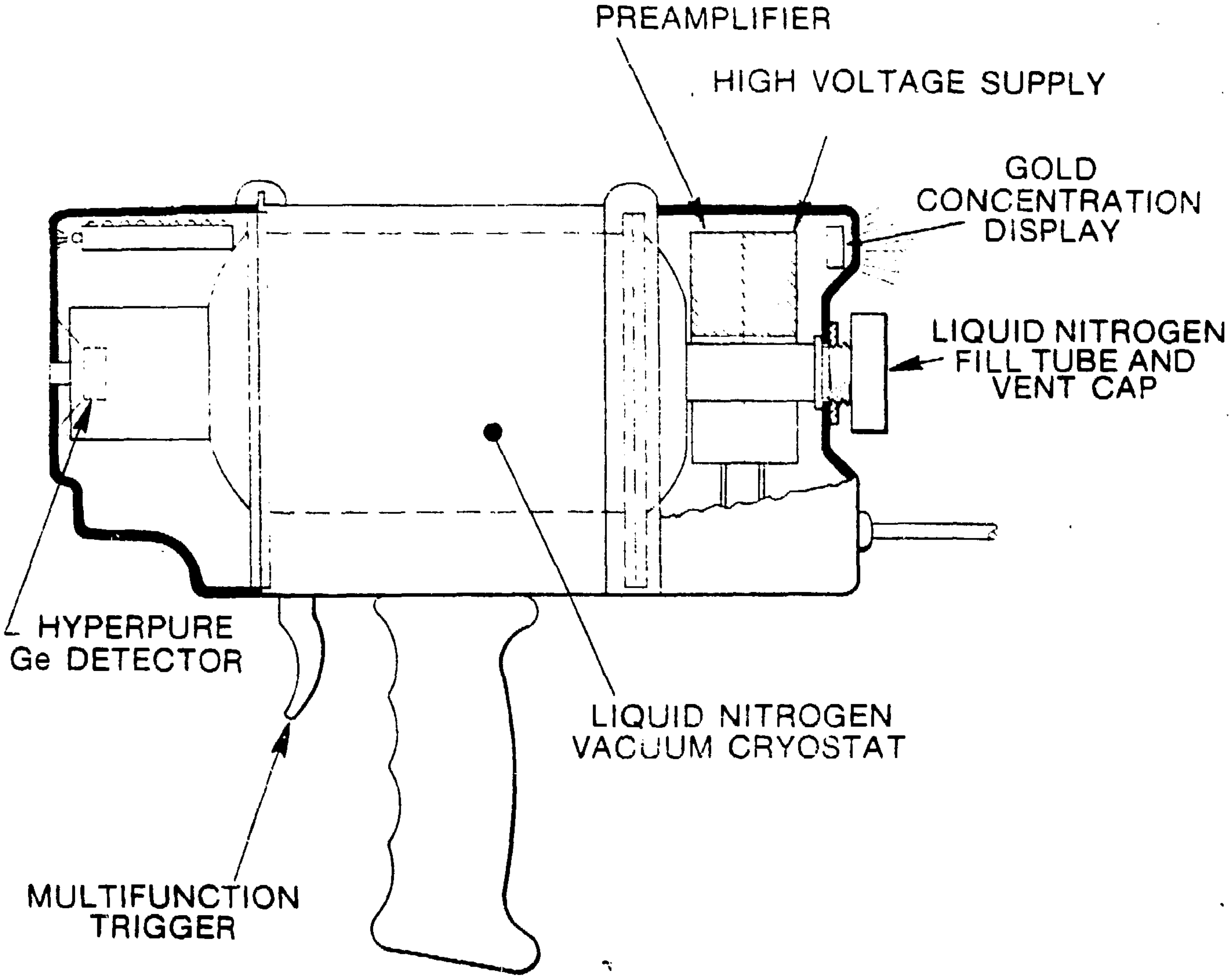
We have discussed both a fully portable and a semi-portable high resolution gamma-ray spectrometer. These instruments give the resolving capabilities that are inherent to germanium spectrometers and also give the portability needed for the applied health physicist.

The instrument has been shown to be usable as either a gamma-ray spectrometer or as an X-ray fluorescence spectrometer. We believe there are many other applications for this type of instrumentation in the health physics area, nuclear materials safeguards, and general materials analysis.



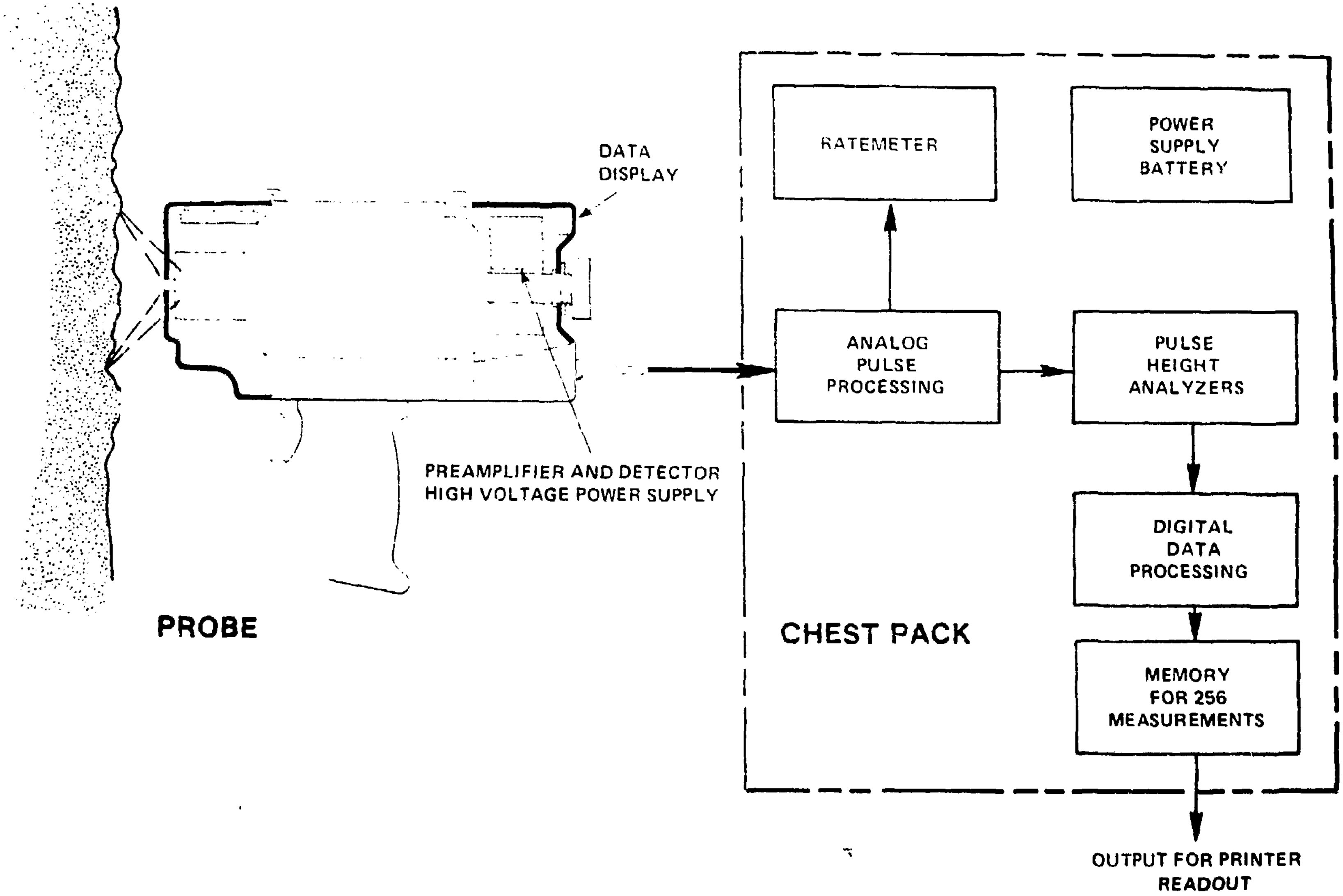
DETECTED SPECTRUM

Figure One



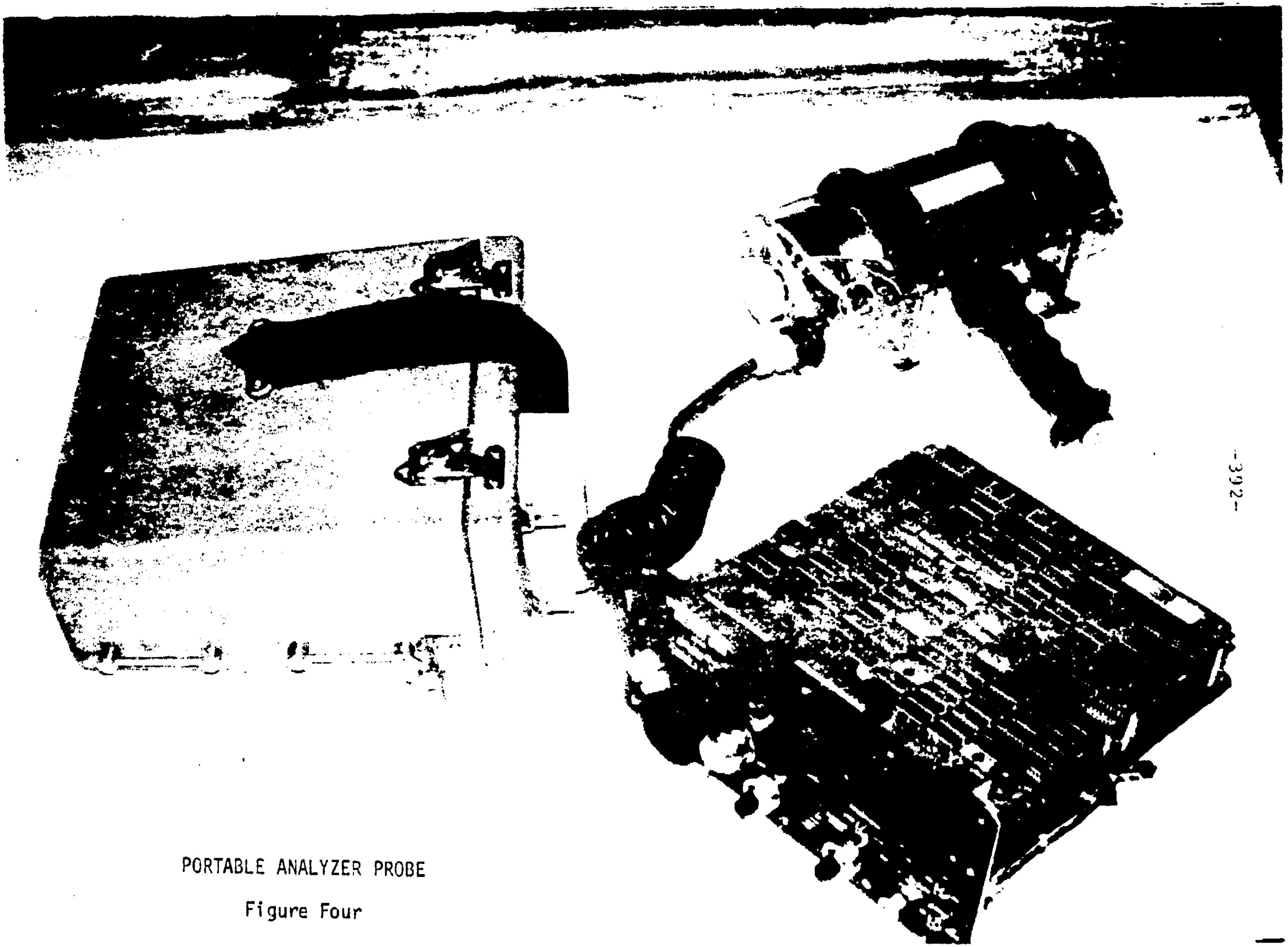
PORTABLE ANALYZER PROBE

Figure Two



ANALYZER ELECTRONICS

Figure Three

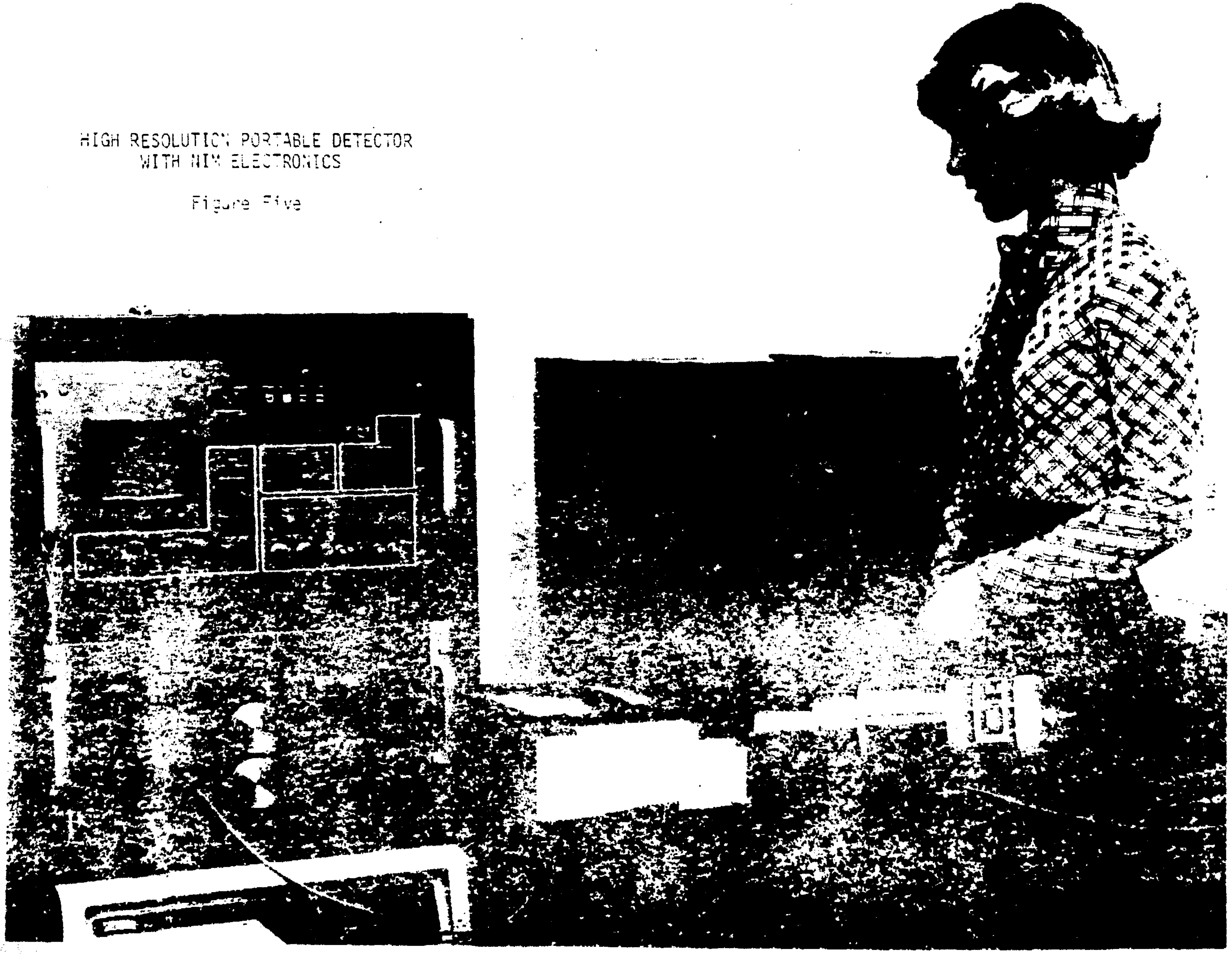


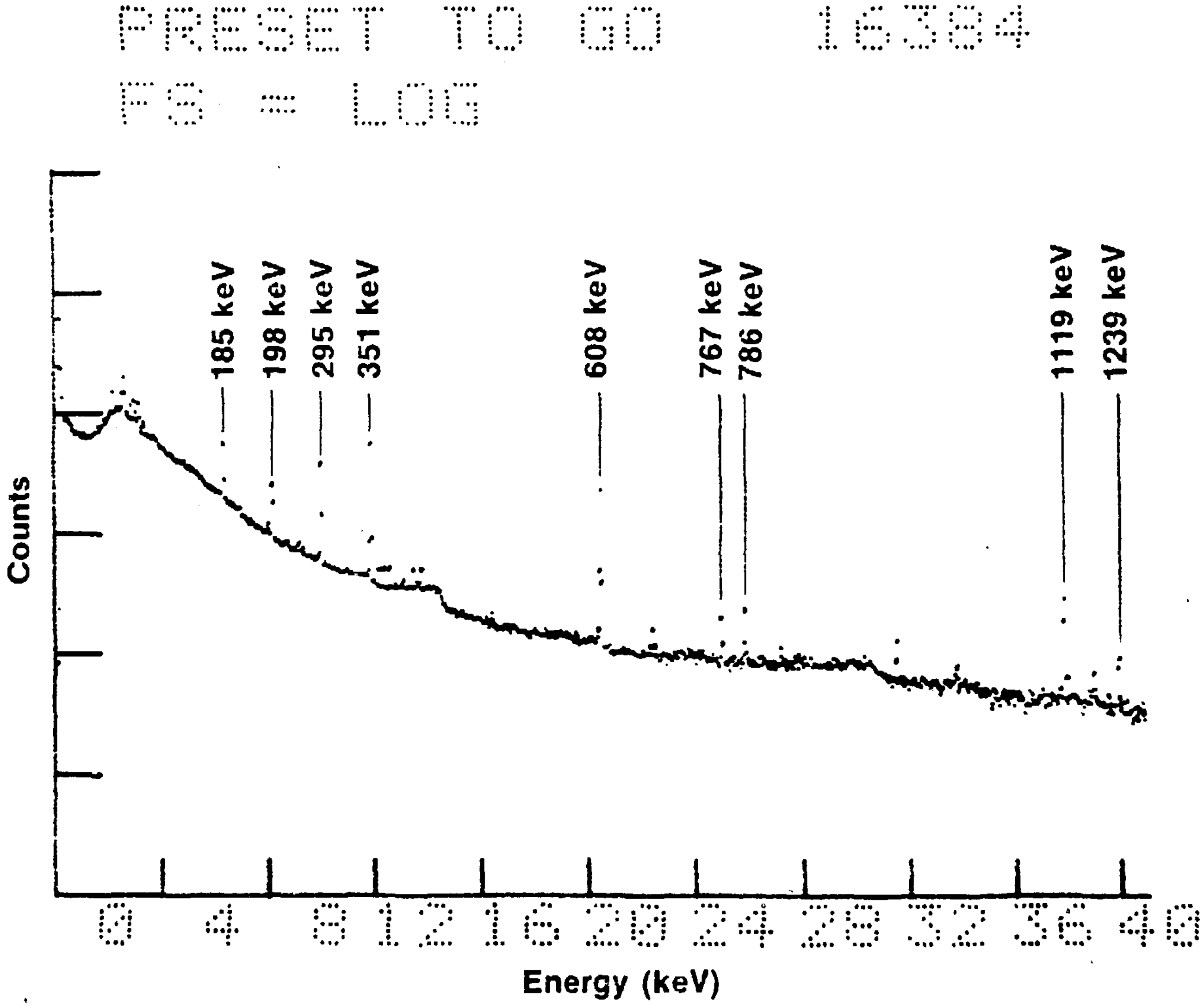
PORTABLE ANALYZER PROBE

Figure Four

HIGH RESOLUTION PORTABLE DETECTOR
WITH NIM ELECTRONICS

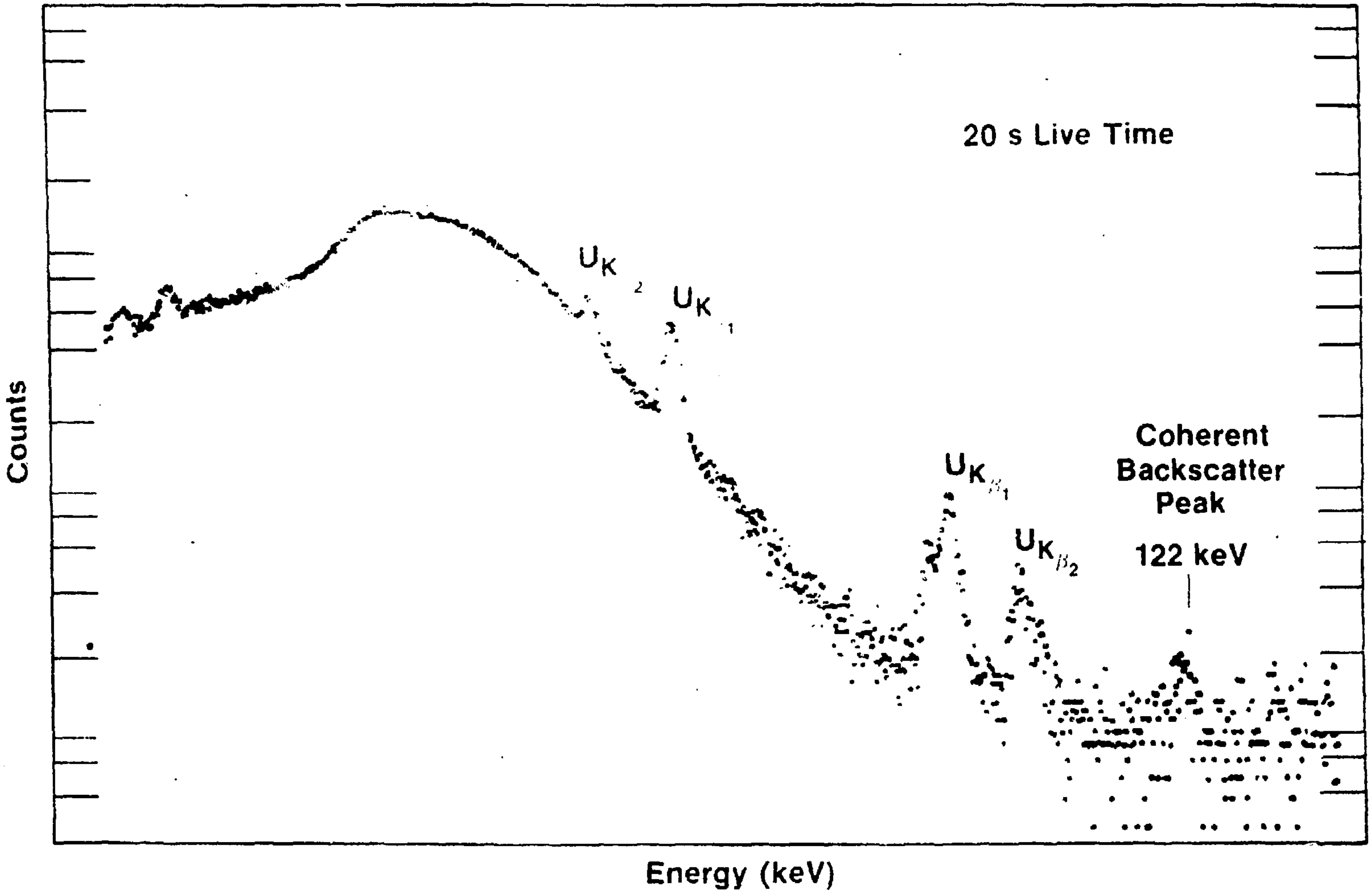
Figure Five





LOW ENERGY GAMMA-RAY SPECTRUM OF URANIUM ORE

Figure Six



FLUORESCENCE SPECTRUM OF URANIUM ORE SAMPLE

Figure Seven

GAMMA-RAY SPECTRAL DETERMINATIONS WITH A PORTABLE ANALYZER

James S. Eldridge

Nuclear and Radiochemical Analysis Section
Analytical Chemistry Division

and

Thomas W. Oakes

Environmental Surveillance Section
Industrial Safety and Applied
Health Physics Division

Oak Ridge National Laboratory*
Oak Ridge, Tennessee 37830

Abstract

A portable sodium iodide spectrometer designed at Lawrence Livermore Laboratory has been used at the Oak Ridge National Laboratory in a variety of field applications. The spectrometer is a battery-operated system complete with a 5x5 cm NaI(Tl) detector, 1024 channel memory divisible into four quadrants, special summation and calibration circuits, and a telemeter output for data transmittals. The portable spectrometer has been used to make *in situ* measurements around a burial ground. Typical spectra of ^{137}Cs and ^{60}Co were easily discernible at one such site. Uptake in vegetation near a sealed trench could be observed by placing the NaI(Tl) probe in the foliage. The extent of the spread of low-level contamination of ^{60}Co was determined in an area that had previously been decontaminated. The unique response of the two ^{60}Co gamma-rays could be obtained by subtracting the local background response within the computing spectrometer. The ability to perform qualitative identification and to assess hazards in emergency situations has been demonstrated.

Introduction

Environmental surveillance activities at a large multidisciplinary institution like Oak Ridge National Laboratory require radioactivity determinations of many types. Qualitative determinations are an essential facet of the routine surveillance programs as an aid in quality assurance procedures (Oa77). Such determinations allow the selection of representative samples in case of mixed radionuclide contamination sites.

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*Oak Ridge National Laboratory is operated by Union Carbide Corporation for the Department of Energy.

Emergency response plans for contamination incidents include procedures whereby early identification of the released radionuclides may be made (E177). The ability to perform gamma-ray spectral measurements for qualitative identifications at the contamination scene results in a more rapid hazard assessment.

As an adjunct to both the surveillance programs and the emergency response capabilities, a specially designed, computing gamma spectrometer was acquired. This portable, complete spectrometry system was one of thirteen such systems built at Lawrence Livermore Laboratory under the auspices of the United States Energy Research and Development Administration (M^{c}76). The purpose of the current investigation is to obtain operational experience and to determine the applicability of the computing gamma spectrometer to field measurements.

Equipment

The computing gamma spectrometer is a second generation design of a truly portable gamma-ray spectrometer. It is intended to be used anywhere a man can carry it. The device is a completely self-contained unit with detector, power-supply, amplifier and 1024 channel memory housed in a luggage-type container with dimensions of 20 x 43 x 56 cm. The spectrometer will fit under the seat of a commercial airplane and has a weight of 17Kg with its included accessories. The special battery system is rechargeable with a nominal operational life of 15 hours but only 8 hours with maximum use of the display at 25°C. The battery life is decreased at colder temperatures.

The spectrometer system includes a 5x5 cm NaI detector housed in a special compartment. The analyzer memory of 1024 channels may be operated as a single group or as four 256-channel subgroups. The system design permits addition, subtraction, and overlay of selected quadrants. Three separate display markers may be positioned within the spectral regions for energy determinations and for defining summation zones.

A standard model SX70 Polaroid camera is included to record the spectral responses and to provide a read-out of the eleven light-emitting diode (LED) displays of the experiment parameters. These LED registers provide an indication of the time of day, day of year, live time, quadrant displayed, sample identification, and integral counts between the display markers. In addition, the counts per channel and the energy of the marker channels are displayed on the LED registers.

A special telemetry output via an audio coupler provides access to the channel-by-channel spectral data as well as the LED registers. This output mode requires implementation with a modem to permit conversion to computer or data terminal. We have not added the necessary peripheral equipment to fully utilize this feature at the present time.

Figure 1 shows the operational panel of the portable spectrometer. The 3x5 cm display and LED registers are seen in the black rectangle on the left. The center portion shows the 36 keyboard, dead time meter, and battery meter. The detector compartment (cover has been removed) on the right contains the 5x5 cm NaI in a white shock and thermal shield. Other operational controls for the analyzer, amplifier, and clocks are contained within this compartment.

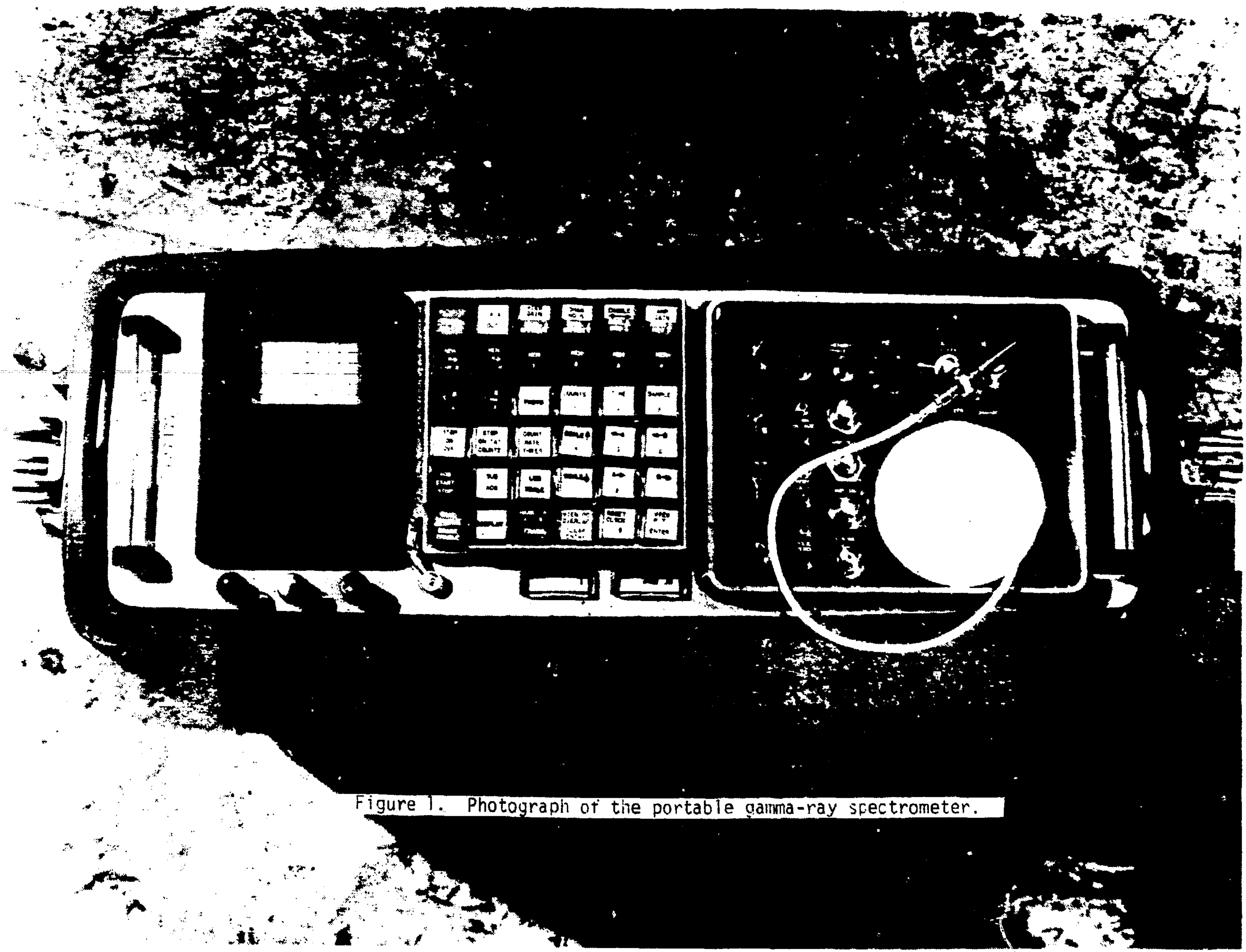

Figure 1. Photograph of the portable gamma-ray spectrometer.

Field Studies

To date, the portable spectrometer has been utilized for three field expeditions. The first trial involved the study of the intensity and quality of gamma radiation in the vicinity of one of the older sealed trenches at the Oak Ridge National Laboratory (ORNL) waste disposal site. Gamma-ray spectra were acquired around the periphery and upon the asphalt cap of the sealed trench. At this old burial site, the major gamma spectral response was from varying proportions of ^{60}Co and ^{137}Cs. By placing the detached sodium iodide probe directly in the foliage of trees growing at that location, it was possible to demonstrate the uptake of both ^{60}Co and ^{137}Cs in those trees. The ambient temperature at the time of the measurements was about 32°C and the relative humidity was very high. The spectrometer system operated without a problem. Difficulty was encountered, however, in reading the LED registers in the brilliant sunshine.

The second field investigation involved the qualitative determination of the major radioactivity in a stainless steel neutron chopper destined for shipment to an overseas laboratory. The chopper was contained in a wooden shipping crate of 13x13x200 cm. The crate was temporarily stored in a high radiation level vault at a reactor site at ORNL. The portable spectrometer was positioned about 6 meters from the unshielded crate, perpendicular to its long axis. By means of such spectral determinations, it was possible to determine that essentially all of the residual γ-activity of the chopper was due to induced ^{60}Co. These field studies were made in an air-conditioned facility with normal room lighting. No operational difficulties were encountered.

The third and most recent field application of the portable spectrometer involved its use in a radiation survey at a contaminated facility located at the eastern boundary of Oak Ridge, Tennessee. A survey of inlets along the eastern shore of Melton Hill Lake (Oak Ridge city limits) was undertaken to determine the location of any residual radioactivity from earlier decontamination efforts. The inlets were reached by boat launched from the Oak Ridge Marina, about 800 m downstream. The portable spectrometer was carried in the open boat for the purpose of on-site examination of sediments from the inlet bottoms. The ambient temperature at the time of the measurement was about -8°C and the relative humidity was very low. The spectrometer operated satisfactorily, but little information was obtained due to a lack of contained radioactivity. Samples were collected for subsequent analysis at ORNL.

Figure 2 is a detailed map of the eastern boundary of Oak Ridge, Tennessee, showing the sampling locations (1-8) in the upper right corner. Locations 1 and 2 show the inlets reached during the boat survey.

Subsequent visits to the area were made by land, and field surveys and sample collections were obtained at locations 3-8. The source of the contamination was the building indicated by the black symbol just south of the numeral four in the figure. Contamination in the vicinity of the facility is mainly ^{60}Co with smaller quantities of ^{137}Cs. Radioactivity has been transported from the facility to Melton Hill Lake by waters of Braden Branch which flow from east to west through the property. Gamma-ray spectral determinations were collected at all sampling locations with the portable spectrometer. Qualitative determinations performed in situ were valuable adjuncts to the quantitative determinations subsequently performed in the laboratory on samples collected in the field.

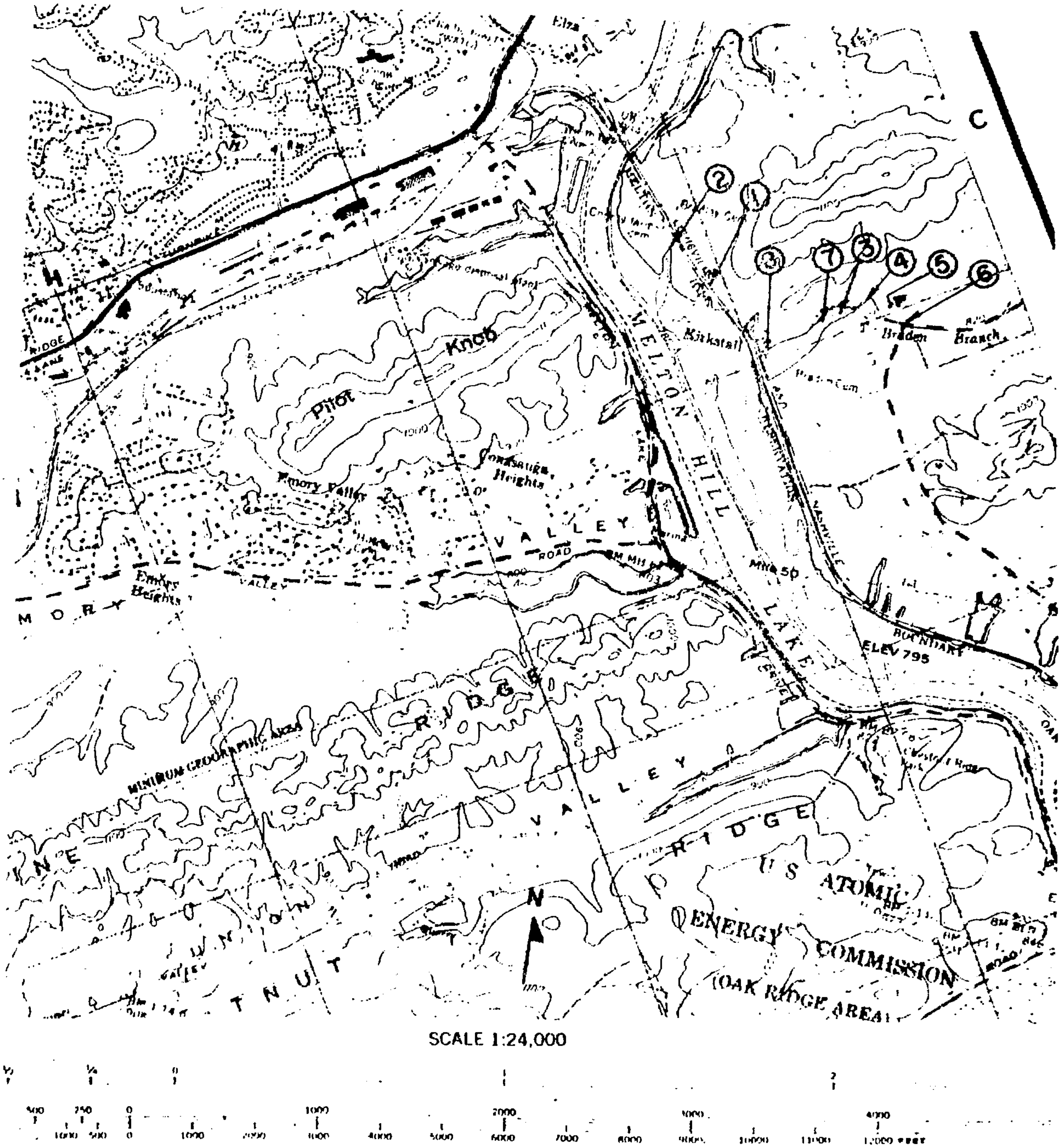

FIGURE 2 Map of eastern edge of Oak Ridge showing the sampling locations in the field studies

Table I shows quantitative results of laboratory measurements along with sample descriptions.

Discussion

The portable spectrometer has been found to be an extremely useful field instrument. It performed satisfactorily under field conditions where the temperature ranged from +32°C to -8°C along with relative humidity extremes. Difficulty was encountered in carrying and operating the instrument in rugged terrain when only one person was present. The package weight, 17Kg, becomes burdensome after a period of 20-30 minutes when it is suspended over the shoulder of the operator. With the multitude of accessories e.g., camera, film, adapter, etc., it is desirable to have an assistant for all field determinations. The addition of a portable transport device such as a modified golf cart would be a useful accessory to relieve the physical effort required for transport. In addition, the cart might prevent accidental contamination of the instrument package in those cases where the device is used on contaminated areas. The package could be operated on the cart with the sodium iodide detector removed from its well.

Table I. Laboratory Analysis of Samples Collected During the Third Field Exercise

Sample Location*	Description	^{60}Co**	^{137}Cs**
1	Sediment at inlet	1.4	-
2	Sediment at inlet	<0.3	-
3	Wet sediment from center of Braden Branch	515	-
3	Soil from bank of Braden Branch	75	1.8
3	Water sample from Braden Branch (pCi/l)	200	1000
4	Water sample from spring north of Braden Branch (pCi/l)	<1	<1
4	Water plant from spring north of Braden Branch	<0.2	<0.2
5	Cow manure from grazing beef cattle***	5.4	0.8
6	Grass at entrance gate	0.3	1.3
7	Sediment from entrance of Braden Pond	58	-
8	Sediment from exit of Braden Pond	9.4	-

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*Refers to sampling locations marked on Figure 2.

**Units are pCi/g unless noted.

***Contained 1.9 pCi/g ^{95}Nb, 1.2 pCi/g ^{144}Ce and other fallout nuclides.

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H

TWO NEW PORTABLE SURVEY INSTRUMENTS: THE FIELD PHOSWICH DETECTOR AND THE WEE PEE WEE

by

C. John Umbarger and Michael A. Wolf
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ABSTRACT

As part of a continuing program to upgrade the health physics survey instrumentation at Los Alamos, we have recently developed two new portable instruments. The first is a fully portable phoswich detector for low energy photons from small amounts of plutonium and americium in the field. The instrument has a background that is 2-3 times lower than an equivalent thin NaI detector. The instrument features an aural popper, analogue rate meter, and timer/scaler with liquid crystal display. The second instrument, called the "Wee Pee Wee," is an alpha air proportional probe with complete electronics and readout package mounted on the probe itself. The entire package has a mass of 0.66 kg (1.45 lb) and is carried and operated in one hand. For monitoring shoes and other places where it is difficult to read the count-rate meter, the meter is made detachable for clipping to a shirt pocket, etc. An audio popper, range scales to 100 K cpm, and visual checks for high voltage and battery levels are also included.

INTRODUCTION

The first part of this paper describes the design of battery powered electronics to allow a phoswich detector to operate as a true portable field survey instrument. In the second part of this paper we describe the design of the "Wee Pee Wee," which is based on new miniturized electronics coupled to a standard alpha air proportional probe for the monitoring of alpha contaminated surfaces. The miniturized electronics, which are directly applicable to other types of detectors, allow a significant advance in portable instrumentation where size, weight, and battery life are paramount.

I. The Field Phoswich Detector

A. Background

For some time the standard instrument for detection of low energy photon emitters in the field has been the FIDLER detector[1)], a thin, large area (126 cm^2), NaI(Tℓ) detector with portable electronics. However, Compton events from high energy photons are one major source of detector backgrounds and place a lower limit on the detectability of true low energy events. A popular way of reducing this background is the use of a phoswich crystal[2] and pulse shape analysis (PSA)(3,4). This technique not only reduces Compton backgrounds, but reduces phototube noise as well[5]. Such systems have been used in the laboratory for such diverse measurements as small sample counting and in-vivo chest monitoring.

We began operating two truck-based phoswich monitoring systems[6] at Los Alamos two years ago for field survey use. The systems have operated very reliably in all extremes of weather conditions and have proved themselves superior to the field FIDLER in detection limits. Their one drawback, however, was their dependence on the trucks via a long (30m) umbilical cord. The phoswich instrument described here operates independently of the trucks and is fully portable.

B. Phoswich Detector Characteristics

The physical design for the phoswich detector (commercially available) is shown in Fig. 1. As show in the figure, the phoswich is composed of two different scintillators coupled (one upon the other) to a single photomultiplier tube (PMT). The two crystals have sufficiently different phosphor decay times to allow the spatial origin of the photon interaction to be determined using PSA. Hence, it is possible to differentiate between desired low energy photons that stop only in the thin front crystal and those gamma rays, though similar in energy, that interact through other processes (such as Compton scattering) and deposit energy in both crystals. The detectors used in this work have a front 1 mm thick NaI(Tℓ) and a back 38 mm thick CsI(Na). The crystals have a 96 x 96 mm square face. The phosphor decay times for the front and back crystals are 0.23 μs and 0.63 μs,[7] respectively. Shown in Fig. 2A are the pulse shapes from the two different crystals. These pulses are then amplified and shaped by an integrating preamplifier giving the pulse shapes shown in Fig. 2B. It is this difference in shape that is sensed electronically and used for event discrimination and, hence, background reduction.

C. Phoswich Electronics: General Description

The major problem in implementing the phoswich for field work was the replacement of the large amount of laboratory electronics normally used[6,8-13] for this purpose. Besides miniturizing the electronics package, the system also had to be made rugged for typical field conditions and have low power consumption.

The pulse analysis method chosen is basically the same as that used in the ORTEC 458[14], which measures the fall time of an input pulse from a shaping amplifier. However, the electronics in this design are original. Fig. 3 is the Block diagram for the instrument. The basic method of pulse shape analysis is to shape the linear pulses with a simple delay line amplifier, catch the peak value in a pulse stretcher, attenuate this voltage to 90% and 10%, then compare these voltages with the decay time of the linear pulse which has been slightly delayed to allow for circuit delays and settling times. The result is a logic pulse whose width is equal to the fall time of the shaped linear pulse (rise time of the integrated pulse)[9]. This pulse is fed to a time-to-amplitude converter (TAC) and converted to a pulse whose height is determined by the original decay time. This TAC pulse, as well as the linear pulse, is analyzed by single channel analyzer (SCA) windows. The SCA outputs are "anded," so if a pulse meets both windows, a logic pulse is sent to a scaler/timer, a log rate meter, and an aural popper[15]. The popper emits a "squeak" for each true NaI event with the frequency or pitch of the squeak a function of the average count rate, hence facilitating qualitative surveys without constant reliance on the count rate meter. The details of all of the electronics contained in Fig. 3 have been reported elsewhere[16].

D. Phoswich Physical Arrangement

The phoswich field survey instrument is shown in Fig. 4. The entire instrument has a mass of 8.8 kg. The detector package itself has a mass of 4.4 kg, mostly due to the phoswich detector. The remaining electronics are split between the front and back of the operator. The rate meter and timer/scaler package is located in front as shown in Fig. 4. The battery pack and main electronics are carried in the backpack. The four component packages of the system are shown in Fig. 5. The battery pack is on the extreme right while the main electronics package is on the extreme left. The readout module and detector packages are in the middle. Shown on the main electronics package are calibration controls for the TAC and energy SCA's as well as the phototube high voltage. Signal output connectors are also provided for signal monitoring. The detector package can be lowered, via a longer detector - backpack cable, into deep holes and ditches while still being monitored from the surface since the readout module stays with the operator.

E. Phoswich Performance

The instrument is easy to calibrate and adjust in the field using the different signal outputs available to the user. If a portable oscilloscope or, better yet, a field pulse height analyzer (PHA) is connected to the gated TAC output, a full time spectrum (counts vs time) can be collected. Fig. 6 shows such a spectrum from a ^{241}Am source which emits only low energy photons. The regions characteristic to the NaI and CsI crystals are indicated. For field surveys of low energy photon emitters, the NaI region is selected via a TAC single channel analyzer. Fig. 7 shows the gated linear energy output from the same ^{241}Am source. Here, the spectrum displays counts vs photon energy. The two main photon energy regions of ^{241}Am are clearly indicated. To select any particular energy region, the energy single channel analyzer is adjusted while monitoring this output. Because of the added TAC electronics, the phoswich sensitivitiy is reduced ∿11% compared to an equivalent FIDLER.

The instrument has been tested for sensitivity variations as a function of temperature. Between 0-40 degrees centigrade, the main limitation is the detector itself. The measured gain changes followed very closely those known[7] for the scintillator crystals. With the instrument calibrated at 25°C, a variation of ±15°C produced less than a 25% change in sensitivity. This includes the temperature effect on the batteries and regulated power supply. Battery life is approximately 16 hours. The "gel cel" batteries can be completely recharged overnight. The charger is included in the system.

Background count rates for the field phoswich system are reduced significantly compared to a comparable thin NaI system. Table 1 shows typical backgrounds for the two systems, both totally unshielded. As shown, the improvement over the FIDLER is $\sim$2.6 in the energy region 7-70 keV. The actual reduction depends on the energy region chosen. The FIDLER used in the comparison was somewhat thicker (1.6 mm vs 1.0 mm) than the NaI section used in the phoswich. This would add some extra background to the FIDLER spectrum. However, the FIDLER had a much thicker quartz light pipe (5 cm vs 3 mm) than the phoswich. Also, NIM electronics were used on the FIDLER as opposed to portable instrumentation. Both of these factors would favor the FIDLER. Therefore, we feel the comparison is valid.

Table 2 shows that the detection limit of the field phoswich is a function of many variables. In a field search mode, where we feel we can detect an added signal equal to area background, we can detect activity approximately 2-3 times lower (better) than a comparable field FIDLER. For typical Los Alamos soils, we can detect $\sim$1 nCi/g soil for ^{239}Pu and less than 0.1 nCi/g for ^{241}Am. Using the timer/scaler and long counting times, we can now work at the 3σ level, or similar, of background. In this instance the detection limits improve dramatically.

Since July 1977, the field phoswich has been used extensively in routine surveys at Los Alamos. The continued field use has pointed out several modifications that were needed to improve the ruggedness of the instrument. Most of these involved the packaging of the system for weather tightness and shock mounting. The system has proved reliable in all types of field and weather conditions.

The Wee Pee Wee

A. Detector

The common alpha air proportional probe is commercially available and is routinely refurbished at Los Alamos as necessary (e.g., electrodes restrung, mylar cover window replaced, etc.). The probe has a surface dimension of approximately 23 x 5 cm with an active area of $\sim$70 cm^2. The intrinsic counting efficiency of the probe is approximately 33%. The electronics, described below, are general in nature and are applicable to other detector types commonly found in portable survey instrumentation.

B. Electronics

The electronics, shown schematically in Fig. 8, utilizes hybrid circuitry developed in the Electronic Division at Los Alamos. It is the hybrid circuit that is responsible for the small size of the electronic package as well as its lower power consumption. The hybrid circuit contains the components shown within the box in Fig. 8. The high voltage supply is efficient for low power applications and combines an inverter driven by a separate oscillator (U4A in Fig. 8), a voltage multiplier, and a high impedence feedback circuit to regulate the output. The separate oscillator enables the frequency and duty cycle of the high voltage drive waveform to be varied for most efficient operation. The input amplifier is a conventional charge sensitive FET circuit followed by a low power operational amplifier (U1A). A low power comparitor (U2) with adjustable threshold acts as a discriminator and shaper. This drives two one-shot multivibrators (U3), one with its pulse width adjustable by the range switch and serving as the ratemeter. The three ranges are 1 K, 10 K, and 100 K cpm. The second "one-shot" gates a CMOS oscillator (U4B) which drives a ceramic transducer acting as an audible beeper for each count. A meter switch selects either one of the three ranges, relative probe high voltage, or relative battery voltage. An earphone jack is also provided. The entire package is powered by a single 5.6 volt mercury battery and draws approximately 5 ma. Battery life is 100 hours continuous use, much longer with intermittent use.

C. Physical Arrangement and Performance

The entire instrument has a weight of 0.66 kg and is carried and operated in one hand. Probe changes in the field are easy and take only a few seconds and no tools. The probe is held to the electronics package with three spring flanges and our standard Series "C" connector that mates, with a light press fit, with its female counterpart (without the locking ring) located in the bottom of the electronics case. As shown in Fig. 9, the instrument has a detachable meter on a coiled miniture "phone" cable for use when the probe is placed in positions such that the normal meter position is difficult to see. The meter snaps into the end of the electronics case and is also easily removed without tools. The calibration pots are located under panel holes for ease of calibration. Our latest prototype, not shown in Fig. 9, has these calibration ports located behind the removable meter in order to improve weather tightness. A separate battery compartment located on the back of the electronics package is easily opened by the removal of two screws. If a NiCd battery were used instead of the mercury battery, an external battery charger could be included. We feel this is not necessary with the 100 hour continuous-use battery life of the mercury cell.

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TABLE 1. TYPICAL BACKGROUND COUNT RATES FOR UNSHIELDED FIDLER AND PHOSWICH DETECTORS

ENERGY REGION (keV)	FIDLER[A] BACKGROUND (cpm/cm^2)	PHOSWICH[B] BACKGROUND (cpm/cm^2)	RELATIVE (FIDLER/PHOSWICH) BACKGROUNDS
12-29	3.5	1.4	2.5
12-40	5.9	2.6	2.3
51-70	10.6	3.8	2.8
7-70	20.8	8.0	2.6

[A] FIDLER: 1.6 mm x 127 mm dia NaI(Tℓ); 0.25 mm (10 mil) Be Window.

[B] Phoswich: 1.0 mm x 96 mm x 96 mm NaI(Tℓ) and 38 mm thick CsI(Na);
0.25 mm (10 mil) Be Window.

TABLE 2. FIELD PHOSWICH DETECTION LIMITS

DETECTION LIMIT (D) = F{ISOTOPE, SAMPLE MATRIX, ENERGY WINDOW, DETECTOR CHARACTERISTICS, AREA BACKGROUND, AND SAMPLING TIME}

A. FIELD SEARCH MODE

1. D DEFINED @ SIGNAL = BACKGROUND.
2. THEREFORE, D(PHOSWICH) ~2-3 LOWER (BETTER) THAN D(FIDLER).
3. FOR TYPICAL LOS ALAMOS SOILS, D(PHOSWICH) ≤ 1 NCI/G, ^{239}PU
   ≤ 0.1 NCI/G, ^{241}AM

B. TIMER/SCALER MODE

1. D DEFINED @ 3σ, ETC., OF BACKGROUND.
2. THUS D ≪ 1 NCI/G, ^{239}PU, ETC.

FIGURE CAPTIONS

FIG. 1. Physical arrangement for typical phoswich detector[7].

FIG. 2. A. NaI(Tℓ) and CsI(Na) light output shapes.

B. Same output pulses from preamplifier.

FIG. 3. Electronic block diagram for the field phoswich systems. The circled numbers refer to pulse shapes described in Ref. 16.

FIG. 4. Field phoswich instrument in use. Readout and controls are worn on the operator's chest. Calibration controls are housed in a module in the backpack along with the batteries.

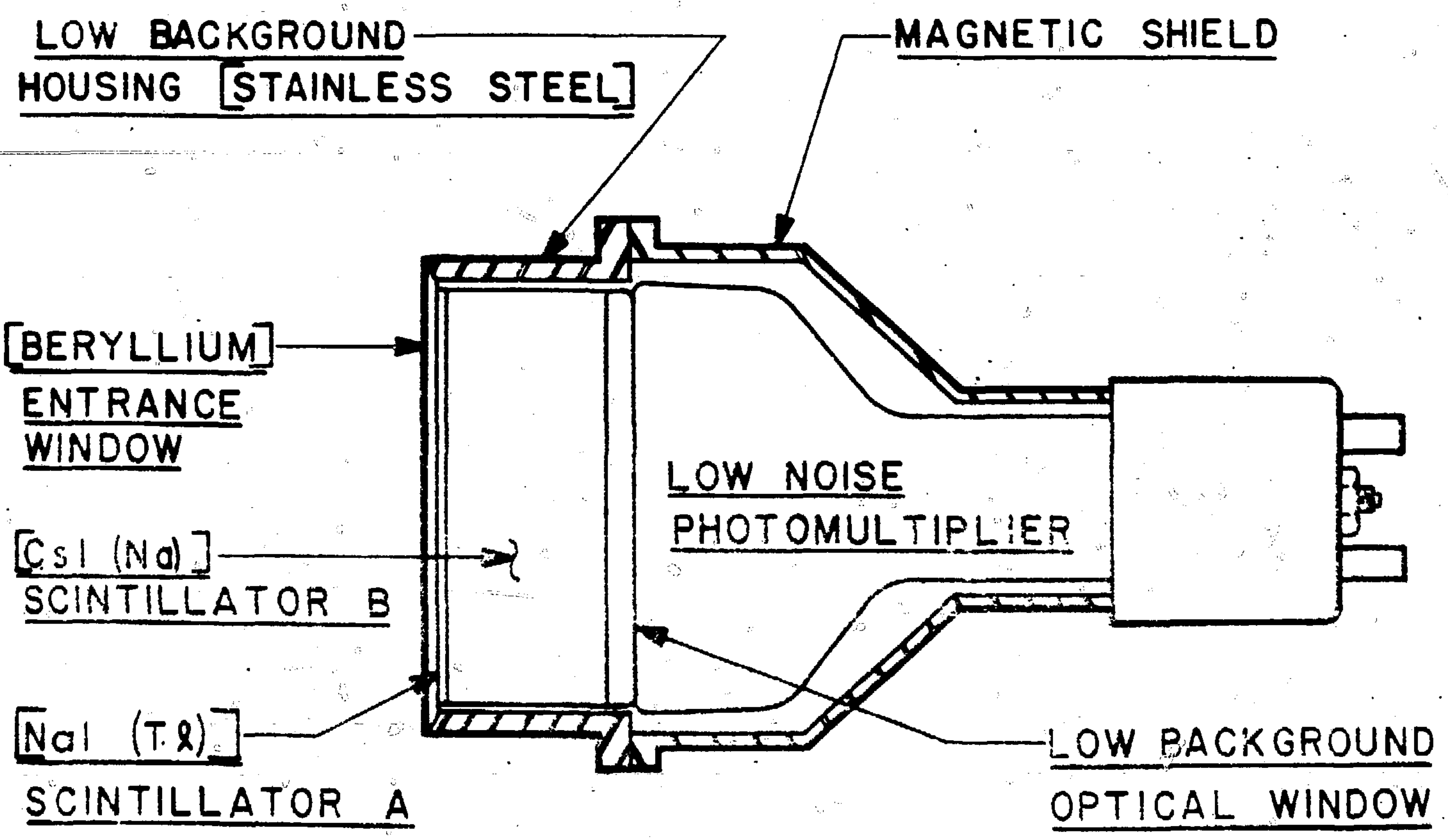
FIG. 5. Components of the field phoswich system, from left to right; (1) main electronics package showing calibration pots and signal output connectors; (2) data read out modual showing rate meter and covered liquid crystal display for timer/scaler; (3) phoswich detector package; (4) battery pack.

FIG. 6. Pulse height analyzer spectrum of the phoswich TAC output from a ^{241}Am source. The NaI and CsI regions are labeled. The time axis is only relative.

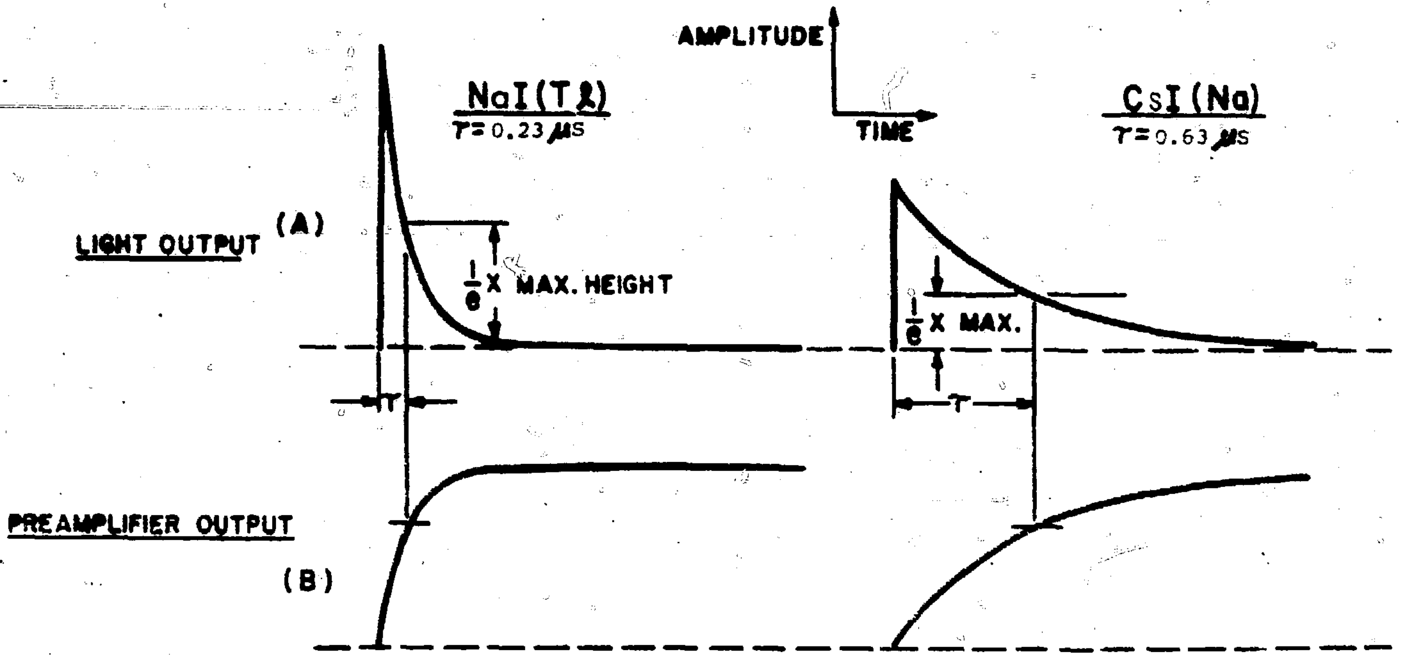
FIG. 7. Pulse height analyzer spectrum of the phoswich gated linear energy output from the ^{241}Am source. The energy axis is in keV.

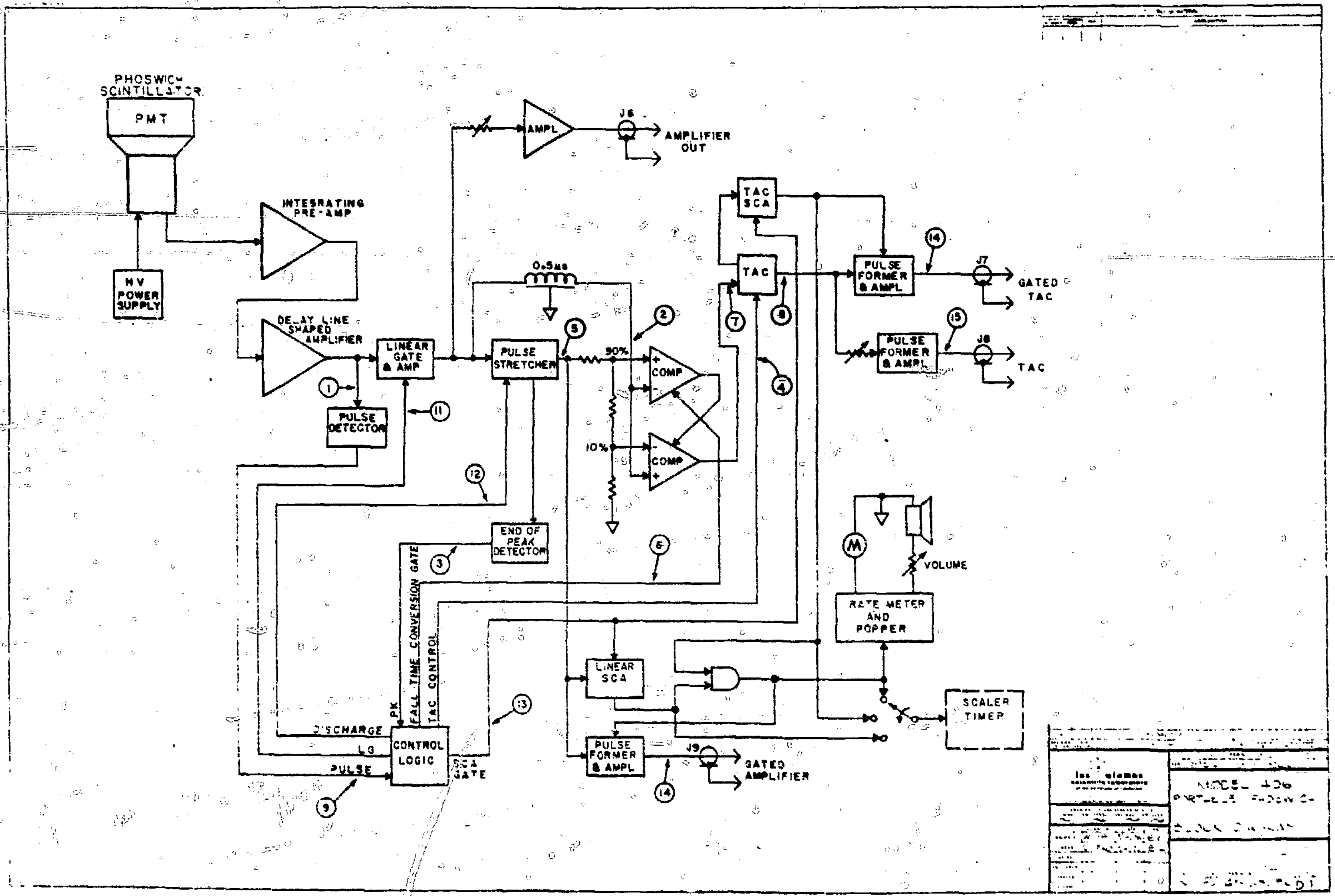
FIG. 8. Wee Pee Wee electronics schematic.

FIG. 9. Wee Pee Wee alpha survey instrument with meter detached from the case.



LOW BACKGROUND HOUSING [STAINLESS STEEL]
MAGNETIC SHIELD
[BERYLLIUM] ENTRANCE WINDOW
LOW NOISE PHOTOMULTIPLIER
[CsI (Na)] SCINTILLATOR B
[NaI (Tℓ)] SCINTILLATOR A
LOW BACKGROUND OPTICAL WINDOW


AMPLITUDE
TIME
NaI(Tl)
τ=0.23 μs
CsI (Na)
τ=0.63 μs
LIGHT OUTPUT
(A)
1/e X MAX. HEIGHT
1/e X MAX.
τ
PREAMPLIFIER OUTPUT
(B)


PHOSWICH SCINTILLATOR
PMT
HV POWER SUPPLY
INTEGRATING PRE-AMP
DELAY LINE SHAPED AMPLIFIER
LINEAR GATE & AMP
PULSE DETECTOR
AMPL
J6
AMPLIFIER OUT
0.5μs
PULSE STRETCHER
90%
10%
COMP
COMP
END OF PEAK DETECTOR
TAC SCA
TAC
PULSE FORMER & AMPL
J7
GATED TAC
PULSE FORMER & AMPL
J8
TAC
M
VOLUME
RATE METER AND POPPER
SCALER TIMER
LINEAR SCA
PULSE FORMER & AMPL
J9
GATED AMPLIFIER
CONTROL LOGIC
DISCHARGE
LG
PULSE
PK
FALL TIME CONVERSION GATE
TAC CONTROL
SCA GATE
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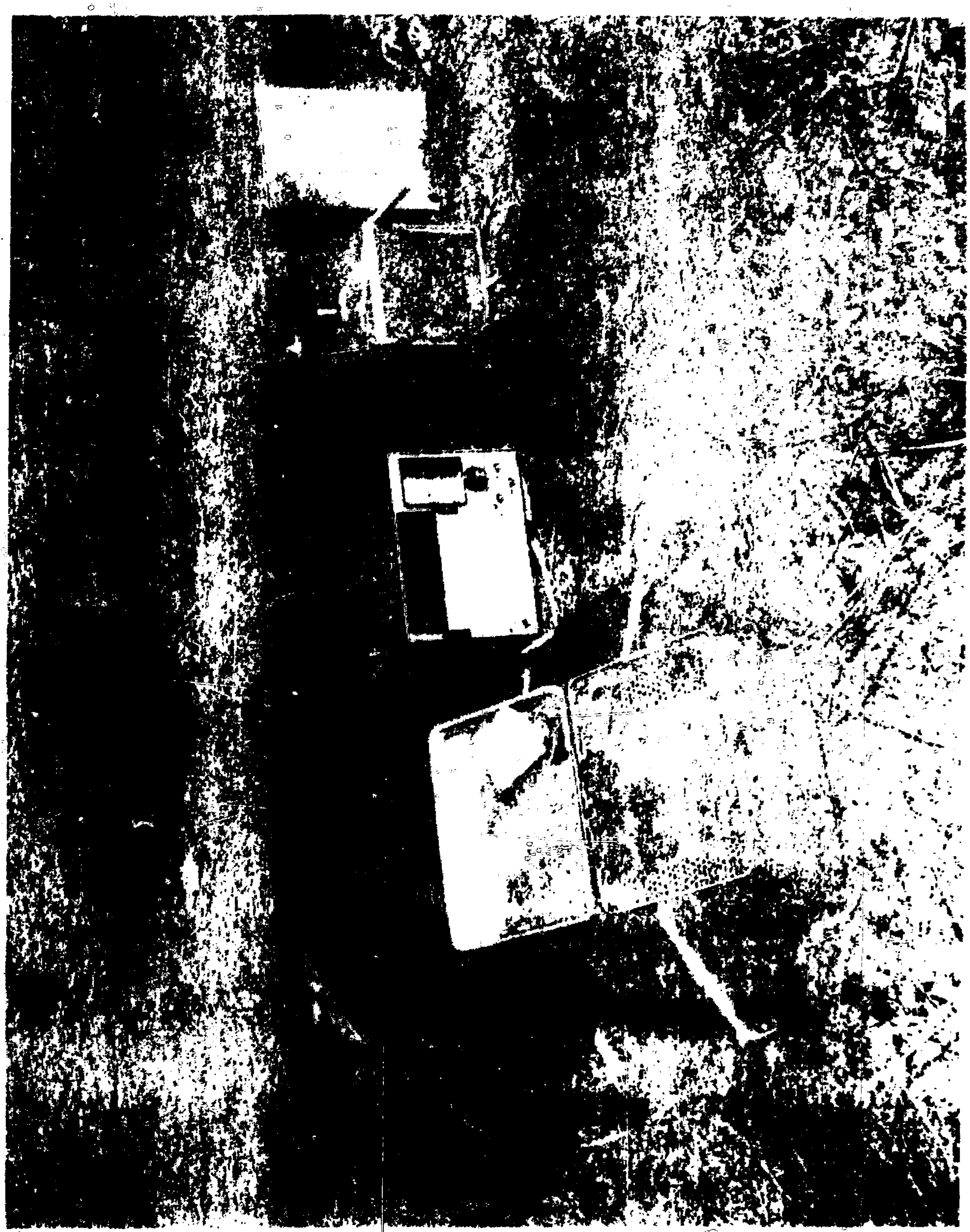
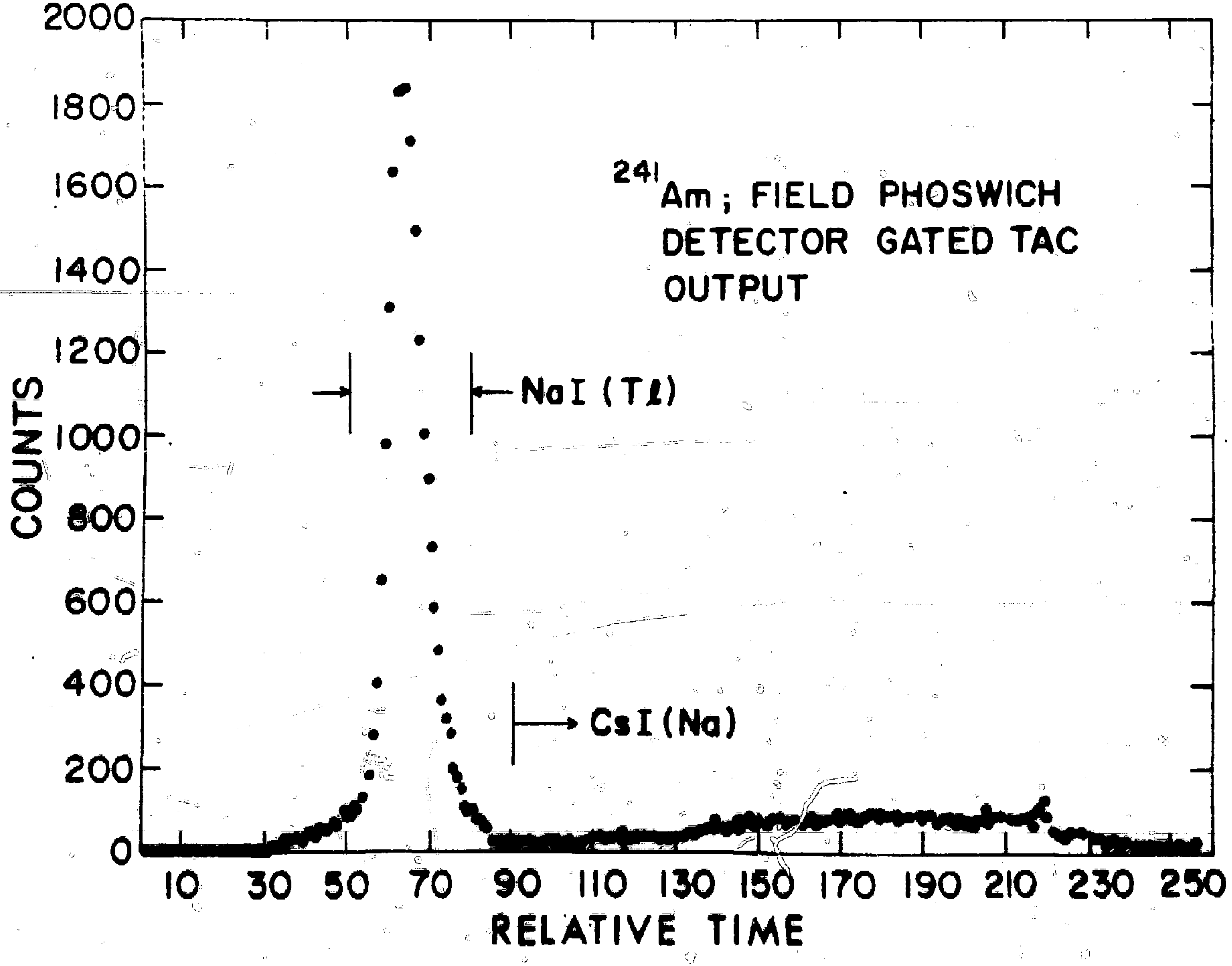
los alamos scientific laboratory
MODEL 406
PORTABLE PHOSWICH
BLOCK DIAGRAM




241Am ; FIELD PHOSWICH DETECTOR GATED TAC OUTPUT
NaI (Tℓ)
CsI (Na)
COUNTS
0
200
400
600
800
1000
1200
1400
1600
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2000
RELATIVE TIME
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30
50
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150
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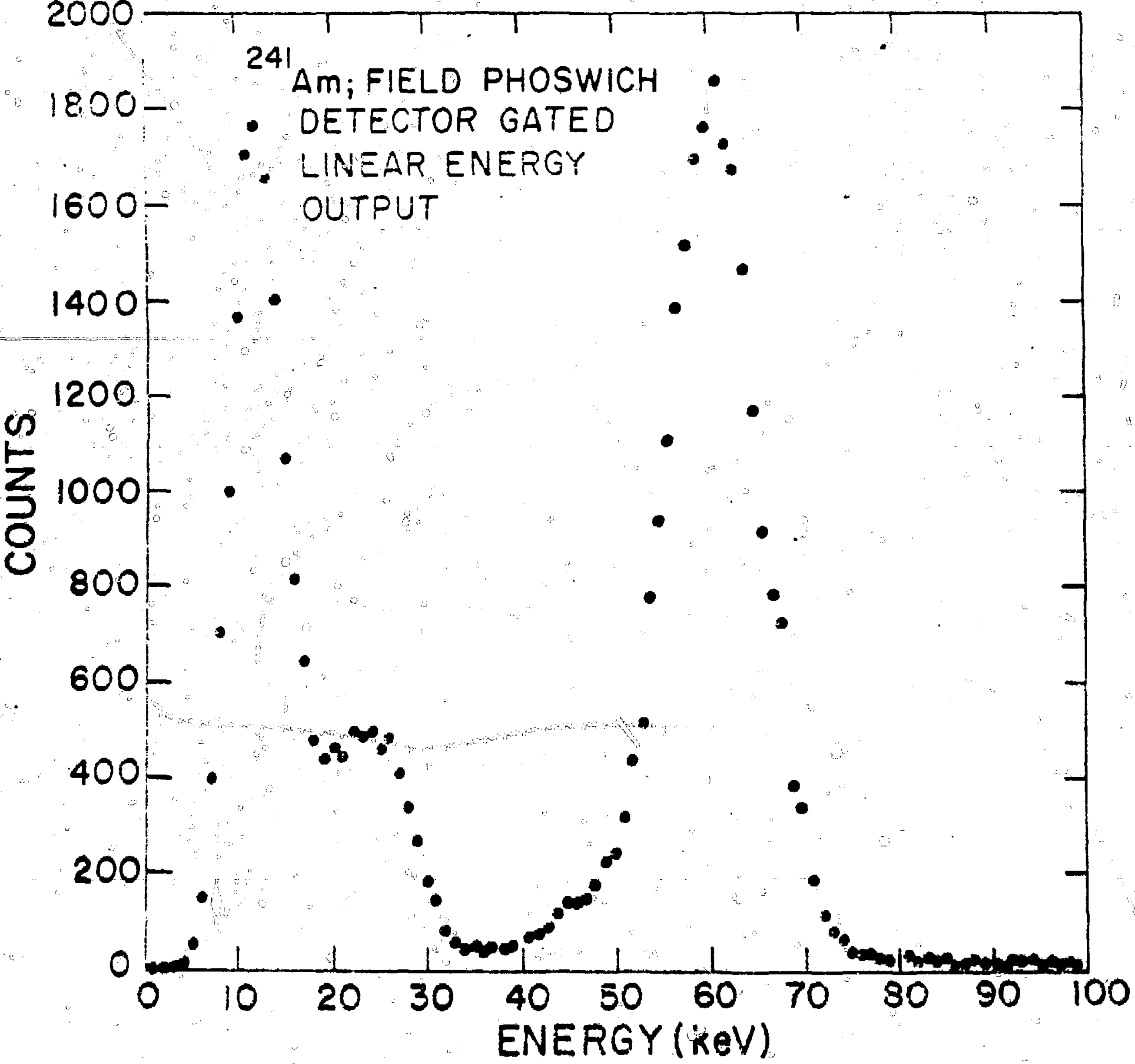
210
230
250


^{241}Am; FIELD PHOSWICH
DETECTOR GATED
LINEAR ENERGY
OUTPUT
COUNTS
0
200
400
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40
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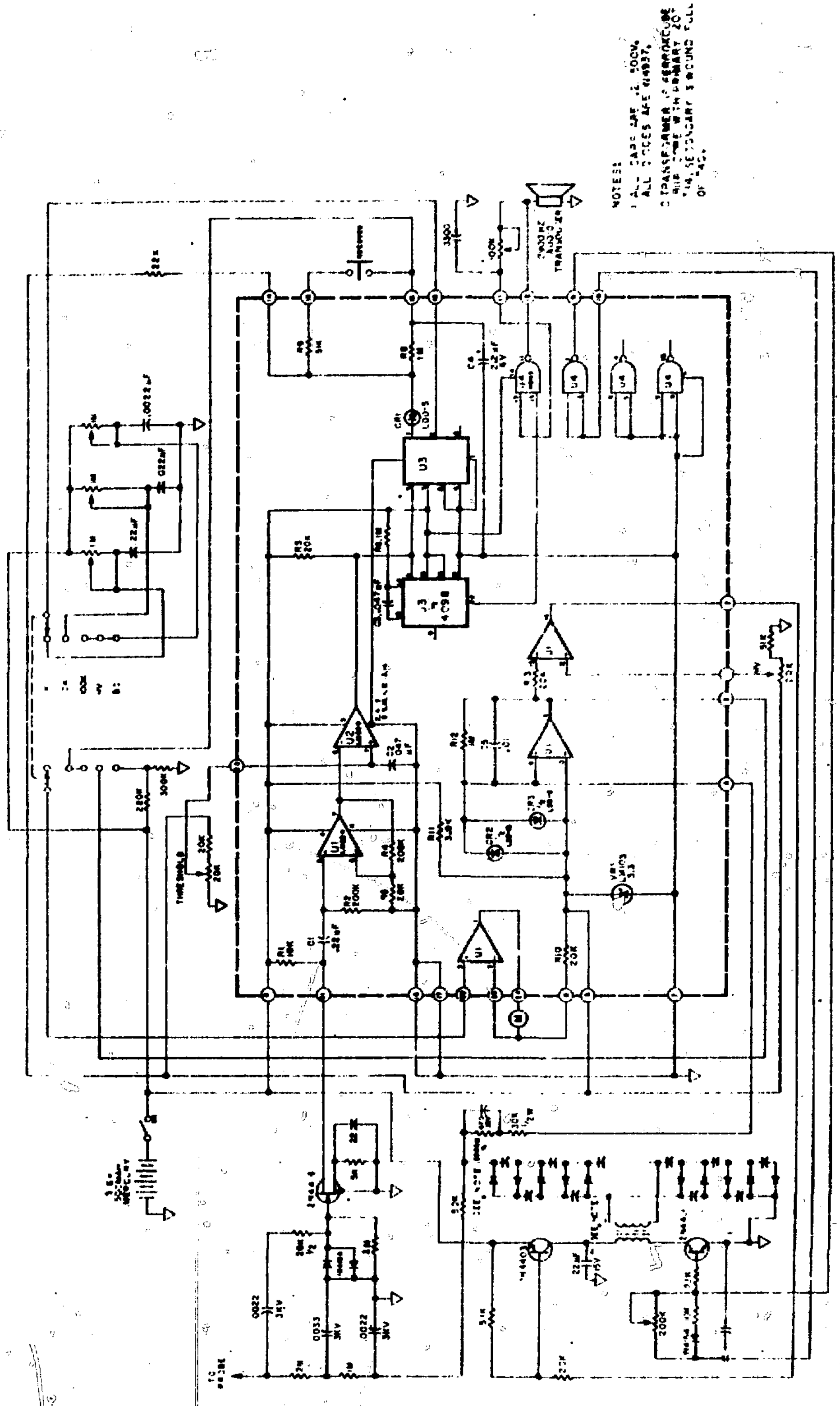
80
90
100
ENERGY (keV)


NOTES:
1. ALL CAPS ARE ... 50CV. ALL DIODES ARE 1N4937.
2. TRANSFORMER IS FERROXCUBE ... WITH PRIMARY 20T ... SECONDARY ... FULL OF #40.
AUDIO TRANSDUCER
THRESHOLD
TO PROBE
U1
U2
U3
U4
4098
CR1
R1 10K
R2 200K
R3 20K
R4 200K
R8 1M
R10 20K
R11 3.9K
R12 1M
VR1 3.3
22K
220K
300K
.0022uF
.0033
.22uF


IN. 1 2 3 4 5 6
Los Alamos Scientific Laboratory
OF THE UNIVERSITY OF CALIFORNIA
CM. 1 2 3 4 5 6 7 8 9 10 11 12 13 14 15

Question:

What is the response time?

Answer:

Response time is primarily digitally related to the display period, for dose rate is two seconds, at higher levels it automatically increases as part of the auto ranging function of display. That low level is two seconds.

A MINIATURE DIGITAL RADIACMETER

by David Raymond[1] and Joseph Nirschl[2]

ABSTRACT

A prototype instrument for monitoring radiation levels has been developed under contract to U.S. Army Electronics Research and Development Command, Fort Monmouth, N.J., for technical evaluation. This instrument by MDH Industries measures gamma dose-rates from 10 mrad/h to 1000 rad/h using a small pressurized ion chamber, which contains a hybrid integrated circuit current-to-frequency converter. The autoranging digital readout uses a liquid crystal display. Audible and visual alarm are incorporated, to alert the operator when a preset level of dose-rate has been exceeded. A non-volatile total dose memory can be read on command. Control of the instrument is via a sealed front panel keyboard. Battery life exceeds 100 hours continuous operation between recharging.

The instrument also includes provisions for a plug-in Geiger-Mueller probe that extends the sensitivity to .01 mrad/h for use as a frisking probe. Both detectors are designed to be sensitive to beta radiation as well as gamma.

Data is presented on the evaluation to date of the performance of the prototype instrument over a wide range of temperature and dose-rate.

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[1] with MDH Industries, Inc., Monrovia, CA

[2] with U.S. Army ERADCOM, Fort Monmouth, NJ

INTRODUCTION

The measurement accuracy of radiation survey meters (radiacmeters) is usually not so high (say ± 15% of true dose-rate) as to require the accuracy and resolution of a digital display. But there are other considerations and incentives for going digital in a radiac design, namely:

(1) Autoranging is facilitated, a definite advantage for a wide-dynamic range instrument (e.g. 5 decades).

(2) The readout device can potentially become much smaller in size and more rugged than a conventional (D'Arsonval) indicating meter without sacrifice of readability; it may also be expected to become available at lower cost.

(3) If a sufficiently compact detector is employed, the instrument's physical size can be significantly reduced, particularly if large-scale integration (LSI) is brought to bear on the digital electronics complexity.

(4) From a human factors viewpoint, the reading of numerically displayed radiation levels, is less confusing, particularly to the untrained operator and, therefore, safer in use. This advantage, among others inherent in digital display, was well described by Jones$^{1)}$.

Apart from the cited reasons for a digital radiacmeter, the prototype instrument to be described offers the added advantage of accumulated dose readout capability, without undue added complexity, thereby enhancing its operational versatility.

The present miniature instrument (Fig. 1) developed under contract to US Army ERADCOM (Contract DAAB07-76-C-0894) followed a larger size brassboard model phase which demonstrated the basic concept.

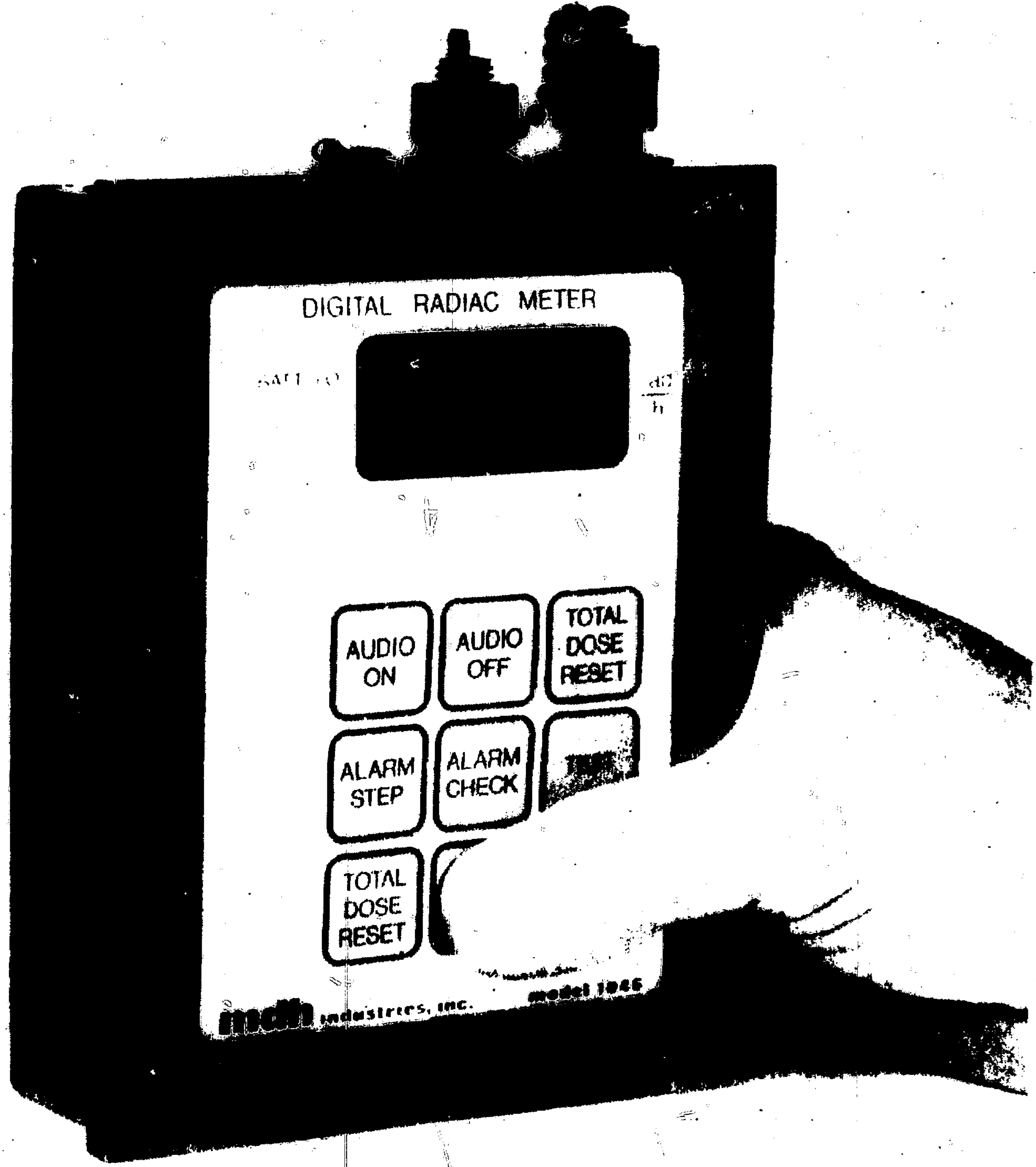


Figure 1 Prototype of Miniature
Digital Radiacmeter

Even in its present compact size this instrument represents but an intermediate step in the eventual development of a small radiac meter utilizing the latest (LSI) microcircuitry to provide measurement of dose-rate over a wide dynamic range as well as integrated dose reading. Indeed, the possibility of a cigarette-pack (king) size digital radiacmeter appears to be feasible now and within reach of practical realization.

DESCRIPTION OF PHYSICAL CONSTRUCTION AND FUNCTIONAL FEATURES

The MDH Model 1046 Digital Radiac Meter is contained in a rectangular aluminum housing, 4.5 x 5.75 x 1.5 in., except for the attachable probe intended for detection of low-level radiation from surfaces and personnel. This probe is contained in a protective housing 1.5 in. in diameter by 8.0 in. long. Extending from this cylinder is a connecting cable 5 ft. long. The meter weighs 1.7 lbs. and the remote probe and cable weigh 0.94 lbs.

The keyboard controls on the face of the radiac are all simple single pole, single throw, normally open pressure switches. The unbroken surface of the keyboard permits controls to be exercised by means of a waterproof panel, eliminating the need for individual protection around each switch.

The digital readout is a 3 1/2 digit liquid crystal display (LCD) unit mounted on a circuit board and visible through the clear water-proofing surface of the keyboard. Auxiliary indicators are also part of the LCD. The display operates from about -15°C to +48°C. Below -15°C the display freezes and whatever number was present at that time remains displayed. Above 48°C the display turns dark. It recovers when cooled down again. The instrument continues to function beyond these extremes.

Behind the keyboard, which is attached to the removable front panel, three circuit boards, the battery, and the ion chamber neatly fill the volume. Two of the circuit boards have a total of 55 integrated circuit packages arranged as tightly as is feasible. The third circuit board contains a power converter, a detector bias supply, and the GM tube interface circuits. A test mode is available to verify proper performance of the current-to-frequency (I/F) converter, time-gate logic and display.

The test mode inputs a signal to the electronics on command which approximates 250 rad/h. This value is somewhat variable and it is only used as a gross check on electronics performance. Operating the TEST key has no effect on the accumulated dose stored in the memory.

The radiac is turned on by pressing the POWER ON/OFF key. After an initial settling time, which might last for up to approximately 30 seconds, depending on whether radiation is present or not, the display will indicate the dose-rate in rad/h, from .000 to 1000 rad/h. With radiation (above 10 mRad/h) present, the settling time is typically within 10 seconds. The same key is pressed again to turn the instrument off.

The remote probe, a Geiger-Mueller tube with connecting cable, covers the range from 0.01 millirad/h to 10 millirad/h. With the probe connected to the instrument, the operating mode is dose-rate only. The display reads directly in millirad/h. An audible click is generated for each pulse from the GM tube which corresponds to one digit or 0.01 mrad/h.

When the probe is not connected, the internal detector, a pressurized ionchamber incorporating a current-to-frequency (I/F) converter, covers measurement from approximately 10 mrad/h to 999 rad/h. Several choices of information are available using this detector.

The primary operating mode using this detector is dose-rate, although the high-impedance I/F converter basically quantizes charge, i.e. dose, so as to produce a pulse for a quantity of radiation dose. In order to cover the wide range of radiation, the dose-rate is measured by a dose-rate counter under control of time-gate logic. The time-gate logic selects an integration interval for the dose-rate counter based upon the number in the dose-rate counter at the end of each interval. If the number exceeds 80, then counting stops and the decimal point is properly located. For example, a dose-rate of 20 rad per hour, which produces a frequency of 10,000 pulses/sec, would accumulate 20 counts in 2 ms, the shortest allowed interval. Since this is below 80, counting would continue until a total of 20 ms has elapsed, at which time 200 counts would be accumulated, and the counter would stop. The display would read 20.0 from the accumulated counts and location of the decimal point because of the interval used. At lower dose-rates, the counting interval might continue to 200 ms or even 2 seconds. One dose-rate measurement is made each 2 seconds. If the rate is equal to or greater than 1000 rad/h, a full scale reading of 1000 rad/h will be displayed.

NON-VOLATILE DOSE MEMORY/READOUT

The radiac incorporates a memory which accumulates integrated dose. Each sampling interval the count accumulated in the dose-rate registers is not only used to display dose-rate, but is added to the integrated dose register. The value stored in this memory, up to 999 Rad, can be non-destructively read out by pressing the appropriate key. If this range is exceeded, the full scale reading of 999 Rad will be retained and displayed on demand, while a plus sign on the left of the display will indicate this fact (overflow) regardless of the operating mode.

To read the integrated dose which has accumulated since the last reset, the READ DOSE key is depressed and kept down until the operator has had time to read the stored and displayed value. When released, the instrument automatically reverts to reading out dose-rate.

Since the integrated dose value is not lost, neither by readout nor by turning the instrument off (assuming a charged battery), a separate operation is used to start the accumulation of dose from zero. This is done by simultaneously pressing both TOTAL DOSE RESET keys. Two keys are used to minimize the likelihood that the dose reset operation is inadvertently activated and prior accumulated dose readings lost.

VISUAL AND AUDIBLE ALARM WITH PRESETTABLE THRESHOLD

When the dose-rate exceeds a preset threshold, the alarm modes incorporated in the radiac are activated and remain active until reset by the operator. The visible alarm is a blinking display. The audible alarm, which can be enabled or disabled from the keyboard, is a pulsing tone.

The alarm threshold is set by pressing the ALARM STEP key. Pressing the ALARM STEP key advances the threshold level in order of magnitude steps from .01 Rad/h to 1000 Rad/h (actually 999 Rad/h at this level). Each operation of the key causes a one decade step advance, no matter how long the key is held down. From 999 Rad/h, pressing the key causes the threshold to return to .01 Rad/h and starts the step sequence again for further activations. Each time the key is operated the newly-selected threshold appears on the display, and the alarm is reset, if it had been previously activated. Turning power off does not lose the last setting for the alarm threshold.

To determine whether the alarm function is operating, the ALARM CHECK key is pressed. When this is done, the visual alarm (a blinking display) starts operating. The audible alarm operates only if previously enabled by pressing AUDIO ON. While ALARM CHECK is pressed the alarm threshold level will be displayed. When released, the alarm functions are reset to trigger when the measured dose-rate exceeds the previously displayed threshold level.

As noted above, the audible alarm must be turned on by pressing the AUDIO ON key after pressing the POWER ON-OFF key to turn the unit on. While on, the colon of the display blinks to indicate this condition. The colon is used to take advantage of properties of existing LCD's. Pressing the AUDIO OFF key disables the audible portion of the alarm system.

MINIATURE IONCHAMBER DETECTOR

The pressurized ionchamber with built-in hybrid integrated circuit (I.C.) current-to-frequency (I/f) converter represents the heart of the instrument. This detector is calibrated for gamma radiation above 100 keV and over the range from approximately 1 mRad/h to 999 Rad/h. The detector consists of ionchamber electrodes, the hybrid I.C. I/f converter, and an outer housing to contain the pressurized argon atmosphere. The choice of a volume and pressure for the ionchamber is governed by our ability to measure small currents and by limits on allowable pressure. A practical lower limit of current measurement is about 10^{-14} A @ 50°C. Using a gas ionization constant of 1 R = 3.34×10^{-10} coul/cm^3 for air at STP we find that a 1 atmosphere chamber will give 0.93×10^{-16} A/cm^3 @ 1 mR/h. We thus need a factor of about 100 increase in the quantity of gas (or pressure x volume) in order to effectively measure this current. A limitation on pressure that occurs at high radiation rates comes from an increasing probability that free electrons will recombine with positive ions before they reach the positive electrode. As the dose-rate and gas pressure increase the density of positive ions goes up, increasing the likelihood of a recombination within the gas that causes a loss of current. Other factors that affect the probability of recombination are the electrode spacing and the electric field(bias voltage) between the electrodes, but the current degradation is not as strong a function of these variables as of pressure. Based on the performance of previous chamber designs the volume used is about 6 cc at 16 atmospheres pressure. By operating the chamber at 500 V bias, good ion collection is obtained over a wide range of dose-rate, with signal reduction amounting to approximately 20% at 1000 Rad/h.

Chamber materials determine the energy response of the detector, and the first step toward achieving tissue-equivalence is to use materials similar to tissue in atomic weight for the electrodes. The outer wall, however, is stainless steel, for physical strength, which contributes excess secondary electrons at energies below 200 keV. Compensation shields around the detector can effectively flatten the energy response.

Because the ionchamber collects electrons produced from gammas, it would also be sensitive to electrons if introduced into the chamber. This is made possible by making one end of the chamber a thin membrane backed by a perforated outer cap to hold the pressure. The membrane allows 400 keV betas to pass into the sensitive volume in which case more secondary electrons may be produced and detected. The beta window permits the basic instrument to be checked with a portable beta (Sr-90) source such as TS-784 or AN/UDM-2.

The handling of the signal current from the ionchamber electrodes is critical to the successful operation of the detector. Since currents of 10^{-14} A are to be measured, most ordinary insulators are not effective, particularly if any significant voltage is placed across them. Even the best insulators are susceptible to degradation unless their environment is carefully controlled. Placing the signal processing electronics inside the chamber minimizes the exposure of the signal current to leakage paths as well as controlling the environment around the signal path. Hybrid I.C. technology permits this to be accomplished within the miniature-size ionchamber, so only non-critical external wire connections are involved.

SENSITIVE CHARGE QUANTIZER (I/F CONVERTER)

The circuitry used to convert the ionchamber signal current to a useful analog is a sensitive current-to-frequency (I/f) converter chosen for its simplicity, accuracy, and utility. This converter (see block diagram, Fig. 2) behaves much like an operational amplifier in that a feedback current is generated which exactly cancels the signal current, and the amplifier input, which connects to the junction of these currents, is held at a virtual ground. With the signal input held at ground it becomes relatively easy to guard against leakage paths to the input, with grounded conductors surrounding the signal path. The feedback current is generated as a sequence of discrete amounts of charge. A MOSFET operational amplifier with capacitive feedback integrates the signal current to form a voltage ramp. When this voltage reaches a threshold, a pulse is generated which causes a discrete amount of charge to be fed back to the input amplifier. Each pulse that is

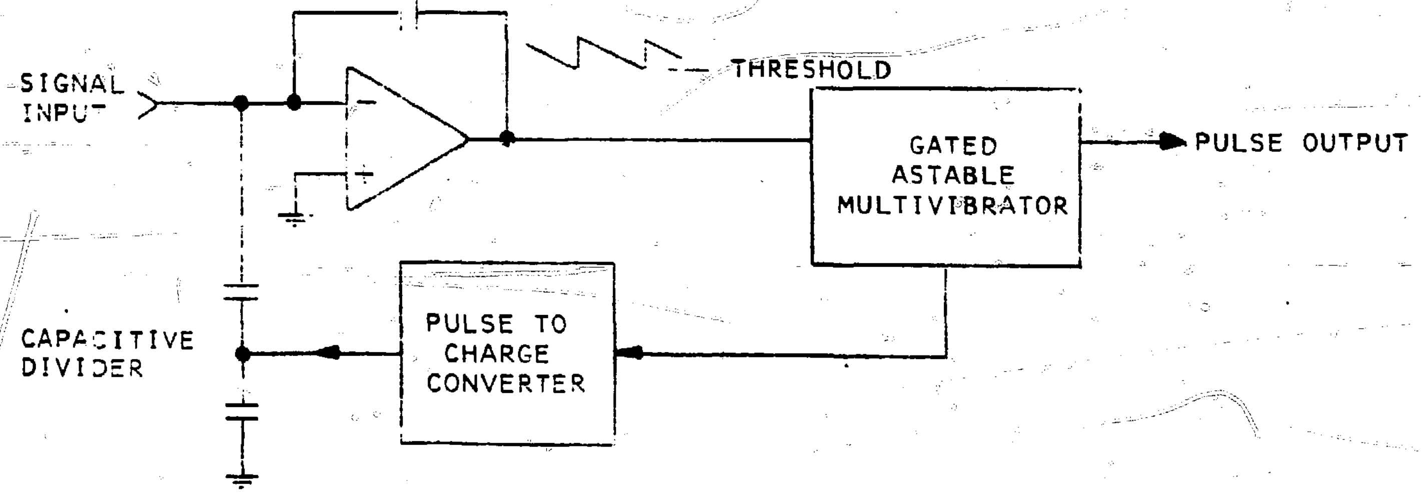
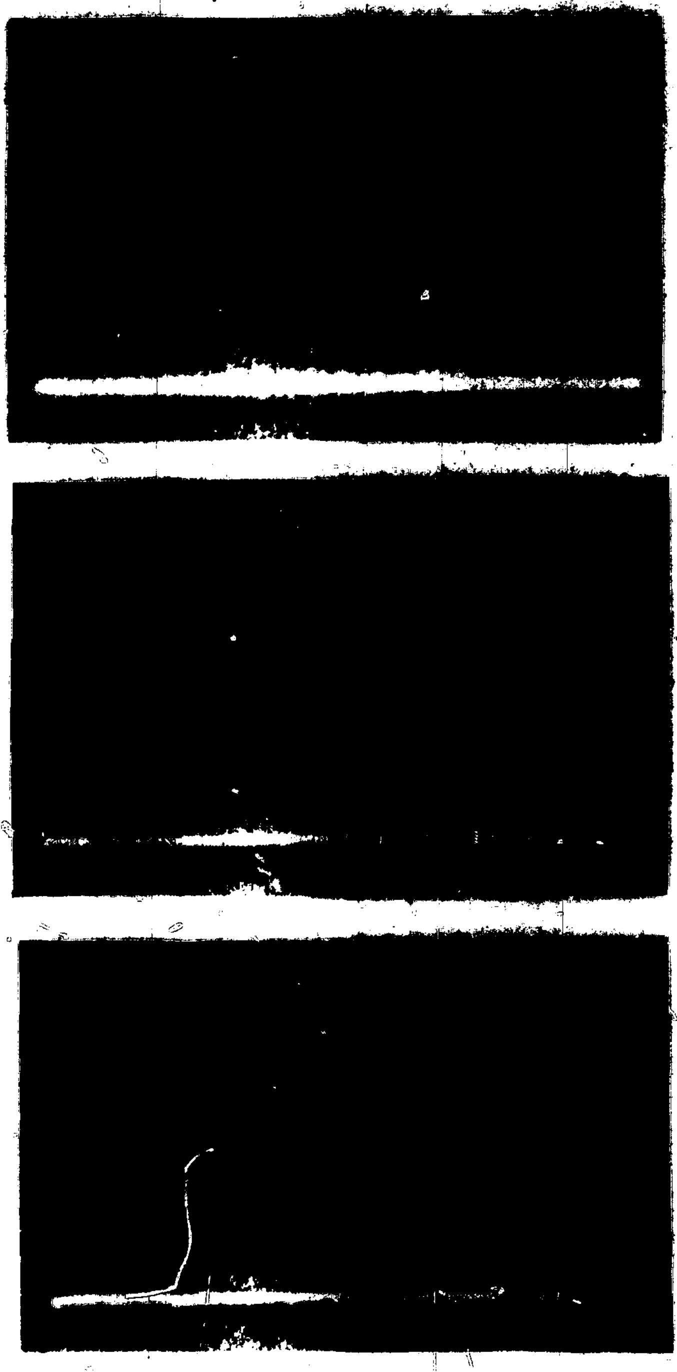


FIGURE 2
BLOCK DIAGRAM
CURRENT TO FREQUENCY CONVERTER

generated thus corresponds to an increment of charge from the signal current (and hence dose) and the frequency of these pulses is an analog of the signal current (and hence dose-rate) in useful digital form.

The accuracy of this conversion technique is determined primarily by the accuracy of the pulse-to-charge conversion scheme. In this circuit we switch a regulated voltage across a stable capacitor as a means of generating stable charge pulses. These charge pulses are much too large to feed back directly to the input because of minimum size limitations on both the capacitor and the switched voltage, so the pulses are attenuated capacitively in order not to introduce leakage, before they reach the input. The remaining accuracy is determined by components which introduce leakage currents at the input to add an offset term in the current-to-frequency transfer function. Low leakage capacitors are used and the voltage applied across them is kept very small wherever possible. The principal source of leakage current is the MOSFET itself because the source and drain must be connected to sizeable voltages in order to bias it properly. Leakage currents at 50°C are about 10^{-14} A so they are right at the threshold of the signal current measurement. The capacitive feedback scheme employed here is believed to offer more sensitivity than earlier I/f converter designs (e.g. ref. 2,3). Unlike Geiger-Mueller detector methods for digital display, the ionchamber plus I/f converter approach provides fairly steady data display in the digital readout, which is considered an advantage from a human factors viewpoint. The periodicity and proportionality to dose-rate, of the converter output pulse frequency, is illustrated in Fig. 3.



(A) 1 Rad/h

Horizontal
Scale : 1 msec/Div.

(B) 10 Rad/h

Horizontal
Scale : 100 µsec/Div.

(C) 100 Rad/h

Horizontal
Scale : 10 µsec/Div.

Note : Bright periodic dots are from signal pulses (at 20 µsec pulse period). Higher frequency pulse train is from reset action.

Figure 3 Current-to-Frequency (I/f) Converter Output Pulses For Decading Dose-Rates Of Radiation (1, 10, 100 Rad/h). Vertical Scale : 2 Volt/Division

MODEST POWER CONSUMPTION - LONG BATTERY LIFE

An outstanding feature of the instrument is its low power drain of about 10 mW. The rechargeable (6 volt) battery operates the unit for typically more than 100 hours before requiring recharging. When the battery charge is insufficient to operate the unit at full accuracy, an arrow in the display points to the BATT LO mark on the front panel. The use of the LCD display enables long operating time and battery life(as well as outdoor readability), without the "push-to-read" procedure usually required with light emitting diode (LED) type displays. The temperature limitation of the LCD, on the other hand, is presently recognized as a shortcoming, particularly when considering the military environment. There are, fortunately, development trends discernible, which promise a digital display with low power demand and improved temperature performance to become available in the future. Alternatively, an auxiliary LED display (with push-to-read button) could be added for the temperature extremes.

PERFORMANCE TESTS

A number of basic performance tests were conducted at the Combat Surveillance and Target Acquisition Laboratory (US Army ERADCOM) on both the prior large-size brassboard model of the digital radiacmeter and the miniature *prototype model (MDH Model 1046, Fig. 1). These tests involved mainly* dose-rate and dose accuracy tests, temperature stability tests, battery life test and evaluation of ancillary operational features (digital display, alarm).

Although the extent of this evaluation was limited in scope and duration so far, test results from the brassboard unit as well as the model 1046 unit are considered indicative of the design integrity and merits of this hybrid (ionchamber plus Geiger-Mueller counter) digital radiac approach. This is particularly true after a corrective feature was incorporated into the brassboard model design (a "keep alive" optoelectronic pulser on the I/f-converter input). This was found necessary after ERADCOM testing of the brassboard model disclosed a tendency for the I/f-converter to latch up, due to negative polarity offset charge in the MOSFET input stage on power turn-on and capacitive coupling to the input. It should be noted that the I/f-converter is capable of quantizing charges as small as .04 picocoulombs per pulse. Transient charge phenomena on the MOSFET input producing a negative offset charge on the input, would occasionally cause paralysis of the system for a considerable period of time, due to the long time constants involved for charges to leak off. Since introduction of the optoelectronic pulser no further problems of this type have been observed. This corrective feature is also embodied in the model 1046 miniature instrument.

DOSE-RATE MEASUREMENT (CALIBRATION) ACCURACY AND STABILITY

The basic radiac response depends on the integrity of the pressure seal of the pressurized ionchamber for measurement accuracy and stability. This has been a matter of considerable interest and concern to ERADCOM on account of earlier experience of difficulty with sealing a pressurized ionchamber (ref 4). However, design and construction (with stainless steel enclosure) of the miniature pressurized chamber used in the digital radiac are such that a high degree of confidence exists in pressure retention. The brassboard model instrument was tested over a period exceeding one year (including some temperature testing) and no pressure loss or change in calibration was observed. Calibration accuracy (for Cs-137) was typically within 10% (except for 1000 Rad/h point being 19% low) and Nov. 77 average readings (from .01 to 100 Rad/h) were within 1% of Oct. 76 readings. With regard to the two miniature prototype models, available data suggest that the Serial 02 instrument maintained its calibration for an available observation period of about two months (Table 1). A question arose with regard to the Serial No. 01 instrument as it showed some evidence of change in calibration, possibly due to pressure loss in the ionchamber. This matter is still under observation as the change in calibration data may have been brought on by a fault condition (cell polarity reversal) ofthe (NiCd) battery. Calibration stability (pressure retention) will, of course, continue to be checked on the two prototype models. Initial calibration accuracy of the Serial 02 unit is also reflected in the data of Table 1. It is within approximately 10% for most of the range (to 100 Rad/h), but tends to drop off at the high intensity of 1,000 Rad/h, by approximately 20%. This is comparable to the brassboard model performance. It is expected that this drop off can be further reduced by appropriate design modifications in the geometry of the chamber and/or applied voltage (presently 500 V for both ionchamber and Geiger-Mueller probe) or electronic compensation. The calibration of the instrument can be adjusted

Table 1

Comparative Gamma Radiation (Cs-137) Calibration Data for Miniature Model of MDH Digital Radiacmeter (Serial 02)

Exposure Dose-Rate R_γ (Rad/h)	Average Response1) at Time2)			
	A		B	
	R_A	Δ_r (%)3)	R_B	Δ_t (%)4)
.010	.00973	-2.7	.010	+2.8
.050	.0549	+9.8	.0549	0
.10	.0963	-3.7	.0952	-1.1
.50	.509	+1.8	.506	-0.6
1	1.027	+2.7	1.018	-0.9
5	Not Taken	-	5.134	-
10	10.36	+3.6	10.30	-0.6
50	54.06	+8.1	54.29	+0.4
100	110.	+10	111.	+0.9
500 5)	431.14	-13.8		
1000 5)	799.6	-20		

NOTES:

1) Average of typically 8 sample readings (AN/UDM-1A Calibrator).

2) Time A: 15 September 1977
Time B: 1 November 1977

3) Response (calibration) Error $\Delta_r = \dfrac{R_A - R_\gamma}{R_\gamma} \times 100$ (%)

4) Calibration stability $\Delta_t = \dfrac{R_B - R_A}{R_A} \times 100$ (%)

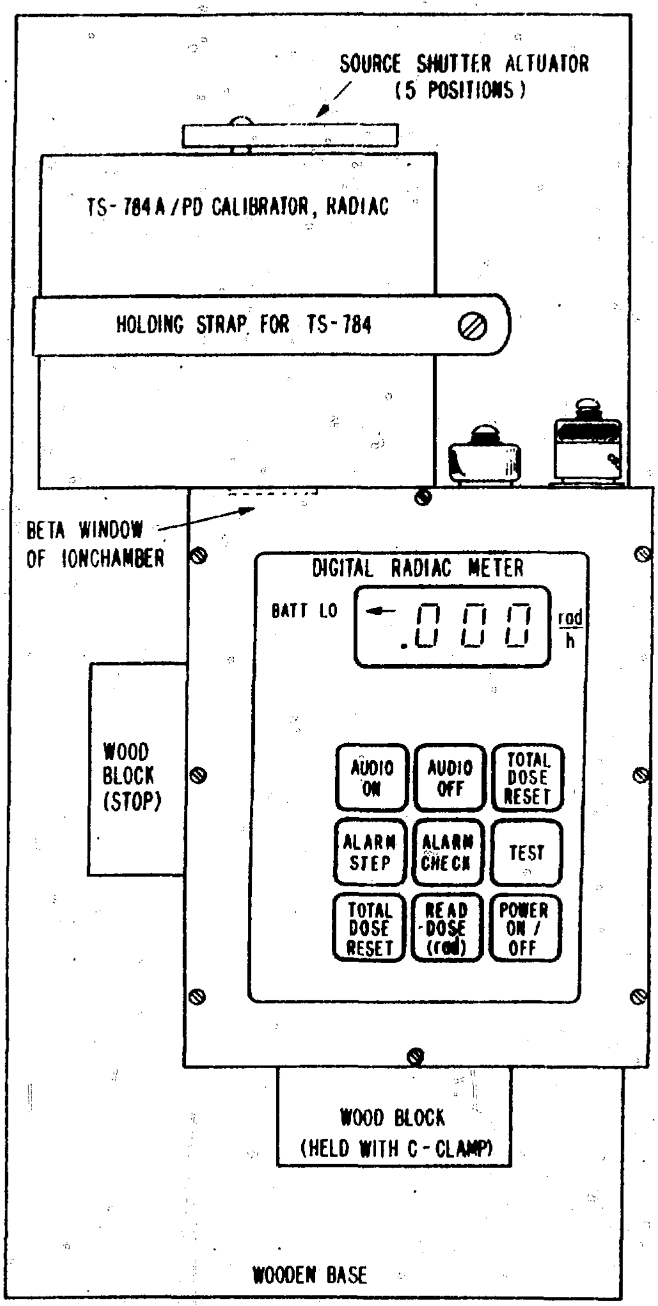
5) Cobalt 60 Expsoure.

by a single variable capacitor and from the data of Table 1 it seems that by "error splitting" the instrument accuracy across the five decade dynamic range could be improved to some extent (this was not done as yet, as emphasis is on continued observation of original calibration). The GM probe accuracy was also checked (Serial 02 instrument) with low intensity gamma (Cs-137) exposure and found to be within 6% in the exposure range from 0.2 to 1.0 mR/h; but a reduced response was observed in the range from 2 mR/h (-11%) to 10 mR/h (-27%). This is attributed to a lack of deadtime compensation of the BS-1 GM tube response, which could be corrected by appropriate circuit modification. Actually, the ionchamber overlaps the GM probe in this range, as it will measure reliably to approximately 4 mRad/h.

Temperature Performance (Hot/Cold)

Although no rigorous MIL standard type temperature test sequence was performed as yet, results of preliminary ("quick look") temperature tests on both dose-rate and total dose indication stability, are encouraging (see Tables 2 and 3). The test fixture used for irradiating the digital radiacmeter during temperature and dose tests is shown in Figure 4. Except for one dose-rate (.4 mR position of beta source, Table 2), where a 22% shift (drop) was noted at cold temperatures, dose-rate stability in both the hot and cold tests remained within ±10% of room temperature values, which is considered acceptable (over the range from +48°C to -26°C). It should be noted that the TS-784 calibrator positions are only used for obtaining relative dose-rate readings, depending on the contact geometry between the instrument's beta window and source aperture.

The most obvious defect in the temperature performance of the radiacmeter is the limitation of the Liquid Crystal Display (LCD) as indicated in the data sheets. For the more extreme temperature environment of any military application



TEST FIXTURE WITH TS-784
BETA SOURCE IN CONTACT WITH
DIGITAL RADIACMETER (BETA WINDOW)

Figure 4

ERADCOM
NOV 77

Table 2

Results of Heat Test, Compact Digital Radiacmeter (Serial 01)

TS-784 Beta Calibrator Positions				4	40	400	4K	40K	Remarks
Time	Temperature Setting	Chamber Temperature	RUN						Readings are averages (Rad/h), $\bar{R}$.
10:42	Off	27°C (Room)	1	.0273	.4153	8.4	81	468.5	Heat ON at 13:01, 20 Jul 77, temperature set to +49°C.
13:25	+49°C	47°C	2	.0275	.4108	8.25	81	462.5	Continued in 40 mR position.
14:01	+49°C	49°C		LIQUID CRYSTAL DISPLAY (LCD) BLACK					Opened chamber, let test item cool off, observed LCD recovery.
14:32	Off	Decreasing	3		.4065				Closed chamber, set temp. to +60°C at 14:33.
15:32	+60°C	+60°C		LIQUID CRYSTAL DISPLAY (LCD) BLACK					Opened chamber, instrument case too hot to touch. Keyboard panel slightly warped. Let cool off.
15:46	Off	Decreasing							LCD starts recovering.
15:48	Off	+46°C	4	.0283	.4054	8.37	80.3	454.3	Instrument case still hot.
				+3.7	-2.4	-3.6	-0.9	-3	% change in average reading of run 4 referred to run 1

$$\Delta\bar{R} = \frac{\bar{R}_4 - \bar{R}_1}{\bar{R}_1} \times 100 \ (\%)$$

Table 3

RESULTS OF COLD TEST, COMPACT DIGITAL RADIACMETER (SERIAL 02)

				TS-784 Beta Calibrator Positions (mR)					
Time	Temperature Setting	Chamber Temperature	RUN	4	40	400	4K	40K	Remarks
									Readings are averages (Rad/h), $\bar{R}$.
08:28	Off	23°C (Room)	1	.0255	.180	4.173	45.75	260	Room ambient readings. Cold ON at 08:34, 21 Sep 77, temperature set to +5°C.
09:45	+5°C	-2°C	2	.024	.177	4.108	46.26	260.8	LCD still readable (through chamber window). At 10:15 set temperature to -20°C. Left in 400 mR position of TS-784.
12:05	-20°C	-24°C	3			4.16			To keep chamber closed, only one position taken. LCD still readable. At 12:10 set temperature to -25°C.
13:40	-25°C	-26°C	4			4.18			LCD apparently frozen. Opened chamber to let recover (frost formed).
14:09	Off	-8°C	5			4.13			LCD digits reappear.
14:11	Off	-8°C	6	.0198	.1723	4.12	47.9	280	Decimal point poorly visible.
				-22	-4.3	-1.3	+4.7	+7.7	% change in average readings of run 6 referred to run 1 $\Delta \bar{R} = \dfrac{\bar{R}_6 - \bar{R}_1}{\bar{R}_1} \times 100$ (%)

an improved LCD or comparable (low power) digital display device would be required, as discussed earlier. The inherent temperature stability of the I/f converter and digital electronics is further evidenced by the dose (heat) data presented in the following.

Dose Integration Accuracy

This was tested repeatedly, for different dose-rates and integration times, as tabulated in Table 4. It is noteworthy that indicated dose-rate was typically integrated with a 1% or better accuracy, based on measured time. Also, dose-rate stability is seen to be within about 1%, except for the very low rate (test run 1, 5%). Dose integration was also checked in a high temperature environment, as summarized in Table 5. Again good stability prevailed, even for a temperature of +60°C (140°F).

Energy Dependence Test

The graph of Figure 5 represents the results, for two directions of incident (X-ray or isotopic) radiation namely (1) exposure through the beta window of the ionchamber (.004 inches of stainless steel) and (2) through the aluminum wall of the instrument case (adjacent to ionchamber). It is apparent that, in normal usage, a shielding cap should be applied to the beta window of the ionchamber, to suppress the enhanced low energy response; or else the ion chamber module should be completely enclosed within the instrument case, since the beta sensitivity is mainly required for frisking purposes, via the attachable Geiger Mueller frisking probe and its beta window.

Table I

CUMULATIVE RADIATION DOSE TEST DATA
MINIATURE DIGITAL RADIACMETER (MDH MODEL 1046, SER. NO. 02)

Test Run	Start Date Time	End Date Time	Time, T Elapsed (hours)	Av. Exposure Dose-Rate, $\bar{R}_B$ (Rad/h) at start (at end, % change)	Start Dose (Rad)	Dose 1) (Calculated) (Rad) $D_c = \bar{R}_B \times T$	Dose (Indicated) (Rad) D_i	Difference (%) $\frac{D_i - D_c}{D_c} \times 100$
1	21 Sep 77 17:18	22 Sep 77 08:09	14.85	.02617 (.0249, - 4.9%)	0	0.389	0.39	+ 0.3 2)
2	22 Sep 77 08:14	22 Sep 77 13:32	5.3	4.695 (4.665, - 0.6%)	0	24.88	24.7	- 0.72
3	22 Sep 77 13:36	22 Sep 77 16:23	2.78	269 (271.9, + 1.1%)	34.3	782.12	787	+ 1
4	4 Oct 77 15:26	7 Oct 77 12:50	69.4	.02767 (.02786, + 0.7%)	0	1.922	1.93	+ 0.42
5	12 Oct 77 17:12	18 Oct 77 16:28	143.267	.2722 (.2748, + 1%)	0	38.997	39.2	+ 0.52
6	28 Oct 77 13:32	28 Oct 77 15:18	1.7667	494.6 (498.2, + 0.7%)	0	873.79	875	+ 0.14
7	5 Nov 77 15:06	7 Nov 77 13:10	46.0667	8.1167 (8.14, + 0.3%)	0	373.91	371	- 0.78

Notes: 1) Based on average exposure rate, $\bar{R}_B$, at start
2) Within digital resolution of instrument

Table 5

Results from Dose/Temperature (Heat) Experiment

(Prototype Digital Radiacmeter, Serial 01)

Time (20 July 1977)		10:49	2:15	13:01	13:26	13:40	14:32	15:49
Temperature in Test Chamber	(°C)	27	27	16[3]	+47	+40	+50	+60
Average Dose-Rate, R_β [1]	(Rad/h)	.4153	.4172	.3978	.408	.4098	.4065	.4054
Time Lapse	(hr)	0	1.4333	2.20	2.6167	0	0.8667	2.15
Dose Increment, calculated[2]	(Rad)	0	0.595	0.318	0.173	0	0.36	0.893
Total Dose, calculated	(Rad)	0	0.595	0.913	1.09		5.77	6.303
Digital Dose, observed	(Rad)	000 (reset)	0.60	0.91	1.13	5.41[4]	5.75[5]	6.27[5]
Then Temperature Set to	(°C)			+49		+55	+60	
Error (%)			0[6]	0[6]	+3.7	(new reference time)	-0.35	-0.48

NOTES:

1) Beta exposure from TS-784; 2) All dose calculations based on initial (room temperature) rate (.4153 Rad/h) and

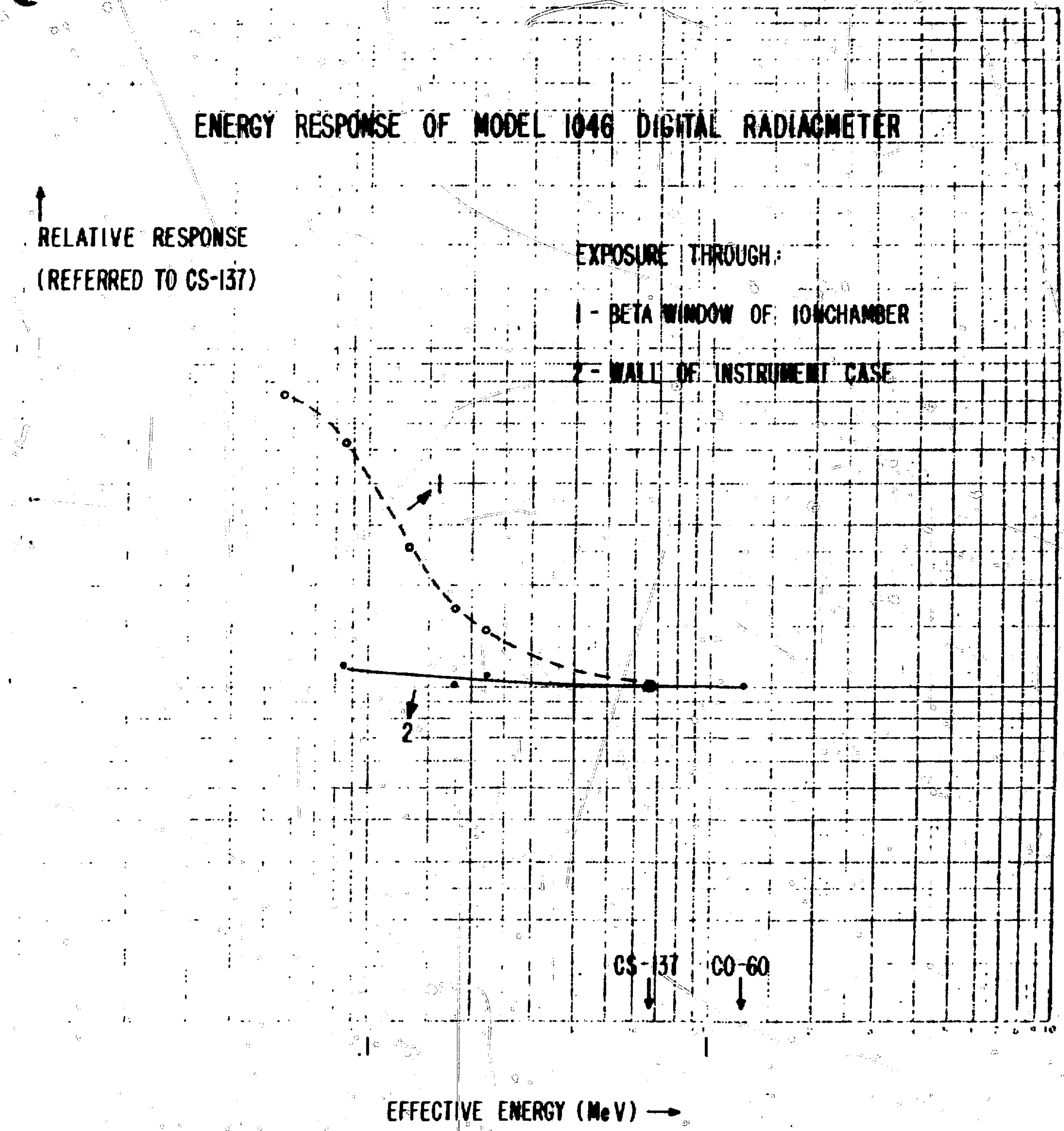
$$D = \int_0^T R_\beta dt \approx R_\beta \cdot \Delta t \text{ (for } R_\beta = \text{const.);}$$

3) From a brief cold temperature phase (between 12:21 and 12:43, -20°C).

4) Because of dose-rate check at various TS-784 intensities, this point used as new reference time and dose.

5) Upon brief recovery period for Liquid Crystal Display (LCD) (cooling).

6) Within digital dose resolution.



ENERGY RESPONSE OF MODEL 1046 DIGITAL RADIACMETER

Figure 5

Instrument Battery Life (Continuous Operation)

The model 1046 digital radiacmeter contains a rechargeable (NiCd) battery and one of the two external connectors permits connection to a battery charger. Several tests were performed to assess battery life for continuous operation (measurement of radiation dose-rates, no alarm activation). The tests were started after the battery was charged, although not necessarily for the same length of time in each case. This may, at least in part, account for the varying results (Table 6), another variable being radiation intensity. At low intensities the current from battery was measured at 1.5 mA, which rises by about 50% (to 2.3 mA) at high intensities. This is a remarkably low battery drain and a definite asset for purpose of miniaturization. The data of Table 6 also reconfirm the excellent (~1%) signal stability noted earlier, in terms of dose-rate displayed at the beginning and end of the test runs, for constant radiation exposure.

Table 6

RESULTS FROM BATTERY LIFE TEST FOR CONTINUOUS OPERATION OF MINIATURE DIGITAL RADIACMETER 1)

(MDH Model 1046, Ser. No. 02)

Test Run	Start Date Time	End Date Time	Average Exposure 2) Dose-Rate, $\bar{R}_B$ (Rad/h) At start (At end, % Change)	Recorded Total Operating Time 3) (hours)	Temperature Remarks
1	21 Sep 77 08:13	26 Sep 77 17:25	Variable	129.2	Included a cold and hot temperature test cycle.
2	4 Oct 77 14:00	7 Oct 77 16:48	Variable	74.8	Room ambient temperature
3	12 Oct 77 17:12	18 Oct 77 16:28	.272 (.275, + 1.1%)	143.26	Room ambient temperature
4	5 Nov 77 15:02	9 Nov 77 16:02	8.117 (8.20, + 1%)	97	Room ambient temperature
5	23 Nov 77 08:16	28 Nov 77 13:47	.290 (.2877, - 0.8%)	125.5	Room ambient temperature

Notes:
1) Upon charge of NiCd Battery
2) From TS-784 Beta Calibrator
3) Based on last observed proper performance; instrument failure due to battery exhaustion occurred somewhat later (e.g. during night).

References

1. A. R. Jones, "A Versatile $\beta\gamma$ Survey Meter with Numerical Display", IEEE Transactions on Nuclear Science, Volume NS-24, No. 1, February 1977.

2. F. M. Glass, "Ionization Chamber Current Quantizer for Radiological Environmental Monitoring System", Oak Ridge National Laboratory (ORNL) TM-3906, August 1972.

3. E. G. Shapiro, "MOSFET Current-to-Frequency Converter with a Linear Sub-Picoampere-to-Microampere Range", IEEE Transactions on Nuclear Science, Vol. NS-18, No. 1, February 1971.

4. J. C. Nirschl and G. F. Healey, "Multidecade Hybrid Radiation Survey Meter", Nuclear Instruments and Methods 65, 1968.

A DIGITAL READER FOR CONDENSER IONIZATION CHAMBERS

K. Stuermer

The Condenser R meter ion chambers or dosimeters and their accompanying readout instruments have enjoyed a long history of stability and reliability. The simplicity of their operating principles and the uncomplicated nature of their measuring techniques and mechanism, involving only simple and quite stable optical and mechanical constants, has contributed greatly to their continued acceptance over the last forty years.

There was some hesitation, then, in the consideration of the development of a reader for condenser chambers which, by the very nature of the additional features felt desirable, would necessarily become much more complex and hence perhaps less reliable or precise. However, three features which were felt very desirable in a new reader far outweighed these possible considerations. The three features which are thought to be important to the more precise and efficient use of condenser chambers are:

1. A completely automatic reading/charging operation.
2. A modern digital readout presentation.
3. Two full decades of exposure readout for each dosimeter type.

Figure 1 shows a picture of the instrument which was visualized. The reader consists of a Formica case with handle and separable lid, about 30 by 24 by 18 centimeters - small enough to fit under an airplane seat. The readout instrument itself is in the center of the case with space for the power cord on one side and a disappearing holder for three dosimeters on the other. Four additional dosimeters can be stored in the lid.

At the top of the panel is the socket for dosimeter insertion. A readout panel which is tiltable anywhere from horizontal to vertical is in the center and a bank of push-buttons to select the dosimeter to be read and recharged is at the bottom.

In operation, the proper range for the dosimeter to be read/recharged must first be pushbutton selected to set up appropriate readout units. The buttons are interlocked and the selected button changes color to insure awareness of the correct selection. The dosimeter is inserted in the well after its back cap is removed. It is merely allowed to drop or slide into the well. The bayonet tab on the side of the dosimeter does not engage in this reader since the dosimeter cannot be inserted that far. The recharging process automatically occurs in an interval of less than one second and the exposure previously measured by the dosimeter is displayed on the readout panel along with decimal point and proper units. The dosimeter is then removed, its cap replaced, and another exposure may be performed.

Two of the push-buttons on the panel provide for special functions. The "V" button, when depressed, instantly causes the charging voltage of the reader to be displayed to the nearest tenth of a volt. The "C" button is used to check the capacitance of a thimble chamber. Depressing this button and then inserting a completely discharged chamber into the well causes its capacitance to be presented on the display in picofarads.

The readout has four-and-a-half digits, enabling a 25R chamber to be read out to the nearest 0.01R or 10 mR (a 100R thimble also can be read out to this precision). This provides an increase of over one decade of readout for the same dosimeter compared with quartz fiber type readers.

The reader has the dynamic range to read condenser chambers even if they are completely discharged. However, if a reading of more than rated full scale (for instance 25.01R on the Model 70-5) is obtained, an "Over-Range" indication appears on the display, as well as the nominal quantity with units, to indicate that the reading may or may not be valid depending upon the exposure and the exposure rate at which it was accumulated.

Figure 2 is a block diagram of the recharging circuitry. At the top left is the dosimeter socket. The recharging circuitry connected to its center electrode consists of three integrated circuits with auxiliary components: an integrator with integrating capacitor C1, a comparator, and a gated multivibrator. These components are all floated at +525.0 volts above ground, generated by a precision power supply. A dosimeter inserted into the well, upon making contact with the center pin causes a charging current, Ic, to flow. This current puts a small charge and voltage on C1 with the polarity indicated. This current is an exponentially decaying one with a time constant of 0.05 seconds or less, depending on the dosimeter capacitance and the value of R1. This voltage change at the output of the integrator causes the comparator to gate the multivibrator into operation and causes rectangular pulses of magnitude V and duration t to be generated. These voltage pulses, impressed upon R_{fb} cause similarly shaped current pulses to discharge C1 back below the trip level of the comparator. The action thus is for the circuitry to keep C1 at zero voltage and, in the process, recharge the dosimeter to 525.0 volts.

Each current pulse represents a charge of $V/R_{fb}xt$ coulombs and the circuitry is sized to produce 20,000 pulses at full rated exposure of each dosimeter type. The output of the charging circuit is a counter or register which accumulates and displays the number of these pulses, not a digital voltmeter. For convenience in displaying this information the 20,000 pulses for rated exposure are divided by either 2,4, or 8 by the circuitry to directly produce, 10,000, 50,000 or, 2,500 output pulses which directly give digits suitable for 100, 50, or 25 unit full-scale dosimeters.

Calibration of the instrument depends only upon R_{fb}, V, t, and the pulse count registered. It is independent of the value of the integrating capacitor C1. In practice, the magnitude of the peak value of the voltage pulse, V, is varied with internal adjustments to produce proper response for each dosimeter type.

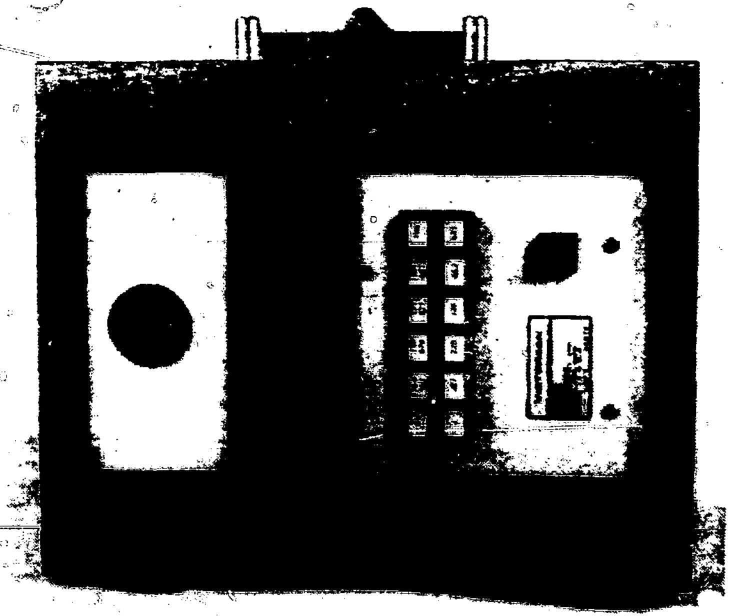
An infra-red source has been placed near the top of the well to shine an infra-red beam across the well to a photo-transistor. The interruption of this beam, located about 2.5 centimeters above the point where contact to the charging pin is made, occurs about 30 milliseconds before such contact is made and serves to initiate an automatic zeroing operation that takes place as the dosimeter is being inserted into the well.

Figure 3 shows the details of this zeroing circuitry. Interruption of the infra-red beam causes switches S1 through S5 to close for about 10 milliseconds. The dosimeter has not made contact yet at this time. The closing of these switches essentially makes a voltage follower out of both the integrator and comparator and forces the offset voltages of these operational amplifiers to appear on capacitors C2 and C3 respectively. Then, when these switches again open 10 milliseconds later (again before the dosimeter is contacted) the offset voltages of both the integrator and comparator are held for a few seconds at least on C2 and C3 respectively, and the negative input of the integrator and the positive input of the comparator are at exactly zero volts, dynamically ready to accept any signal without offset errors that would otherwise exist while the offset voltages are overcome. Actually, S1 is a reed switch because of the impedance levels required at the input stage and S2 through S5 are all contained in a single integrated circuit.

The measurement and presentation of the precision charging voltage also is accomplished by switching when the "V" button is depressed. Figure 4 shows the detailed circuitry for this feature. Actuation of button "V" causes S6 to remove R6 from across C6 and connect this discharged capacitor to the input. The circuitry then charges C6 to 525.0 volts and presents the result, properly scaled, as 525.0 volts on the display. S6 actually consists of two reed switches, each rated at one kilovolt to handle the high voltage in this portion of the circuitry.

The precision voltage supply is referenced against a National Semiconductor type LM 299 precision reference device which has a maximum long term stability of 20 parts per million for 1000 hours of operation, and a maximum temperature coefficient of 0.0001% per degree centigrade. This extreme stability, together with a matched temperature coefficient resistance divider and an operational amplifier with offset voltage drift less than 0.5 millivots has resulted in drifts in this voltage of less than ±0.05 volts over three months of almost continual operation. This represents about ± .02% of full scale exposure on a dosimeter and opens up the possibility of permitting charging on one reader and reading out and recharging on another.

The linearity of the reader is as good as the number of digits presented, because the charge lost due to leakage currents in the integrator operational amplifier, the integrating capacitor C1, and the reed switch S1 are held to less than one significant digit of the readout. This is accomplished by the auto-zeroing operation previously described and by stopping the acquisition of pulses one second after interruption of the infra-red beam. The recharging of even a completely discharged dosimeter has been completed to within 0.005% of its final value by this time, so any additional pulses are caused by leakage currents. This disabling of the counter one second after the infra-red beam is interrupted has however resulted in improper operation of the reader when a dosimeter is inserted extremely slowly. In this case none, or only part, of the recharging pulses may be counted before the one second cut-off and the exposure reading is either zero or is too low. This shortcoming in this bread-board version of the reader is presently being attacked in two ways. One is the upgrading of the integrator operational amplifier to a lower leakage current variety and the extension of the cut-off time to , say, 5 seconds. The other is the modification of the one second timing function to start when the dosimeter is actually contacted and recharging starts. This latter method is possible since there is a very short voltage pulse which appears at the output of the integrator when contact is actually made and this signal can be used to start the counter enabling time.



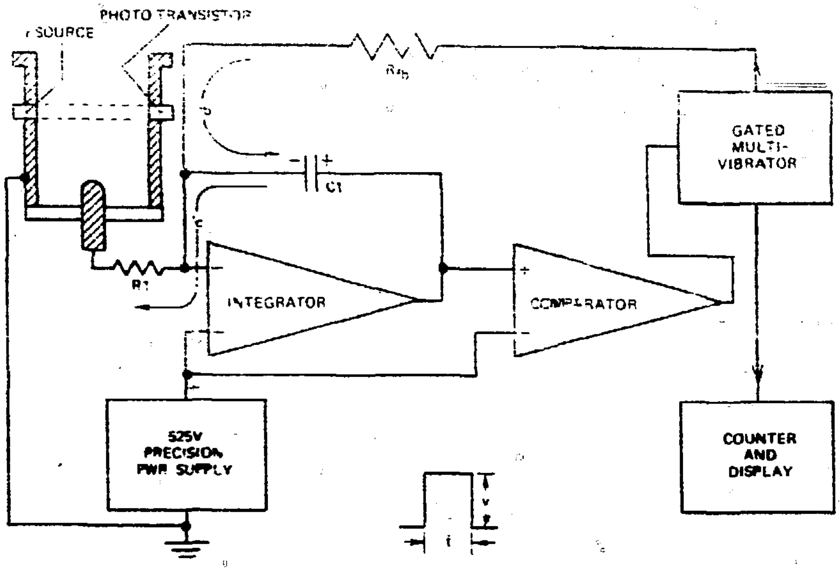


FIGURE 2. BLOCK DIAGRAM

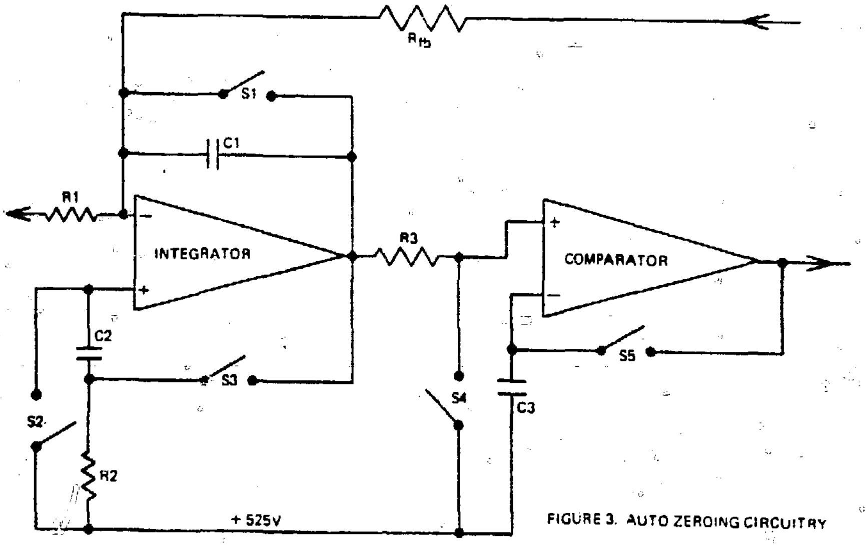


FIGURE 3. AUTO ZEROING CIRCUITRY

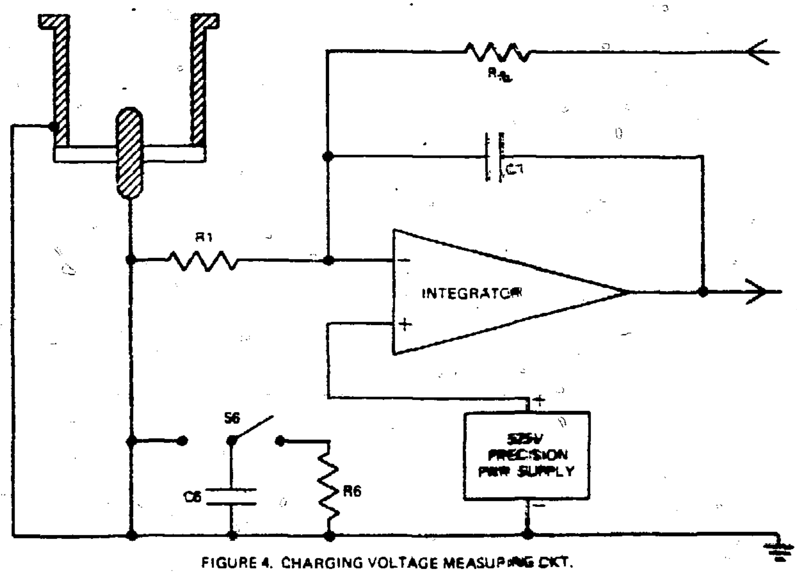


FIGURE 4. CHARGING VOLTAGE MEASURING CKT.

MODIFICATION AND USE OF A VICTOREEN MODEL 550 ELECTROMETER FOR CALIBRATION

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ABSTRACT--The calibration system consists of a Victoreen Model 550 Electrometer, with ion chambers, which has been modified to incorporate a digital timer which is tripped at a predetermined value set on the electrometer. This system provides a precision, standardized gamma and x-ray calibration system which is calibrated at a Regional Calibrations Laboratory accredited by the AAPM. The modified system is designed to be portable for use in laboratories throughout the University and can be easily taken apart for ease of handling. The electronic modification is diagrammed and discussed as to ease of installation and compatibility with the entire system. Comparisons are made between a Standard versus Modified Model 550 as to accuracy, precision and utility of operation. The comparisons are based on measurements taken of a ten curie Cobalt Sixty Calibration Well and several x-ray units. The calibration well exposure rates were varied over the range five mr/hr to 20 R/hr and x-ray exposures varied between several R/min to several hundred R/min and for a number of energy spectra.

INTRODUCTION

Many experimenters have had the annoying occurrence of setting up an experiment with R Chambers only to have the timer or stop watch not being set, leaving chambers overexposed, removing and replacing chambers, etc. The advent of semi-automated secondary standard ionization chamber measurement systems, such as the Victoreen 555 and 550, MDH, NEI and other commercial units solved the problems of removing the R Chamber after each exposure and leaving the chamber in too long, etc.

However, the major problem of timing was not fully addressed by commercial manufacturer, Victoreen models, for example, have a preset exposure module but no timer and no latch on the readout. For the busy Health Physicist it is impractical to sit around for five to twenty minutes per exposure, then reset the electrometer and take another reading. A commercially available electrometer and ion chamber modified in house and coupled to a commercial timer was considered as the best compromise solution to the problems inherent to current systems and R chambers.

The present work has been undertaken then to develop an exposure measurement system which insures ease of measurement, timing, latched readout reproducibility in measurement, minimal exposure to the operator and ease of transport of the system. Optimization of system reproducibility (precision) over long periods of time is required to insure repeatable calibration points for TLD, survey instruments, and various experimental measurements. This optimization has been accomplished for a system consisting of a Victoreen Model 550 electrometer with ion chambers and a Datel digital timer for primary use in a seven meter calibration well, as well as other gamma and x-ray facilities on the San Jose State University campus.

The first section of this paper will describe the electronics modifications. The second section contains a description of the calibration well, the calibration procedure, and the various calculations needed to determine the precision of this system. Independent measurements were made with various chambers and compared.

EXPERIMENTAL METHODS
PART I

The modifications of the Victoreen Model 550, Radicon III are in the function switchs and the exposure trip module. The rest of the circuitry stays the same. A block diagram of the modified system is shown in Figure 1.

The function switch modification involved one trace cut between the center contact of the left section of the hold push button switch and U5 pin 2 (see Figure 2). Six wires were added to the function switch. An amphenol 9 pin (J4) hex connector was added to connect the lines to the interface.

The Exposure Push Button (P.B.) switch, right section, is used to generate the signal, $\overline{\text{Exposure}}$. The center contact is connected to pin E of the hex connector (J4) and the left contact is grounded (see Figure 3). The right contact is not used.

The Hold Push Button switch, right section, has the right contact connected to J4 pin B for the signal $\overline{\text{Tout}}$. The left and center contact stay the same.

In the left section of the Hold P.B. switch, the center contact generates the signal $\overline{\text{Hold P.B.}}$ and it is connected via J4 pin E. The left and right contact stay the same, except a wire is added from the left contact to J4 pin H for the $\overline{\text{Print}}$ signal.

The Zero Push Button, right section, the center, left and right contact stay unchanged. However, a wire is added to the center contact to J4 pin A for signal $\overline{\text{Zero}}$.

U5 pin 2 is connected to J4 pin F to provide the signal $\overline{\text{Print}\cdot\text{Tout}}$ + $\overline{\text{Hold P.B.}}$

The Exposure Trip Module modification utilizes the unused amplifier (U4(LM324) (see Figure 4). A +3V reference voltage was added to the inverting input (-). The output of the Trip Module (pin 9) connects to the non-inverting input (+) of the unused amplifier (U4). This amplifier is used as a comparator. The output of the comparator pin 7 is connected to J4 pin D (Figure 5).

For the logic circuit in the timer external circuitry, the +5V power supply and ground of the Victoreen 550 is used. The connections are J4 pins K and J.

The timer consists of a Datel 8200, with manual and electrically operated start-stop-reset functions. A 5 mHz crystal controlled oscillator is used for precise timing. The external circuitry is used to interface the timer to the Victoreen 550.

The external circuitry consists of a latch, monostable multivibrator, and gates (see Figure 6).

The latch is emulated by two "Or gates" U3 (7400). They are cross coupled, one input is connected to the $\overline{\text{Exposure}}$ and the other is $\overline{\text{Zero}}$.

The monostable multivibrator U5 (74121) is connected to the output of the latch and it generates the signal $\overline{\text{Start}}$. The signal $\overline{\text{Start}}$ is 1 μsec. wide.

The Trip (max counts) signal is inverted to produce the signal $\overline{\text{Stop}}$ which will stop the timer and $\overline{\text{Tout}}$. This condition holds the digital readout of the Model 550.

The $\overline{\text{Zero}}$ signal is buffered by U4 (7404) to produce the signal $\overline{\text{Reset}}$.

The signal $\overline{\text{Tout}}$ and $\overline{\text{Print}}$ are "Anded" by U1 to produce the signal $\overline{\text{Print}} \cdot \overline{\text{Tout}}$.

The signal $\overline{\text{Print}} \cdot \overline{\text{Tout}}$ or $\overline{\text{Hold P.B.}}$ are "Ored" by U2 (7408) to generate the signal $\overline{\text{Print}} \cdot \overline{\text{Tout}}$, and $\overline{\text{Hold P.B.}}$ S1 is a 5 contact rotary switch which produces the decimal point of the timer. S2 is also a 5 contact rotary switch which multiplexes the time base out signal to the time base in signal. S1 and S2 are ganged mechanically to present the right time in seconds.

The description of the external circuitry or the interface is presented as follows (refer to Figure 6):

When $\overline{\text{Zero}}$ is active, it enables the latch to be in a ready state, resets the timer and enables the zero coil at the Victoreen 550 to be able to zero the D.P.M. (Digital Panel Meter).

When the Exposure P.B. is active, it produces the signal $\overline{\text{Exposure}}$. This signal generates the start pulse and triggers the timer to start timing. It also commands the Victoreen 550 to start counting. When the preset number in the Exposure Trip Module is reached, the signal $\overline{\text{Trip}}$ is generated. This means the counted event is equal to the preset event. The $\overline{\text{Trip}}$ signal generates the $\overline{\text{Stop}}$ signal for the timer and $\overline{\text{Tout}}$ for the Victoreen 550 (this holds the displayed number).

$\overline{\text{Print} \cdot \text{Tout}} + \overline{\text{Hold P.B.}}$ are used to enable the zero coil. This gating function was designed not to degrade the integrity of the operation of the Victoreen 550.

PART II

The source used in this study was an ICN Co-60 source, 10 ci ± 10% in September 1973, identified as Model 378. The Co-60 wire source is doubly encapsulated in 17-4PH stainless steel (.735 inches long and .276 inches diameter) with sufficient ferrite to allow for handling with an electro magnet. The source is housed in an aluminum bucket with a De Staco clamp holding a thin aluminum cover to prevent the source capsule being ejected during any sudden stops, etc. The complete housing is shown in Figure 7. The source housing is raised and lowered by gear driven linked chain. The source position is obtained on a digital read out. A fifty turn pot is mounted on a gear shaft and the output signal is digitized.

The source housing is located in a 7.5 meter well. The well is covered with a four inch steel sliding door filled with lead shot. The attenuation factor of the door is approximately 200. The usable calibration range is from 7 M to 0.25 M, corresponding to an exposure range of approximately 1 mr/hr to 60 R/hr, utilizing both the closed and open door conditions. The maximum personnel exposure is 100 mr/hr at the calibration well console with the source at .25 M, and with the well door open. The console location for the Victoreen 550 is behind a six inch concrete wall. The personnel exposure level at this console is minimal.

The San Jose State calibration well permitted replication of the majority of exposure rates and beam qualities encountered in many other well type calibration facilities. In this study, exposure rates from 1 mr/hr to 60 R/hr were measured. Intensity was controlled by raising and lowering the source in the well. Beam quality was assumed to be uniform for all measurements. However, it is noted that there is a small difference in the energy spectrum of the beam, a result from scatter, and this factor is inherent in most well type facilities. Exposure measurements were initially made with a standard Radicon III Model 550 equipped with model .01, .1, 1, 10 MA probes with equilibrium caps.

These probes were secondary standards for San Jose State University whose calibrations were made at the certified (pending) AAPM Regional Calibration Facility, Cleveland, Ohio. The beam field size was 30 cm and was constant for all exposures. In all measurements, each probe was centered over the well 4.0 cm above a safety glass plate which covers the well opening. Initial measurements were made by simultaneously depressing a stop watch and the Exposure button on the Victoreen 550. At the end of the exposure period, the stop watch and Hold button were simultaneously depressed. The combined errors due to the timing and start/stop methods were estimated from the precision of the measurements to be in the range of ± 1.5 to 3.5%.

The positioning of the source had an uncertainty of ± 1 cm, the resultant maximum uncertainty at the closest position was ± 4%. Reported inaccuracies by Victoreen for the system excluding the probes were estimated by the author to be approximately ± 0.6%. The uncertainty of the probes traceable to NBS was ± 1%

for the 0.1, 1, 10 MA probes and ± 2% for the 0.01 MA probe at Co-60 energies. An uncertainty of 1.5% for geometry factors stated by Victoreen appears to be high and are realistically less than ± 0.5% for use in the SJSU calibration well. The uncertainty for changes in beam quality due to scattering as a function of source position in the well was estimated to be on the order of ± 1%. The value was estimated from the well configuration and Monte Carlo calculations reported by Berger & Raso(1). The maximum uncertainty for the system was calculated to be 7.5% (.01 MA probe) and 6.5% for the others for the unmodified system under the worst geometry and exposure rate conditions.

For measurements with the modified system, the uncertainty for the .01 MA probe is ± 6.4% and ± 5.4% for the other probes for the same conditions as was previously stated.

For measurements with the modified system, the uncertainty at one meter or more and under nominal conditions is 4% for the .01 MA probe and 3% for the other probes.

RESULTS

The precision of the the system with the .01 MA probe is shown in Table I for all ranges. Table II shows the comparison between probes at 2 meters. Within the error limit of the study, no non-linearity effects were observed on any of the scales except for low dose rates and short exposure periods on

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(1) Radiation Research 12, 20-37, 1960.

TABLE I
MEASUREMENTS AND PRECISION .01 MA PROBE - Co-60

Mean Exposure Rate (mr/hr)	Well Position #	Percent Error of Set
1.25	7-closed door	1.5
1.62	6	0.8
2.40	5	2.0
3.64	4	0.6
6.35	3	0.4
13.65	2	0.3
45.67	1	0.4
109.7	0.5	0.1
168.5	7-open door	0.7
230.	6	0.1
335.	5	0.1
563.	4	0.1
960.5	3	0.2
2283	2	0.1
7738	1	0.1
23,560	0.5	0.2
58,220	0.25	0.5

TABLE II
COMPARISON OF MEASUREMENTS OF 4 PROBES AT Co-60 ENERGY, 2 METERS

Chamber	Mean Exposure Rate (mr/sec)	% Error Variance of Probe to Mean*
X.01	.5368	4.8
X.1	.5177	0.8
X1	.5210	0.8
X10	.5207	0.8

*Mean based on the mean of each set of the .1, 1, 10 probes

the most sensitive scale. There is a discrepancy between the .01 MA probe calibrated at Victoreen and the other three probes calibrated at the AAPM facility of about 4%. With the cooperation of Victoreen, we rectified the problem.

Our observations indicate that over a reasonable timing period, one to two minutes and dose rates greater than 1 mr/sec, that the precision and accuracy of the modified system and a a standard system with a stop watch are comparable within 2%. The precision within each set of both methods are not statiscally different. However, for long duration exposures (low dose rate) the major sources of error were source position and timing, three to four percent greater total error with a stop watch.

The presented data represent averaged values except when only one data point was used. The data were fitted using a Variance and S.D. program on a SR-52 calculator.

DISCUSSION

The data for the exposure points indicates that the precision and accuracy of this system are better than ± 5% for the majority of exposures. This accuracy must presume that the beam quality, exposure rates and exposure fields are within a normal variance of those parameters in which the data points were obtained.

It was determined in field use that the system could be easily transported by a single person and completely set up and ready for operation within thirty minutes.

CONCLUSION

The results of the modification to the Victoreen Model 550 Electrometer, ionization chamber system provide a secondary standard system for measuring exposures or exposure rates from 25 KeV x-ray units to Co-60 sealed sources. The measurements can be made easily and accurately throughout a university complex. The electronics modifications can be accomplished in a short period of time for only a modest expediture.

NOTE: The mention of commercial products, their sources, or their use in connection with material reported herein is not to be construed as either an actual or implied endorsement or adverse evaluation of such products by San Jose State University or the California State University and College system.

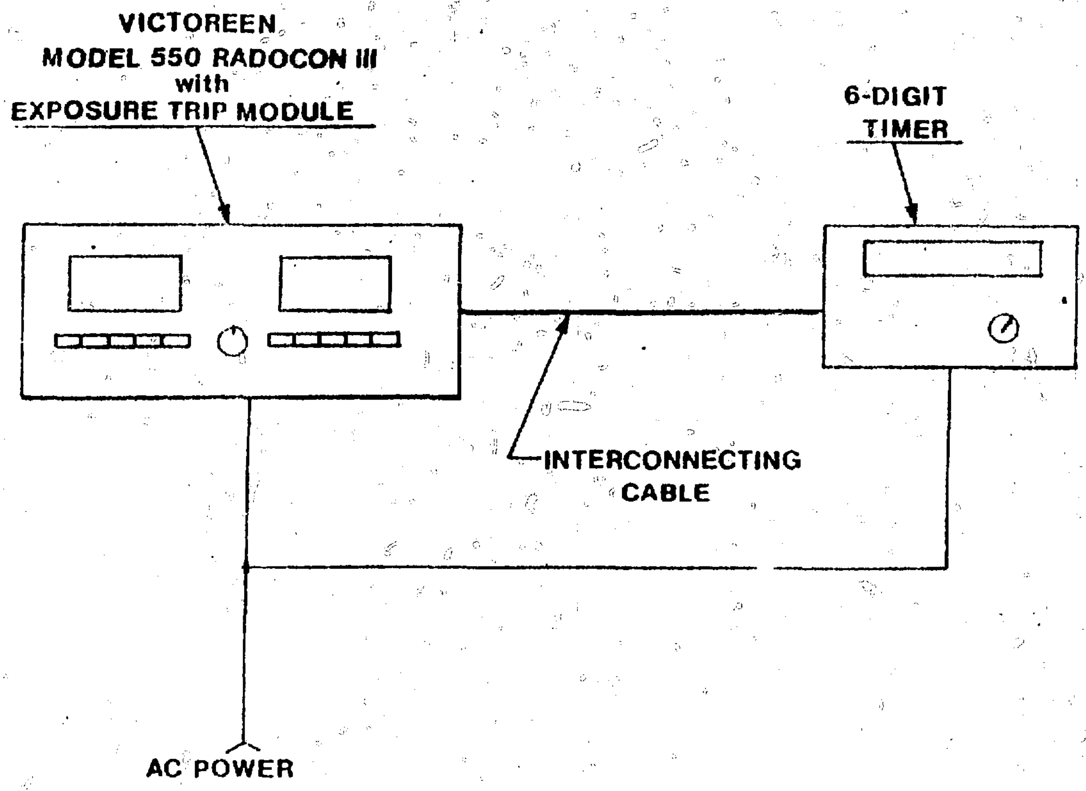


fig 1. SYSTEM BLOCK DIAGRAM

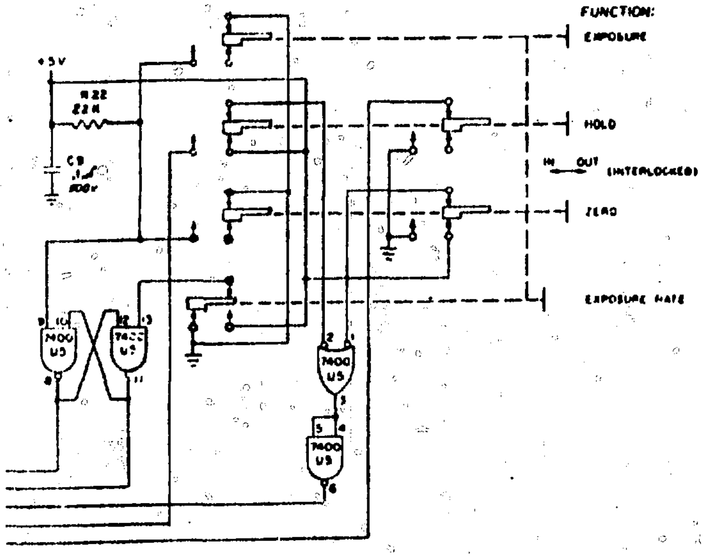
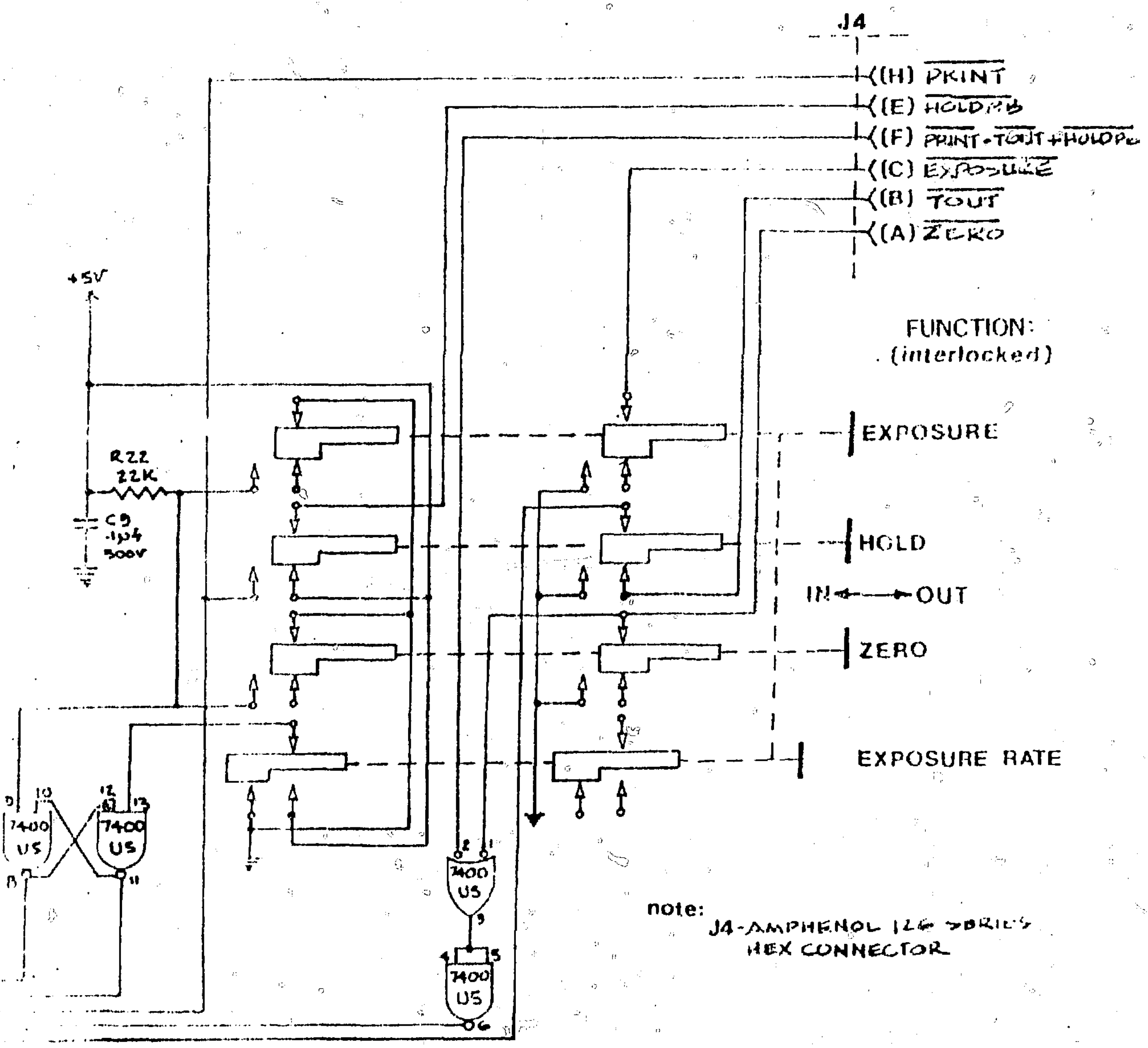


fig 2. UNMODIFIED 550 FUNCTION SW.



note:
J4-AMPHENOL 126 SERIES
HEX CONNECTOR

fig 3. MODIFIED 550 FUNCTION SW.

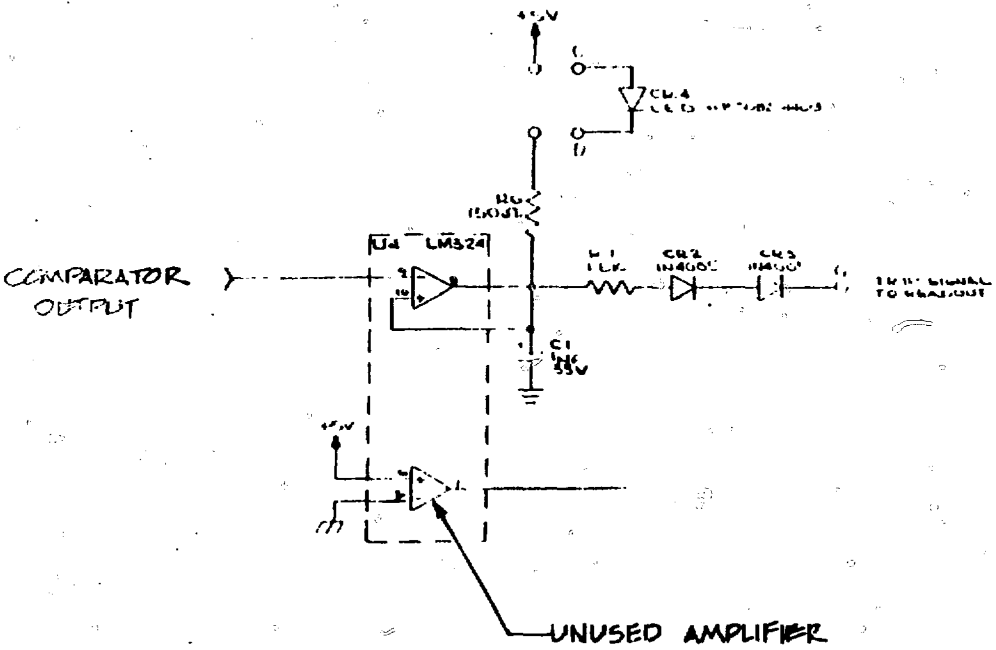


fig 1. UNMODIFIED TRIP MODULE OUTPUT

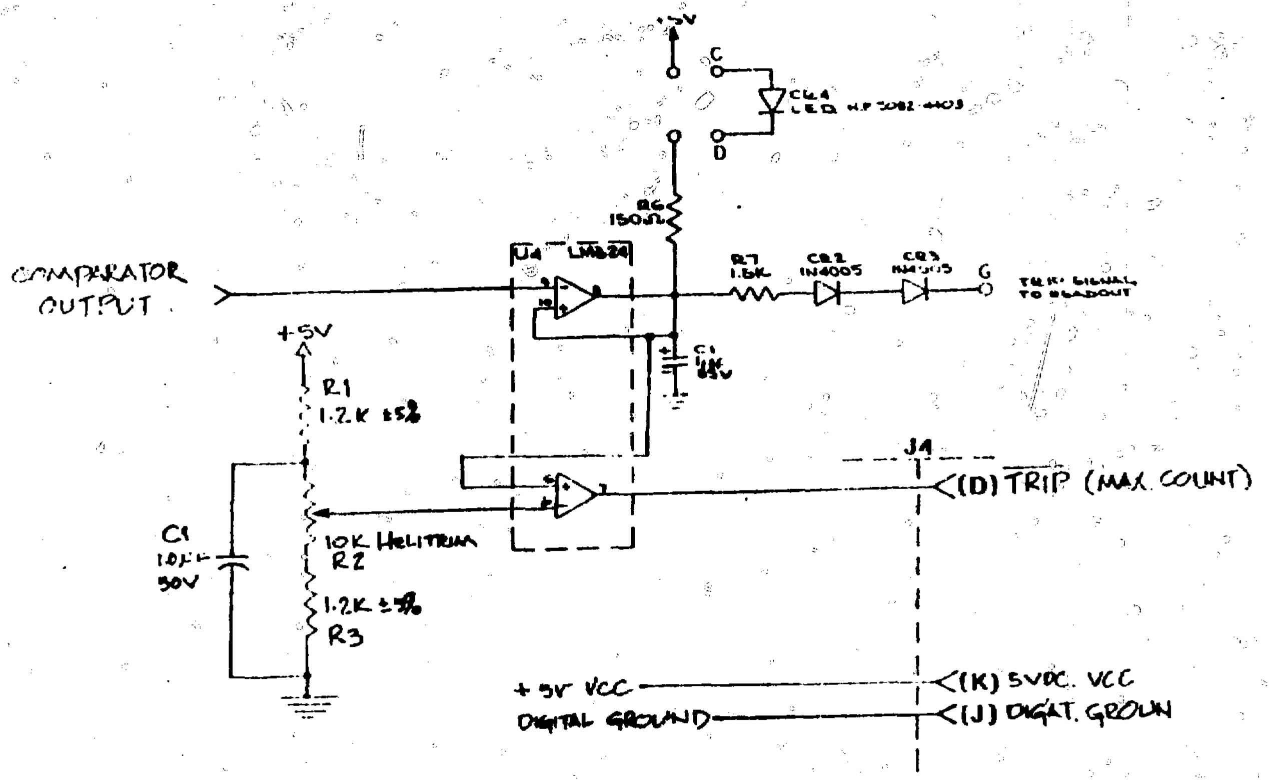
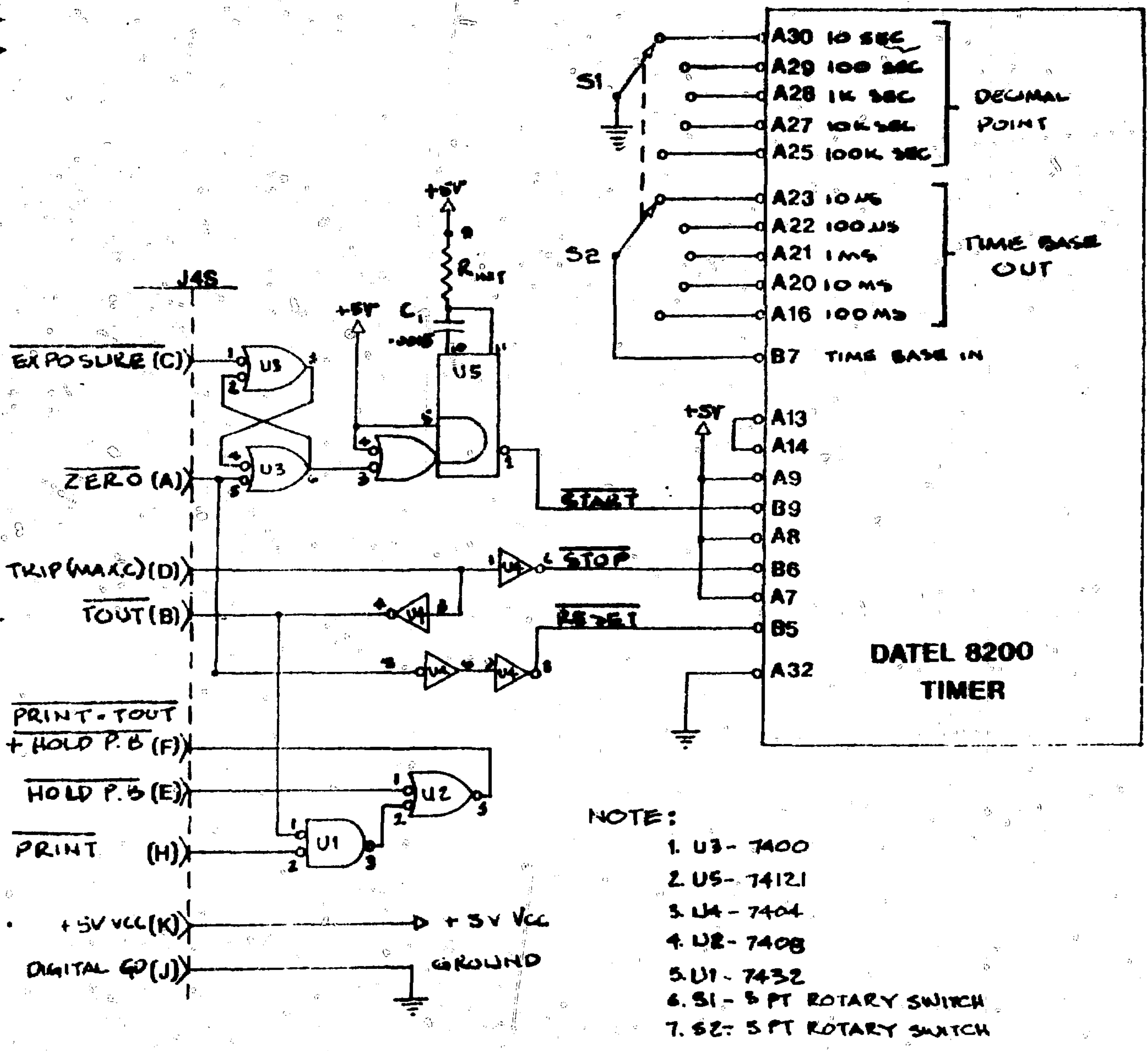


fig 5. MODIFIED TRIP MODULE OUTPUT



NOTE:
1. U3 - 7400
2. U5 - 74121
3. U4 - 7404
4. U2 - 7408
5. U1 - 7432
6. S1 - 5 PT ROTARY SWITCH
7. S2 - 5 PT ROTARY SWITCH

fig 6. TIMER SCHEMATIC

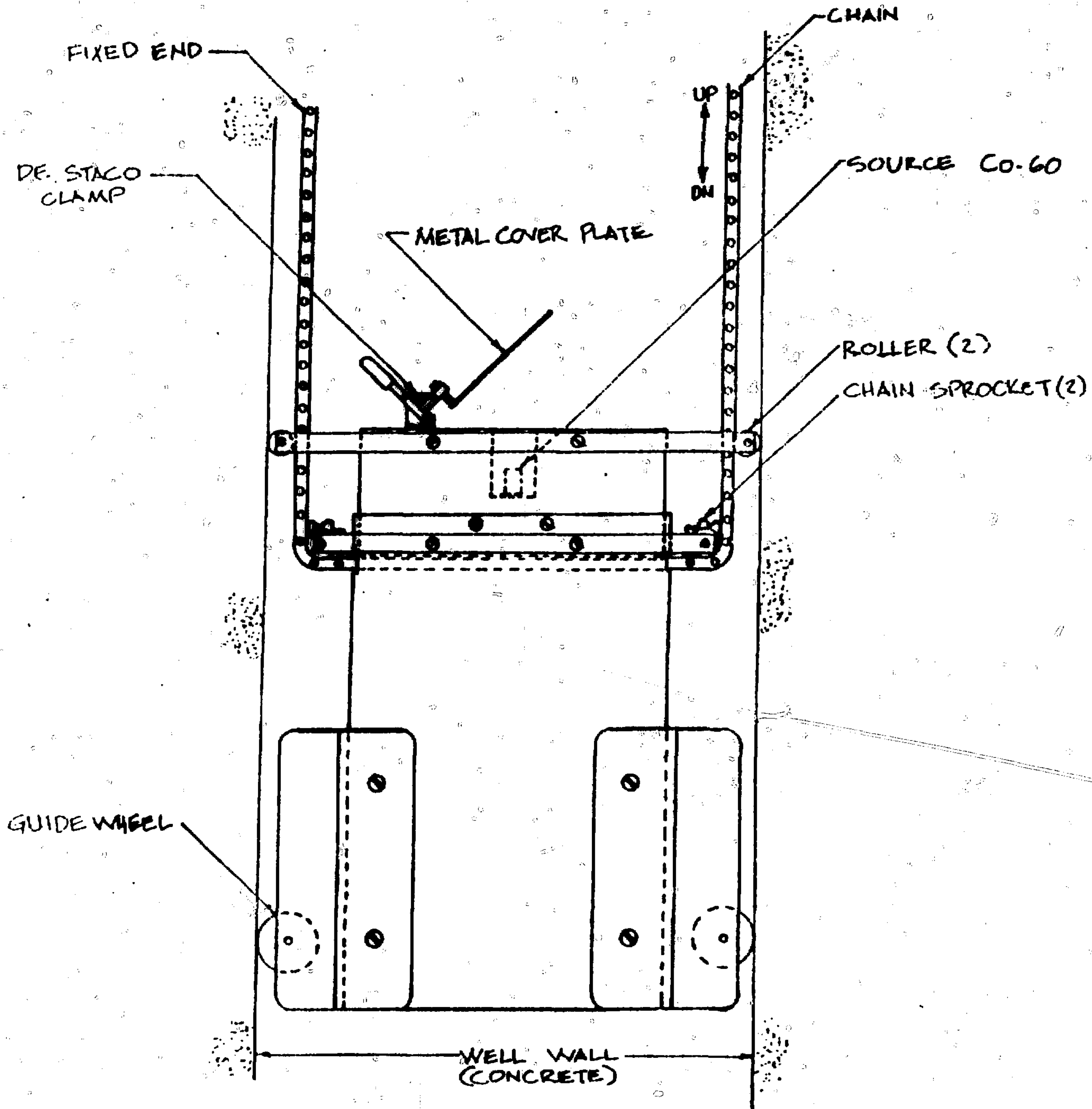


fig 7. SOURCE CARRIER

A COMPLETELY NEW APPROACH TO PORTABLE INSTRUMENTATION

by J. Little
Eberline Instrument Corporation

ABSTRACT

Historically, survey instruments have been hand held, battery-operated with limited detector interchangeability, limited computation capability and limited data presentation. In general, the instruments were ratemeters that used either jeweled or taut-band meter movements. Invariably, these "meters" were the weakest link in the instrument's operating life. Each instrument also required complex calibration for each detector type. Usually, each survey meter was married to a specific type of detector and was used for only one application.

A new type of portable instrument has been developed. It does not use meter movements, has a very simple calibration procedure, will operate with a great variety of detectors with simple interchange of detector probes, provides more flexibility in digital presentation of the data, and is capable of doing many things that in the past were done by calculators or computers. These new instruments also tend to be smaller and more rugged than their predecessors. This paper is intended to show some of the possibilities of usage and the ease of maintenance and calibration of these new instruments.

DISCUSSION

Portable survey instruments have been limited primarily to ratemeters dedicated to a specific application. Because of statistical fluctuation and built-in instrument time constants, these instruments are not very successful in low radiation fields. Their use for alpha and neutron monitoring is limited because of this. As new requirements for lower and lower readings have evolved, new methods of monitoring have become necessary.

One of the first steps for better low level monitoring instrumentation was the portable scaler. It has the ability of monitoring low fields by counting or collecting the data over long periods of time. These instruments are straight pulse counters so they have no internal time constants limiting them. Thus the accuracy of scalers is limited by how much time you want to spend collecting data. However, a drawback to scalers is that they provide data in terms of counts per time only and the user must convert the data into a more useful format.

Another step has been taken by combining the advantages of the portable scaler with internal data handling for digital display in the units of interest. The resulting ratemeter-scaler allows the operator to preselect the desired precision of measurement and engineering units for digital display, with all calculations performed by the instrument at the time of measurement. The portable ratemeter-scaler is similar to the portable scaler in that it is a pulse counter and it is similar to the ratemeter in that it displays data in a rate format. It uses all solid-state circuitry including the readout which gives it much better reliability and durability.

The portable ratemeter-scaler also has a very sensitive input with a wide range of adjustment so that it may be used with a large variety of detectors; scintillation, gas proportional, Geiger-Mueller and other types of detectors may be used.

The greatest improvements offered by the ratemeter-scaler are in the data handling abilities and in its accuracy. The data handling allows the user to select a method of readout; i.e., mR/hr, dpm per 100 cm^2 (see Figure 1), and then dial in the conversion factor for the particular type of detector. The instrument then displays in the readout format desired. This is accomplished by using a calculator in the instrument and by doing either a multiplication or a division on the raw data. Almost all conversion factors can be broken down into either a simple multiplication or division operation.

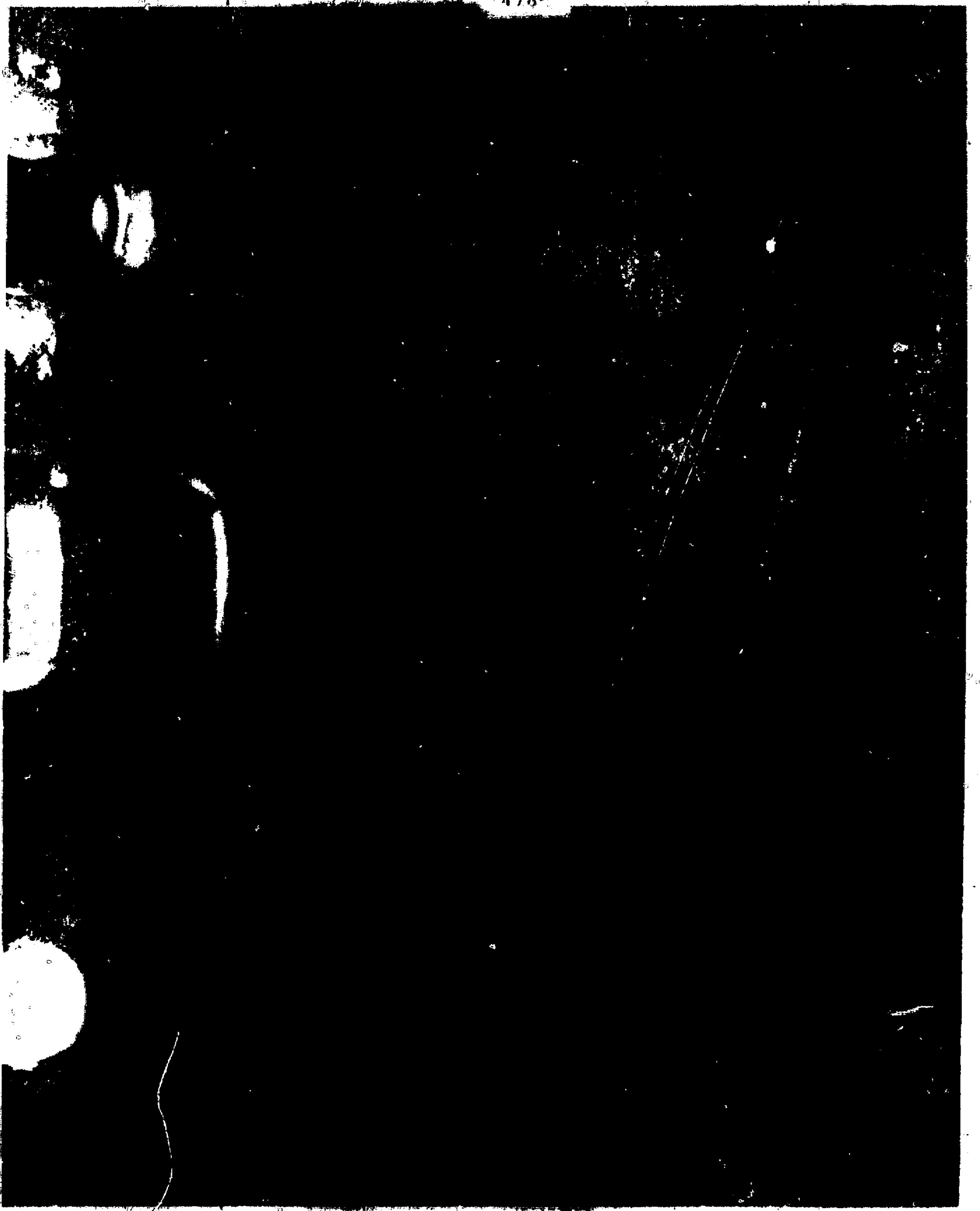
For example, a common energy compensated G-M detector has a conversion factor of approximately 1200 cpm per mR/hr. Using this detector with the ratemeter-scaler multiplier board, you would first do the conversion 1/1200 = .00083 and then dial .00083 into the multiplier board (see Figure 2). The instrument then uses this factor and displays the readout as a multiple of it. In this example, the readout would be in mR/hr. If the divider board is used, then 1200 would be dialed in and the readout would be displayed in mR/hr.

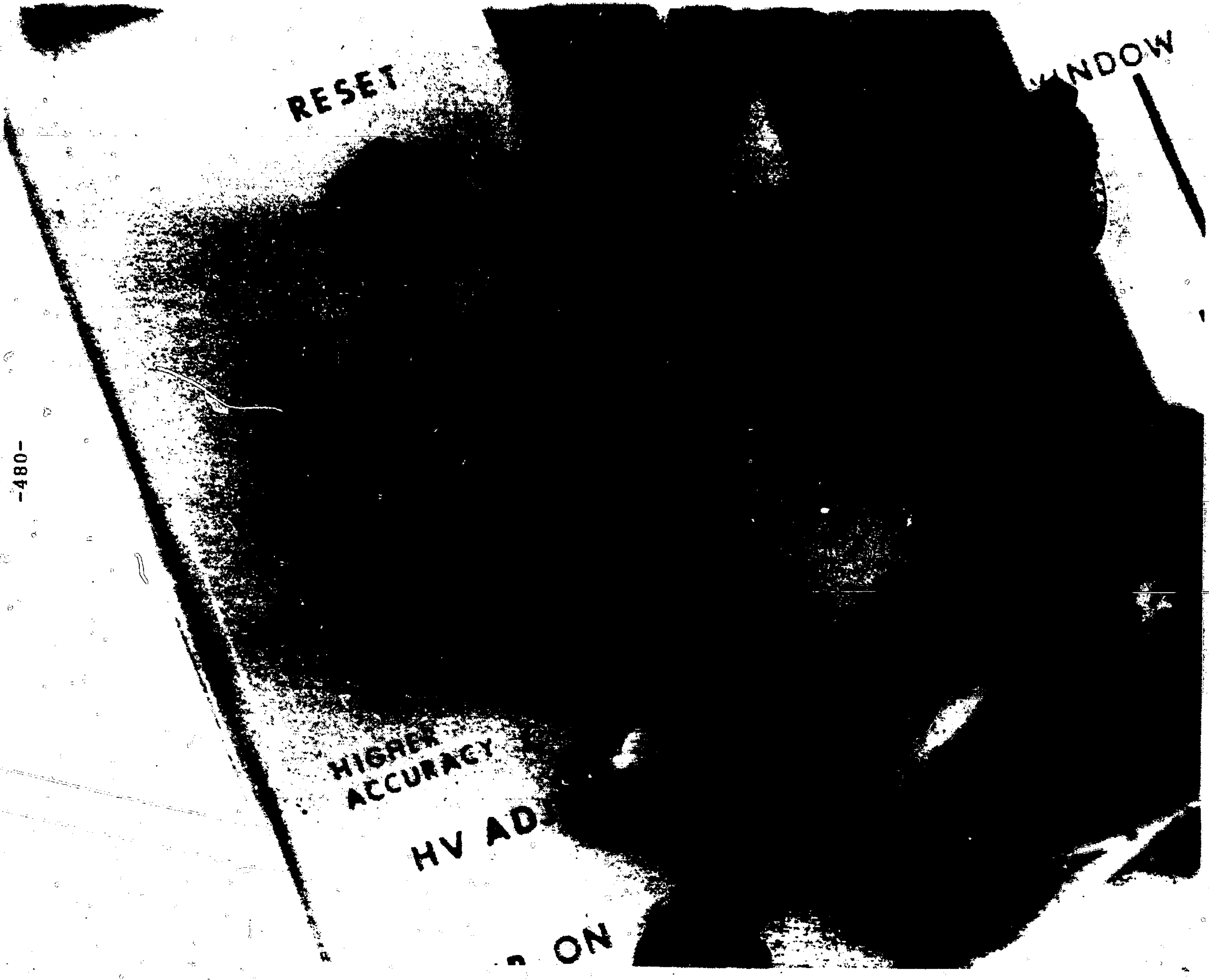
The accuracy that you want from a particular reading is also selectable by the user. The ratemeter-scaler has an accurate clock built in and it uses this to determine the time to accumulate a certain amount of data. The user can select the amount of data needed. The user can select 10, 100, 1000 or 10,000 counts, depending on precision desired and how long the operator is willing to wait to achieve the desired precision. The standard deviation for 10,000 counts is a much smaller precentage of total counts (1%) than the deviation for 10 counts (31%). Thus, if you want good precision, you select the largest number of counts available. If you want speed and don't need high precision, you select the smallest number of counts available. You may select one of the intermediate positions if you want a compromise between speed and precision. These selections are marked A, B, C or D. A would give the fastest result and the poorest accuracy. D would give the slowest result but the highest accuracy (see Figure 3 and Figure 4).

Because of the ratemeter-scaler's ability to operate with a large variety of probes, high voltage also is displayed on the readout. This allows the user to quickly interchange probes by merely readjusting the high voltage to the setting for the specific probe, make a change in the conversion factor and continue operating. These data can be predetermined for the individual detectors and attached to the instrument for ready access. In some cases, a change of instrument input sensitivity may be necessary. These few steps consist of the entire calibration of the ratemeter-scaler. There are no ratemeter range calibrations. Because the ratemeter-scaler is strictly a pulse counter, verification of operation is also extremely simple. You only have to check that it counts pulses and that the instrument operates on the pulses giving the proper digital display. A radioactive source is needed only to verify that the detector operation is consistent with the previously determined response factor for that type detector.

SUMMARY

The ratemeter-scaler has several advantages over portable instruments previously available: ease of operation, ease of calibration, ease of verification, direct readout in meaningful format and interchangeability with most detector types. This means that you can now have one instrument that does what it took many ratemeters or scalers to do previously. Each type of measurement can be displayed in the format most usable for that measurement: alpha in dpm/100 cm^2, beta in cpm/100 cm^2, gamma in mR/hr and neutron in mrem/hr. In one portable suitcase, you can have the equipment to do alpha, beta, gamma and neutron measurements (see Figure 5).




RESET
NDOW
HIGHER ACCURACY
HV AD
ON

Accumulated Data	Standard Deviation	% of Accumulated Data
10 counts	3.16	31
100 counts	10	10
1,000 counts	31.6	3.1
10,000 counts	100	1

Figure 4. Accumulated Data, standard deviation and deviation as % of accumulated data

TECHNICAL SESSION I

PERSONNEL MONITORING

Chair: Gene Tochlin

Session Chair, Eugene Tochlin, filling the gap of M. Moreno paper.

I'm going to spend a little time discussing what two health physicists are doing at a place called Varian. Actually, we're doing quite a bit and the two of are Bill Laraveer and myself who have been at Varian for two or three years. By way of introduction, we are in the radiation division and the main product is our medical linear accelerators which are used for cancer therapy. In addition there are also a line of industrial linear accelerators extending from the medical x-ray accelerators have full time energies extending through 4 to 25 mv with electron energies' capabilities as high as 35 mev with one of our machines. At the same time the various industrial units have x-ray energy ranges from two to ten MeV. The maximum dose rate from the highest energy machine is the rather staggering 6,000 rads per minute at one meter, which is kind of spooky. If you're concerned with radiation safety, it's certainly something to be concerned with. Now the type of work that Bill and I do is more recently classified as medical health physics or since we're not really associated with a hospital, perhaps paramedical health physics would be more appropriate. The type of problems that we have gotten into are somewhat the following:

First of all, you may be aware of the fact that xrays at these very high energies are highly directional and therefore, in order to provide a uniform field for radiation measurements, one needs a beam flattening filter. The design of beam flattening filters can be either a trial and error procedure or one that is based on lot of accumulated data which allows one to design for whatever energy one has. This is something that we have been doing in the past. It has been done by consultants; now we have taken this upon ourselves. Another large part of our work has to do with leakage measurements from the xray machine per se. Regulations specify that leakage not exceed 1/10 of 1% of the primary beam at one meter, and of course this has to be carefully evaluated in order to be sure there aren't any hot spots. For this purpose we have built what we call a meter arc, which is a rather nifty arrangement. It looks somewhat like a crescent moon. I'm sorry I didn't bring any slides, but it consists of a semi-circular unit which is suspended around the isocenter of the machine, suspended below and above the machine so that we have this arc one meter from the target itself. The unit contains 14x17 films; in addition it contains TLD docimeters and two are ion chambers. Our goal is to make exposures which will give us leakage radiation the order of 1 to 2R in this general region. We calibrate the film, the TLD's, the ion chambers with cobalt 60. We find amazingly good reproducibility in the dosimetry corner where are three dosimeters are inter-compared. General speaking the agreement within 3 docimeters is better than 20%, and this is something better than we had expected, primarily with the film itself. At these higher energies the film behaves very orderly. As I say, we have one dosimetry corner which in effect calibrates the system, then we can look for gradients in the films and other areas which would define the hot spots, and since the film is calibrated now we can define the higher leakage areas in terms of percent of the primary beam. This meter arc is rotated through about 15 positions until we're all through we have hundreds of films to look at. One of our goals and this is rather interesting, I've always been used to working in a laboratory type of environment, here this is a factory and the goal is to get machines out once they've been tested, evaluated and found to meet specs. They leave in a hurry, and much of our work has to be done on the spur of the moment. We literally have to be thrown out of rooms sometimes because machines have to leave. So we have to work fast, we have to devise dosimetry the techniques that take a minimum of machine time. Perhaps a somewhat longer period than that after the fact, but since getting the machine tested by an engineering group is the prime requisite here, we kind of have to work around this and we learn to move rather swiftly for this reason. Now another thing we do quite a bit of is room shielding design. This is one of the original purposes of the medical health physicist at Varian. We provide new shielding designs for new rooms which is a large part of it, and in many cases redesign, readapt, rooms with the old, existing units. Perhaps old CO^{60} or betatrons, rooms, to adapt to our machine. We have also made measurements of the TVL of the leakage radiation of our some of our machines.

PHOTON AND BETA RESPONSE OF A NEW THERMOLUMINESCENT DOSIMETER BADGE

by

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George J. Littlejohn
Ellery Storm

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ABSTRACT

The Los Alamos Scientific Laboratory is in the process of converting from a film to a thermoluminescent dosimeter (TLD) badge. The new badge is made of Cycolac plastic and contains an aluminum card in which are mounted three TLD-700s and one TLD-600 LiF chips. One TLD-700 and the TLD-600 are within a cadmium filter and are used to determine the neutron exposure. The photon response of the TLD chips was measured over an energy range of 10 to 1000 keV. A TLD-700 chip, covered by a 90 mg/cm^2-thick copper filter embedded in a 250 mg/cm^2-thick Cycolac, was selected to measure penetrating radiation. A 60 mg/cm^2-thick Cycolac filter is used to measure nonpenetrating radiation. The photon energy and angular response of the TLDs under the penetrating and nonpenetrating filters is presented. The response of the badge to beta radiation, varying in maximum energy over the range of 0.77 to 2.3 MeV, is given. Finally, the accuracy of the new TLD badge in measuring photon radiation from plutonium is discussed.

I. INTRODUCTION

The Los Alamos Scientific Laboratory is in the process of converting from a film to a TLD badge because our measurements indicate the TLD badge provides an improved energy and angular response to both photons and electrons. In addition, a paper that will be given by Blackstock shows that neutron doses from the cadmium cutoff (.4 eV) up to 1 MeV can be estimated. The presentation that follows gives a physical description of the badge, discusses the method of processing the badge, presents measurements of the energy and angular badge response to both photon and electron radiation, and discusses the method of evaluating the dose.

II. BADGE DESCRIPTION

A. TLD Chips

The TLD-700 and TLD-600 LiF chips are fabricated by the Harshaw Chemical Company. TLD-700 is depleted in ^{6}Li and thus insensitive to neutrons. TLD-600 is enriched in ^{6}Li to obtain high thermal neutron sensitivity by the ^{6}Li $(n,\alpha)^3$ H reaction. Each chip is 0.32 x 0.32 cm in area and 0.089 cm thick.

B. TLD Cards

Figure 1 shows the Harshaw type BGN-6777 TLD card which holds three TLD-700 and one TLD-600 chip. The TLD chips are encapsulated between two sheets of Teflon 0.0063 cm thick. The Teflon, in turn, is held in place by two sheets of aluminum riveted together. The aluminum TLD card is the size of a dental film and 0.16 cm in thickness. The TLD-700 in position one is used to estimate the dose due to penetrating (gamma) radiation. The TLD-700 in position two is used to estimate the dose due to nonpenetrating (x-,beta-) radiation. The TLD-700 in position three and the TLD-600 in position four are used to estimate the dose due to neutron radiation. A seven-digit identification number on the card is both visual and machine-readable.

C. TLD Badge

A cutoff corner on the TLD card ensures proper orientation in the Cycolac plastic badge, shown in Figure 2. Two types of badges, referred to as the "photon badge" and "neutron badge", are in use. The two-piece photon badge contains a 90 mg/cm^2-thick copper filter embedded in 250 mg/cm^2-thick Cycolac that is used to evaluate the penetrating dose. A 60 mg/cm^2-thick Cycolac window is used to estimate the nonpenetrating dose. The advantages of an open window in evaluating beta doses were sacrificed for the advantages of a dirt-free badge; 60 mg/cm^2 was the minimum Cycolac thickness that could be achieved in badge fabrication. There are two circular compartments, 0.7 cm in diameter, that hold a backup TLD-700 chip, a glass rod, and an indium foil. The backup TLD will be used in the event of card damage or loss; it may be read annually to compare with the monthly annual sum. The glass rod and indium foil will be used only in the event of a radiation accident.

In addition to these features, the three-piece neutron badge contains a cadmium "cup" and slip-on "piggyback". The cadmium cup is a 3.3 x 2.1 x .30 cm slotted filter into which the TLD card is inserted. This filter covers the TLD-700 and TLD-600 chips in positions three and four and is used to evaluate low energy (<1 MeV) neutron doses. The Cycolac piggyback attachment, holding an NTA film, slips over the neutron badge, permitting fast (>1 MeV) neutron dose evaluation by track counting.

III. BADGE PROCESSING

A. Reader

The four TLD's in each card are read by the Harshaw Model 2276 automated TLD system. As shown in Figure 3, the system consists of three units: a logic module that controls the system; a mechanical transport module into which cards are loaded, read, and unloaded at a rate of one per minute; an integrating picoammeter that integrates the light output and presents it in digital form. Output is fed to a Texas Instrument Model 700 ASR electronic data terminal, permitting storage on magnetic tape cassettes and a printout.

B. Anneal

A variety of TLD annealing procedures are in use at various laboratories and several of these were tested. Figure 4 shows the glow curves obtained with: no anneal; a post-exposure, room temperature, three-day anneal; a post-exposure 100°C anneal for 10 and 30 minutes; a pre-exposure 80°C anneal for 17 hours. With the latter anneal, a nearly gaussian, single-data peak is observed, with no significant low temperature peak. A 120-day study indicated fading of <1% over this period.

C. Testing

Instead of measuring and storing correction factors for each TLD in the computer, it was decided to determine an average TLD reading and insist that every TLD fall within ±15% of the average. Ten thousand cards, each containing four TLD chips, were calibrated with a ^{60}Co source. If all four TLD's on a card fell within the 15% limit, the card was accepted. If one or more TLD's on a card exceeded the limit, it was exposed a second time. Cards with TLD's that failed twice were returned to Harshaw for replacement. Of the 10,000 cards, approximately 850 were returned for replacement.

An average TLD card residual of approximately 4 mrem was measured by reading each TLD twice after exposure. An average background buildup, caused primarily by cosmic radiation, of 0.4 mrem per day was measured.

IV. BADGE RESPONSE

A. Photons

ICRU Report 25 distinguishes between penetrating and nonpenetrating doses. The true penetrating dose is obtained from a standard chamber reading at a depth of 1 cm in a phantom; the approximate penetrating dose is measured by a filtered TLD badge on the surface of a phantom. Similarly, the true nonpenetrating dose is obtained from a standard chamber reading at a depth of 7 mg/cm^2, and the approximate nonpenetrating dose is measured by a lightly filtered TLD badge on the surface. The response is defined as the ratio of the TLD readings to the appropriate true dose readings, as defined above.

Figure 5 shows the response of the Los Alamos TLD badge that is designed to approximate the correct measurement of penetrating and nonpenetrating doses; neither response is ideal. However, the TLD chip filtered by 90 mg/cm^2-thick copper, plus 250 mg/cm^2 Cycolac, gives the true penetrating dose within 25% for photon energies above 35 keV. As mentioned previously, to obtain a dirt-free badge, a thickness of 60 mg/cm^2 Cycolac covers the "unfiltered" badge window. Figure 5 shows the response of the TLD chip covered by 60 mg/cm^2 Cycolac; also shown is the response of the TLD covered by 7 mg/cm^2 Teflon. In either case, below 100 keV, the nonpenetrating dose is over-estimated by 25 to 50%, depending on the photon energy.

Figure 6 shows the angular response of the TLD badge for both the penetrating and nonpenetrating filter areas for photon energies of 22, 60, and 100 keV. At zero degrees, the badge is facing the source in the normal calibrating position. Except for edge effects, the response of the TLD badge is relatively flat with angle.

B. Electrons

The response of an unfiltered TLD as a function of maximum beta ray energy is shown in Figure 7. The response changes gradually with energy, decreasing 50% from 2.3 to 0.77 MeV. Figure 7 also shows the response of the TLD in the Los Alamos badge filtered by 60 mg/cm^2 Cycolac; absorption in the plastic results in a sensitivity that is reduced by factors of 1.5 and 9 at 2.3 and 0.77 MeV, respectively.

Figure 8 gives the angular response of the TLD badge to beta rays from a ^{90}Sr(^{90}Y) source. There is a sharp decrease in response as the badge is rotated from the normal exposure position.

V. BADGE EVALUATION

A. Penetrating Dose

A ^{60}Co source is used to calibrate the TLD's and the sensitivity of the reader is adjusted so that one nanocoulomb corresponds to one mR. The penetrating dose P is given by the copper-filtered TLD reading R_{cu},

$$P \simeq R_{cu} \qquad (1)$$

Figure 5 shows that Eq. (1) gives the correct penetrating dose within 25% only for photon energies >35 keV.

B. Nonpenetrating Dose

The nonpenetrating dose NP is given by the Cycolac-filtered TLD reading R_{cyc},

$$NP \simeq R_{cyc} \qquad (2)$$

Figure 5 shows that when the badge is exposed to photon radiation, Eq. (2) over-estimates the nonpenetrating dose by 25-50% for energies below 100 keV. Figure 7 shows that when the badge is exposed to beta radiation, Eq. (2) underestimates the nonpenetrating dose for all energies below 2.3 MeV. For beta rays with maximum energies of 2.3 MeV, the most frequent case encountered at Los Alamos, the true nonpenetrating dose is a factor of 2.4 higher than that given by the badge.

C. Plutonium Dose

The dose received by personnel handling plutonium is of particular interest at Los Alamos. Assume the plutonium source emits two monoenergetic photon beams, one at 17 keV and one at 60 keV. If ${}_{17}D$ is the dose due to the 17 keV line and ${}_{60}D$ the dose due to the 60 keV line, then the copper-filtered and Cycolac-filtered TLD readings are

$$Rcu = {}_{17}S_{cu} \times {}_{17}D + {}_{60}S_{cu} \times {}_{60}D \qquad (3)$$

$$Rcyc = {}_{17}S_{cyc} \times {}_{17}D + {}_{60}S_{cyc} \times {}_{60}D, \qquad (4)$$

where ${}_{17}S_{cu}$ and ${}_{60}S_{cu}$ are the 17 and 60 keV TLD responses filtered by copper, and ${}_{17}Scyc$ and ${}_{60}Scyc$ are the 17 and 60 keV responses filtered by the Cycolac. The responses are given by Figure 5, reducing Eqs. (3) and (4) to

$$R_{cu} = 0.33 \times {}_{17}D + 1.25 \times {}_{60}D \qquad (5)$$

$$R_{cyc} = 1.10 \times {}_{17}D + 1.33 \times {}_{60}D \qquad (6)$$

If most of the dose is due to the 17 keV line, Eq. (5) shows that the penetrating dose would be underestimated by as much as a factor of 3. On the other hand, if most of the dose is due to 60 keV line, the penetrating dose would be overestimated by as much as a factor of 1.25. Empirical measurements at the Los Alamos plutonium-handling facility indicate the softer keV component is usually attenuated by shielding and that 3/4 or more of the dose is due to the 60 keV line. As a result, the TLD badge gives the true penetrating dose within 25%.

Equation (6) shows that the nonpenetrating dose is always overestimated by a factor of 1.10 to 1.33.

VI. SUMMARY

A physical description of the new Los Alamos badge has been presented. The selection procedures for the TLD's were discussed. The energy and angular responses of the Los Alamos badge to both photons and electrons was described. A brief discussion of the evaluation of penetrating and nonpenetrating radiation has been included, with particular emphasis on the evaluation of plutonium doses. Although no comparisons have been given, our measurements indicate that the new TLD badge will result in more accurate estimates of personnel doses.

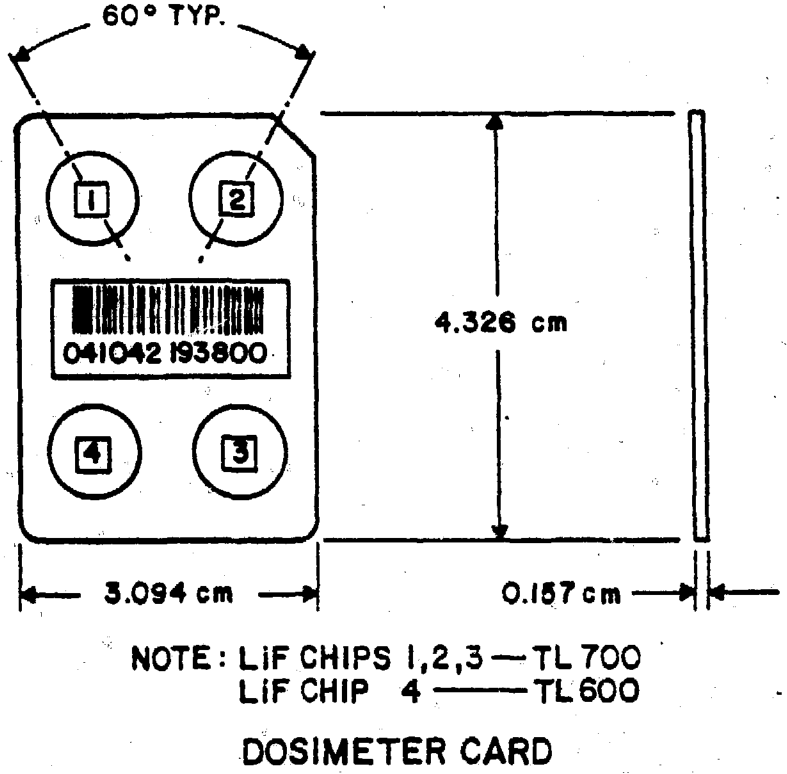


FIGURE 1. HARSHAW BGN-6777 TLD CARD

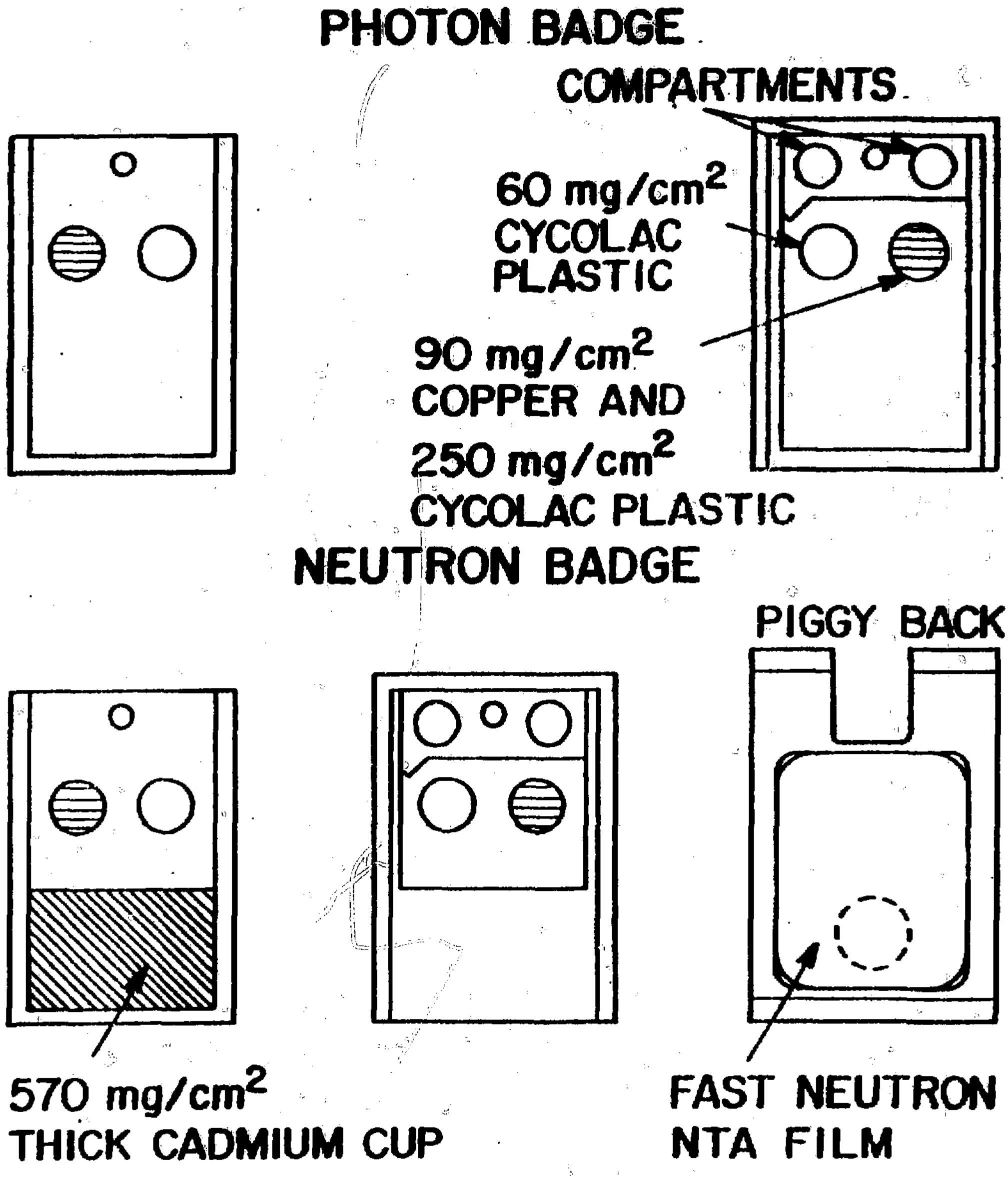


FIGURE 2. LASL CYCOLAC TLD BADGE

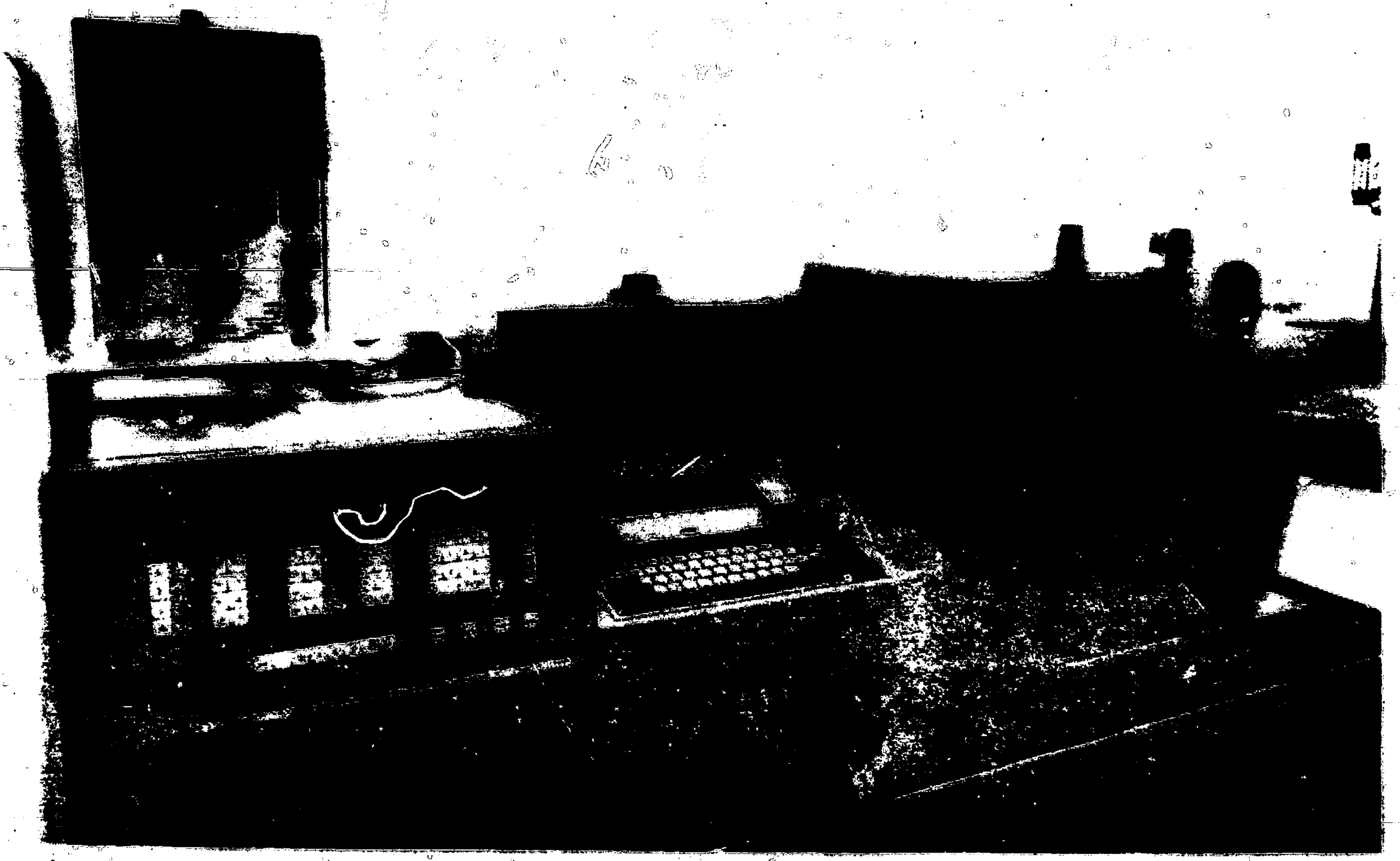


FIGURE 3. [illegible] 2276 AND SYSTEM

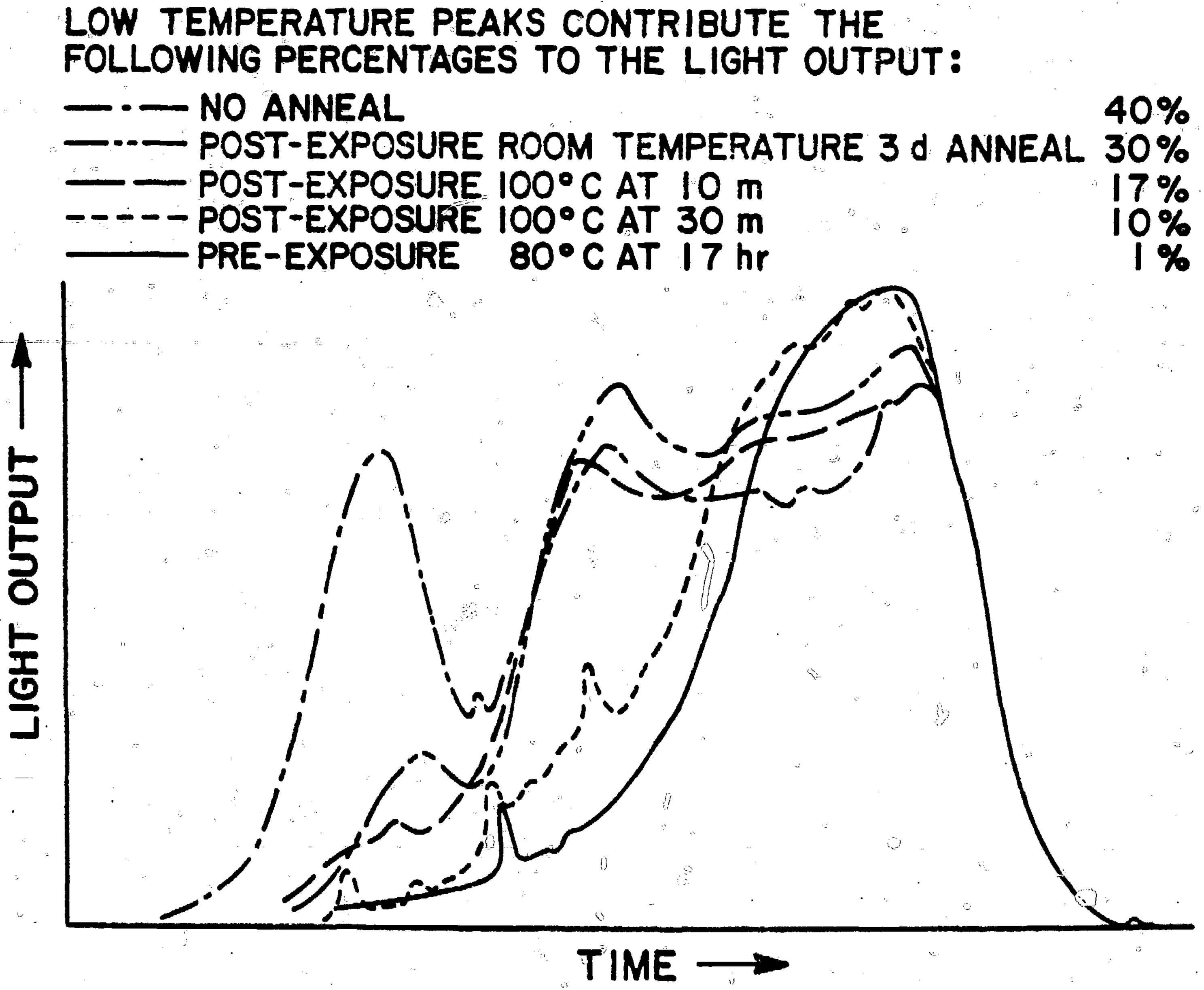
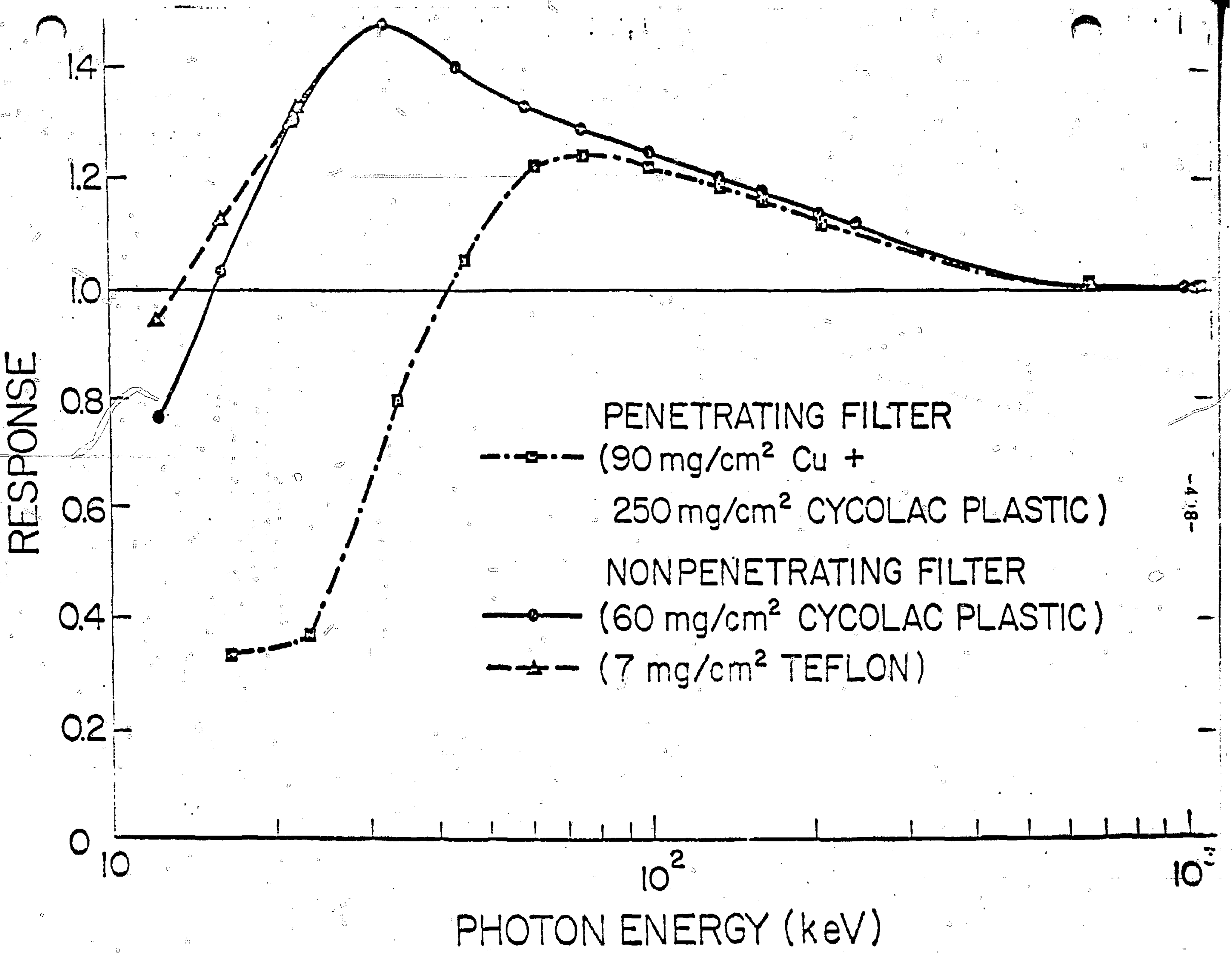


FIGURE 4. COMPARISON OF GLOW CURVES OBTAINED WITH DIFFERENT ANNEALS


RESPONSE
1.4
1.2
1.0
0.8
0.6
0.4
0.2
0
10
10²
PHOTON ENERGY (keV)
PENETRATING FILTER
(90 mg/cm² Cu +
250 mg/cm² CYCOLAC PLASTIC)
NONPENETRATING FILTER
(60 mg/cm² CYCOLAC PLASTIC)
(7 mg/cm² TEFLON)

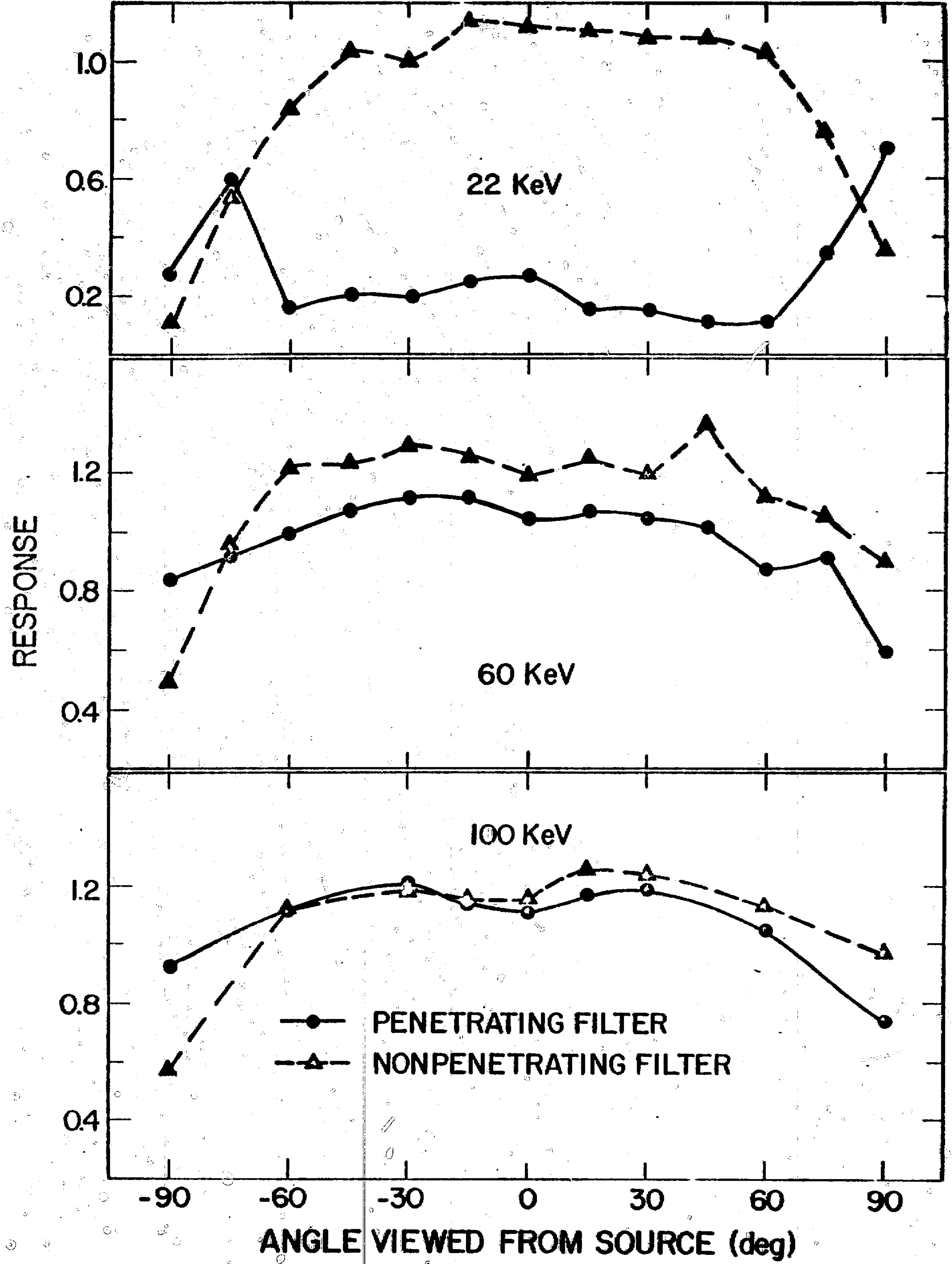


FIGURE 6. PHOTON ANGULAR DEPENDENCE OF THE TLD BADGE

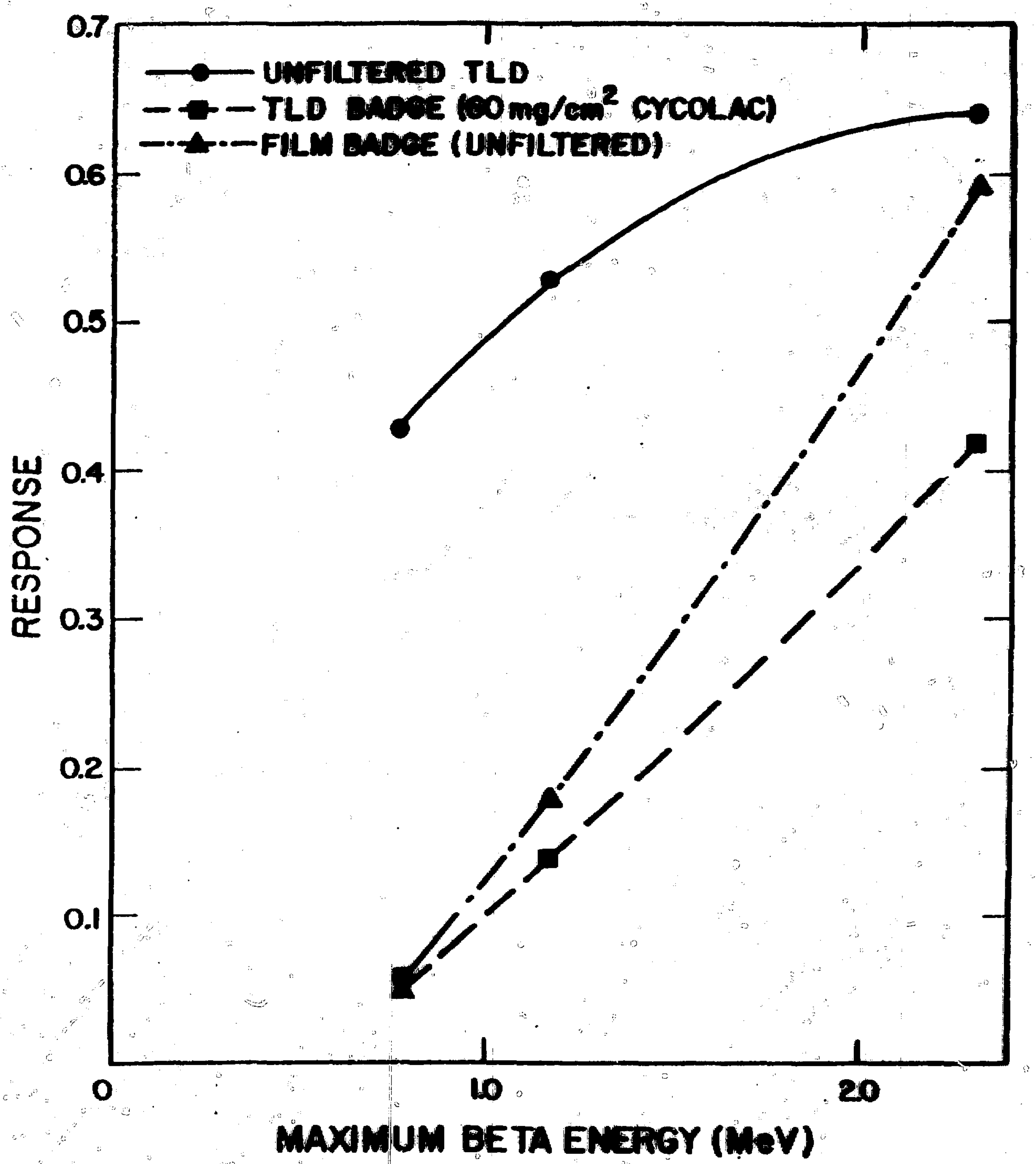


FIGURE 7. TLD AND FILM BADGE ENERGY RESPONSE TO BETA RADIATION

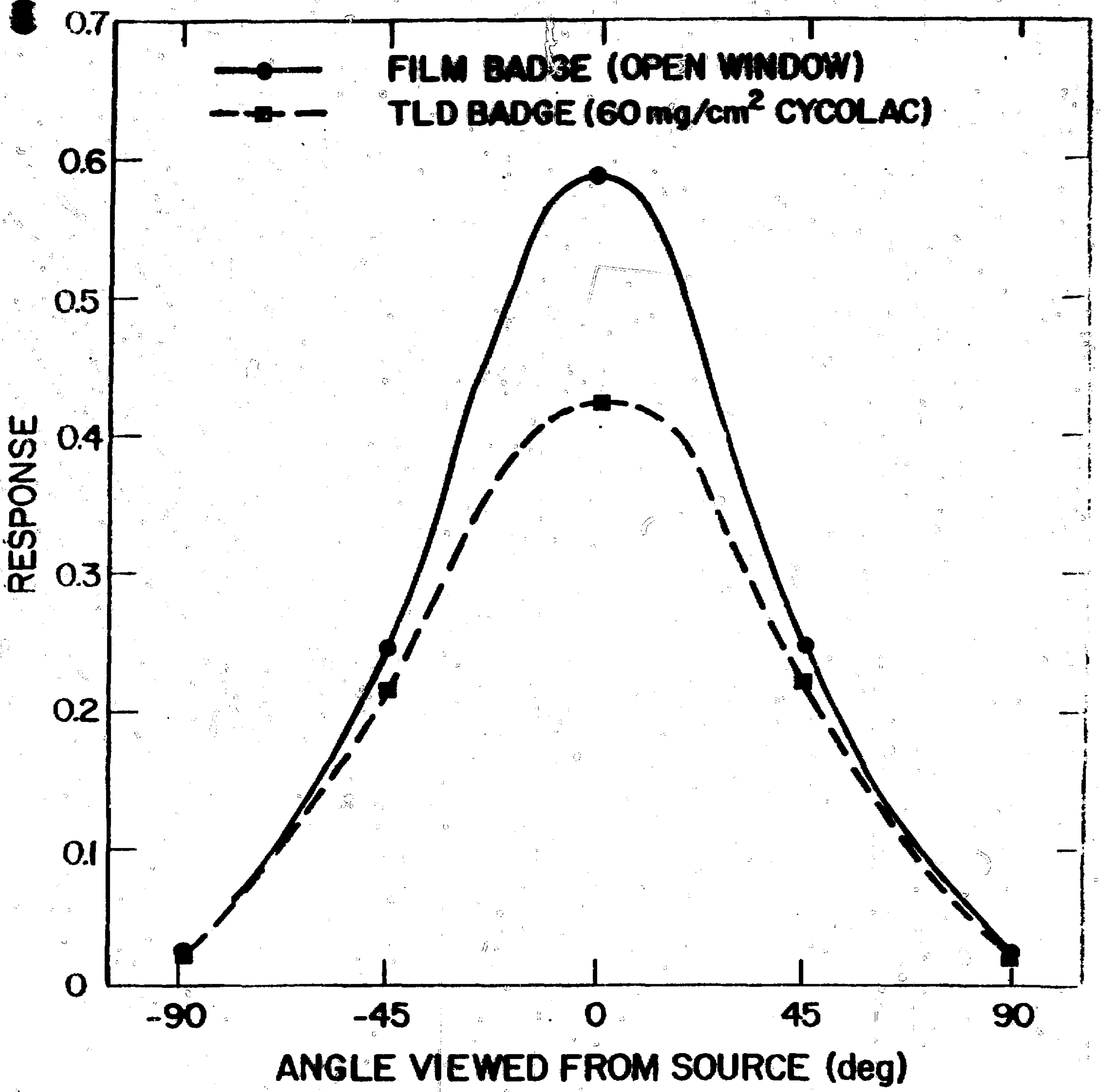


FIGURE 8. TLD AND FILM BADGE ANGULAR RESPONSE TO ^{90}SrY BETA RADIATION

Question (Murray Walsh, Ontario Hydro:

I have two questions to ask. I wonder if you could describe briefly how you perform your beta energy dependence tests, what sources you used.

Answer:

Yes. We used strontium-Yttrium-90, which has 2.3 Mev, we used a thallium 204, and we used Bismuth 210, which gives you the three energies that I mentioned

Question (Walsh):

How did you measure the field coming from these sources?

Answer:

We used a thin-walled ion chamber.

Question (Walsh):

Secondly, have you looked at the longterm effect of possible degradation of the TLD by annealing at 80° C for 17 hours?

Answer:

Yes, I've known that was presented here but I have had some TLD's that have gone through quite a number, over 100 annealings and there's no effect to my knowledge.

Question:

Did you notice any decreasing sensitivity?

Answer:

No sir, I sure haven't.

Question (Ken Skrabble, University of Lowell:

I understand you measured the beta dose with an ion chamber--how would you classify this measurement of dose then--would you classify it as the surface dose to the live skin, or an average dose. I didn't see any beta energies below 0.8 Mev in terms of response. What were you plotting as response? On the ordinant--were you plotting average dose, or surface dose to the live skin?

Answer:

Well, the responses reading the ratio of the TLD and the appropriate true dose reading which I've described in my paper. A true dosereading is defined as a standard chamber reading at a depth of 1 cm on a phantom. That's the true penetrating; the true non-penetrating dose is the standard chamber at a depth of 7 milligrams per square centimeter on a phantom.

Question:

O.K., so then you are referencing here the dose below 7 milligrams per square centimeter at the surface of the live skin?

Answer:

Yes, sir.

Question:

O.K. We've been concerned about this. Of course, it's very difficult to measure instantaneous quantity of the surface dose from a practical standpoint. In order to do this with the dosimeter of course, the thickness of the detector would have to approach zero thickness to measure an instantaneous quanitity. We'd get no output then. I think more practical approach would be to try to measure directly the average dose to the live skin. Since we do this anyway in other dosimetry problems such as internal dosimetry. For beta this can be done quite easily, and one need not be concerned about the response of very low beta energies because, if you use a detector which has thickness approximately equal to the live skin layer and you cover that with an absorber equivalent to the dead layer of skin, 7 milligrams per square centimeter, you have a system which you could automatically, if it had the same atomic composition as tissue, would automatically respond as an average dose to live skin. I'm making this point because we seem always to try to get to the point where we're approximating this instantaneous dose and I don't know why we always try to measure the instantaneous dose at the surface of the live skin. I don't think that's meaningful from a biological standpoint, and I just don't know why we're designing dosimetry systems the way we are. To try to measure something that's almost impossible to measure in the first place. Because if you had shown response at lower beta and you'd have factors of 10 to 100 in terms of variation response.

Answer:

You're right, you're right, thank you.

Question (Paul DeWitt, University of Wisconsin):

When you look at the sensitivity change with repeated annealings, was this in the Harshaw card mount or was this with bare TLD?

Answer:

Well, we're using strictly a card, when I say a bare card I mean a card as I showed in the first slide.

Question:

That's with the depth line mounting?

Answer:

Yes, sir. As you see on the card.

"IMPROVEMENT OF A COMMERCIAL THERMOLUMINISCENT (TL) ANALYSER FOR LOW RADIATION DOSE DETERMINATION".

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Nacional Autónoma de México,
México 20, D.F., México.

Abstract:

"The thermoluminiscent response of a commercial analyser, (Harshaw Chemical Co., serie 2000) is improved for determination of radiation dose - and TL - dating by the simple changes of: 1)linear direct light transmission of the TL - signal instead of mirror reflection method ; 2) optical filter with - light transmission of 320 to 680 n m and 3) EMI-PMT 9635 QB.

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*Work supported by the INEN, México.

1. Introduction.

There are two main applications for a commercial TL-reader namely:

1) Low radiation dose determination and

2) TL-dating.

The purpose of this paper is to present some simple changes of the commercial TL apparatus to improve its sensibility and response.

Such changes are:

1) linear direct light transmission of the TL - signal, instead of mirror light reflection method.

2) optical filter with light transmission of 320 to 680 n m for TL - dating work and

3) use of a EMI-PMT 9635 QB or a similar photomultiplier tube.

2. Methods and materials.

A new photomultiplier tube chamber that is named "B" in this paper, was built to achieve the changes mentioned above. A sketch is shown in figure 1 in which it can be noted that the Harshaw original thermoelectrically cooled PMT Chamber, thermistor and thermocouple was used; a photograph of PMT chamber "B" is shown in figure 1 a. The EMI PMT 9635 QB (2), optical transmission filter and heating planchet center are in a straight line. The --

chosen quantum efficiency from 200 to 650 n m of the PMT is adequate for both applications: low radiation dose determination and TL dating.

The out puts from the "PMT chamber B", the same as those of the Harshaw PMT original chamber, are indicated in figure 1 and connected according to the instruction manual of the model 2 000 TL analyser.

The PMT 9635 QB voltage divider network is built on the teflon socket and the photomultiplier operated in the --- integrating mode.

The Harshaw original PMT chamber called "A" in this work, use TL signal mirror reflection method; infrared optical filter with transmission from a 330 to 680 n m - and from 690 to 1 000 n m, and a PMT identified as - - P 4272 T. (1)

3. Results.

The thermoluminiscent response versus dose at - -- differents PMT voltage, for TLD - 100 irradiated with ^{90}Sr using the Harshaw original PMT chamber "A",- is presented in figure 2 and corresponding results -- using PMT chamber "B" are shown in figure 3. Both figures 2 and 3, allow to get the ratio of response of - both PMT chambers at the recomended operating - --

voltages.

The blue filter with transmission from 320 to 680 n m, 4 mm. thick, is particularly useful to avoid the red and infra-red emission that make very difficult to obtain – data from glow curves of ceramic materials on TL-dating (3) (4).

Such situation is ilustrated in figure 4 in which the – — unwanted IR emission of natural TL from quartz grain – is presented and cut it out by using the blue filter, see– figure 5.

Finally low dose irradiation response of BeO [illegible]c, — 0.25 inch diameter treated as suggested by Gammage(3), irradiated with a calibrated gamma source of ^{60}Co and reader out using PMT chambers "B" and "A" are shown in figure 6 a an b, respectively.

Conclusions:

The use of a simple accessory device as the PMT — Chamber B give to the commercial analyser more – — flexibility and better working conditions in low radiation dosimetry and TL-dating.

References:

1. Instruction manual Model 2000 TL - Analyser. -- Harshaw Chemical Co. Ohio, U.S.A.
2. Index of Photomultiplier Tubes & Windowless Electron Multipliers, EMI - Electronics Ltd. Electron Tube - Division, Middesex, England.
3. Topics in Radiation Dosimetry. Frank H. Attix. -- Academic Press.
4. Bulletin CF - 1, Glass Color Filters. Corning Glass Works. N.Y. U.S.A.
5. R. B. Gammage etal, Radiation Effects, 1976, Vol.29, 171-176.

Figure Captions

Figure 1	Schematic representation of the "PMT Chamber B"
Figure 1a	Photograph of the "PMT Chamber B"
Figure 2	TL response vS dose at differents PMT voltages for TLD 100, irradiated with ^{90}Sr source, 12 rads/sec "PMT Chamber A" (Harshaw)
Figure 3	TL response vs dose at differents PMT voltages for TLD-100 irradiated with ^{90}Sr. source, 12 rads/sec. "PMT Chamber B"
Figure 4	Glow curves of 44-106 μm quartz grains, showing natural TL, using a IR filter, and "PMT Chamber A"
Figure 5	Glow curves of 44-106 μm quartz grains, showing natural TL, using a blue filter and "PMT Chamber B"
Figure 6	TL response vs exposure of 0.25 inchs diam. thermalox 995 BeO disc. Curve a) with "PMT Chamber B" and -- b) with "PMT Chamber A"

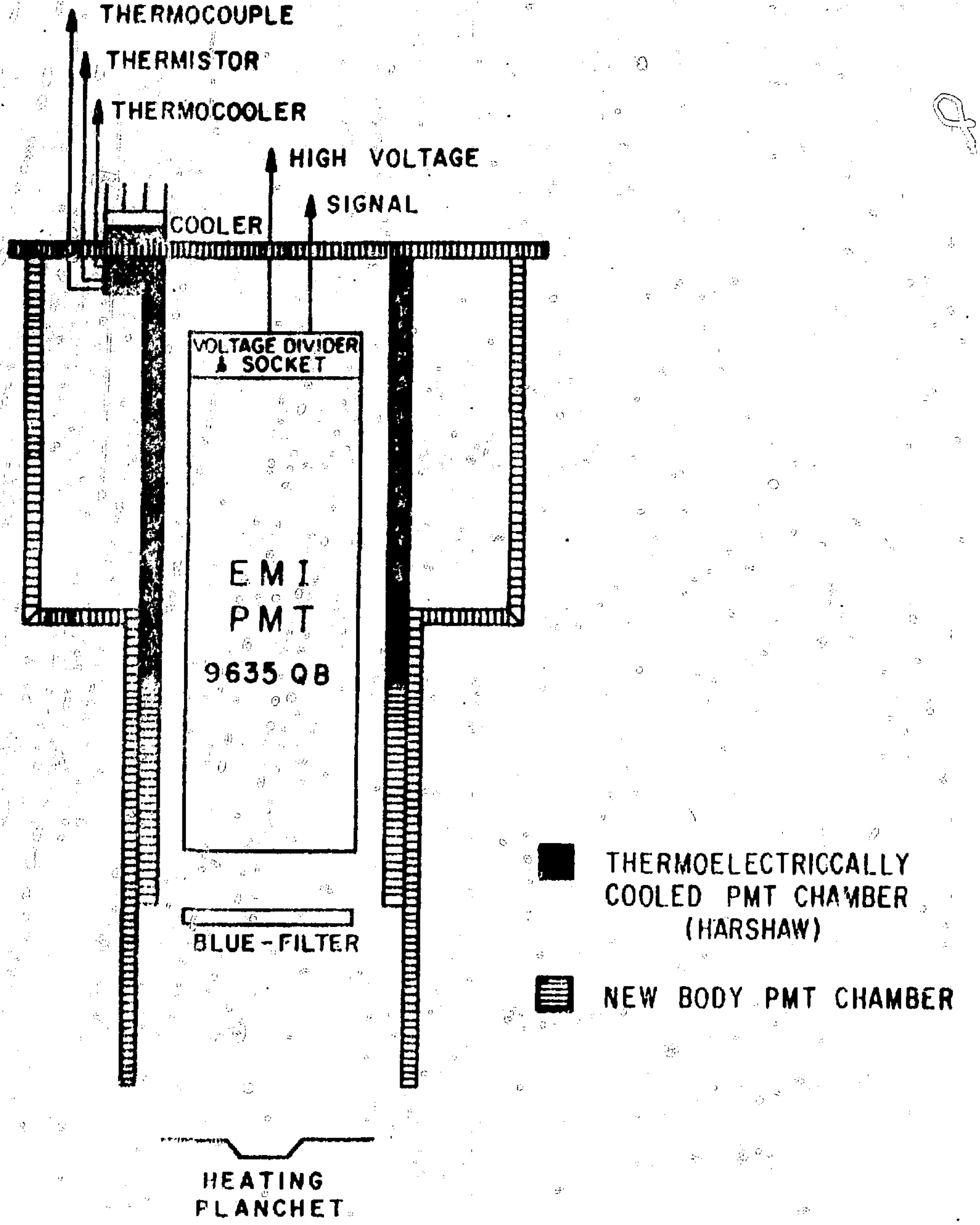

THERMOCOUPLE
THERMISTOR
THERMOCOOLER
HIGH VOLTAGE
SIGNAL
COOLER
VOLTAGE DIVIDER & SOCKET
EMI PMT 9635 QB
BLUE-FILTER
HEATING PLANCHET
■ THERMOELECTRICALLY COOLED PMT CHAMBER (HARSHAW)
▤ NEW BODY PMT CHAMBER

FIG. 1

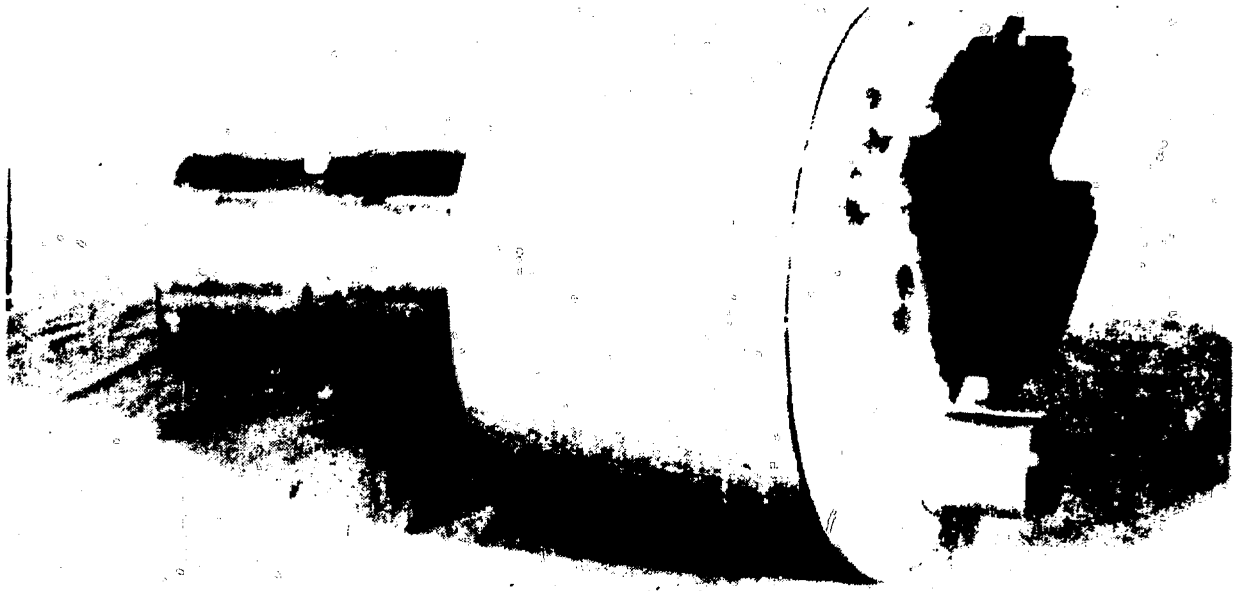


FIGURE 1a. PMT Chamber B

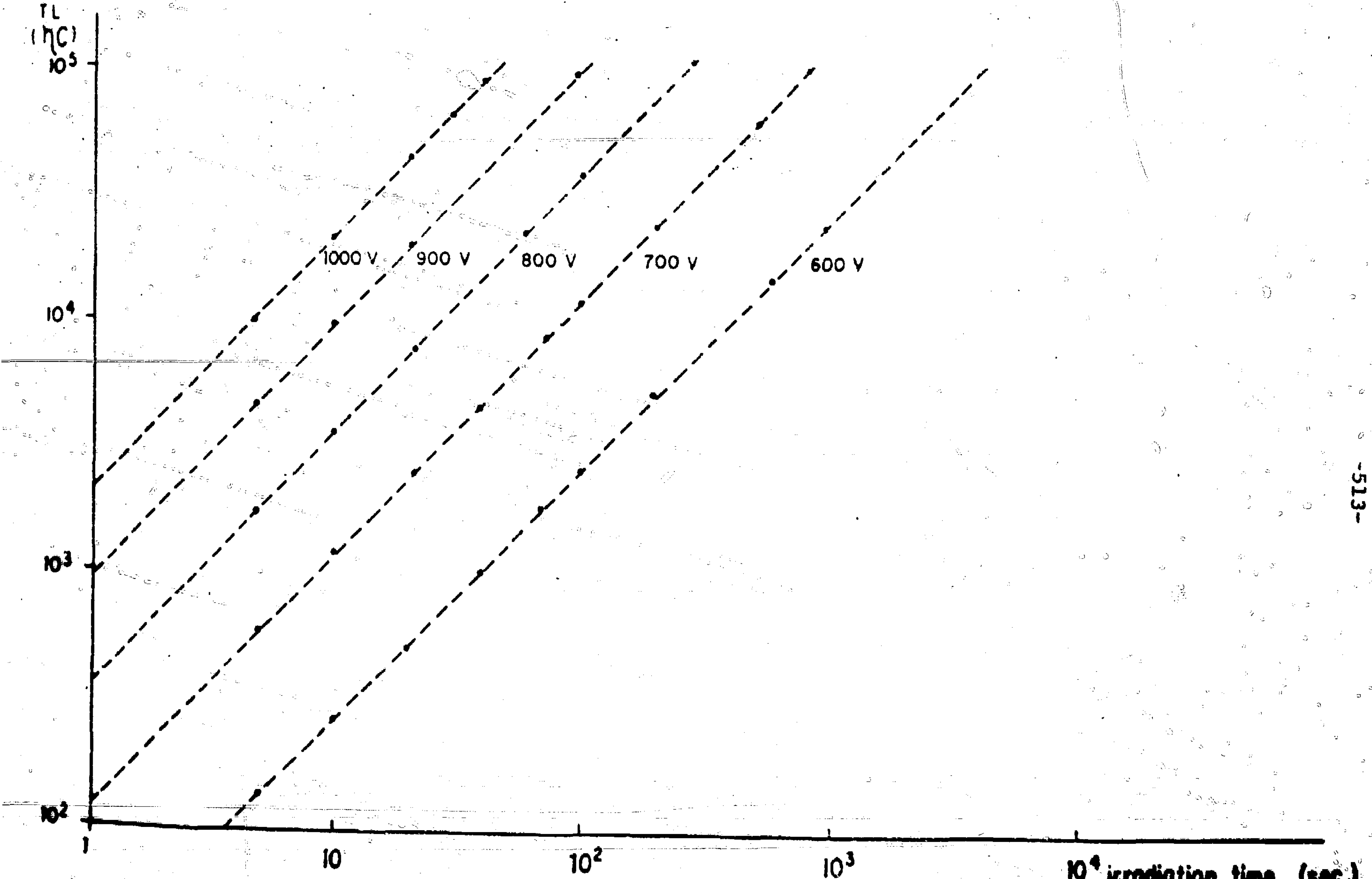

TL
(nC)
10^5
10^4
10^3
10^2
1
10
10^2
10^3
10^4 irradiation time (sec.)
1000 V
900 V
800 V
700 V
600 V

FIG. 2

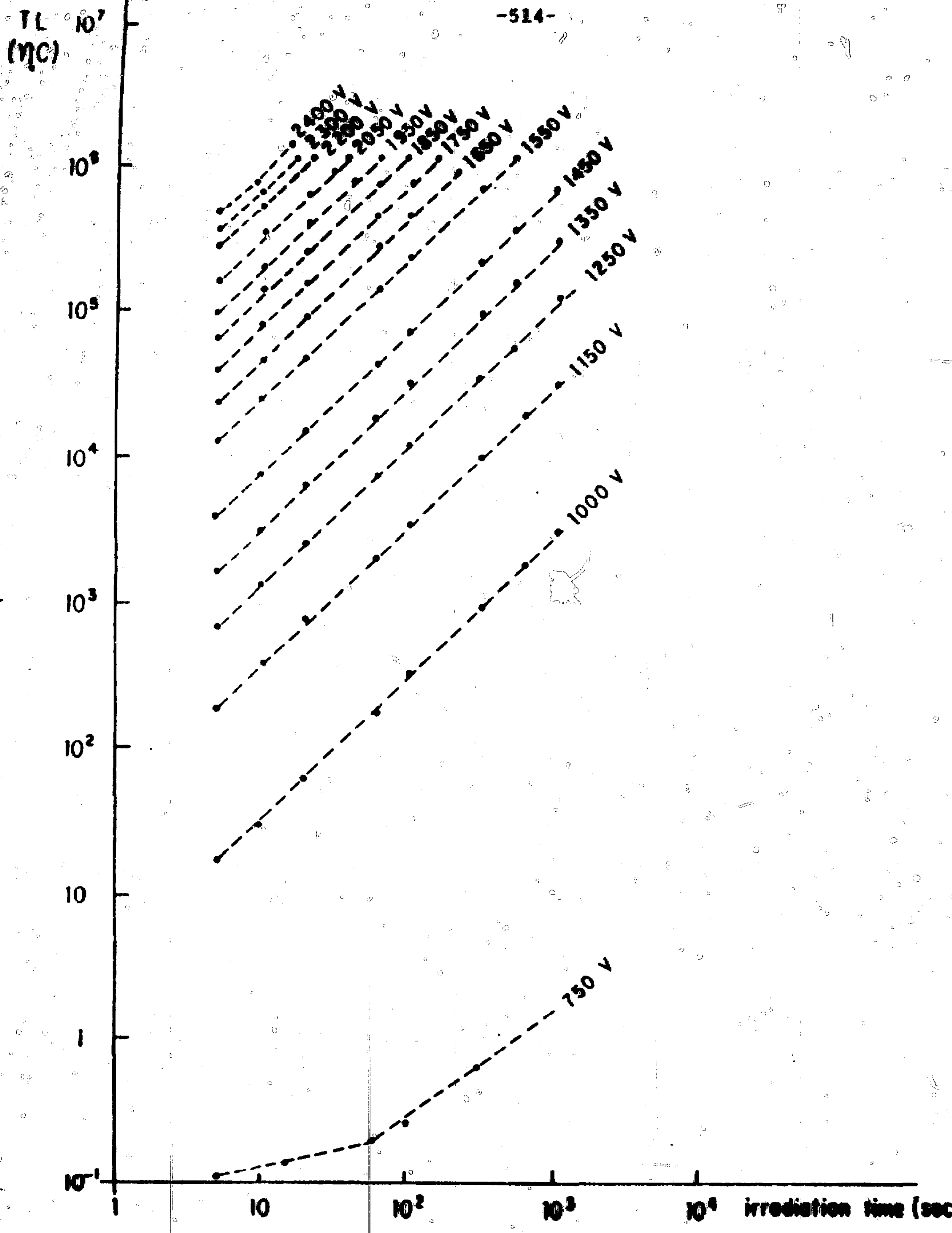

TL
(ηC)
10^7
10^6
10^5
10^4
10^3
10^2
10
1
10^-1
1
10
10^2
10^3
10^4
irradiation time (sec
2400 V
2300 V
2200 V
2050 V
1950 V
1850 V
1750 V
1650 V
1550 V
1450 V
1350 V
1250 V
1150 V
1000 V
750 V

FIG. 3

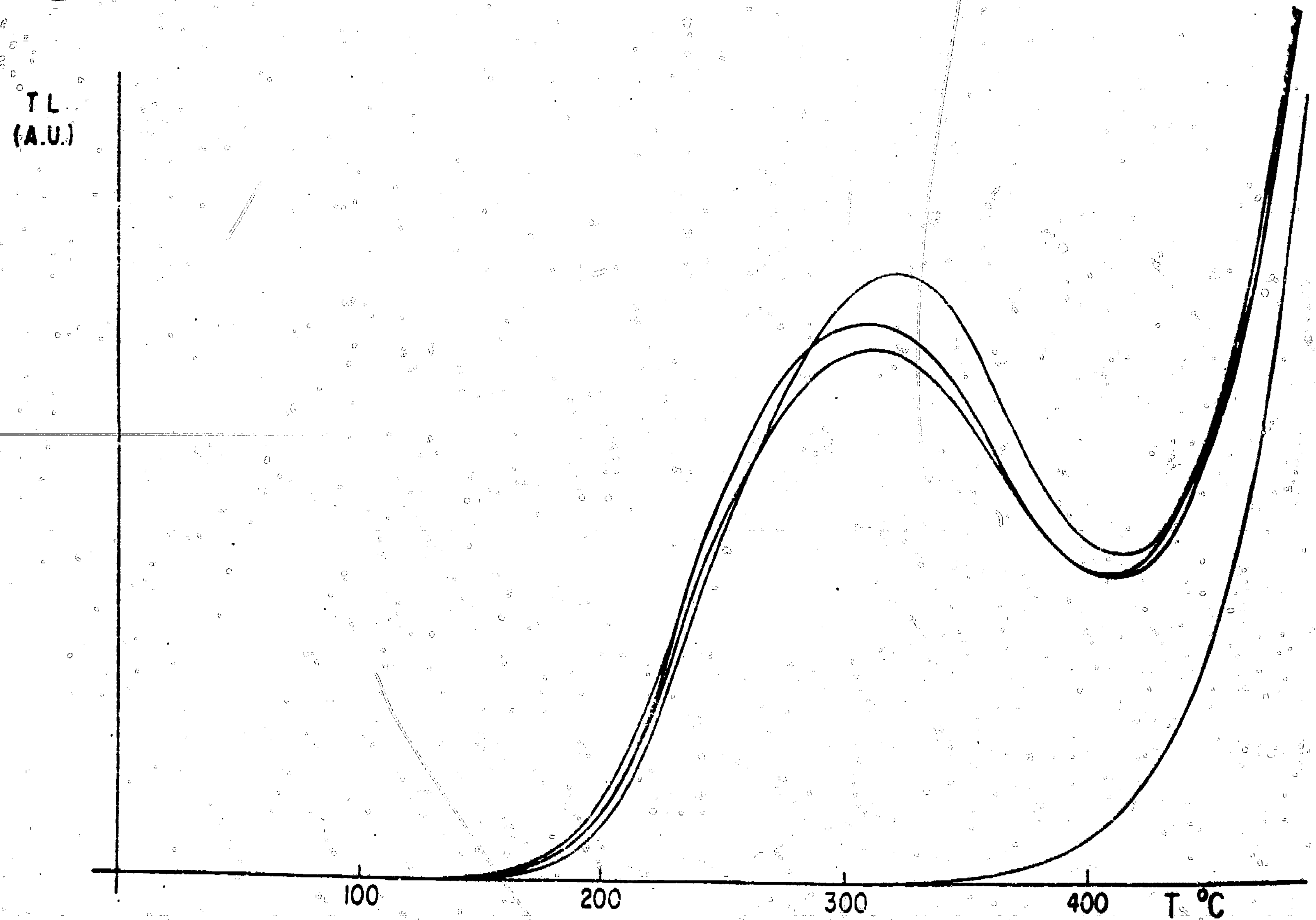


FIG. 4

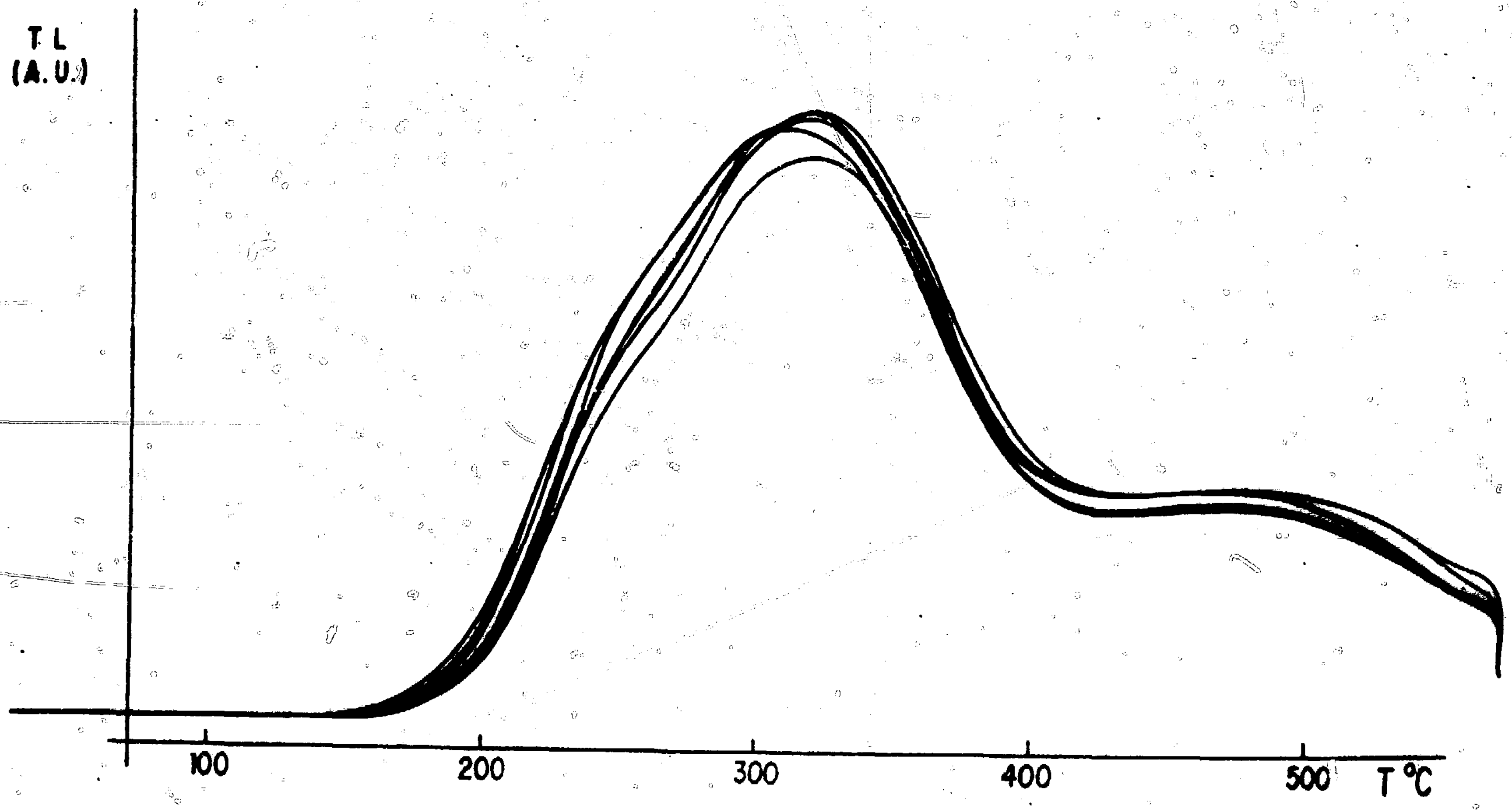
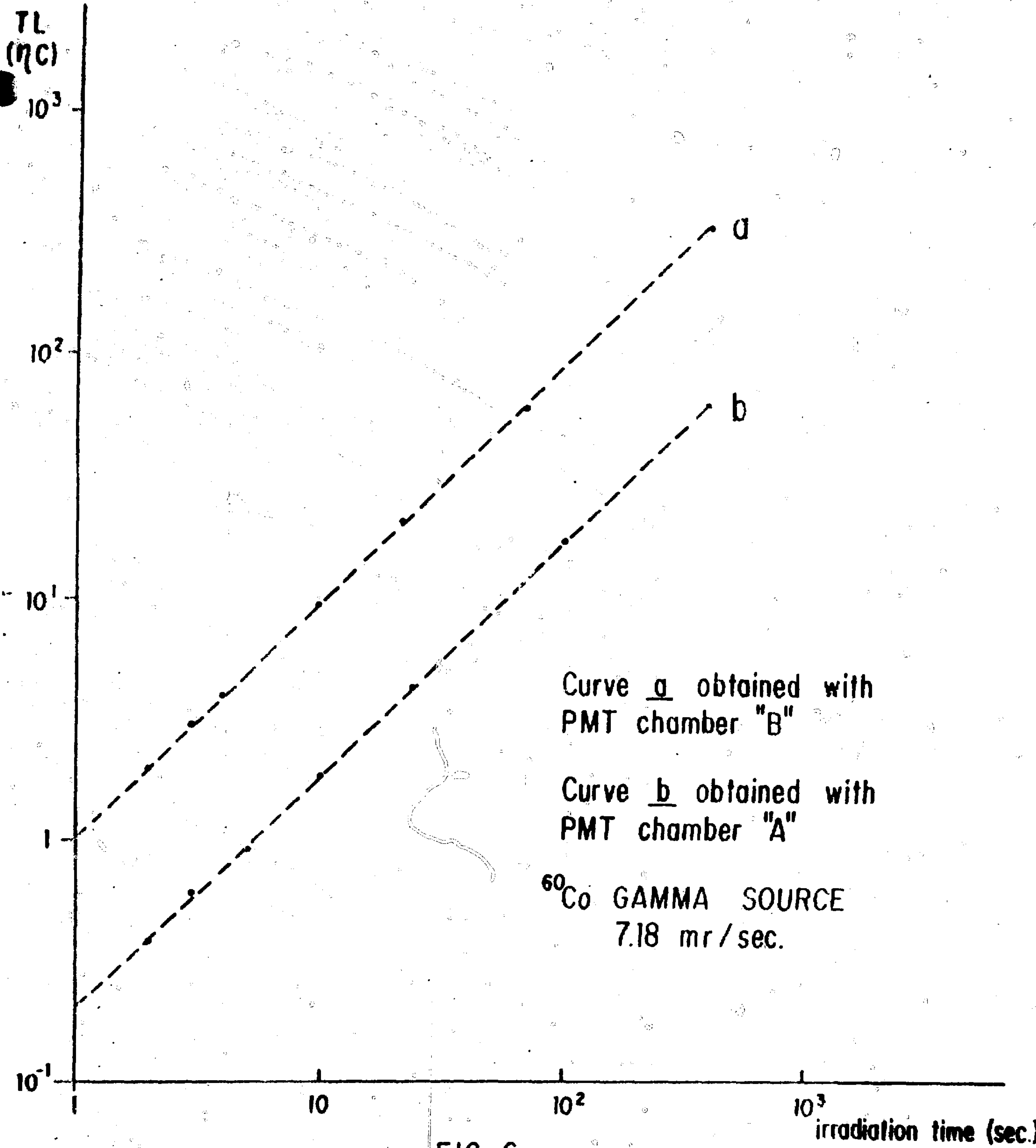


FIG. 5



Curve a obtained with PMT chamber "B"

Curve b obtained with PMT chamber "A"

^{60}Co GAMMA SOURCE
7.18 mr/sec.

FIG. 6

MEETING REGULATORY STANDARDS WITH BeO CERAMIC TLD*

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ABSTRACT

Measurements of exposures below 1 mR are possible with BeO ceramic TLD by signal recording that discriminates against an interfering pyroelectric incandescence. Performance under environmental monitoring conditions is considered in light of current regulatory criteria. Factors such as reproducibility and batch uniformity are satisfactory. An anomalous energy dependence causes an over-response that will probably require use of an energy compensation shield.

---

*Research sponsored by the Department of Energy under contract with Union Carbide Corporation.

Introduction

Our opinion was stated in a recent publication (1) that BeO ceramic TLD, trade name Thermalox 995 of diameter 12.5 mm*, should by now have found widespread acceptance as a personnel and environmental radiation dosimeter. For all its touted advantages, BeO ceramic has found quite limited acceptance for fieldwork (2,3,4). A principal reason is that a nonradiation induced emission called pyroelectric incandescence (1) interferes with the radiation induced TL. Incomplete resolution of the two types of emission has limited the minimum level of detectability to a few mR (5). Since most personal and environmental exposures are at or below a few tens of mR this difficulty of resolution needs to be resolved. The principal purpose of this communication is to describe our efforts to discriminate against pyroelectric incandescence during recording of the glow curve such that meaningful measurements can be made at a few tenths of an mR.

An evaluation is also made of how well BeO ceramic TLD is able to meet regulatory performance criteria in environmental radiation monitoring.

Thermoluminescence Reader

Previously reported results (1) noted the difficulties caused by faulty or inappropriate electronic components that produced too high a background and, in particular, an inability to resolve the TL glow curves caused by ionizing radiation and pyroelectric incandescence. By incorporating a high voltage supply and low noise picoammeter into the reader together with state-of-the-art analog and digital integrated circuits, the noise level has been reduced from 20 to 2 mV peak to peak.

---

*Ceramic BeO of 99.5% purity by Brush Beryllium Co., Elmore, Ohio.

To improve discrimination between the radiation induced TL and the overlapping pyroelectric incandescence, the signals are differentiated electronically (see Figure 1) to cause the composite signal to cross the baseline. A zero crossing detector senses and allows clipping of the unwanted negative signal. The positive TL signal is integrated by counting the output of an analog to digital converter as the BeO is heated within a preset temperature range. The circuit also serves to block changes in the dark current from the PM tube that arise from temperature cycles. The result is a very stable background signal.

Reduction of Interference from Pyroelectric Incandescence

Without the benefit of the circuit shown in Fig. 1, a trough appears in the glow curve between the incompletely resolved radiation induced TL and the pyroelectric incandescent TL. Small peak-to-trough ratios, Fig. 2, then limit the accuracy of measurements at a few tens of mR. Some improvement can be achieved by spectral discrimination using a thicker UV filter as shown in Fig. 3. The critical improvement is brought about by the peak shaping and clipping, and stabilization of the background. Superior sensitivity and reproducibility ensue with a minimum detectable exposure (MDE) of 0.1 mR (MDE $\cong 3\sigma$). The response and percent variation characteristics as a function of exposure are shown in Figs. 4 and 5, respectively.

The previously reported statement that the intrinsic intensity of the pyroelectric peak is of variable magnitude (1) is in error. With good reproducible thermal contact between the detector and the heater planchette, the pyroelectric glow curve for a given detector has lately been shown to have invariant intensity. Without this invariance very low exposure measurements would be impossible.

Ability to Meet Regulatory Performance Criteria for Environmental Monitoring

The requirements and recommendations for performance specifications are generally more severe in environmental than personal monitoring. The BeO ceramic TLD is being assessed for the more stringent application of environmental work with the expectation that success will ensure suitability for personal monitoring also.

The regulatory positions being considered are those contained in the American National Standard "Performance, Testing, and Procedural Specifications for Thermoluminescence Dosimetry (Environmental Applications)," known as ANSI N545-1975, and the U. S. Nuclear Regulatory Commission, Regulatory Guide 4.13, also having the same title.

Standard Exposure

A field cycle of 3 months in a natural background radiation field (10 uR/h) results in an accumulated exposure of about 20 mR. Tests of the detector uniformity and reproducibility have been made at a standard exposure of 20 mR from either ^{137}Cs or ^{60}Co.

Uniformity

A batch of 50 randomly selected thermalox 995 disks were exposed and read repeatedly. For the whole batch, the relative standard deviation was 14%. If seven of the 50 detectors that gave the most deviant readings were rejected, then the percentage deviation for the remaining 43 detectors would fall to 7%. This degree of selection allows conformity with the NRC guide 4.13 which requires that the chosen batch of detectors show a deviation of less than 7.5%.

Reproducibility

The average coefficient of variation for each of the 50 detectors given three repeat exposures and readings was 1.3%. The percent variation for the worst single detector was only 2.7%. The performance standard (NRC guide 4.13) is for a relative standard deviation of less than 3%, hence the reproducibility of the BeO is satisfactory.

Directional Dependence

The angular dependence of response to ^{60}Co gamma radiation, using a low scatter geometry, is shown in Fig. 6. The directional response at lower photon energies is contained in Fig. 5 of ref. 1. The average deviation in response due to angular variation from the standard orientation (where the face to be read is normal to the incident radiation) was 18% for a single BeO disk. The directional dependence can be reduced to 9% by employing a sandwich of two disks and reading the two outside faces of the composite. Both ANSI N545-1975 and NRC 4.13 require that the directional dependence not exceed 10%. For BeO ceramic this proviso requires sandwiching together of BeO disks during irradiation.

Energy Dependence

Although the effective atomic number of BeO is only 7.1, there is an anomalous over-response (2,5,6,8), instead of the expected under-response (7), at energies below a few hundred keV. The enhancement, however, is not more than a factor of two which is the limit set in regulatory performance specifications for photons with energies of less than 80 keV. This unexpected, and unexplained, over-response constitutes the major drawback to using this nearly tissue-equivalent phosphor in generalized environmental monitoring.

Our past (8) and present (3rd International Intercomparison of Environmental Dosimeters, Oak Ridge, 1977) experiences with BeO in environmental field testing show that the measured exposure can exceed the correct value (measured by a high pressure ionization chamber) by as much as 30-50%. This property constitutes a serious deficiency for environmental applications, since section 3.3 of ANSI N545-1975 states that the experimentally determined value of field exposure shall differ from the correct value by less than 30%. Reluctantly one must recommend the incorporation of energy filters into the dosimetry package in order to reduce the response at low energies, as has been done recently by Busuoli et al. (2).

Other regulatory provisions and qualifications can be met more easily. Moisture, for example, does not impede performance, nor is self-irradiation any problem. Light must be excluded during the field exposure, and handling of the phosphors should be conducted under yellow or red lights (5). This safeguard poses a relatively minor detriment to routine use.

Acknowledgment

The help of J. H. Thorngate is recognized in designing the circuit shown in Fig. 1. The assistance of P. Mellor, ORAU Student Trainee from Kenyon College, Gambier, Ohio, is also recognized in the taking of data for directional dependence.

REFERENCES

(1) R. B. Gammage and J. S. Cheka, Health Phys. 32, 189 (1977).

(2) G. Busuoli, I. Sermenghi, O. Rimondi and G. Vicini, Nucl. Instrum. and Methods 140, 385 (1977).

(3) T. Yamashita, Y. Yasuno, and M. Ikedo, Health Phys. 27, 201 (1974).

(4) K. Puite, G. Scarpa and J. J. Broerse, Proc. 4th Int. Conf. on Luminescence Dosimetry, Vol. 3, pp. 963-969 (Krakow, Poland: Institute of Nuclear Physics), (1974).

(5) K. W. Crase and R. B. Gammage, Health Phys. 29, 739 (1975).

(6) G. Busuoli and H. W. Julius, Proc. 5th Int. Conf. on Luminescence Dosimetry, Sao Paulo, Brazil, 225 (1977).

(7) H. Kriks and V. Siegel, Proc. 4tn Int. Symp. on Exoelectron Emission and Dosimetry, Liblice, Czechoslovakia, 241 (1973).

(8) R. B. Gammage and F. F. Haywood, "Operational Health Physics, Proc. 9th Midyear Topical Symposium of the Health Physics Society, Denver, Colorado, 324 (1976).

FIGURE CAPTIONS

Fig. 1. Circuit for the TL reader to shape, clip, and block D.C. signals from the photomultiplier tube.

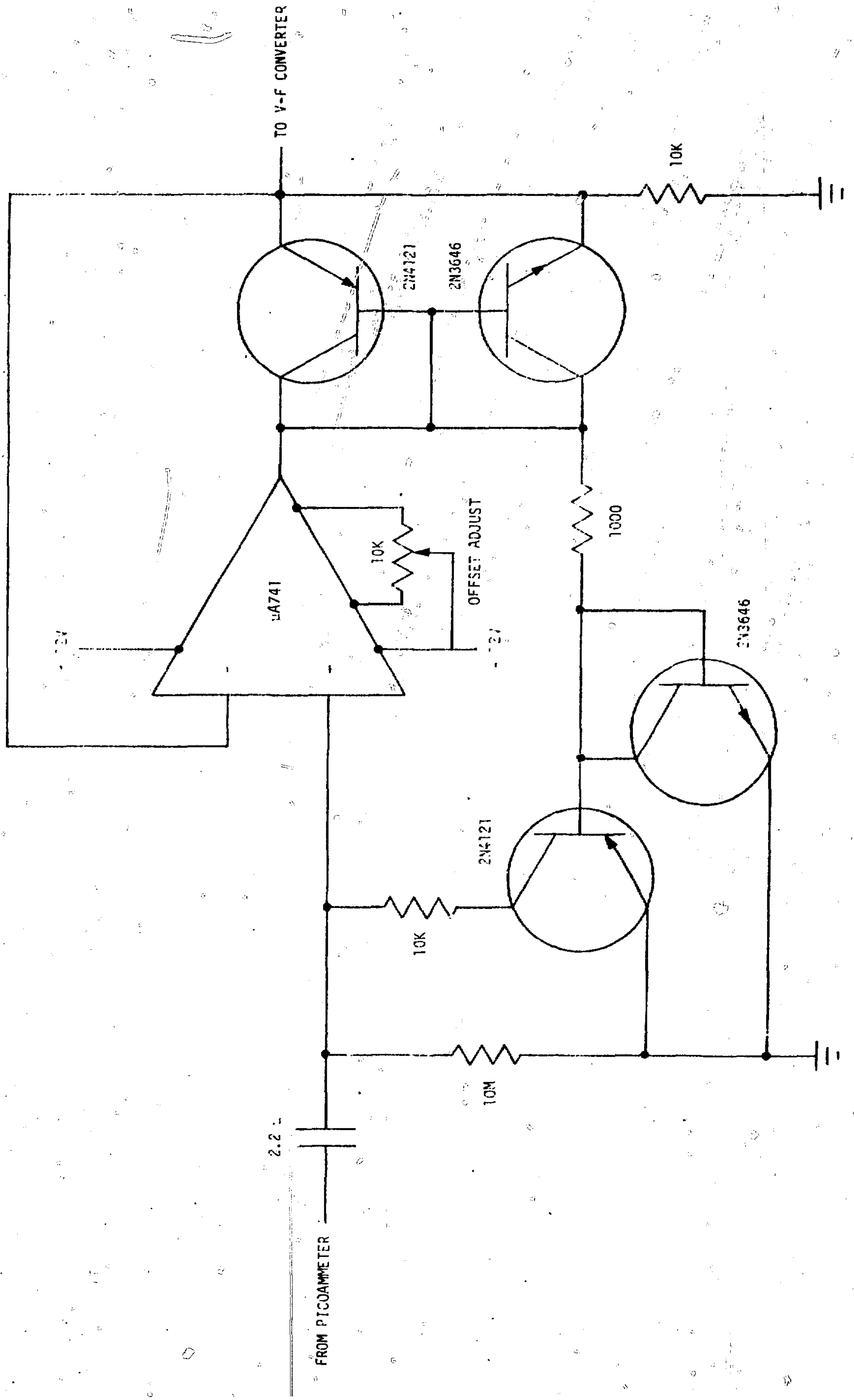
Fig. 2. Thermoluminescence peak-to-trough ratio with a uv-filter thickness of 6 mm.

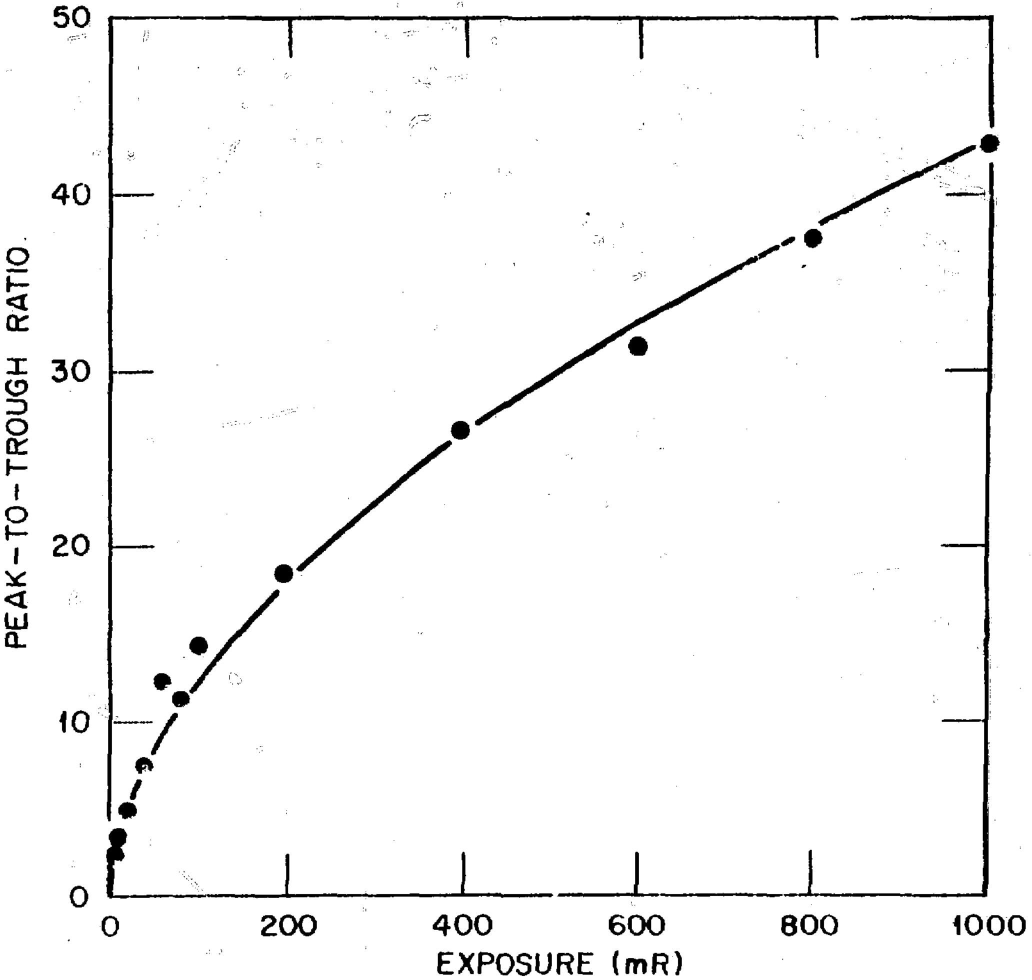
Fig. 3. Thermoluminescence peak-to-trough ratio at two different uv-filter thicknesses and at low exposures.

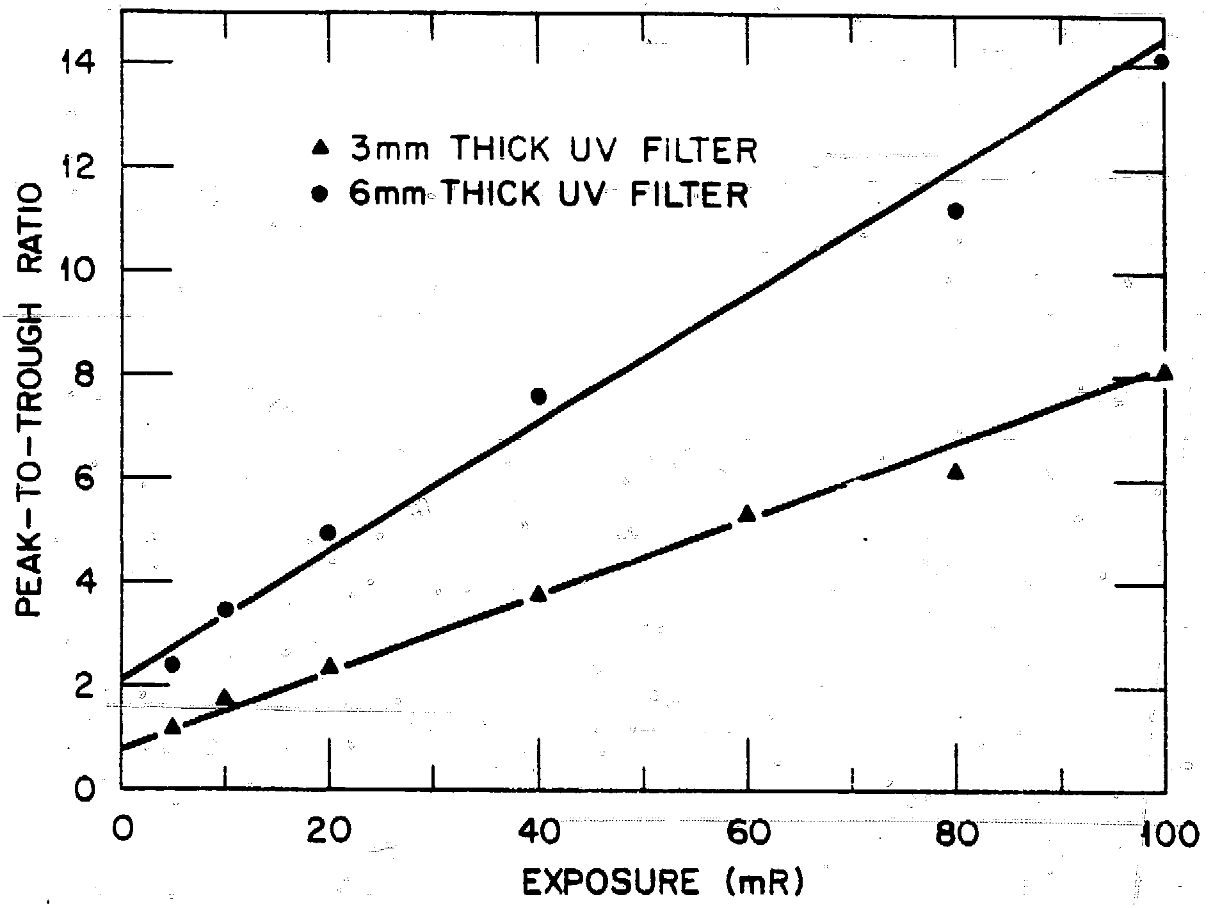
Fig. 4. Response of BeO ceramic, Thermalox 995 to gamma radiation (^{137}Cs) after incorporating the circuit shown in Fig. 1.

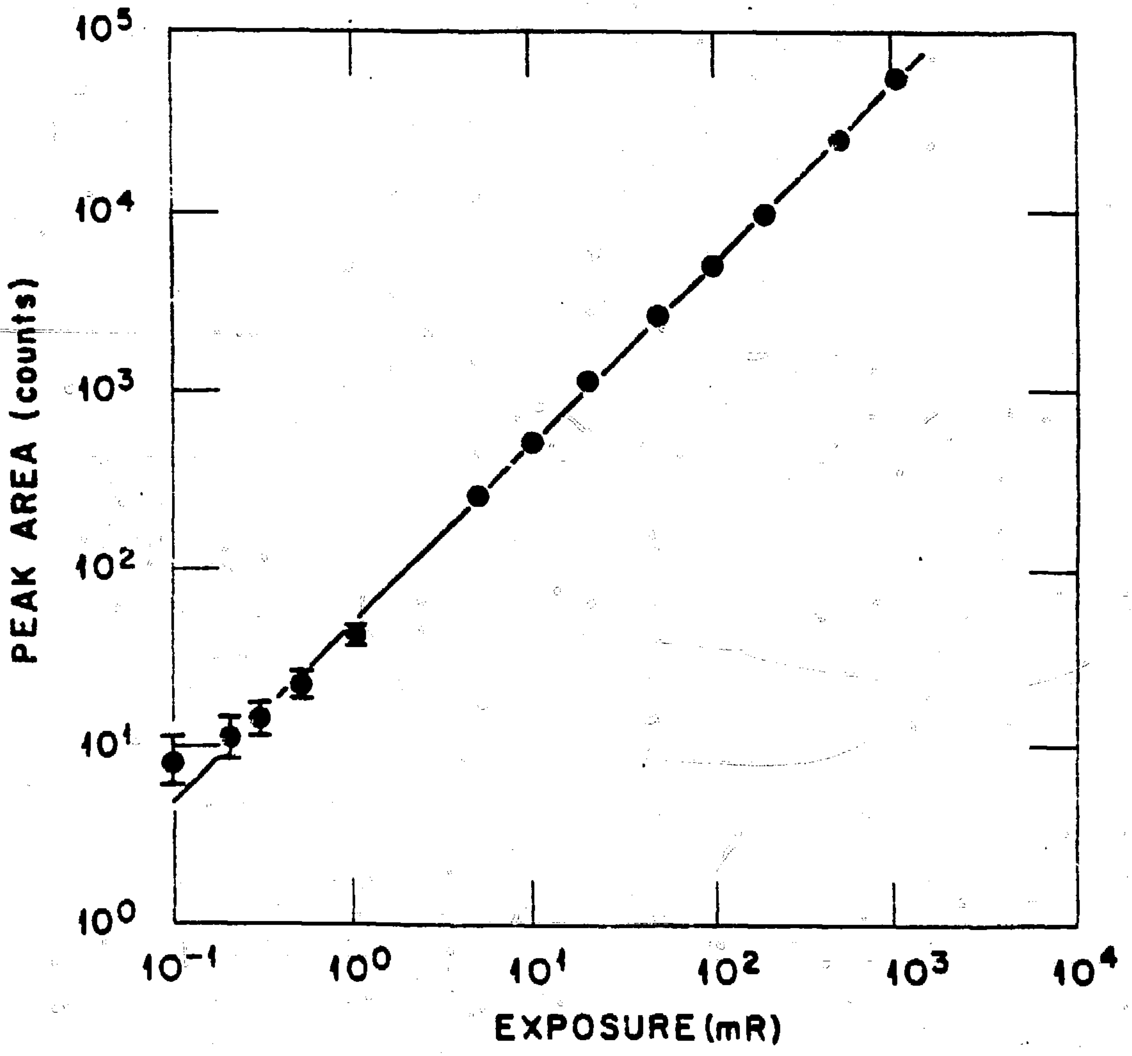
Fig. 5. Percent variation in detector response as a function of exposure to gamma radiation.

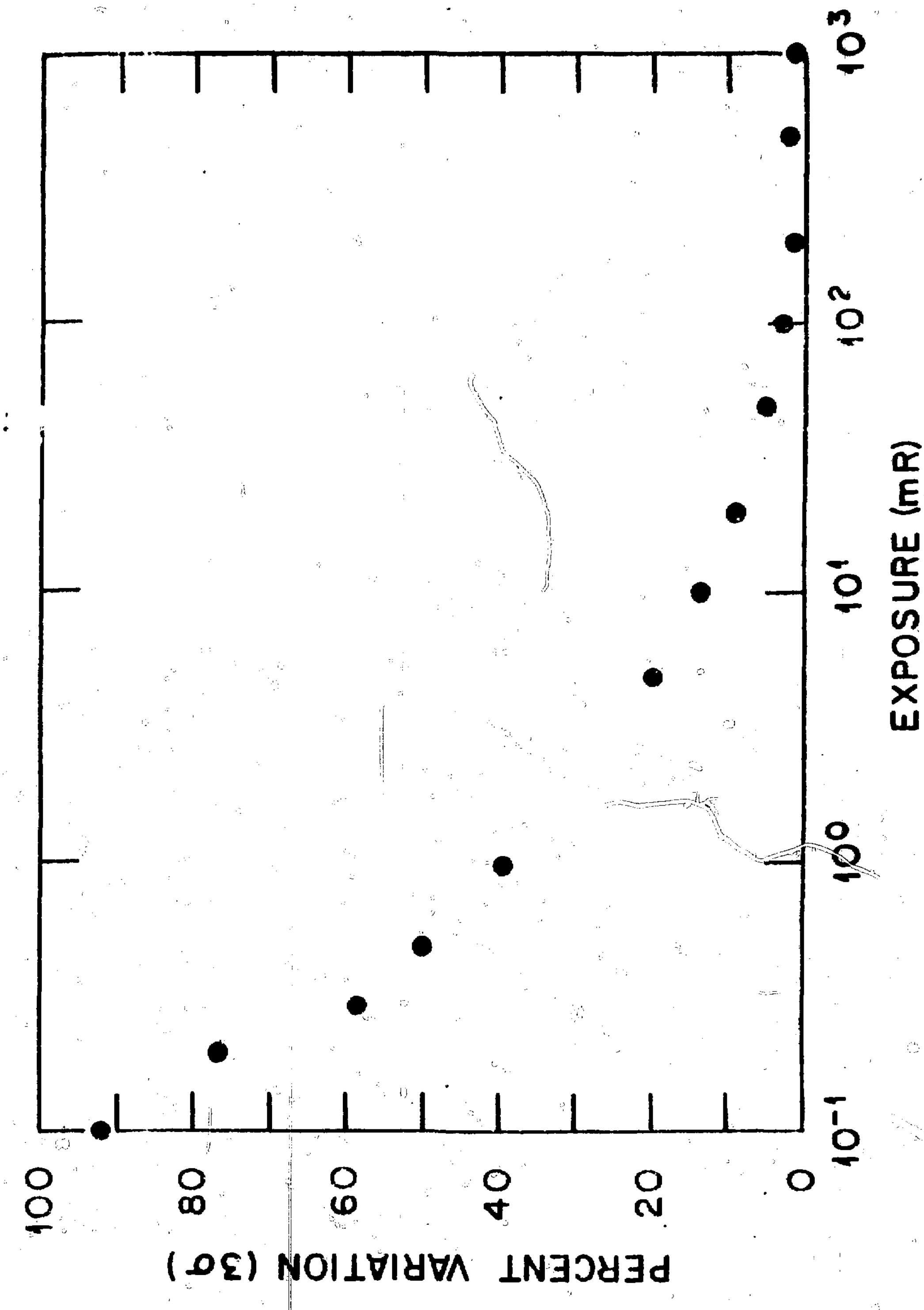
Fig. 6. Directional response of Thermalox 995 BeO ceramic, 12.5-mm in diameter and 1.5-mm thick.


TO V-F CONVERTER
10K
2N4121
2N3646
μA741
10K
OFFSET ADJUST
+12V
-12V
1000
2N3646
2N4121
10K
10M
2.2 μ
FROM PICOAMMETER


50
40
30
20
10
0
PEAK-TO-TROUGH RATIO
0
200
400
600
800
1000
EXPOSURE (mR)


PEAK-TO-TROUGH RATIO
0
2
4
6
8
10
12
14
0
20
40
60
80
100
EXPOSURE (mR)
▲ 3mm THICK UV FILTER
● 6mm THICK UV FILTER


PEAK AREA (counts)
10⁵
10⁴
10³
10²
10¹
10⁰
10⁻¹
10⁰
10¹
10²
10³
10⁴
EXPOSURE (mR)

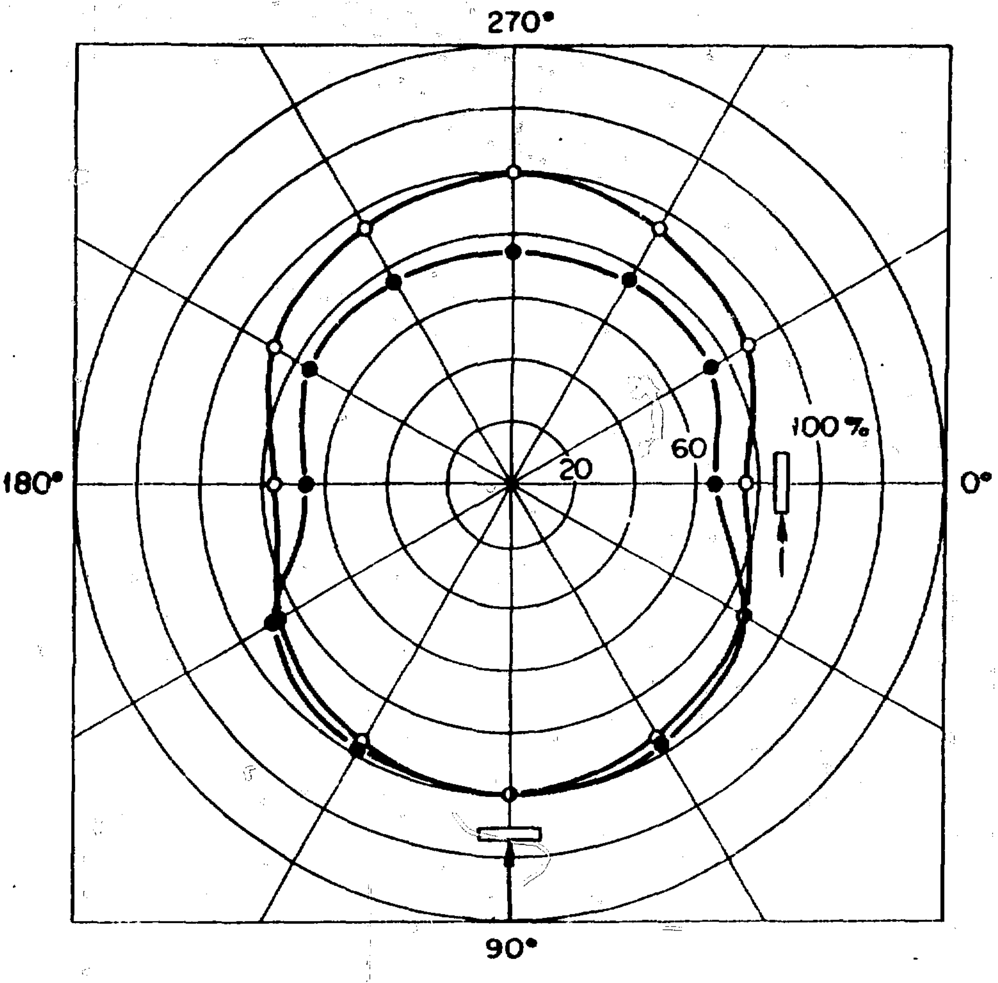

PERCENT VARIATION (3σ)
100
80
60
40
20
0
EXPOSURE (mR)
10⁻¹
10⁰
10¹
10²
10³

3

BeO THERMALOX 995, 12.5 mm DIAM

- ● SINGLE DISK
- ○ TWO DISK SANDWICH

^{60}Co GAMMA RADIATION



Question (Paul Ziemer, Purdue University):

Dr. Gammage, on your studies on light fading, in order to inter-compare the various types of light, do you normalize your data in some way to a standard light unit--lumens, or some such thing?

Answer (Gammage):

No, we look at it just from the very practical viewpoint of what optical lightings we have available to us in the various lab rooms. Very practical - very applied - - -

Question:

It's not necessarily clear then that the particular type of light would give you more fading per unit time had you normalized? I was trying to get some feel - it looked like one type of light gave you more fading than another.

Answer (Gammage):

Generally the shorter wavelength.

Question (Ziemer):

But the actual intensities of the light may not have been equivalent.

Answer (Gammage):

No.

Dup

NEUTRON RESPONSE OF A NEW ALBEDO-NEUTRON DOSIMETER

by

A. W. Blackstock
J. R. Cortez
G. J. Littlejohn
E. Storm

University of California
Los Alamos Scientific Laboratory
Los Alamos, New Mexico 87545

ABSTRACT

The Los Alamos Scientific Laboratory is converting from a film badge to a badge containing thermoluminescent dosimeters (TLD's). The new dosimeter consists of a plastic badge holding an aluminum card which contains three TLD-700 and one TLD-600 Harshaw LiF chips. In the albedo-neutron portion of the badge, based on Hankins' design, the TLD-600 and one TLD-700 chips are shielded from thermal neutrons by 0.026-inch-thick cadmium. From the difference in readings of these two TLD's, the neutron dose is estimated. There is also a detachable holder for NTA film if needed to cover the neutron energy range not covered by the albedo-neutron badge. The neutron energy response of the new badge was measured, using monoenergetic sources over the range from 0.1 to 14 MeV and is compared with a calculated response over this range. Since NTA film cannot detect proton recoils from neutrons with energies less than 0.7 MeV, a primary advantage of the TLD albedo-neutron badge is its ability to record neutrons down to the cadmium cutoff. However, a major disadvantage is its sharp decrease in sensitivity with increasing energy. Data are given showing this. Using a BF_3 detector inside 9-inch and 3-inch polyethylene spheres containing cadmium, Hankins developed a method for applying energy calibration factors to the albedo-neutron TLD badge readings. This method is based on two premises: first, that the response in rem of the 9-inch sphere is independent of energy and second, that the response of the 3-inch sphere is similar to that of the TLD albedo-neutron badge. Measurements to test the validity of these premises will be reported. The method requires initial 9-inch to 3-inch sphere ratios to be measured in each area where neutrons are encountered and periodic checks to detect changes which may occur.

I. INTRODUCTION

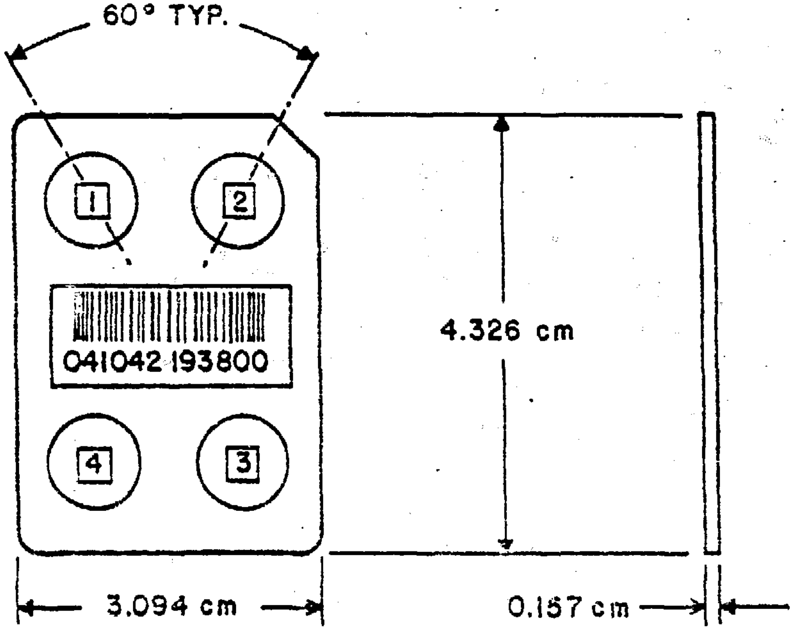
The Los Alamos Scientific Laboratory (LASL) is now changing from a film badge to a badge containing thermoluminescent dosimeters (TLD's). In this paper, we briefly describe the new LASL TLD badge. We also discuss the response of the TLD badge to continuous spectra neutrons and to monoenergetic neutrons as a function of energy. We discuss a calibration method that does not require a detailed knowledge of the neutron energy spectrum. In addition, the accuracy to be expected from the dosimeter in neutron measurements and its limitations are presented.

II. DESCRIPTION OF THE ALBEDO-NEUTRON DOSIMETER

In Paper #I-1, earlier this morning, J. R. Cortez described the TLD's, the TLD card, the badges, and the TLD reader system. To refresh your memory, I will describe, very briefly, the TLD card and badge. Figure 1 shows the TLD card supplied by the Harshaw Chemical Company. The card contains three TLD-700 LiF chips (depleted in ^{6}Li) and one TLD-600 LiF chip (enriched in ^{6}Li), with the latter opposite the cutoff corner.

Figure 2 shows the two types of badge, the "photon" and the "neutron" badge, the latter being the albedo-neutron dosimeter. A 0.066 cm thick cadmium pocket shields the TLD-700 chip in position 3 and the TLD-600 chip in position 4 from thermal neutrons. From the difference in readings of these two TLD's, the neutron dose is determined. As a backup for fast neutron measurements, an NTA film can be inserted in the holder shown and slid onto the neutron badge. Our albedo-neutron dosimeter is based on one designed by Hankins[1] but with some modification. The difference, and its effect, will be discussed later.

A Harshaw Model 2276 automated TLD system, described earlier in Paper I-1, is used to read all four TLD's on a card.



NOTE: LiF CHIPS 1,2,3 — TL 700
LiF CHIP 4 — TL 300

DOSIMETER CARD

FIGURE 1

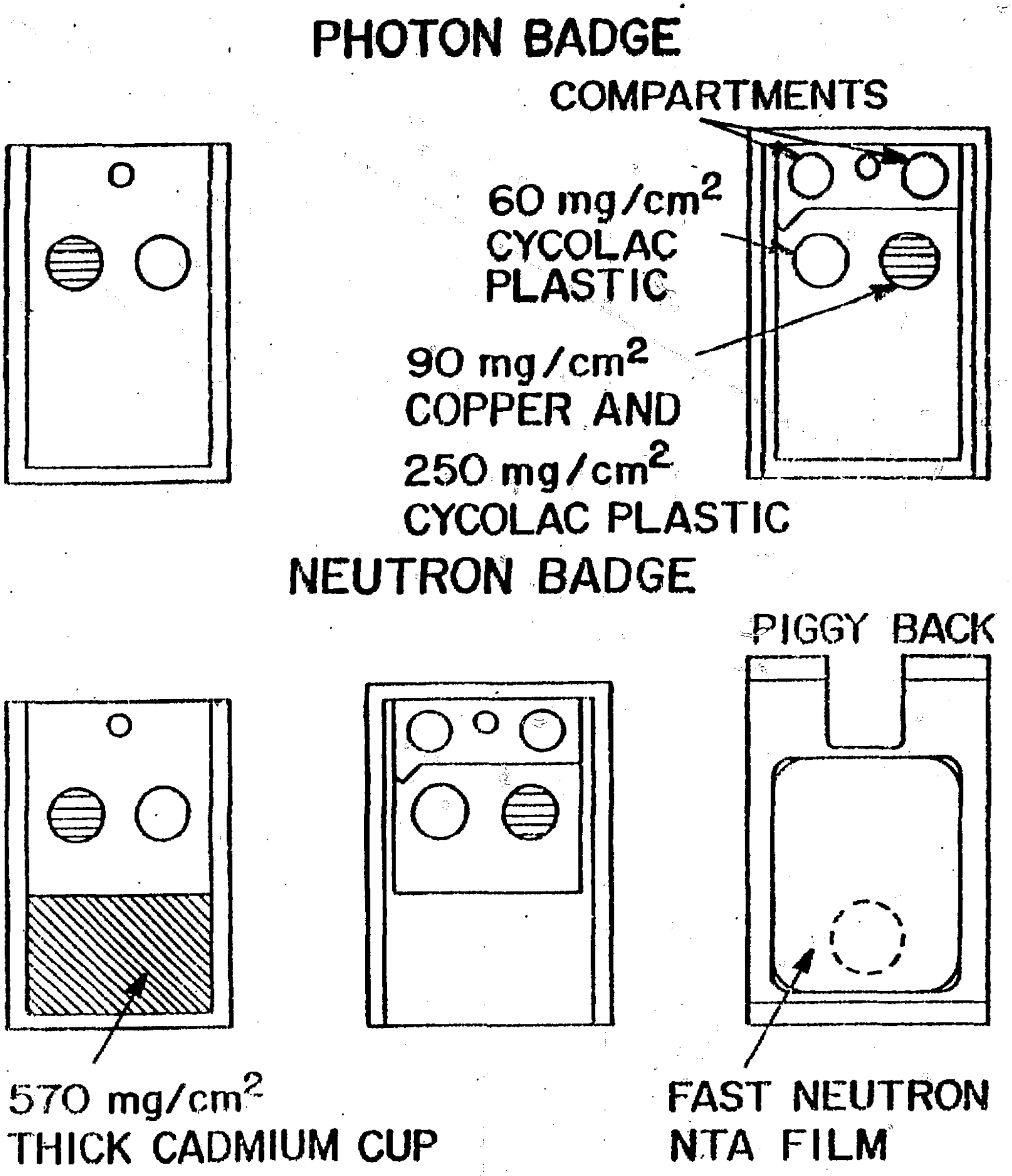

PHOTON BADGE
COMPARTMENTS
60 mg/cm^2
CYCOLAC
PLASTIC
90 mg/cm^2
COPPER AND
250 mg/cm^2
CYCOLAC PLASTIC
NEUTRON BADGE
PIGGY BACK
570 mg/cm^2
THICK CADMIUM CUP
FAST NEUTRON
NTA FILM

FIGURE 2

III. NEUTRON SOURCES

A number of different neutron sources were used in making the measurements. Continuous spectra (α,n) neutron sources used are given in Table I.

TABLE I

(α,n) Neutron Sources

Source	Average Neutron Energy (MeV)
^{238}PuLi	0.5
^{238}PuF	1.3
^{238}PuB	2.4
^{238}PuBe	~4

In addition, we made measurements with the LASL monoenergetic neutron sources listed in Table II.

IV. EXPERIMENTAL RESULTS

Figure 3 shows the NTA film response in the present LASL film badge vs neutron energy, where it can be seen that the response changes by a factor of 10 between 1 and 14 MeV. A serious limitation on the NTA film is that it cannot detect proton recoils from neutrons with energies less than ~0.7 MeV. A major advantage of the LASL TLD albedo-neutron badge over NTA film is its ability to measure neutrons down to the cadmium cutoff.

Alsmiller and Barish[2] calculated the neutron response as a function of energy of an albedo-neutron dosimeter similar to one studied by Hankins[3], which in turn is similar to ours. An average of their results is plotted in Figure 4 and compared to our measurements with a prototype albedo-neutron dosimeter. Their results were normalized to the experimental measurements at 1 MeV. Our measurements, which were made using monoenergetic neutrons over the range from 50 keV to 14 MeV, follow the shape of the calculated curve. The measured thermal neutron response is also shown.

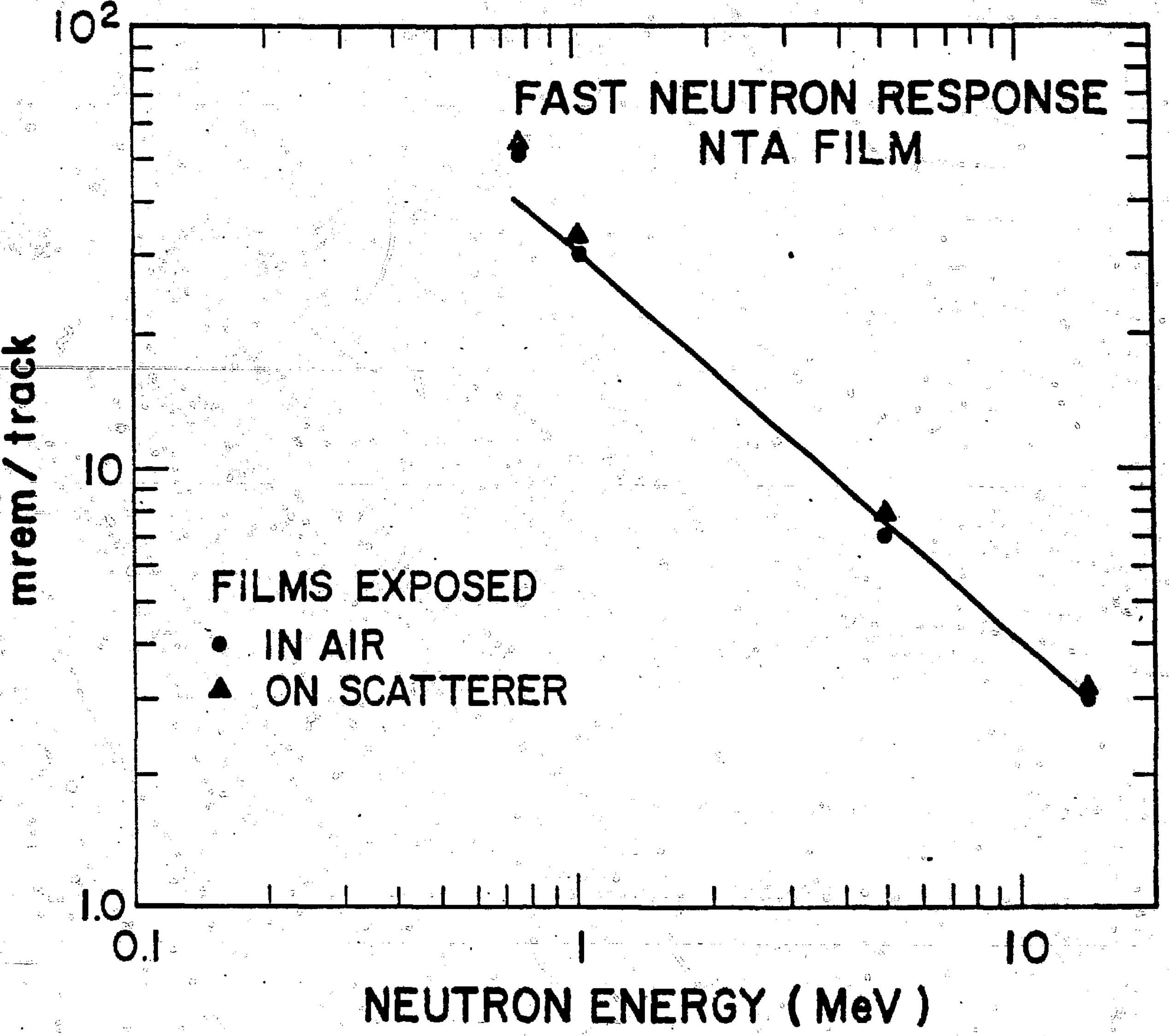

10²
10
1.0
0.1
1
10
mrem / track
NEUTRON ENERGY (MeV)
FAST NEUTRON RESPONSE
NTA FILM
FILMS EXPOSED
● IN AIR
▲ ON SCATTERER

FIGURE 3

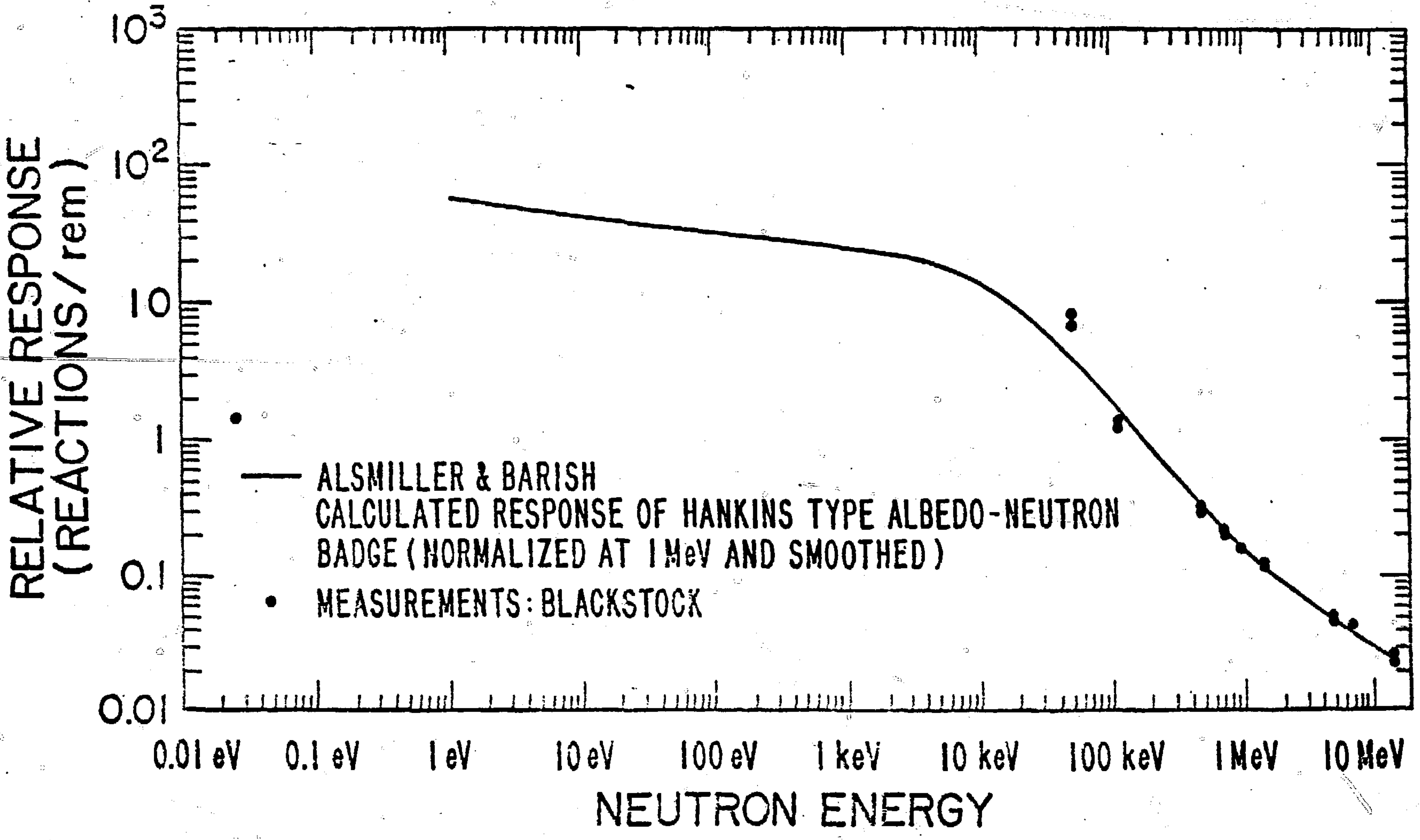

RELATIVE RESPONSE
(REACTIONS/rem)
10³
10²
10
1
0.1
0.01
0.01 eV
0.1 eV
1 eV
10 eV
100 eV
1 keV
10 keV
100 keV
1 MeV
10 MeV
NEUTRON ENERGY
ALSMILLER & BARISH
CALCULATED RESPONSE OF HANKINS TYPE ALBEDO-NEUTRON
BADGE (NORMALIZED AT 1 MeV AND SMOOTHED)
● MEASUREMENTS: BLACKSTOCK

FIGURE 4

TABLE II

Monoenergetic Neutron Sources Used at LASL

Facility	Neutron Energy (MeV)	Reaction	Neutron Flux Detector
Cockcroft-Walton Accelerator	2.5	$D(d,n)\ ^3He$	Associated Particle Detector
	14	$T(d,n)\ ^4He$	
3.75 MV Van de Graaff Accelerator	$0.05 \leq E_n \leq 1.5$	$^7Li(p,n)\ ^7Be$	Fission Chamber
Vertical Van de Graaff Accelerator	$1 < E_n \leq 6$	$T(p,n)\ ^3He$	Proton Recoil Telescope
	$4 \leq E_n < 10$	$D(d,n)\ ^3He$	
	$14 \leq E_n \leq 20$	$T(d,n)\ ^4He$	
Omega West Reactor	Thermal	-	Gold Foil Activation

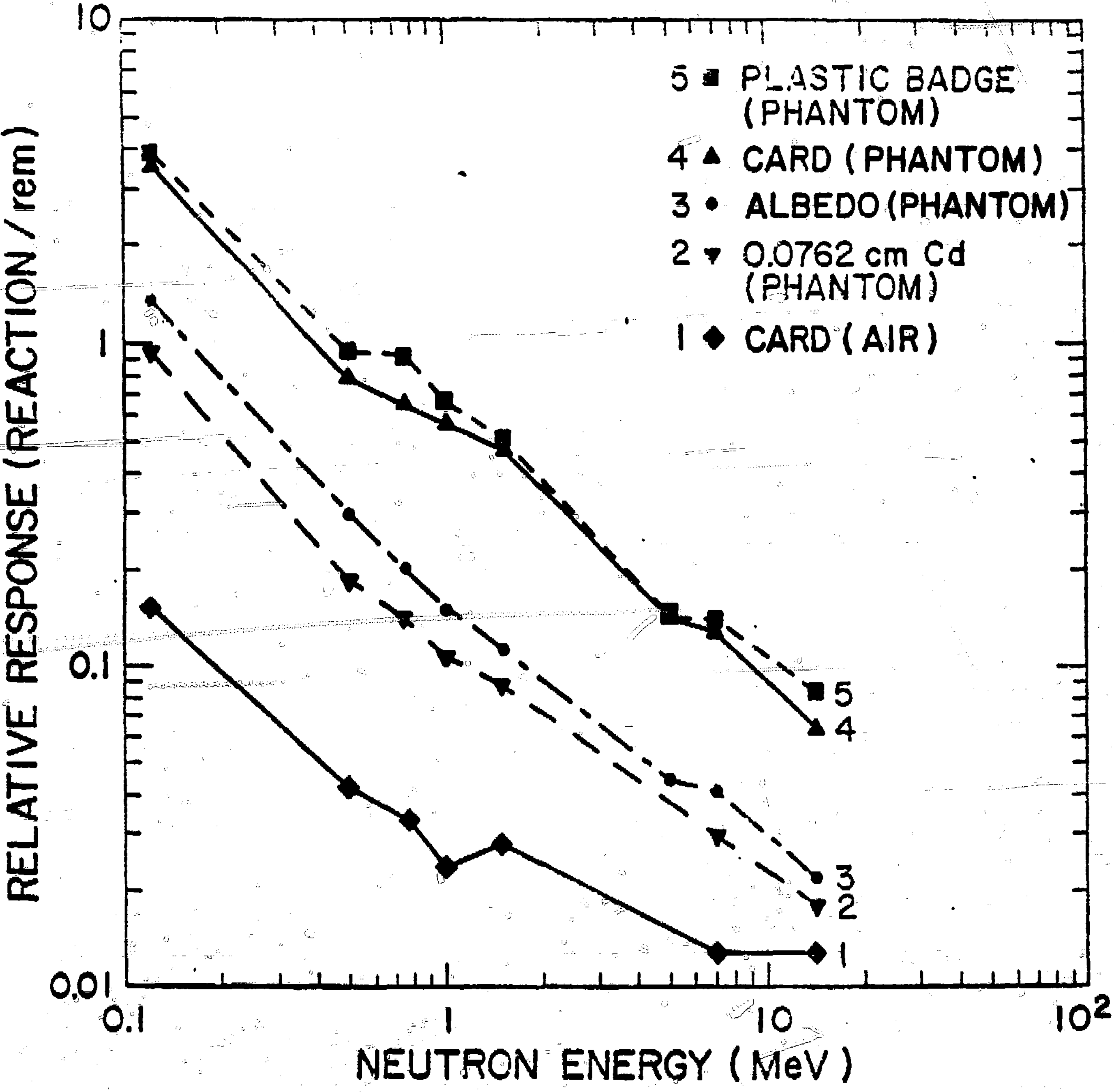
Figure 4 also shows the primary limitation of the TLD albedo-neutron dosimeter - its sharp decrease in sensitivity with increasing neutron energy. The calculated response is fairly flat from 1 eV to 10 keV, but decreases by a factor of ~1000 from 10 keV to 14 MeV.

In Figure 5, the response vs neutron energy of several different configurations is shown. The response of a prototype bare TLD card in air (Curve 1) is compared with its response on a water-filled manikin (Curve 4); neutrons reflected from the phantom increase the response by an order of magnitude. Placing the card in a plastic badge with no filters increases the response only slightly (Curve 5).

Hankins' prototype albedo-neutron dosimeter, on which the new LASL badge is based, had a layer of 0.32 cm thick polyethylene on both sides of the TLD-600 and TLD-700 chips inside the 0.076 cm thick cadmium shield. As Figure 5 shows, the polyethylene increases the response by about 40% (Curves 2 and 3). However, it was decided to eliminate the polyethylene from the new LASL badge because the increase in sensitivity did not justify the added bulkiness and cost of fabrication. Without the polyethylene, the prototype neutron badge response, relative to ^{60}Co γ rays, is approximately 1 at 100 keV, but decreases to about 0.02 at 14 MeV. If 10 mrem is the minimum detectable dose at 100 keV, then the minimum detectable dose for 14 MeV neutrons is 500 mrem. The LASL albedo-neutron badge is designed so that NTA film can be added to complement the TLD measurements for high energy neutrons.

V. METHOD OF CALIBRATION

In their conclusions, Alsmiller and Barish state: "This [the rapidly varying response as a function of incident neutron energy] does not mean that albedo-neutron dosimeters cannot be used, but it does mean that they

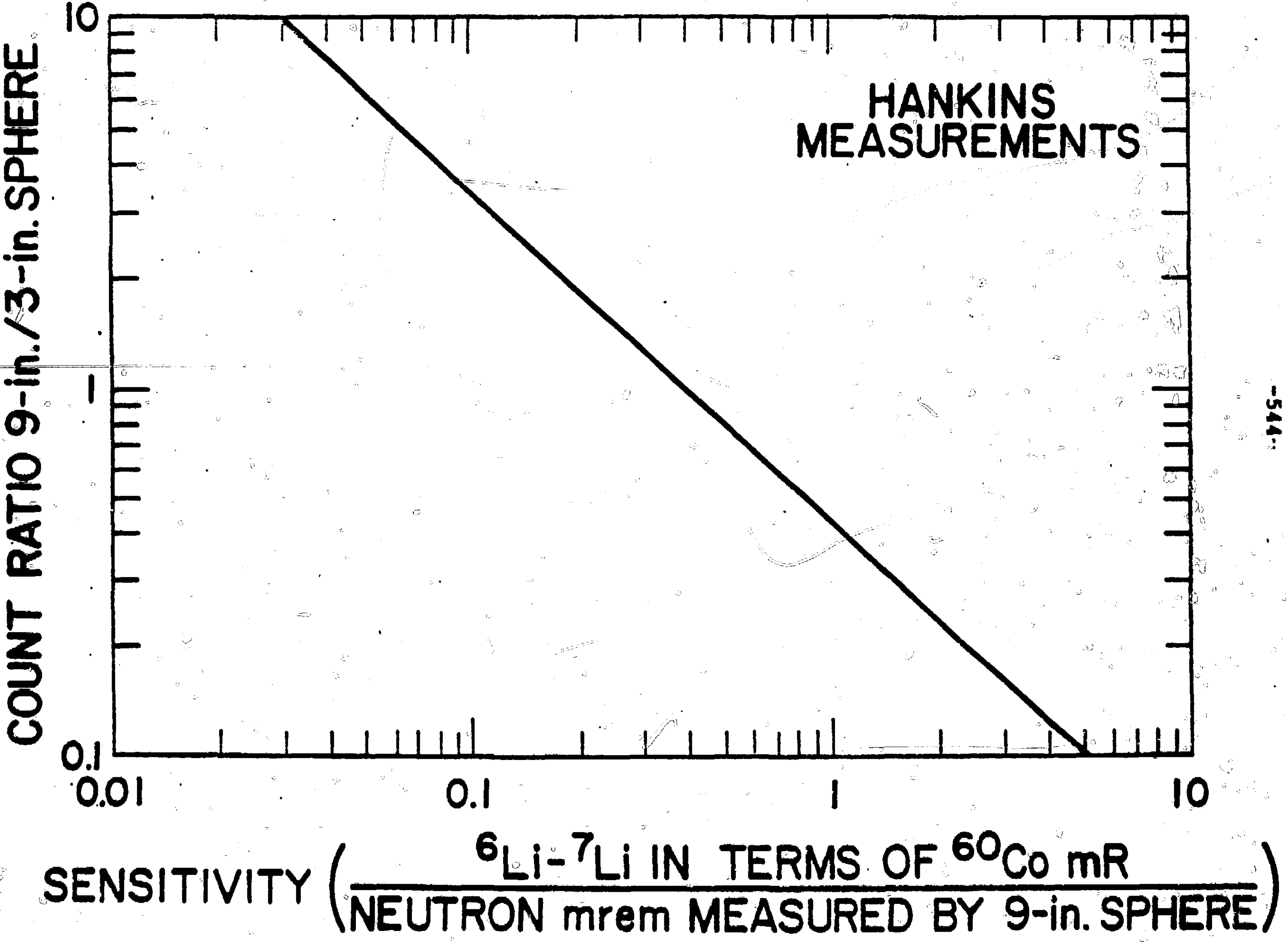


10
1
0.1
0.01
0.1
1
10
10²
RELATIVE RESPONSE (REACTION / rem)
NEUTRON ENERGY (MeV)
5 ■ PLASTIC BADGE (PHANTOM)
4 ▲ CARD (PHANTOM)
3 ● ALBEDO (PHANTOM)
2 ▼ 0.0762 cm Cd (PHANTOM)
1 ◆ CARD (AIR)
5
4
3
2
1

FIGURE 5

can be expected to give reliable results only when used with spectra that are very similar to the spectra for which the dosimeter has been calibrated." However, Hankins[1] developed a method for calibrating the albedo-neutron dosimeter that does not require a detailed knowledge of the neutron energy spectrum. Using a BF_3 proportional counter centered inside 9-inch and 3-inch polyethylene spheres containing thin shells of cadmium, he measured neutrons from a variety of sources. He then plotted the ratio of the 9-inch to 3-inch sphere counting rates vs sensitivity, which is the ratio of the TLD-600 response less the TLD-700 response in units of ^{60}Co mR to the neutron mrem as measured with the 9-inch sphere. As shown in Figure 6, he obtained a linear relationship on a log-log plot. Therefore, to determine neutron exposures using this method, the ratio of 9-inch to 3-inch sphere count rates is measured in each potential neutron exposure area. Then the calibration factor, the inverse of the sensitivity, is obtained from a graph such as Figure 6, generated for the LASL TLD albedo-neutron badge. Periodic checks of the 9- to 3-inch sphere ratios will be required to detect any changes in the ratios which may occur due to changes in the neutron spectra in areas where the badge is worn.

This calibration method is based on two premises: First, that the 9-inch sphere closely approximates the neutron dose in rem independent of neutron energy, and second, that the response of the 3-inch sphere as a function of neutron energy is similar to that of the TLD albedo-neutron dosimeter. Figure 7 shows that the response of the 9-inch sphere is quite flat between 0.5 and 8 MeV, but increases by a factor of ~2 down to 0.1 MeV and drops off by a factor of 2 at 14 MeV. Also, the 3-inch sphere response is similar to the albedo dosimeter response over the whole range



COUNT RATIO 9-in./3-in. SPHERE
10
1
0.1
0.01
0.1
1
10
HANKINS MEASUREMENTS
SENSITIVITY (^{6}Li-^{7}Li IN TERMS OF ^{60}Co mR / NEUTRON mrem MEASURED BY 9-in. SPHERE)

FIGURE 6

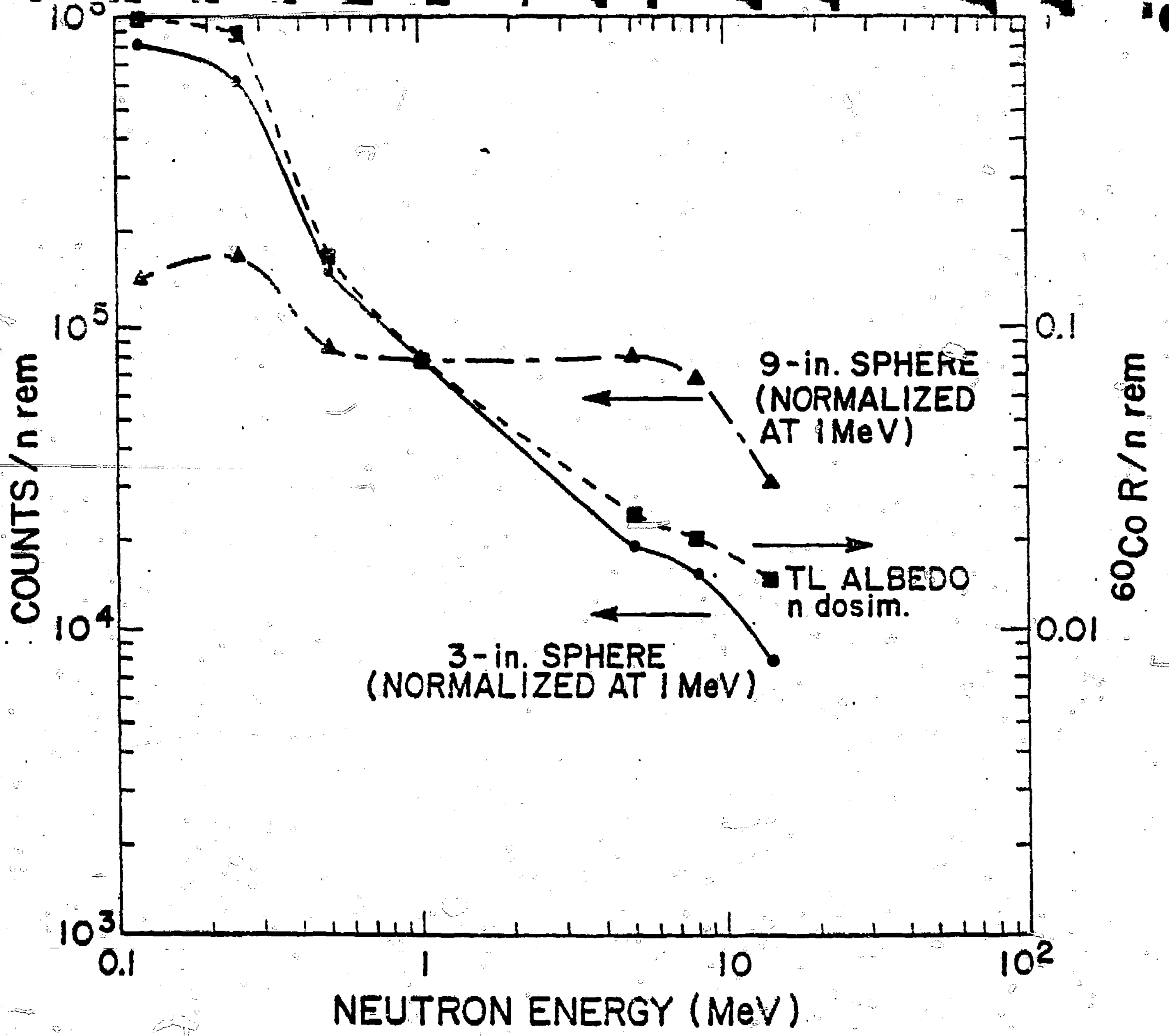

10⁵
10⁵
10⁴
10³
0.1
1
10
10²
0.1
0.01
COUNTS / n rem
⁶⁰Co R / n rem
NEUTRON ENERGY (MeV)
9-in. SPHERE (NORMALIZED AT 1 MeV)
TL ALBEDO n dosim.
3-in. SPHERE (NORMALIZED AT 1 MeV)

FIGURE 7

from 0.1 to 14 MeV, the largest deviation occurring at 14 MeV, where the albedo dosimeter response is 46% higher. Both the 9-inch and 3-inch sphere responses were normalized to the albedo badge response at 1 MeV in Figure 7.

In Figure 6, the 9-inch to 3-inch sphere ratio is plotted vs the albedo badge response, and Figure 7 shows the albedo badge response as a function of neutron energy. From these data, we can plot the 9- to 3-inch ratio vs neutron energy, which is done in Figure 8. Using the 9- to 3-inch sphere ratio technique, we can get an "average" neutron energy for a neutron source from Figure 8. This average energy may be quite different from that measured with other detectors or by other techniques.

VI. FIELD MEASUREMENTS

We have made a number of 9- to 3-inch sphere ratio measurements in two potential neutron exposure areas at LASL: a plutonium chemistry facility and the Los Alamos Meson Physics Facility (LAMPF). At LAMPF, the mean value of the ratio, measured in 18 different locations, was 0.17, with a standard deviation (σ) of 30%. At the plutonium facility, the average value of the ratio measured in 13 different locations was 0.57±35% (1σ). If we assume that the curve in Figure 8 is correct, these ratio values indicate an average neutron energy of <100 keV at LAMPF and of ~200 keV at the plutonium facility. This may explain why few neutron exposures have been observed at these facilities, since NTA film cannot detect neutrons with energies less than about 700 keV.

From a graph similar to that in Figure 6, but generated from our measurements with the new LASL albedo-neutron badge, we obtained a calibration factor (the inverse of Hankins' calibration factor) for LAMPF of 0.91 n mrem/^{60}Co mR, with an overall estimated σ of 35%. From the same graph, we

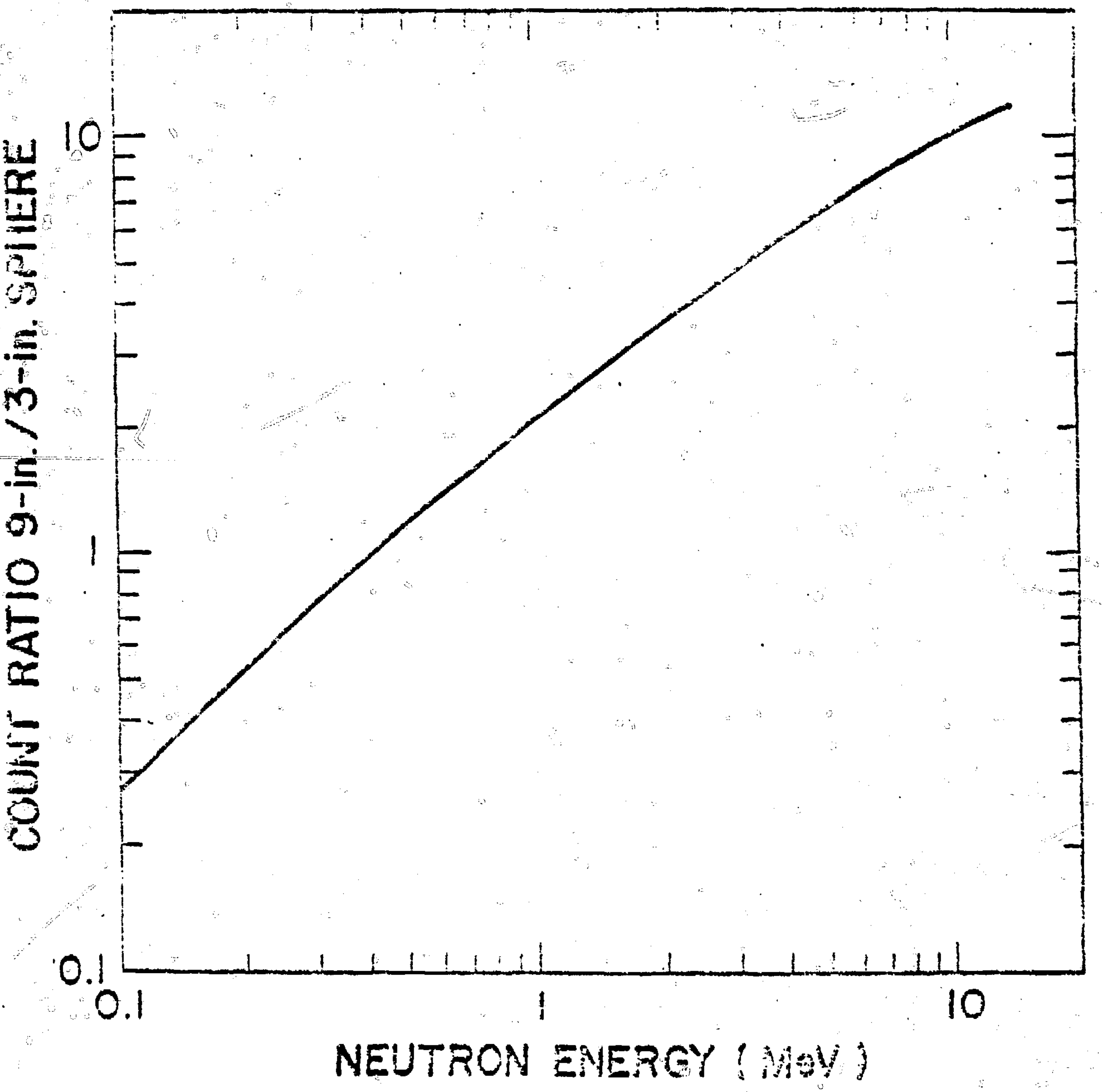


FIGURE 8

obtained for the plutonium facility a calibration factor of 3.2 n mrem/^{60}Co mR ± 40% (1σ). After an adequate number of measurements in each location within a facility, we expect to be able to increase the accuracy of neutron dose determinations providing (1) a person's exposure can be determined to have occurred in one, or a very few, locations and (2) that the ratio at each location does not change very much with time. A difference of 14% was observed in the 9- to 3-inch sphere ratio when the measurement was repeated at one location in the plutonium facility five months later. This is well within the experimental uncertainty indicated above.

VII. SUMMARY

We reiterate here the relative advantages and disadvantages of the NTA film dosimeter and the LASL albedo-neutron dosimeter utilizing TLD's. Although the sensitivity of the NTA film increases with increasing neutron energy, the film cannot measure exposures due to neutrons with energies less than about 700 keV.

The TLD albedo-neutron badge response varies rapidly with incident neutron energy, decreasing with increasing energy. However, it can measure exposures from neutrons with energies < 700 keV, since its response continues to increase down to the cadmium cutoff. The LASL albedo-neutron badge is designed so that NTA film can be added to complement the TLD measurements for higher energy neutrons.

Hankins[1] devised a method for calibrating the dosimeter in spite of the strong dependence of its response on neutron energy. Our measurements thus far indicate that the method gives reasonably accurate exposure measurements in two LASL facilities where neutron exposures may occur.

ACKNOWLEDGMENTS

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The following reports contain numerous references to earlier work.

1. D. E. Hankins, "A Small, Inexpensive, Albedo-Neutron Dosimeter", Los Alamos Scientific Laboratory Report LA-5261 (July 1973).

2. R. G. Alsmiller, Jr. and J. Barish, "The Calculated Response of ^{6}LiF Albedo Dosimeters to Neutrons with Energies $\leq$400 MeV", Health Phys. 26, 13 (1974).

3. D. E. Hankins, "Factors Affecting the Design of Albedo-Neutron Dosimeters Containing Lithium Fluoride Thermoluminescent Dosimeters", Los Alamos Scientific Laboratory Report LA-4832 (January 1972).

Question (Ken Skrabble, University of Lowell):

Is there any preferred geometry of exposure of individuals in these areas? Will you determine the calibration factor or are individuals more or less in any orientation relative to the radiation field?

Answer (Blackstock):

In the plutonium facility they are subject to neutrons from more or less all directions. However, at the LAMPF Facility I guess you could say that they're there also, but we do intend to look further into this. Dale Hankins, when he was at LASL, did make measurements taking into account the directional variation of the field. We still have to do more work in that respect.

Question (Skrabble):

What I was going to suggest, if that's the case, I don't know if you did it or not, but when you radiated your badges on yourphantoms, did you rotate the phantoms on these radiation fields. I think that would probably be appropriate under the circumstances.

Answer:

No, we have not done this yet.

Question (Ziemer, Purdue):

My question is somewhat related to Ken Scrable's --it relates to calibration. What is the importance of the proximity of the badge to the body, relative to wearing these loosely hanging on a lab coat pocket and so on. I assume your calibration of the badge is directly against the phantom, so I'm asking about the clothes interposed on persons.

Answer (Blackstock):

Yes. That is one of the reasons for going to a cadmium shield which shields both sides of the TLD 600 and TLD 700. This configuration Dale Hankins established that the badge does fall away from the body. It's very insensitive to such things as that. That was a great reason for going to cadmium on both sides.

Question (Murray Walsh, Ontario Hydro):

Perhaps I missed it in your presentation, but have you tried to cross-check the calibration factors by either obtaining spectral information by either calculation or measurement and then folding the spectral information with the response of the dosimeter, or by placing these dosimeters on phantoms and and measuring the dose rate with a rem. And what were the results?

Answer (Blackstock):

If I follow you correctly, you want to know whether we made any spectral measurements at first and was it to see how they agree or disagree with that curve I presented?

Question (Walsh):

Well, I was wondering if you did do some spectral information in the places where your people are exposed.

Answer:

Yes, we have done some at LAMPF using Bonner spheres and using only a graphical analysis method. And the energies that we got there did agree fairly well with what I've shown on this graph here. Now, I didn't understand the second part of your question.

Question (Walsh):

What I was asking was whether you have placed the dosimeters on phantoms and measured the dose rate by some other means.

Answer:

Yes, we have done that mostly with the continuous spectra sources and there is fairly good agreement between the dose equivalent that we measure.

Question (Walsh): Excuse me, what do you mean by fairly good–as I'm doing this type of work also.

Answer: Well, in most cases it's 20% or 15%, about that order.

ICPP DIGITAL WIDE AREA PORTAL MONITOR

C. Nichols

I. INTRODUCTION

For over a year the ICPP has utilized a portal monitor which was designed to increase sensitivity to beta radiation and provide increased overall sensitivity for contamination control on personnel exiting a plant. Continued development in the areas of detector technology and geometrical configuration of the portal unit, has produced a prototype portal monitor that is many times as sensitive for detection for radioactive contamination than any monitor previously available. The electronics, developed for the first generation ICPP Portal Monitors, have not significantly changed except for minor improvements. The electronics have been virtually maintenance-free and provide the "State of-the-art" instrumentation for portal monitor application. The planned improvements for the next generation electronics will have refinements to the basic digital logic system with the emphasis on the alarm generating logic. Even though the portal monitor was designed and developed as a contamination control device, testing with 5 to 12 g quantities of U-235 ICPP UO_3 product produced results that easily exceed detection levels of SNM as defined in the standards.

II. EQUIPMENT

Sensitivity of the portal monitor is dependent on three major factors, namely background radiation, detectors, and the geometry between personnel entering the monitor and the detectors. The portal monitor is divided into the following areas which will enhance the sensitivity:

A. Electronics

The electronics utilizes digital logic to provide background subtract compensation to improve the electronic signal to background ratio. Varying background radiation at the ICPP requires setpoints above the maximum background levels in order to avoid false alarms. Once the background is subtracted, the signal should then be caused by the object in the detectors within statistical accuracy. Background radiation levels are provided using the sliding interval technique where background radiation is accumulated in 0.2 sec. intervals over a 100 segment time space or over 20 sec. The digital logic system processes all signals in real time (single bit information rather than averaged blocks of information) reducing hardware requirements and cost.

The electronic system is divided into the following parts.

1. High and Low Voltage Supplies. Proper Electro Magnetic Intensity (EMI) shielding and isolation to prevent noise entering the system via the power sources is important, thus is a standard technique and will not be further explained.

2. Alarm Generating Logic. The alarm generating logic is divided into two sections.

   a. Continuous background updating when the monitor is not occupied, thus background counting with no alarm.

   b. Data manipulation by subtraction of the background counts held in memory from the counts received when the monitor is occupied.

A block diagram of the electronic system is presented in Fig. 1 and 2 for system functions under these two cases. For clarification: in case (a) with the monitor not occupied assume a 1000 count/sec. background; time mode in 0.05 sec. The hold gate is cleared (personnel detector not activated) and counts from the detectors go through the 400 divider to the 100 bit shift register at 1000/400/sec. = 2.5/sec. The shift register advances every 0.2 sec. which then allows a 20 second background storage. The total count in the shift register is 50 for this example (20 x 2.5 = 50). This count is shown on the LED display. The LEDs display the sum of the high level counts in the shift register memories.

In case (b) with the portal monitor occupied, assume background radiation at 1000 c/sec., time mode is the same for case (a), and a person's feet are contaminated at 250 c/sec. The total count rate seen by the monitor is 1250 c/sec. on the foot section channel. The first 1000 c/sec. are subtracted from the 1000 counts held in memory; the remaining counts then are processed in the electronics and appear as 12 on the LED display (1250/400 x 20 sec. = 62.5 counts; 62.5, -50 = 12 counts). These counts then go to the alarm circuitry which compares the counts with the set points. Exceeded counts give alarms that are compared to setpoints. Alarm bits count an up-down counter up as they enter the shift register and count it down as they leave. The up-down counter then has an output equal to the high bits in the shift register. Alarms can be set at 1 of 8, 2 of 8 up to 8 of 8. Any channel can ring the bell but each channel has its own alarm lamp.

3. Personnel Presence Detector. A motion detector is used to indicate to the electronics that the monitor is occupied. This begins the scanning sequence which is determined by the time selector switch i.e., a selected time of 0.05 sec. indicate a scan time of 0.05 sec. intervals. The motion detector was chosen over electric eyes, infrared detectors and foot switches because of the immunity to background light, and electronic noise inherent in other personnel detectors.

4. **Alarms, Audio and Visual.** A bell is used to give an audible alarm with visual alarms to indicate the body location of the radiation detected by the monitor.

B. Radiation Detectors

The type of detectors and their configuration constitute a most significant factor in providing an instrument of maximum sensitivity. Signal versus noise and/or background cannot be improved only manipulated. Wide-area gas-proportional (propane) detectors were chosen not to get a high signal to background ratio, but to get the maximum count rate from the objects in the detectors. The 6" x 14" wide-area detectors have a window thickness of 8-9 mg/cm^2 and use propane as the ionization gas. Up to three detectors are wired in parallel and comprise one channel such as the foot channel, etc. If too many detectors are connected to a channel the background radiation statistically is excessive and sensitivity is compromised. The method then of multiple channels, or signal separation is one of the most effective methods of improving signal to background ratios.

C. Detector Geometry

The wide-area gas-proportional detectors are arranged in a configuration that resembles a partially opened doorway (See Fig. 3). Personnel exiting the plant through the portal monitor approach the doorway, steps sideways (to the right), making sure the feet were scanned by the foot detectors, and proceed through the door. This configuration is most significant for enhancing sensitivity and provides a method for surveying the entire human form in extremely close proximity to the detectors.

III. OPERATING THEORY - CHARACTERISTICS

What is being established in this section is the basis or justification for fabricating frisker electronics that appear to violate the laws of statistics. Therefore a discussion of statistical analysis as it relates to the portal monitor is as follows:

Consider these calculations for false alarm probability:

	Alarm Level	False Alarm Probability
2 = 97.73% or	B + 2	1 - 0.9733 = 0.0227
3 = 99.865% or	B + 3	1 - 0.99865 = 0.00135

Applied to our case, as an example:

background = 500 counts/sec.

scan time = 0.2 sec.

$\overline{B}$ = 100 c/0.2 sec.

2 = 20 alarm level at 120 counts

False alarm probability for 2 is 0.2 sec. x 1/0.0227 = 8.81 sec. between false alarms. Now double the time and T = 0.4 sec., $\overline{B}$ = 200; 2 = 28.28; alarm level at 228 counts.

0.4 x 1/.0227 = 17.62 sec. between false alarms.

Now consider the same time but calculate for two consecutive alarms to trigger an alarm:

(a) Assume alarm setting to give 0.9 reliability or 1 in 10 failure or false alarm;

(b) Two such stages are in series;

(c) On the average 1 of 10 of the first stage will occur when 1 of 10 of second only in 1 of 100 times. This is a reliability value of 0.99.

Mathematically:

$$R(t) = 1 - [1 - r(t)]^n$$

Where R(t) = Reliability of total system

r(t) = Reliability of single stage

n = Number of stages

$$R(t) = 1 - [1 - 0.9]^2 = 0.99$$

for 2 alarm level above background

$R(t) = 1 - [1 - 0.9773]^2 = 0.999485$; 0.2 sec. x 1/.000515 = 388 sec.

Doubling the time as before 776 seconds between false alarms. The conclusion then is an alarm trigger of two consecutive is much better than just doubling the counting time or less false alarms.

Another pertinent question one might ask; "Is there enough time to detect contamination on an individual in 0.2 seconds?"

Consider a realistic background at the ICPP for a wide-area detector of 30 counts/second.

B = 30 counts/sec.

0.2 sec. = 6 counts

= B = 6 = 2.45

4 = 9.8 counts (alarm point setting)

Setpoint = $\overline{B}$ + 4

= 6 + 9.8

= 15.8 counts 0.2 sec.

Time for detection of signal for detection criteria of net signal counts = n x std. deviation of the background:

Mathematically:

$\overline{s}t = n \; \overline{B}t$

$\overline{s}^2 t^2 = n^2 \; \overline{B}t$

and $t = \frac{n^2 \; \overline{B}}{\overline{s}^2}$

4 deviation where = 15.8 counts/0.2 sec. = 79 counts/sec. and $\overline{B}$ = 30 counts/sec.

$t = (4^2)(30)/79^2 = 0.08$ sec. which is less than the 0.2 sec. scan time.

Another question that may be asked that is basic to the frisker electronics is, how can a count for background summed over 20 sec. be subtracted from the count over a 0.05 sec., 0.1 sec., 0.2 sec., etc. interval? (These are the time scan intervals that are set into the electronics). The block diagram, figure 2, of the electronic functions will show that the count for background is divided by different numbers: 400, 200, 100, or 50. The time interval is changed to compensate and give the same number of counts for a given radiation field as is seen directly on the selected time interval switch. For example: 1200 counts/min. field = 1200/60 or 20 counts/sec. = 400 counts over 20 sec. Therefore on T = 0.05 sec. (400) background = 1. This reading then appears as the background reading displayed on the front panel.

IV. SENSITIVITY MEASUREMENTS

Evaluation of the portal monitor sensitivity was conducted for two major areas of concern for personnel exiting the ICPP; namely, the diversion of SNM (Special Nuclear Material) and beta-gamma radiation contamination on the skin and clothing.

SNM Diversion Sensitivity. The walk-through testing generally followed the performance guidelines set forth in a letter dated April 2, 1975, from Robert Thorp, Acting Director for the Division of Safeguards and Security to the Division Directors for ERDA. The performance specifications or sensitivity tests should be made by actually walking through the monitor with a 10 gram sphere of the U-235 isotope, metal carried on the body and located in the minimum sensitivity region (centered head or foot for these monitors). At least 40 walk-throughs should be performed with at least 20 detections. To acquire the required source would have been impractical in the time requirements to complete this report, therefore, the sensitivity tests were made with UO_3 ICPP product 80% enriched and encapuslated geometrically in a 2.5 mm thick glass cylinder and then shielded in a 3 mm thick brass pipe. The end result being a sensitivity measurement much more conservative than the standards required. The frisker performed extremely well as can be seen from Table I.

Beta-Gamma Sensitivity. The walk-through testing for beta-gamma radionuclides required the development of performance standards not available for this type of portal monitor because of the extreme sensitivity. Generally speaking, the least sensitive region of an individual is considered the ceterline from head to foot; therefore, the testing of the portal monitors was evaluated by placing sources in various locations along this centerline. The standards set for the new ICPP portal monitors are: (1) a point source of 0.125 Ci Cs-137 deposited on a glass planchet 1 cm^2 in area and (2) a wide-area source of 0.125 Ci Cs-137 deposited on a plastic plate of 100 cm^2 in area. The number of walk-throughs should be large enough to lend confidence to the results. The detection probability with 95% confidence should be 50% or greater. Using Fig. 4, which is derived from confidence belts given by Dixon and Massey (2) a determination can be made of an acceptable performance by finding the point corresponding to the number of passages and the percentage of passages detected. If the point lies above, then a 95% confidence of 50% or greater detection probability. From Table I and Table II it can easily be seen that the ICPP portal monitor exceeds these performance standards.

V. SUMMARY

The portal monitor designed, built and installed at the ICPP has taken portal monitoring beyond the mere checking of personnel for gross amounts of contamination to the necessary limit of rapid and extremely sensitive automatic radioactivity detection. The new portal monitor will undoubtedly set the standard for sensitivity throughout the nuclear industry. All testing to date indicates that in addition to personnel contamination control, the instrument will serve as an extremely effective SNM detector, capable of detecting U-235 in half the quantities specified in Federal Standards.

TABLE I

UO_3 Product Frisker Alarms - 10 Walk-Throughs Each
Area - 12 grams Product Unshielded. Time
selector switch at 0.1 sec. per scan.

Body Area	No. of Alarms/No. of Walk-throughs
HEAD	10/10
CHEST	10/10
HIP	10/10
ARM-SHOULDER	10/10
POCKET	10/10
KNEE (OUTSIDE)	10/10
ANKLE (INSIDE)	10/10
ANKLE (OUTSIDE)	10/10
BACK (MIDDLE)	10/10

While the SNM standards do not require sensitivity measurements below 10 g of U-235, a 12 g source shielded with 3 mm brass, produce alarms approximately 50% of the time. A 5 g U-235 source encapsulated in an unknown thickness stainless steel would alarm approximately 80% of the time when placed in a shirt pocket or on the top of the head. Alarm conditions were not established for sources located on other parts of the body.

TABLE II

0.125 Ci 1 cm^2 (point source) Cs-137 Portal Alarms with 10 walk-throughs per body area tested. Time selector switch on 0.1 sec./scan.

Body Area	No. of Alarms/No. of Walk-throughs
HEAD	10/10
CHEST	10/10
HIP	10/10
ARM-SHOULDER	10/10
POCKET	10/10
KNEE (OUTSIDE)	10/10
ANKLE (INSIDE)	10/10
ANKLE (OUTSIDE)	10/10
BACK (MIDDLE)	10/10

(The same results were obtained using the 0.125 Ci/100 cm^2 Cs-137 source.)

Question:

Have you had any problems with your monitor:

Answer: (Cliff F. Nichols)

This particular prototype is located, and generally these particular kind of monitors are so sensitive that you have to have them in a clean area quote CLEAN, and our report with this is our guard house, our gate house, and it should be clean before you get out there in the first place. No, we have not had any problems with it yet. and I didn't show that it the slides but we do have a nylon screen over this because we found that buttons when people walked through were getting hung up on , so we got a nylon screen which only cuts our sensitivity by about 3% by putting a nylon screen in there, but we felt we had to do this and the particular mylar is like 8-9 milligrams--very tough--we have not broken one yet, nor have we got any contaminated.

Question (Dwayne Spear, Battelle Northwest):

Are you having any problems with your workers as far as increased sensitivity of the portal monitor and explaining this in connection with the portable instrument that you check out the people on a labor basis.

Answer:

Yes, as a matter of fact we do. I had more trouble with guards, security guards. I don't know if you've ever heard of the Kiss Principle: keep it simple and stupid because if you put a lot of buttons and stuff on there people have a tendency to push and stuff, and I had one button on there called a reset button. You can start your count all over again. We've had problems with the increased sensitivity and people going through, they'll alarm it, but they don't believe it. They keep saying that thing is broke. I hear this and hear this until I keep getting called time and time again to come out and check the guy and we find out that the guy is contaminated but the guards keep pushing the reset button and tell them to go on, that the machine's broke. I hope that answered it.

Question (Dwayne):

That's the majority of the answer. When you go out and you check the people, are you able then to find the contamination with the portable instruments, or do you just rely that the portable monitor is much more sensitive than the portable instruments?

Answer:

It is more sensitive and if you can get a pancake probe and really take a lot of time, we usually can find it, so what we have done, and I don't remember the company but I know that Ludlum makes one, and perhaps others--they're making wide area probe detectors and we've used one of those and we find them quite nicely with that.

Question (Steve Coats, Steve Coats and Smud):

How many detectors do you have?

Answer:

We have--on this particular model there are 13. Now you notice on the slide there are only two on the foot section and I have caught people jumping over both of these and so we're going to put three in the foot section. Really, when a person goes home you'd think they'd want to be free of contamination, but it'd shock you the number of people who try to run through it or jumping over the detectors to get get out.

Question (Coats):

Did you buy these commercially or did you build them yourself?

Answer:

The detectors were bought commercially.

Question:

What is the mechanism for detecting when a person's in it - photocells or something?

Answer:

It uses one of these door opener radar beams called motion detectors. We've found that these make less noise to the electronic system and we've never had a malfunction with those which we've had on electric eyes--you know the sun goes through the windows and you stop updating background, so we use a small radar door opener motion detector.

Question:

You mentioned that you used a source to check out the sensitivity of the .1 microcuries?

Answer:

.125 microcuries, yes.

Question:

That's about 100,000 DPM. Have you tried anything a little less than that? Calibrated source, or something? Have you a little better handle then?

Answer:

Yes. What we did--yes, I can get down to 2-300 counts per minute, with no problems, but when you start doing this I can't check out any of the other monitors that we have in our plants. So we had to get down to something, I could go lower.

Question:

Counts per minute--you mean DPM? or what all, O.K.

Answer:

Everything we had is in counts, if I'm talking about a wide area probe or a pancake probe.

Question:

You mean with a 30% efficiency, or...

Answer:

Yes, when I say 2-300 counts I usually refer to a pancake probe with a 10% efficiency and about a centimeter.

Question:

In one quick last question, do you have any kind of an estimate what this would cost commercially to make?

Answer:

Yes, they're expensive. The more sensitivity the more you're going to pay for it. Approximately 15K.

Question (Bill Lipton, Argon National Lab):

We had problems with our commercial units with radon washout on a rainy day. Do you have similar problems with your unit?

Answer:

No. Are you talking about background raising?

Question (Lipton):

An increase in radon levels--how people see it.

Answer:

Yes, it takes care of it. Yes, it does - we find a lot of alpha occasionally doing this. Again, about all we can do is let them ring it and we have HP come out there and actually start surveying and see what it is and take spectrums to see what kind of problems we do have. We've not had that problem for two or three years, though.

TECHNICAL SESSION J

ENVIRONMENTAL MONITORING III

Chair: Alan Moghissi

A GERMANIUM DETECTOR SYSTEM FOR THE DETECTION OF TRANSURANICS AT LOW-ACTIVITY CONCENTRATIONS IN SOIL

by

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ABSTRACT

A photon spectroscopy system is described which is designed for the detection of plutonium and ^{241}Am in soil samples with a minimum turnaround time. Quantification is based upon the 60-keV gamma emitted in ^{241}Am decay and upon the uranium L x-rays (energies from 13 to 22 keV) emitted by plutonium isotopes during alpha decay. The detector is a single-crystal, intrinsic-germanium-planar detector with a surface area of 21 cm^2. Sensitivity is increased by incorporating a detector window with a larger than normal surface area. This optimized window size was established by Monte Carlo calculations. For small, Petri-dish samples, detection limits at the 3σ level for a 4-hr counting time are better than 4 pCi/g for plutonium and better than 0.05 pCi/g for ^{241}Am. The specifications, performance, and cost of the system are discussed. (This work is being performed under the auspices of the Department of Energy.)

INTRODUCTION

A photon spectroscopy system is described which is designed primarily for the detection and quantification of plutonium and americium in soil samples. This soil assay system is being developed primarily to analyze batch soil samples in support of environmental monitoring of inactive burial grounds once used for nuclear waste disposal. The detection limits of the system are in the few pCi/g region for plutonium in soil, far below the 10 nCi/g retrievable level established for transuranic contaminated waste. A major criterion in the development is that the turnaround time for sample assay be held to a minimum, requiring that there be little or no sample preparation. Additionally, the system is to be capable of detecting uranium isotopes and the natural uranium and thorium radioactive chains. For this purpose, a detector was built with a very large surface germanium crystal and with a new window design more suited to the application of monitoring transuranics in soil.

DETECTOR

The detector (shown in Figure 1) was manufactured by Princeton Gamma-Tech (PGT)[1] and is a single-crystal planar detector of hyperpure germanium (referred to by PGT as an intrinsic-germanium-planar detector). (Liquid nitrogen cooling of a hyperpure germanium detector is not required during those periods when the detector is not in use.) The active area of the germanium is 21 cm^2, making it the largest known detector of this type ever commercially manufactured. The depletion depth of 7 mm was specified to ensure that the detector is relatively insensitive to photons with energies greater than 200 keV. This thin depth minimizes the Compton-scattering contribution to the background in the energy region of 13 to 22 keV (the region of the plutonium decay L x-rays) and yet still provides good energy resolution which degrades rapidly as the detector thickness is decreased below 7 mm. Sensitivity is increased by incorporating a 20-mil beryllium window with a larger than normal surface area of 26 cm^2. This optimized window size (diameter 58 mm) was established by Monte Carlo calculations. The large area and thin depth of the germanium place limitations on the energy resolution, but the improvement in detection limit for plutonium more than overrides the minor restrictions on resolution when compared to thicker, smaller area detectors. Resolution specifications are 924 eV full-width half-maximum (FWHM) at 122 keV of ^{57}Co, 1720 eV full-width tenth-maximum (FWTM) at 122 keV of ^{57}Co, and 814 eV FWHM at 6.4 keV of ^{57}Co. All widths are measured at count rates under 1000 c/s for the source indicated and with a shaping time constant of 6 μs on the main amplifier. The cost is approximately $20,000 for the detector, amplifier, high-voltage supply, and NIM minibin.

ASSAY

Quantification for ^{241}Am and plutonium is based upon the 60-keV gammas emitted in ^{241}Am decay and upon the uranium L x-rays (energies from 13 to 22 keV) emitted during alpha decay of plutonium. After the sample and room background count rate is subtracted from the combined total count rate in the three regions L_α (12.7 to 14.6 keV), L_β (15.8 to 18.1 keV), and L_γ (19.2 to 21.3 keV), a correction is made for the ^{241}Am contribution using a predetermined fraction (obtained via a ^{241}Am soil standard) of the count rate of the 60-keV peak. Figure 2 is a spectrum of a soil sample containing 10 nCi/g of plutonium with an isotopic distribution of 94% ^{239}Pu and 6% ^{240}Pu. The sample holder was the plastic flask described below; the counting live-time was 500 s. The largest peak in this spectrum is due to the uranium L_β x-rays emitted during plutonium decay; the second largest peak represents the 60-keV gammas from the small amount of ^{241}Am in the sample. This spectrum clearly illustrates the significance of the L x-rays (between channels 60 and 110) for assay of plutonium in soil. The peaks near the middle of the spectrum are the K x-rays and gammas emitted by plutonium and by the natural uranium and thorium radioactive series ever present in a soil sample.

The spectroscopy station is set up in a sub-basement of the LASL Health Research Laboratory inside the steel room of an inactive whole-body counter. The steel walls are 18.5 cm thick; additional copper and lead shielding surrounds the detector endcap. Detector, amplifier, high-voltage supply, and NIM minibin are all located inside the steel room. The multichannel analyzer and computer are nearby.

SAMPLE CONTAINERS

In a 1-g soil sample with a ^{239}Pu activity concentration of 1 pCi/g, only 133 plutonium nuclei will decay in 1 hr. With the low intensity of the ^{239}Pu L x-rays,[2] only 6.3 L x-rays will be emitted in 1 hr in such a sample. Sample size and counting time, therefore, are very important.

An L x-ray from plutonium has a mean free path of about 1 mm in soil. An increase in sample thickness beyond 5 mm will have negligible effect upon the number of L x-rays emerging from the sample due to matrix absorption and will have an adverse effect upon the background. If the sample thickness is much less than 4 mm, overall sensitivity is decreased due to loss of valuable counts. The only alternative is to increase the areas of the sample and the detector or to reduce chemically the soil matrix and to concentrate the nuclides of interest.

In the system described here, sample preparation is held to a minimum, with the soil being counted in commercially available, disposable, plastic containers. Two different types of containers have been examined. One is a rectangular, tissue-culture flask with a volume of 250 ml and a thickness of 34 mm. This flask will hold about 400 g of soil and has a screw-on cap, making it relatively easy to fill and seal. The large, flat, plastic sides are essentially transparent to the photons of interest. The second sample holder is a small, plastic, Petri dish with a diameter of 60 mm and a depth of 15 mm. After filling the Petri dish with soil, the two halves are cemented together with a mastic glue to seal the container. To reduce the sample thickness further from 15 mm to about 7 mm, the Petri dish is sometimes padded on the side away from the detector with paper filters to fill the interior volume partially. The flask, with its larger effective sample size, yields a better detection limit for ^{241}Am, while the thinner Petri dish, with a much lower sample background count rate in the L x-ray region, has a better detection limit for plutonium. An ideal sample holder for the detection and quantification of plutonium for this detector would be a disposable, rigid container with at least one thin plastic side, with a diameter of approximately 75 mm and a depth of 5 mm, and which could be easily sealed. We are now looking in the commercial sector for such a container.

DETECTION LIMITS

The detection limit is defined here to be that level which will give a signal that is 3σ above the background, where σ is the statistical uncertainty as determined by the expression

$$\sigma = \sqrt{R_B \cdot T}.$$

The background count rate, R_B, is the rate for the energy region of interest for a blank soil sample; T is the counting time. The 3σ level assures that, if an individual plutonium measurement is at this level, one is more than 99% confident that plutonium is in the sample and that the measurement is not just a background statistical fluctuation. The detection limit, D, is expressed as:

$$D = 3\sigma/(S \cdot T),$$

where S is the signal sensitivity in c/s per nCi/g for a particular nuclide for the same energy region as the background. For a sample in a Petri dish, the detector has $R_B = 4.5 \times 10^{-2}$ c/s and S = 1.52 c/s per nCi/g for the L x-ray region of plutonium. Table 1 contains the detection limits for plutonium and ^{241}Am for counting times of 4 hr.

This detector has a natural intrinsic background in our Laboratory in the L x-ray region of about 2.4×10^{-2} c/s which, in signal strength, is equivalent to about 16 pCi/g of plutonium in a Petri-dish sample. This background, arising from materials used in construction of the detector and shield, represents a lower limit on the background that can be achieved.

In the L x-ray region, a typical blank soil sample in a Petri dish has a background count rate of about 2.1×10^{-2} c/s above the detector background rate. This signal is equivalent to about 14 pCi/g of plutonium in a Petri-dish sample. We attribute most of this natural sample background to L x-rays from the decay of ^{238}U and ^{232}Th and their daughters. The presence of the K x-rays and certain gammas from the uranium/thorium chains in our spectra indicates the presence of at least part of these chains. This extraneous background in the L x-ray region interferes with plutonium detection in the same manner that ^{241}Am does. With the range of uranium and thorium that appears possible for soils in even a limited geographical area, failure to account for the natural radiation in each sample individually can lead to inaccurate plutonium assays at levels below 50 pCi/g of plutonium. We are now looking into means of performing this correction on a sample-by-sample basis using uranium and thorium daughter photons above 40 keV in energy.

WINDOW SPECIFICATION

To determine the optimum sample configuration for a planar detector, an investigation was made of the effect the sample radius has upon overall efficiency of the system. The geometry is defined schematically in Figure 3. The pancake-shaped sample is placed as close as possible to the beryllium window of the detector. For a sample which is much larger in area than the germanium, the window defines the effective radius of the sample. Most planar detectors are made with a window area equal to the active area of the germanium. A window that is smaller than the detector crystal certainly will reduce efficiency by collimating the incident photon flux, but what quantitative effect does a larger window area have on absolute sensitivity of the system? Figure 4 illustrates the dependence of count rate upon window area for 17-keV photons, emitted in a soil sample, into a 19-cm^2 planar detector. The line in Figure 4 is the result of a Monte Carlo calculation with a sample radioactivity concentration [illegible] photon/s per cm^3. The soil sample is 5 mm thick with a radius greater than the beryllium window. The distance between sample and germanium is 6 mm for this calculation. The figure indicates that, if the window area is about 1.2 times the active area, the count rate will increase from 0.48 c/s to 0.53 c/s, an increase of about 10% in efficiency. This requires that the window radius be about 10% larger than the crystal radius and that the crystal mount not act as a collimator. The detector was built to these window specifications by Princeton Gamma-Tech, with the active area as large as could be achieved practically at this time. The 20% increase in window area has the same effect as if a standard detector with 10% greater active area had been utilized. For large planar detectors, it is considerably less expensive to increase the window area by 20% than to purchase a detector with 10% greater active area. Making the window area larger than the detector active area can be cost-effective for both the buyer and supplier of germanium planar detectors.

IMPROVEMENT OF DETECTION LIMIT

The effects of the soil matrix on analysis for plutonium are great, with the x-rays emitted during plutonium decay being severely attenuated and with the matrix contributing to an increased background. By placing emphasis on the development of a straightforward and rapid (less than 24 hr) chemical procedure, it is possible that a quick-chemistry technique may be found which is quite adequate for photon spectroscopy and yet which does not require the sophisticated, time-consuming, sample preparation needed for alpha spectroscopy. We recognize that high-fired plutonium oxides probably cannot be properly accounted for by such a technique, yet with assay uncertainties of approximately 50% near the detection limit, this neglect may be relatively insignificant. Chemical reduction of the soil matrix will have two major operational effects: (1) the desirable signal will be enhanced, and (2) the undesirable background will be reduced. The net effect on plutonium assays will be both to improve the detection limit and to improve the accuracy of the assay. The detection limit may improve by a factor of 5 to 10 for an absolute detection limit for plutonium of around 0.5 pCi/g for a 4-hr count at the 3σ level.

Since the detector described here has an active area that is as large as is practical now for a single-crystal detector, any significant increase in detector area will have to be achieved by using multicrystal arrays. An array may even provide lower background per unit area via anticoincidence gating between the crystals. The cost of an array will increase roughly proportional to the total area, while the detection limit should improve roughly proportional to the square root of the total active area.

Limited improvement in the detection limit for plutonium may be found by using a sample container of optimum size with a very thin film cover and by incorporating Compton suppression techniques to reduce the background. These aspects are being investigated.

SUMMARY

A photon spectroscopy system for soil assay has been described which has detection limits (3σ level, 4-hr counting time, Petri dish sample) of better than 4 pCi/g for plutonium and 0.05 pCi/g for ^{241}Am. These detection limits are achieved by utilizing a very large (21 cm^2) hyperpure germanium planar detector with an even larger (26 cm^2) beryllium window.

ACKNOWLEDGMENTS

We would like to thank Richard J. Olson of the LASL Monte Carlo, Vulnerability, and Weapons Data Group (TD-6) for performing the Monte Carlo calculations. We would also like to thank the LASL Environmental Studies Group (H-8) for their assistance in preparing the soil standards and especially for their financial support of this developmental program.

REFERENCES

1. Princeton Gamma-Tech, Post Office Box 641, Princeton, New Jersey 08540; USA.

2. K. L. Swinth, "Photon Intensities and Their Importance in Counting Transuranic Materials," Battelle Northwest Laboratories report BNWL-1648 (1972).

TABLE 1. Detection Limits for Plutonium and ^{241}Am

Nuclide	Photons	Conditions*	Detection Limits (pCi/g)
Pu**	L x-rays	Petri dish sample < 25 pCi/g ^{241}Am	3.6
Pu**	129 keV gammas	Flask sample < 25 pCi/g ^{241}Am	185
^{241}Am	60 keV gammas	Petri dish sample	0.045
^{241}Am	60 keV gammas	Flask sample	0.025

*All detection limits at the 3σ level with 4-hr counting times. All samples contained approximately 5 ppm natural uranium.

**Isotopic content of 94% ^{239}Pu and 6% ^{240}Pu.

FIGURE LEGENDS

Fig. 1. The largest intrinsic-germanium-planar detector ever built commercially, with an active area of 21 cm^2 and a window area of 26 cm^2.

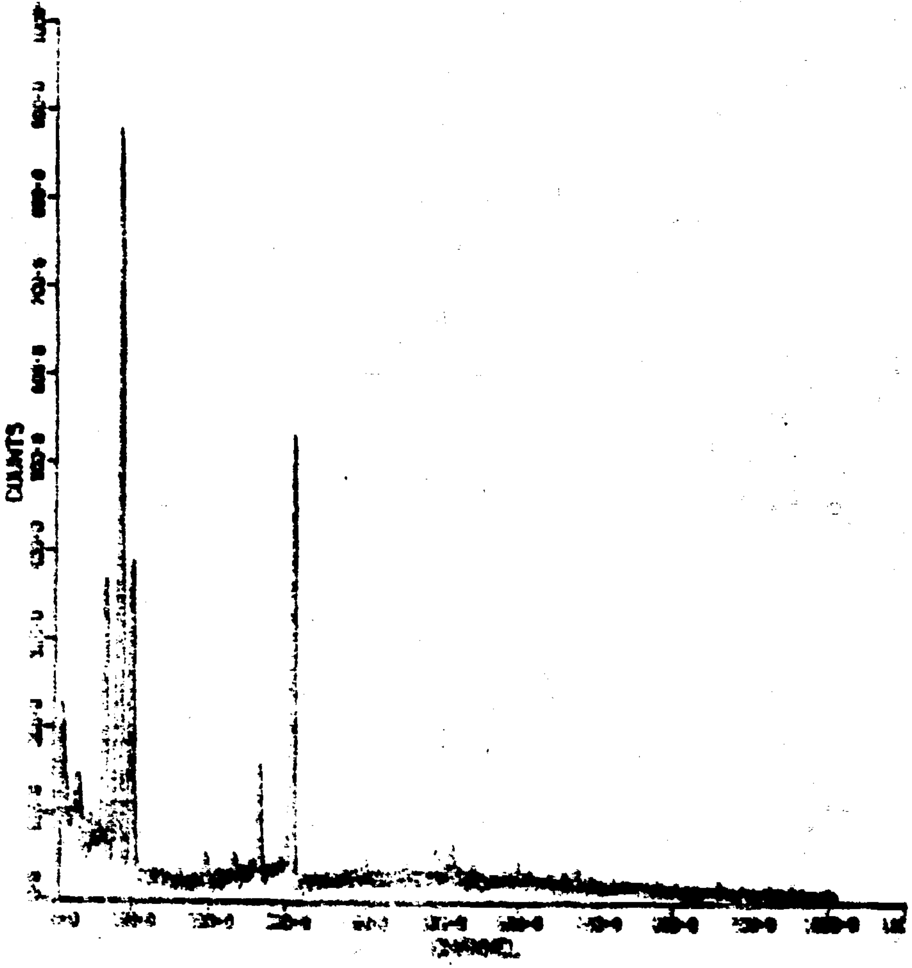
Fig. 2. A photon spectrum of a soil sample containing 10 nCi/g of plutonium. The energy calibration is approximately 0.189 keV/ch, with channel zero corresponding to 1.4 keV.

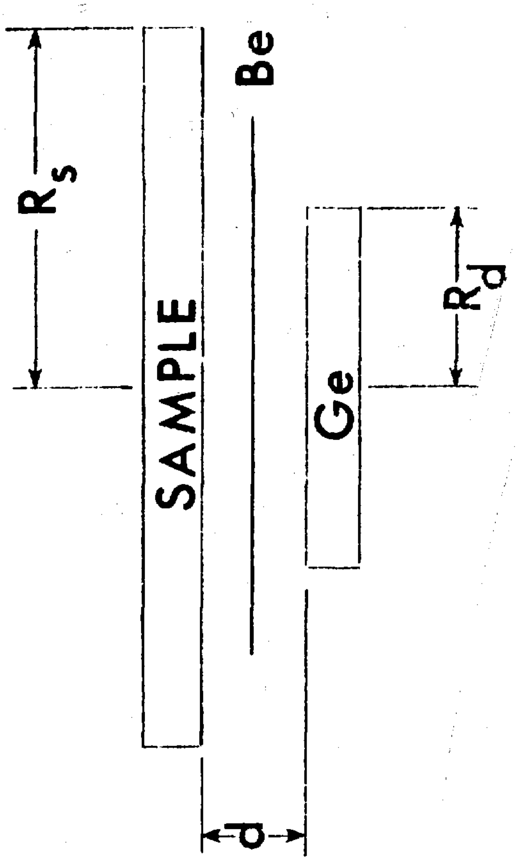
Fig. 3. The relative geometry of the sample, germanium, and beryllium window.

Fig. 4. A Monte Carlo calculation illustrating the dependence of the count rate upon window area for 17-keV photons into a 19-cm^2 planar detector. The soil sample has an activity concentration of 1 photon/s per cm^3. The bars represent the statistical uncertainty of the calculation.


PGT
PRINCETON GAMMA-TEC
INTRINS
GERMAN

SOIL, 10 NCI/G OF PU, 500 S, JAN-78




R_s
SAMPLE
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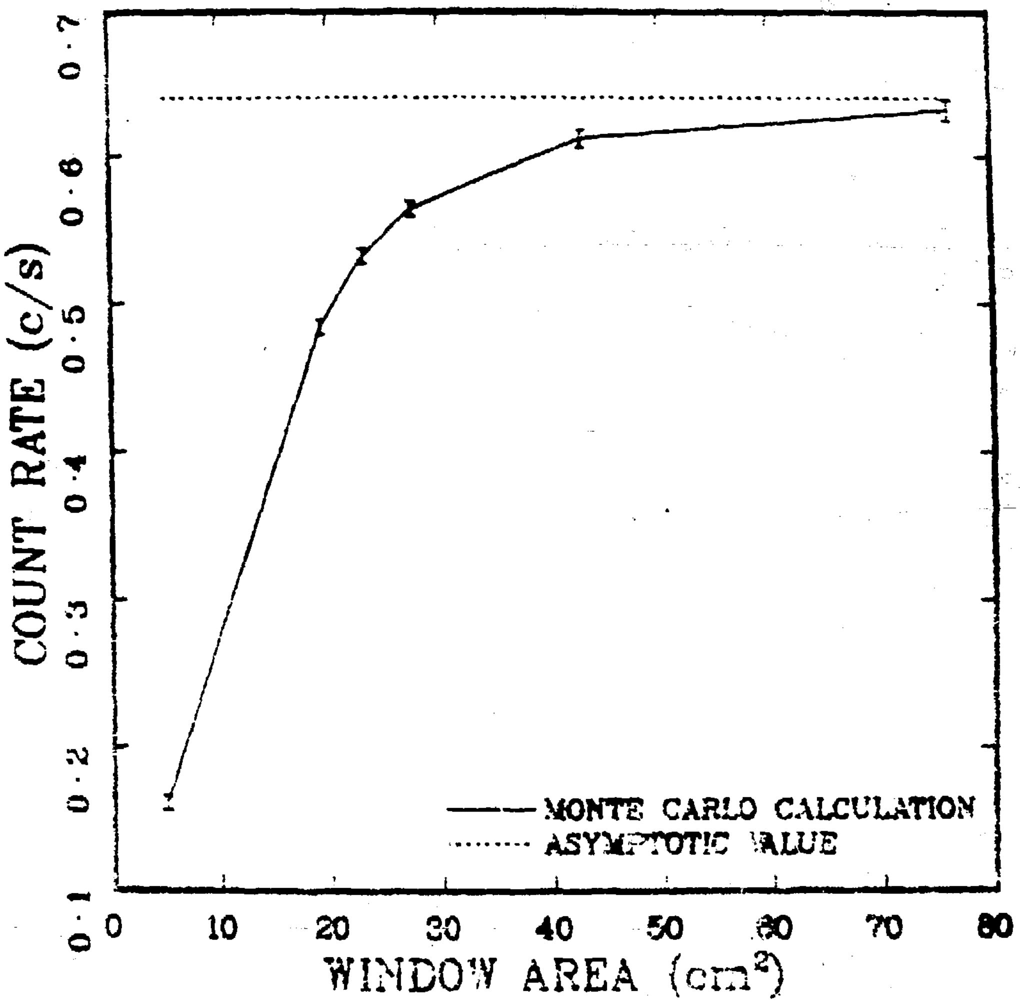
17 keV PHOTONS IN 19 cm^2 DETECTOR



Question (Jim Eldridge, Oakridge National Laboratory):

My question concerns sample preparation. I didn't hear your opening remarks and I'm not sure if you addressed that subject. I have a real concern about the ability to measure these low energy L xrays of plutonium and practical samples. Do you dry the soil to a constant of weight? What is the calibration standard, matrix?

Answer (Leon West):

Those are good points, Jim. We try to hold the sample preparation to a minimum. They are dried at some point. It's not necessary to dry them significantly. What's in there is water. Water's got hydrogen and oxygen in it, that does not attenuate the L xrays nearly as much as silicon. So the primary problem is due to the soil matrix, not the small amount of moisture in the soil. The calibration is accomplished by using soil standards that were prepared by the environmental monitoring group at LASL. They are spiked with known amounts of plutonium, then chemical assays were done on that to verify the activity concentrations.

Question (Jim Eldridge):

Do you know what the actual soil matrix was, was it dunite, for example?

Answer:

The soil was from the Los Alamos Reservation.

Question (Jim Eldridge):

The reason I bring up some of these subjects is that I at least in Eastern Tennessee, we have a much different situation that you have at Los Alamos where you cross the road and find a different soil type.

Answer:

Our major problem is you can cross the road and find a different concentration of uranium and thorium, you can dig a hole and the deeper you go, the more it varies. Anything--you can go near the river and uranium and thorium will vary there. Or activity concentrations in the order of 50 picocuries per gram for plutonium. The uranium and thorium becomes a limiting factor in the accuracy of the assay and also the detection limit. So one needs to be able to account for the L xrays that are emitted by uranium, thorium, and their daughters. And that has to be done on a sample by sample basis.

Question (Jim Eldridge):

That's the reason I raised the question. I feel those of us that are analytical radiochemists need to stress the fact that there are practical limitations, and when we make a presentation like this don't just dwell on the good parts of the thing, but indicate that there are things one has to worry about in the sample presentations.

Answer:

Oh, I agree. This system as a whole is not really designed for long counting times. A 24-hour counting time you can begin to do wet chemistry which requires the order of a couple of days. So in general we would prefer to think that the system is being used for counting times of one hour to four hours, in particular the fourhours is the time limit that we would prefer to use.

CONTAMINATION MONITORING

– PROBLEMS AND SOLUTIONS –

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Paper to be presented at the 11th Midyear Topical Symposium
of the Health Physics Society,
San Diego/Cal.,
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The Case for Contamination Monitoring

Once radioactive substances have entered the human body, they will cause a long-term radiation burden. Even if the presence of radio-nuclides in the body has been detected, it is most difficult to assess the exact dose comittment they are delivering, and it is almost impossible to reduce this dose later on by physical or medical endeavours. Therefore, it becomes imperative wherever open radioactive sources of any kind are handled to prevent the intake of radio-nuclides by the workers themselves or by other members of the population.

To this means, many radiation protection measures are needed and have to be strictly observed. The most prominent amongst them are rigorous cleanliness, use of protective clothing, use of correctly labelled containers, controlled collection of waste materials, and bookkeeping on locations and transfers of radioactive solutions and sources.

Wether, however, such good operating procedures are always observed and effective, and if therefore truly no risk for internal exposure exists, may only be ensured by maintaining a suitable and regular schedule for contamination monitoring. In doing so, three different aims will have to be accomplished:

1. the search for, and the location of, unforessen radioactive contaminations on surfaces, objects and persons **inside the working area** (operational monitoring)
2. the control of objects and persons **leaving the working area** (exit monitoring)
3. the determination of the amount of contamination found in relation to permissible limits, and the verification of effectiveness of decontamination measures.

Consequently, in every radioisotope laboratory one or several monitoring instruments will be indispendable. They have to be not only capable of surveying various objects and plane and rough surfaces as well as hands and clothing of persons. They should also allow for fairly quantitative results in terms of radioactivity per unit area. Such requirements clearly exclude wipe tests as being by far too crude and unreproducible.

Regulations, rules and permissible levels

The safety measures and precautions to be taken when handling open radioactive sources are governed in general by rules and recommendations from international organisations like IAEA (1), WHO (2), and ICRP (3). In particular, more or less stringent national regulations on radiation protection are existing in most countries.

In deriving permissible limits of contamination, however, several assumptions have to be made regarding the radio-nuclides, the type of surface and the physical form of the contamination. Moreover, no quantitative data are available for relating skin contamination levels to the rate of incorporation. Consequently, most permissible limits for contamination to be found in national recommendations or regulations are a) very rough and general values, and will b) differ widely from one country to another. More specified data and considerations are given in (1) and (3).

There is, however, an increasing tendency to harmonize and, together with this reevaluation of data, to restrict the levels of contamination.

A very good example is the new German Radiation Protection Act, issued Oct. 13, 1976, and effective since April 1, 1977 (4).

Section 64 of the current bill not only stipulates compulsory checks **in the controlled area** generally and **on persons leaving** the controlled area in particular, but also for the first time lays down limits for contamination on surfaces and objects which, when exceeded, make decontamination compulsory (Table 1).

Furthermore, the bill calls for "immediate measures in order to avert any danger of spreading or incorporation" if **any skin contamination** is found.

All this has had, in Germany, a rather dramatic effect on the design and use of contamination monitors. On the one hand, special counter tubes had to be developed in order to reach the required ultra-high sensitivity even for such "strange" radio-nuclides like ^{125}I, combined with an effective area large enough to enable hand or shoe checks with sufficient uniformity of response. On the other hand, there was suddenly a need for suitable Exit Monitors which could make the rather tedious procedure of controlling outgoing persons sufficiently reliable and speedy.

Table 1

Surfacs contamination limits
according to the German Radiation Protection Act of October 13, 1976

The values stated are meant to be averaged over an area of 100 cm^2.	**Environment, objects and protective clothing in controlled areas**	**Environment, objects, clothing and laundry in restricted areas**	**Objects, clothing, laundry in general areas**
alpha emitters	**3,7 Bq/cm^2** **(10^{-4} µCi/cm^2)**	**0,37 Bq/cm^2** **(10^{-5} µCi/cm^2)**	**0,037 Bq/cm^2** **(10^{-6} µCi/cm^2)**
All nuclides other than alpha emitters	**37 Bq/cm^2** **(10^{-3} µCi/cm^2)**	**3,7 Bq/cm^2** **(10^{-4} µCi/cm^2)**	**0,37 Bq/cm^2** **(10^{-5} µCi/cm^2)**

The new Xenon-filled detector probe BZ 200 XK–P

What counts on a contamination monitor is – literally speaking – the detector probe. Quite a variety of probes are in use today for contamination monitoring purposes. If the requirements are just surface monitoring with a detection sensitivity of not more than 10^{-3} μCi/cm^2, even the good old end window GM counter tube will do the job. To assess, however, a multitude of radioisotopes including ^{125}I or ^{14}C down to 10^{-5} μCi/cm^2, and to be usuable as hand-clothing monitors, beta-sensitive detectors with window areas of 100 – 200 cm^2 are needed.

For this purpose, large-area proportional detectors have been employed since years in Germany, and recently also in other countries (5, 6). They would be ideal, exept for the necessity of the counting gas supply. To make a sealed counter tube, these was no suitable window foil material available up to now that would be absolutely vacuum-tight, as well as sturdy enough to make windows of several 100 cm^2 area, at the required density for low-energy beta measurements, of less than 4 to 5 mg/cm^2. First attempts had been made with aluminium foils. They proved successful with densities of more than 7 – 10 mg/cm^2 (7, 8). However, a breakthrough was reached by applying Titanium foils which could be used with densities as low as 4 mg/cm^2, thus giving a high enough yield even for ^{14}C.

On this basis, Laboratorium Prof. Dr. Berthold developed the proportional detector probe BZ 200 XK–P (9).

Having a window area of 200 cm^2, those counter tubes are designed in view of checking optimally surfaces, objects and, mainly, the fingers and the palm. The external dimensions are given in Fig. 1.

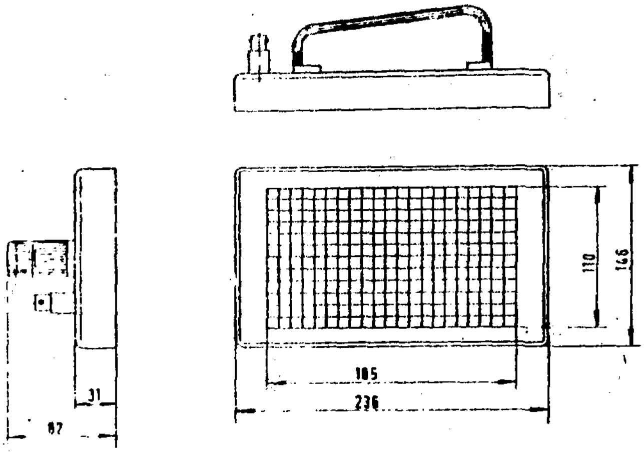


Fig. 1

Dimensions of Xenon-filled proportional counter probe BZ 200 XK-P
The window consists of a Titanium foil of approximately 4 mg/cm^2.
Operating voltage is 1900 Volts.

The probe has a Xenon filling at about 780 KPa, which enhances greatly the counting yield for low-energy quantum radiation. Thus, it will be able to detect 0.5 x 10^{-5} μCi/cm^2 of an extended plane source of ^{125}I. This means an improvement by a factor of nearly 20 compared to a methane-operated proportional counter tube of the same dimensions, and even by a factor of 8 compared to a small scintillation probe as commonly used today in some RIA laboratories.

The count rates and detection limits for various groups of radioisotopes, as compared to other types of detectors, are given in Table 2.

The lifetime of the probe – apart from forced damaging of the window – will be by far more than 2 years. It is, in contrast to GM counter tubes, practically not influenced by the number of counts.

Table 2

Sensitivity of the new Xenon filled proportional detector probe compared with some other detector systems commonly used

		Xenon probe window area 200 cm^2 4 mg/cm^2		**Methane probe** window area 200 cm^2 0.4 mg/cm^2		**Scintillation probe** window area 4.5 cm^2 (23 mm ϕ)		**GM end window probe** window area 6 cm^2 (27 mm ϕ) 2.5 mg/cm^2	
background rate in cpm (approximate values)		**700**		**450**		**100**		**30**	
a) count rate **b) detection limit**		**a) cpm per 10^{-5} µCi/cm^2**	**b) fraction of 10^{-5} µCi/cm^2**	**a) cpm per 10^{-5} µCi/cm^2**	**b) fraction of 10^{-5} µCi/cm^2**	**a) cpm per 10^{-5} µCi/cm^2**	**b) fraction of 10^{-5} µCi/cm^2**	**a) cpm per 10^{-5} µCi/cm^2**	**b) fraction of 10^{-5} µCi/cm^2**
pure γ/X-ray emitters	^{75}Se	240	0.5	40	3	30	1.6	1	30
	^{125}I	230	0.6	12	9	12	4	0.4	70
mixed β-γ emitters	^{60}Co	380	0.4	1050	0.1	5	10	9	3
	^{131}I	520	0.3	1100	0.1	14	4	9	3
	^{137}Cs	610	0.3	1100	0.1	7	7	9	3
pure β emitters	^{14}C	260	0.5	1000	0.1	–	–	6	5
	^{32}P	850	0.2	1080	0.1	–	–	9	3

The count rate values given apply to an extended plane source of 10^{-5} µCi/cm^2. Detection limits are calculated for 3 times standard statistical error of background count rate and 20 sec measuring time or 10 sec ratemeter time constant, respectively. Rounded-off values only are given.

Types of monitors fitted with the new probes

Table 3 shows the different types of monitors that have been developed together with the new probe

As an operational monitor, a portable ratemeter is used that has the detector built into the bottom side of the instrument, so that it can be operated single-handed. In this case, the probe has an effective window area of 100 cm² because, according to the German regulations, the measurement of surface contamination has to be averaged over exactly that area. The ratemeter has two semi logarithmic ranges, an alarm threshold with a acoustic warning signal and an audible single pulse indication. Two luminous diodes are signalling the measuring range choosen, and a oncoming end of battery life.

The same ratemeter unit may serve as a simple exit monitor when inserted into a wall mounting containing the mains connection unit, and when fitted with an external probe of 200 cm² area. The probe itself is easily detachable for checking of clothes.

However, such a multi purpose monitoring system will be sufficient only for truly small laboratories. Installations with more than one or two staff members need anyhow both types of monitors, a portable one for controlling the working areas, and a permanently installed exit monitor.

To facilitate hand checks, a two-channel model is available for separate measurements of both hands at the same time, the probes being equally detachable. Such a device seems to be well suited for the usual size of RIA laboratories.

Larger installations handling activities in the mCi range, or being connected to larger nuclear medical departments, are required to set up a complete hand-shoe-clothing monitor at every exit. For this application, a three-channel monitor is proposed as shown in Table 3.

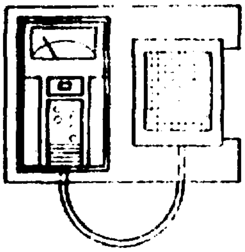
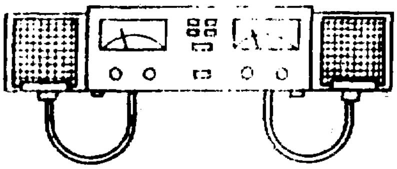
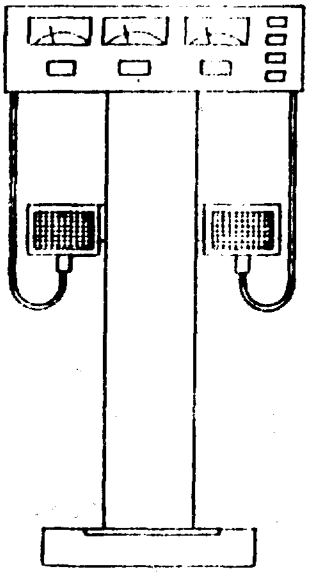
The separation of the indication into three channels was choosen out of two reasons first, it enables a quick location of the contamination perceived, in respect to left hand – right hand – shoes, and second it maintains the required high detection sensitivity by separating the background count rates of the different probes. Furthermore each channel has it's own presettable alarm threshold as well as a failure indication.

Hand probes again are detachable for checking of clothes, and the single pulse clicks are switched on automatically when doing so. Single pulses are also displayed in each separate channel by luminous diodes, thus providing an easy and instantaneous check of the correct functioning of the monitor, as well as means for recognizing immediately rapid changes of pulse rates when searching for clothes contamination.

In order to give the results with sufficient statistical reliability, a certain measuring time is required. The monitor watches the observance of this time which is normally set to be ten seconds.

Where high energy gamma emitters are used besides the ^{125}I, as will be the case in nuclear medical departments, the monitor will need a possibility for compensation against varying ambient gamma fields. Such compensation is usually accomplished by one or several "shield" gamma counter tubes whose pulse rates where subtracted from the output of the measuring detectors. This has the disadvantage that contaminating radio-nuclides with a gamma radiation component are also "compensated away", at least partially so. The new Berthold monitor therefore uses an electronic compensation where background count rate of the measuring detector itself is stored periodically and subtracted on the subsequent contamination measurement cycle. A similar circuitry has been described by Hogan (6).

Table 3. Types of contamination monitors fitted with the new Xenon-filled detector probe

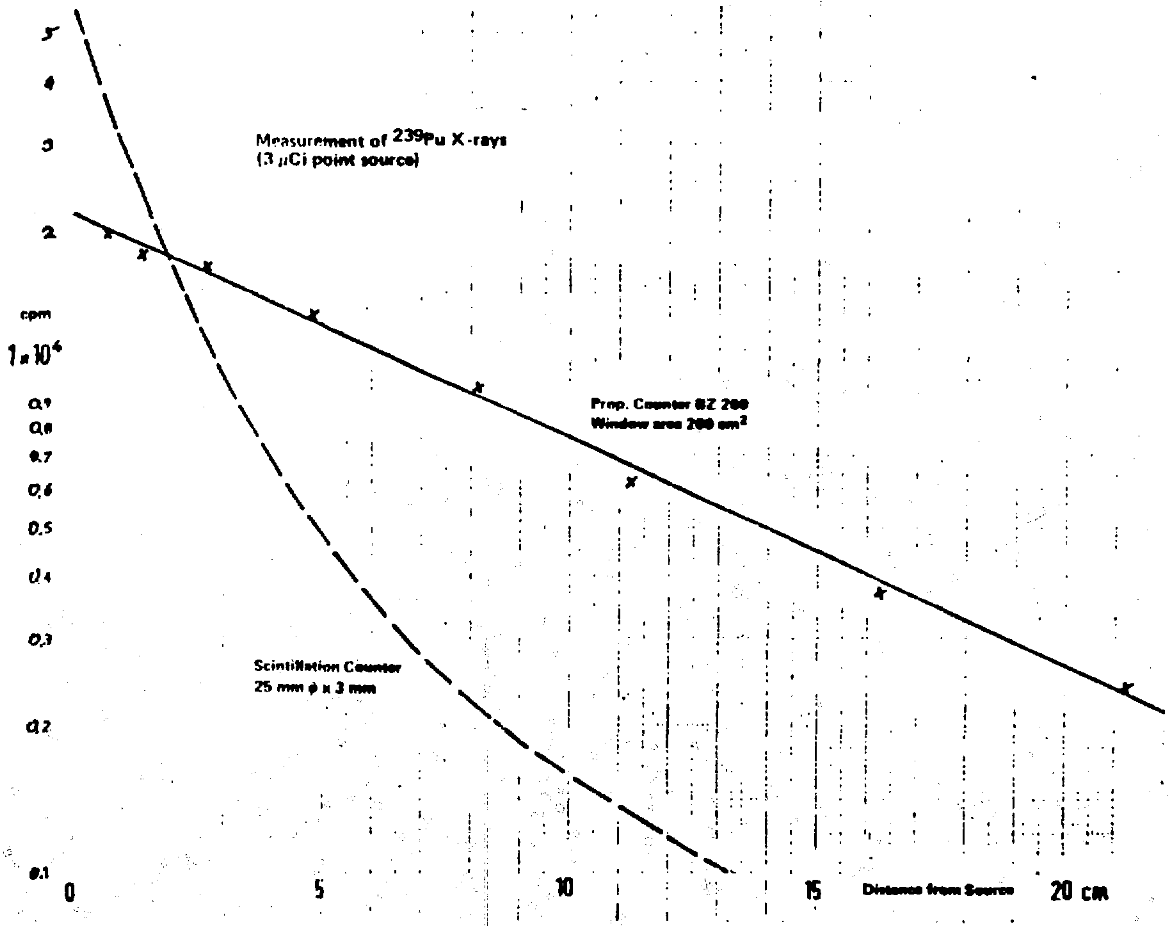
 **LB 1210 B**	**Operational Monitor LB 1210 B** Portable Ratemeter with Xenon filled probe of 100 cm^2 area, built into the bottom face of the ratemeter
 **LB 1210/11 B**	**Hand-Clothing Exit Monitor LB 1210/11 B** The monitor consists of the operational monitor LB 1210 B, together with an external probe of 200 cm^2 area and a wall mounting device containing also a mains connection unit It may be taken off and carried around for measurements in the laboratory.
 **LB 1038 B**	**Hand-Clothing Exit Monitor LB 1038 B** Two-channel monitor for simultaneous but separate checks of both hands. Two probes of 200 cm^2 area, detachable for monitoring of clothes.
 **LB 1041 B**	**Hand-Shoe-Clothing Exit Monitor LB 1041 B** Three-channel monitor, with background compensation, for simultaneous but separate checks of left hand, right hand hand, shoes.

Assessment of Alpha Contaminations

For assessing alpha contaminations either the aforementioned large-area proportional counters, or scintillation probes are used. Proportional counters have the advantage of being capable to measure both alpha and beta radiation with optimum yield. To cope with the gas supply problem, a detector has been developed that operates on liquid fuel supply, – the same principle as used with cigarette lighters, and also the same liquid gas bottles. One "shot" of fuel will last for one days run. Thus, the same operational monitor as shown in Table 3 can be fitted also with an alpha-beta-detector of 100 cm^2 effective area, capable of detecting 2 x 10^{-5} μCi of an alpha-beta-emitting point source, or 2 x 10^{-7} μCi/cm^2 of an extended source.

However, when monitoring rough surfaces as clothing, or even streets or soil in the open, one will be better off by assessing the X-ray component accompaning most alpha emitters. In this case, the Xenon-filled proportional counter again is very well suited to do the job because of it's high yield for low-energy quantum radiation. Figure 2 shows the performance of such a counter tube when measuring ^{239}Pu, compared with a scintillation counter. At higher distances, the proportional counter is superior due to the better geometric factor.

Fig. 2



Assessment of Tritium Contamination

The usual probes including the new Xenon detector, are not able to assess a contamination by ^{3}H. The thickness of their windows will not allow the week energy betas of max. 18 keV to penetrate The maximum range of tritium radiation is 0 6 mg/cm^2 in plastic. It is therefore just possible to use detectors with ultra-thin windows, but this reduces the yield to fractions of a percent The yield also varies extremely with the distance between the detector window and the surface to be monitored.

The only alternative available is given by windowless proportional counter tubes (10) Even if the effective window area cannot be increased beyond approx. 100 mm^2, and the counting gas consumption is rather high, such detectors are in practice the only reasonable means for ^{3}H monitoring directly on surfaces.

The detection limit that can be obtained depends largely on the virtually inevitable absorption of tritium betas in the contaminated material. With smooth surfaces and windowless measurements theoretical detection limits of approx. 3 x 10 ^{5} μCi/cm^2 can be reached.

A ^{3}H contamination probe of this type is manufactored, too, by Laboratorium Prof. Dr. Berthold, featuring a system of interchangeable diaphragms which are held in position magnetically. This makes it possible to adapt the detector simply and quickly to the surface to be checked (11). The probe is available as an optional extra to the above-mentioned portable monitor and can be used alternatively with the standard Xenon probe. In this case, the electronics part is mounted together with the now necessary counting gas bottle on a small trolley.

Such a system seems to be the most suitable solution for operational contamination monitoring in RIA laboratories handling both ^{125}I and ^{3}H the same time.

References

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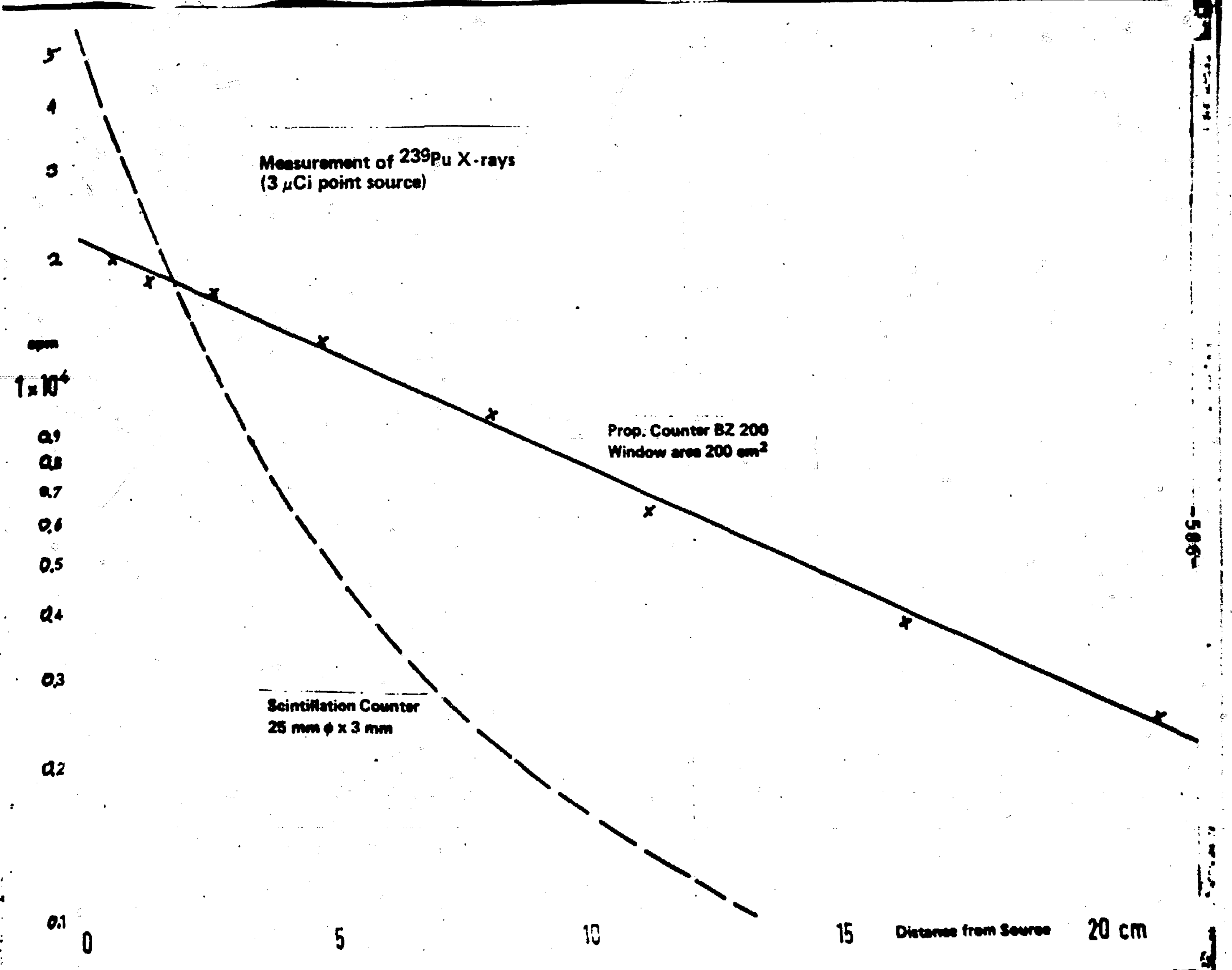

Measurement of ^{239}Pu X-rays
(3 µCi point source)
cpm
5
4
3
2
1 x 10^4
0,9
0,8
0,7
0,6
0,5
0,4
0,3
0,2
0,1
Prop. Counter BZ 200
Window area 200 cm^2
Scintillation Counter
25 mm φ x 3 mm
0
5
10
15
Distance from Source
20 cm

Question (Busick, from SLAC):

I'm concerned about our own ANSI standard on surface contamination which was recently issued and was withdrawn. Most of our monitoring is dependant upon charged particle emission. There are a lot of things that accelerators activation products don't emit charged particles--such as ^{7}Be. I was wondering if you had tried these on accelerators typical of pure gamma emitters such as Beryllium seven?

Answer:

No, I have no exact figures for so short-lived - isotopes submitted for measurement, but at the moment we have such probes in several accelerator installations and I hope to get results you are mentioning in the next few weeks or months.

Question (Tom Corbin, Lawrence Berkeley Lab):

I'm just kind of curious about the damage to the probe, and how it will affect this operation, and also about the repair?

Answer:

I didn't quite understand your question.

Question:

If you punch through an end window tube it's gone, O.K.? Now this operates the same way, I assume, and if so, it's not a very cheap item, I've already priced it, and how much would it cost to repair, and what are turn around times for that?

Answer:

Yes, that's a drawback. You know if you have a flow through column you have damage risks. But the foil we have is a very sturdy thing, I told you it was Titanium which is a very hard metal so it's not liable to rupture. You can destroy it by sheer force , certainly, then it has to be sent to the manufacturer, you cannot repair it yourself. The price for an exchange probe will be about ½ of the price of a new one.

TECHNIQUES FOR INSTRUMENTAL ENVIRONMENTAL SAMPLE ANALYSIS AT THE SAVANNAH RIVER PLANT - N.D.JOHNSON

The Savannah River Plant (SRP) is a nuclear manufacturing complex located in the southeastern United States near Aiken, South Carolina (Fig. 1). The plant occupies an area (roughly circular in shape) of about 780 km^2 and is the major producer of special nuclear material for national defense, for government uses, and for some civilian purposes. Nuclear materials are produced by the transmutation of elements in large reactors that are moderated and cooled by heavy water. Other major operations include heavy water extraction, nuclear fuel and target fabrication, separation of nuclear products from radioactive byproducts, and waste management.

The SRP site is shown in more detail in Fig. 2. Major nuclear operations are performed at facilities located well within plant boundaries, thus providing large exclusion distances (>8 km) between these operations and the plant boundary.

Extensive monitoring was started in 1951 to provide baseline data on radiation levels before plant operations began. Environmental monitoring has been an integral part of plant operations since the start of activities at SRP. Monitoring programs have continued throughout the history of the plant, providing both reliable measurement of releases of radioactive materials at the source and concentrations in the environment.

The environmental monitoring program has expanded greatly in recent years and presently requires measurement of radioactive materials released at approximately 40 emission sources on site requiring 11,500 analyses per year. The environmental program presently samples approximately 500 locations requiring more than

60,000 analyses per year (Fig. 3). We currently perform analyses for about 35 specific nuclides.

RADIOMETRIC INSTRUMENTATION

The radiation monitoring and data handling instrumentation required for analyzing this large number of samples has necessitated extensive automation and instrument modifications for our specific needs. Equipment design allows for rapid conversion from automatic to manual counting for processing priority samples. Descriptions of the various systems will be broken down into the three primary types of radiation; alpha, beta and gamma.

o ALPHA

The majority of alpha analyses are performed by two zinc sulfide alpha detectors (Fig. 4). The detectors are mounted on two Tracerlab^{R} Automatic Flow Counters which were modified by the Savannah River Laboratory to count up to 120 samples in sequence and identify each sample as to type, location, and analysis. Data output is through a Canberra^{R} 1485 Data System Multiplexer and 1488 Teletype Scanner to the Teletype^{R} KSR33. The paper tape from the Teletype is processed by the IBM 360 computer for report generation and record storage.

The samples are mounted on 1½" planchets, placed in specially coded planchet holders (Fig. 5) and counted for alpha activity. Most of the samples are analyzed for total alpha activity, while some are chemically separated for specific alpha emitting radionuclides. Holes drilled in the base of the planchet holder are decoded into five identification numbers. The first two numbers o ALPHA CONTD.

specify the sample type, i.e. stream, river, vegetation, ---. The third number specifies the type of analysis (total alpha, uranium-plutonium) and the last two numbers identifies the sample location.

The decoder consists of a plate containing twenty microswitches (Fig. 6) which positions the sample beneath the detector and decodes the five identification numbers. The identification number is recorded on the paper tape with the detector number, total counts and counting time.

Measurement of ^{238}Pu, ^{239}Pu and other alpha activity is performed on a Northern ^{R} NS-720, 4096 channel analyzer connected to sixteen ruggidized, silicon surface barrier radiation detectors (Fig. 7). Data output is to a Teletype ^{R} KSR33 for manual integration of the peaks and calculation of alpha activity for specific radionuclides.

An interface for this system to a PDP 15/76 computer system is currently being built. Data from the parallel printer connector of the analyzer is interfaced through a DEC kit 11-F into the PDP 11 processor. Software currently being written will provide for data handling and disc storage.

o BETA

Measurement of nonvolatile beta and chemically separated cesium, strontium 89, 90 and strontium 90 is made on two proportional gas flow beta detectors (Fig. 8). These detectors, like the zinc sulfide alpha detectors, are mounted on modified Tracerlab ^{R}

o BETA CONTD.

Automatic Flow Counters. Data are output, in the same format as the alpha counters, to the Teletype ^{R} for a hard copy and punched paper tape. The paper tape is processed by the IBM-360 computer for report generation.

A Motorola ^{R} M6800 microprocessor system is being designed to feed the serial teletype output to the PDP 11 processor through a DEC compatible MDL 11 serial interface. Data will be stored on magtape for processing and report preparation.

A Beckman ^{R} Wide Beta II instrument is used to measure extremely low level beta activity.

Three Beckman ^{R} LS-233 Liquid Scintillation Counters (Fig. 9) are used to measure other specific beta activities; HTO, ^{32}P, ^{35}S, ^{91}Y, and ^{147}Pm, following laboratory preparation and chemical separations. Calculations are performed both manually and by computer. Data output is being converted from Teletype ^{R} hard copy and punched paper tape to the computer system. The parallel Teletype format will be converted to a serial format through a UART (Universal Asynchronous Receiver Transmitter). The serial information will be processed by the same M6800 microprocessor as the alpha and beta system described above. Output from the PDP 15/76 will be on magtape for data reduction and report generation on the IBM 360 computer.

o GAMMA

Gamma analyses on most environmental samples are made with a 9" x 9" sodium iodide well crystal (because of the extremely low levels of radioactivity) (Fig. 10). Routine environmental samples up to a maximum size of 3 in. diameter x 4 in. high can be counted with the large well crystal system. Detector shielding is provided by 4-1/2" thick lead filled steel cave, lined with copper sheathing. The system also has an associated automatic changer which processes up to a hundred samples. Data are recorded on magtape, and computer calculated by solving a set of simultaneous equations equal to the number of radionuclides present. Radionuclides that are common in various matrix solutions are determined from appropriate matrix solutions which have been programmed.

Two 3" x 3" sodium iodide crystals (Fig. 11) are used to measure concentrations of radioactivity in samples containing higher levels of gamma activity than in routine environmental samples. The detectors are shielded by a minimum of 4" of lead. The complex gamma spectra are resolved by computer the same as for the well crystal system.

Production effluent samples, both atmospheric and aqueous, are anlayzed by a 85 cm^3 closed end coaxial Ge(Li) detector with data reduction provided by a PDP 15/76 computer system. Provisions have been made to operate in either a Compton suppressed or unsuppressed mode (Fig. 12). Compton suppression is provided by counting in the anticoincidence mode with a NaI crystal annulus providing Compton photon detection. The annulus assembly consists of two 9" diameter by 5" thick thallium activated sodium iodide crystals mounted on 9" diameter by 4" thick pure sodium iodide light pipes.

o GAMMA CONTD.

They are optically isolated with a 3¼" diameter hole perpendicular to the axis to form the annulus. The GeLi detector is placed below the hole and raised into position within the annulus. Samples (Fig. 13) are loaded from the top and the hole is plugged with a 3" x 3" NaI(Tl) crystal. The assembly is shielded with a minimum of 4 inches of lead.

The sample is identified by location, geometry, collection and analysis date. Also supplied is the ratio of the sample volume to total effluent volume to determine release quantities. At the end of data acquisition the spectrum is analyzed for peak location and area; energies are assigned to the peaks and compared to a library of radionuclide peak energies. Once the radionuclide is identified, a table of absolute gamma abundances is referenced and combined with the detector efficiency curve to calculate total radionuclide activity. Decay corrected concentrations are used in determinations of release quantities.

When counting samples in the Compton suppression mode, the computer references a second set of gamma abundances. These abundances have been calculated based on our unique counting geometry to give correct radionuclide concentrations.

o CRITICALITY NEUTRON DOSIMETER CALCULATIONS

Because of the variety of operations at SRP involved with handling fissile materials, we have the capability in our counting facility to rapidly estimate radiation exposure to personnel in event of a criticality incident.

o CRITICALITY NEUTRON DOSIMETER CALCULATIONS CONTD.

A rapid initial estimate of the neutron exposure is made by counting blood serum with the 3" x 3" NaI (Tl) crystals. Data obtained from the ^{24}Na spectrum are entered manually in the PDP 15/76 computer and the decay corrected ^{24}Na concentration (microcuries per cubic centimeter) is calculated. Final estimates of exposure doses are based on data obtained from criticality neutron dosimeters (CND) worn by employees. *(If no CND was worn, exposure dose must be based on the SRP personnel monitoring badge, blood sodium, and any other available data.)* The purpose of the CND dosimeter is to establish: (1) the relative abundance of neutrons in the various energy sections of the spectrum, (2) the corresponding dose to the individual from this spectrum and (3) the primary direction from which the bulk of the radiation was received. This information is obtained from various components of the dosimeter (Fig. 14). The data, in conjunction with blood sodium and Thermoluminescent Dosimeter badge results, permit the most accurate determination of an individual dose. Calculations are made by computer. The elements of the criticality neutron dosimeter used for the neutron dose determination are (1) bare indium foil, (2) cadmium covered indium foil, (3) sodium fluoride, (4) copper foil and (5) sulfur. The automatic gas flow beta counters and the 3" x 3" NaI detectors are used in counting the dosimeter components. Each component is counted twice and checked by the computer for agreement to within 10%. If this agreement is not signified by the computer, the component is recounted.

o CRITICALITY NEUTRON DOSIMETER CALCULATIONS CONTD.

The same computer program which checks for 10% agreement between duplicate counts also calculates the activity of the component at the time of the first count. These activity values, component weights, and elapsed time since the incident are entered into a second program along with the blood sodium activity for final calculation of radiation exposure (Fig. 15) (including flux and dose) contributed by each neutron energy interval. By comparison of sodium activity in blood and in the sodium fluoride component of the dosimeter, the orientation of the dosimeter on the wearer at the time of exposure is also calculated.

The integration of our counting facility and PDP 15/76 computer allows a rapid determination of dose in less than four hours after an incident.

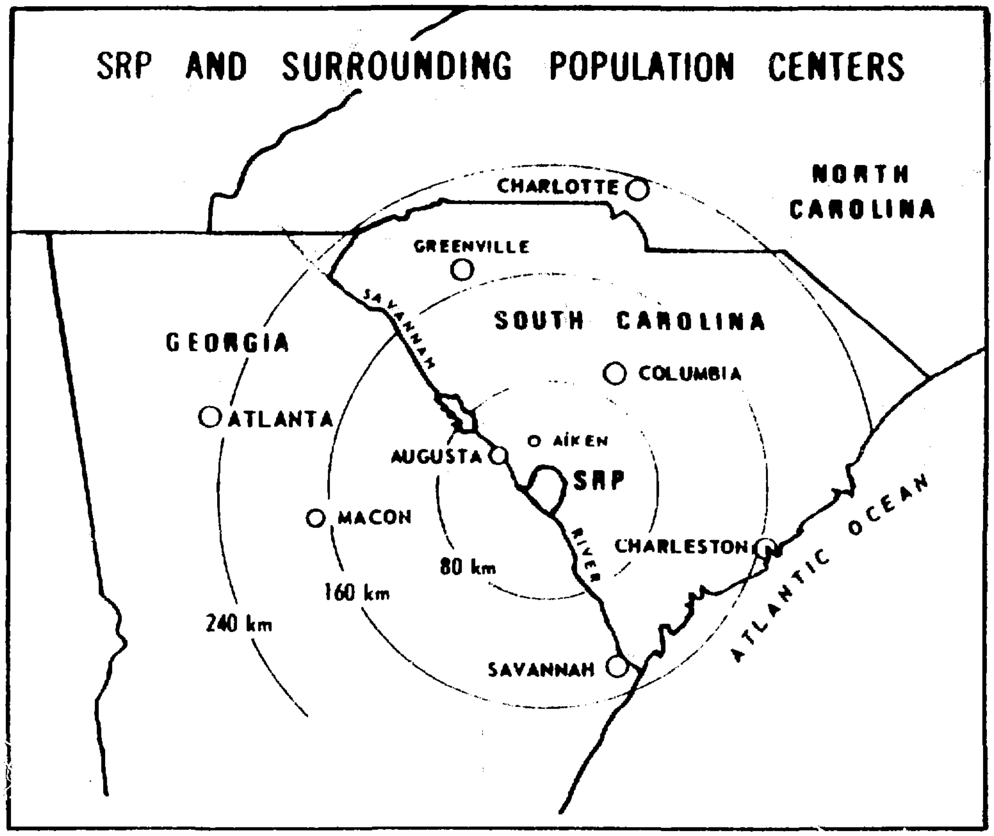

SRP AND SURROUNDING POPULATION CENTERS
NORTH CAROLINA
CHARLOTTE
GREENVILLE
SOUTH CAROLINA
GEORGIA
COLUMBIA
ATLANTA
AIKEN
AUGUSTA
SRP
MACON
SAVANNAH RIVER
CHARLESTON
80 km
160 km
240 km
SAVANNAH
ATLANTIC OCEAN

Figure 1.

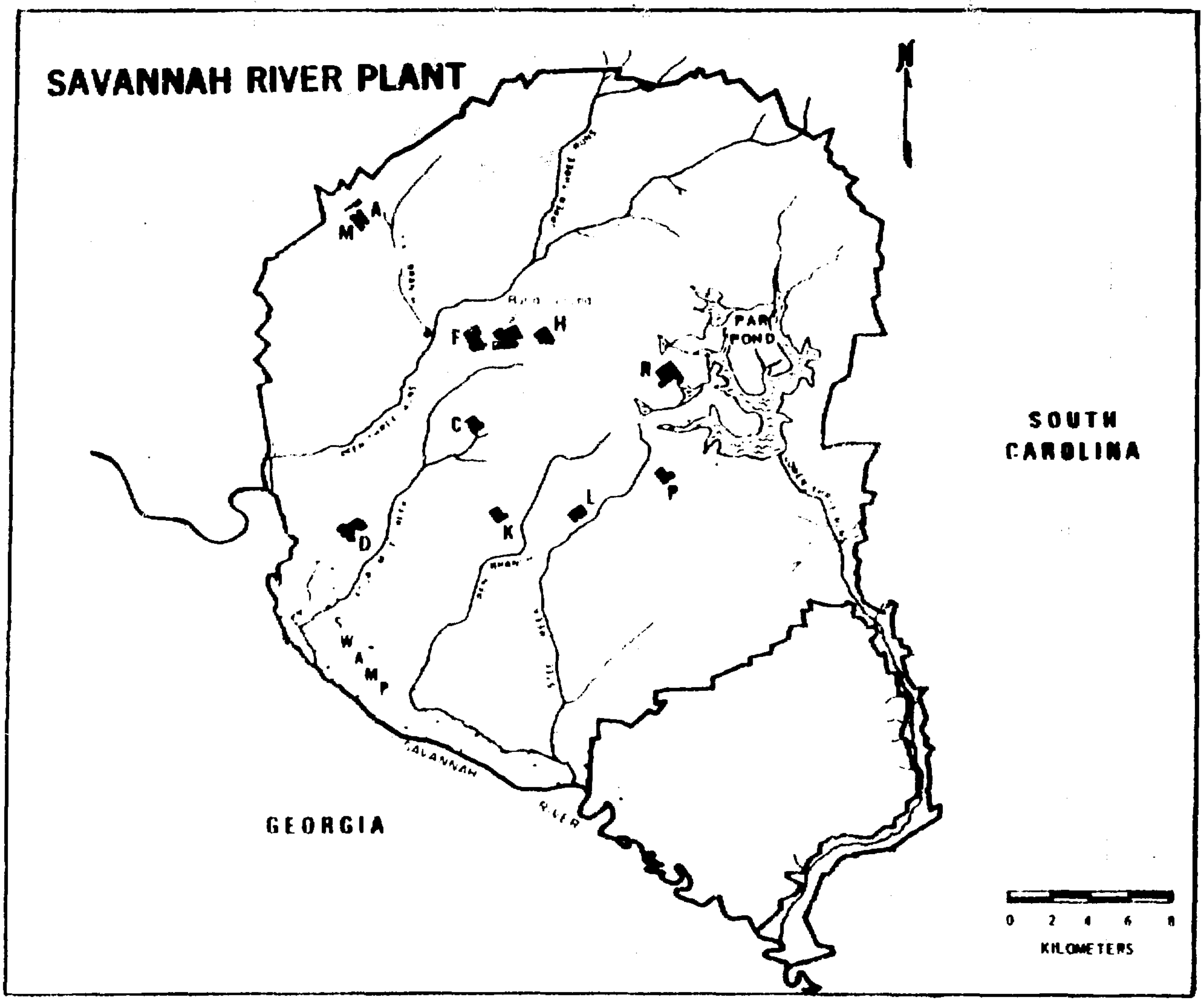

SAVANNAH RIVER PLANT
N
M
A
F
H
PAR POND
R
C
SOUTH CAROLINA
D
K
L
P
SWAMP
SAVANNAH RIVER
GEORGIA
0 2 4 6 8
KILOMETERS

Figure 2.

ANNUAL ENVIRONMENTAL MONITORING

	SAMPLES	ANALYSES
PROCESS EFFLUENT (AIR AND WATER)	2,000	11,500
WATER, WELL AND SURFACE	4,000	16,000
ATMOSPHERE	1,600	3,000
VEGETATION, FOOD, AND SOIL	900	2,000
BIOLOGICAL (TERRESTRIAL AND AQUATIC)	2,500	2,500
SPECIAL AND CONTROL	10,000	25,000
TOTAL	21,000	60,000

Figure 3.

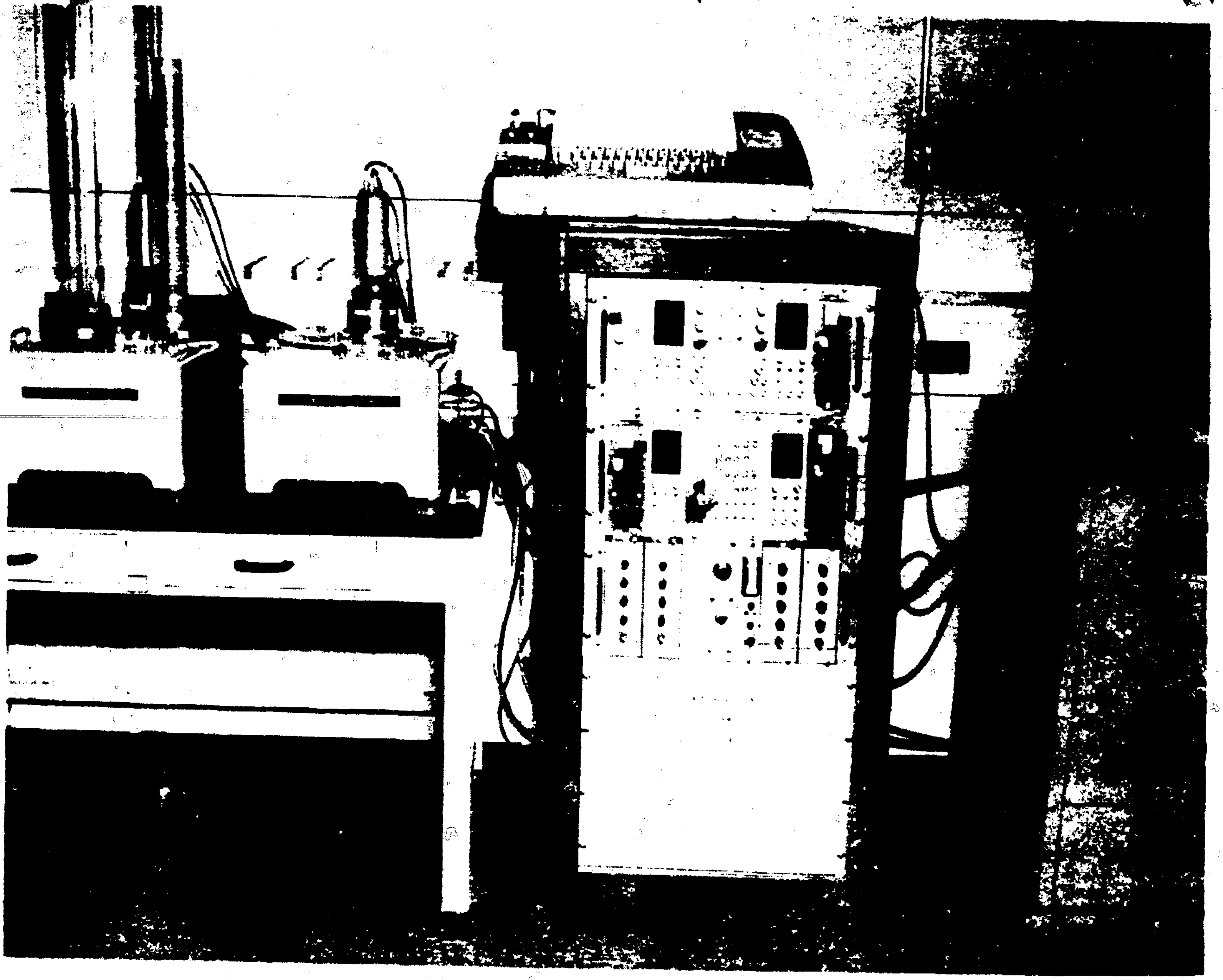


Figure 4.

Figure 3.

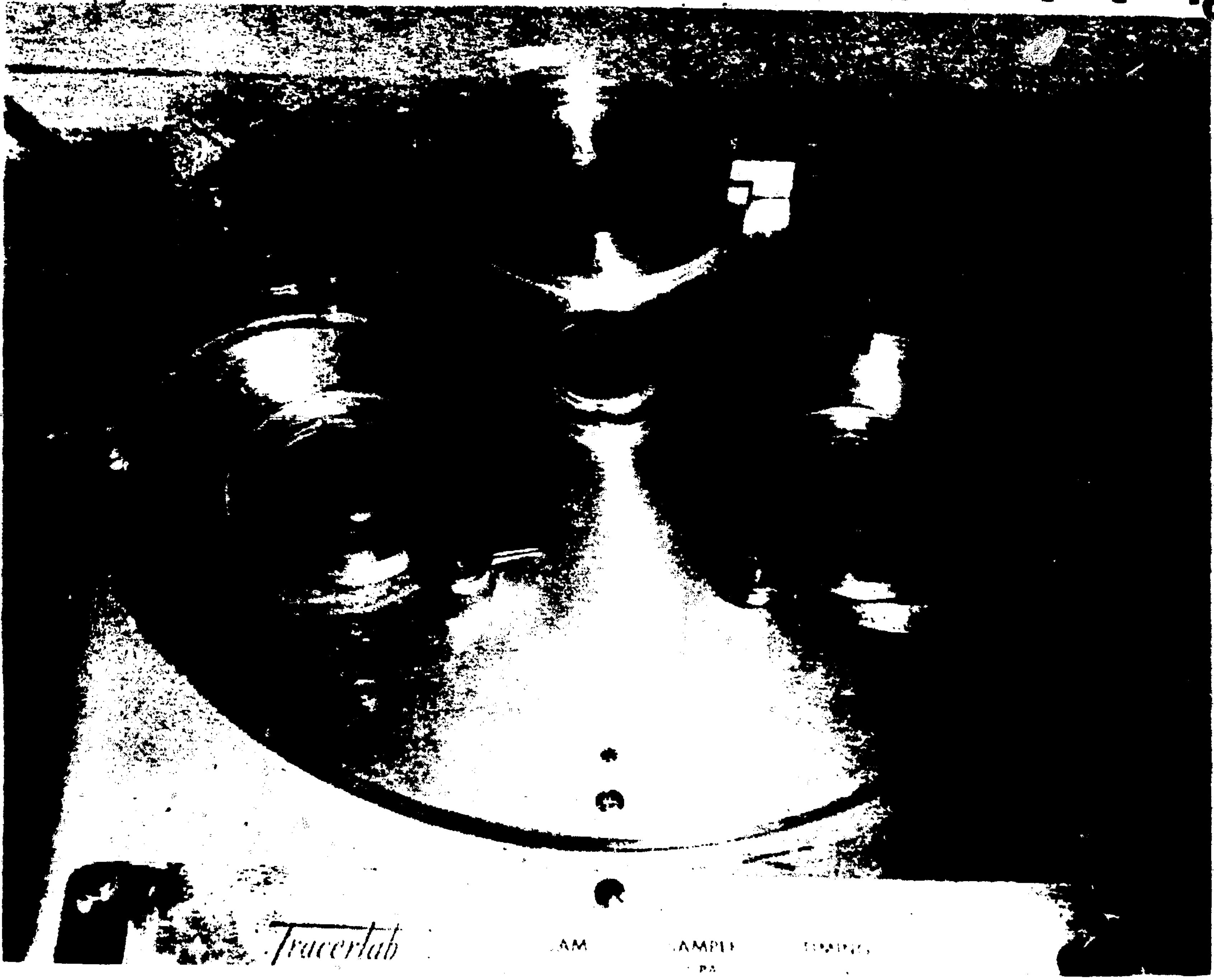


Figure 6.

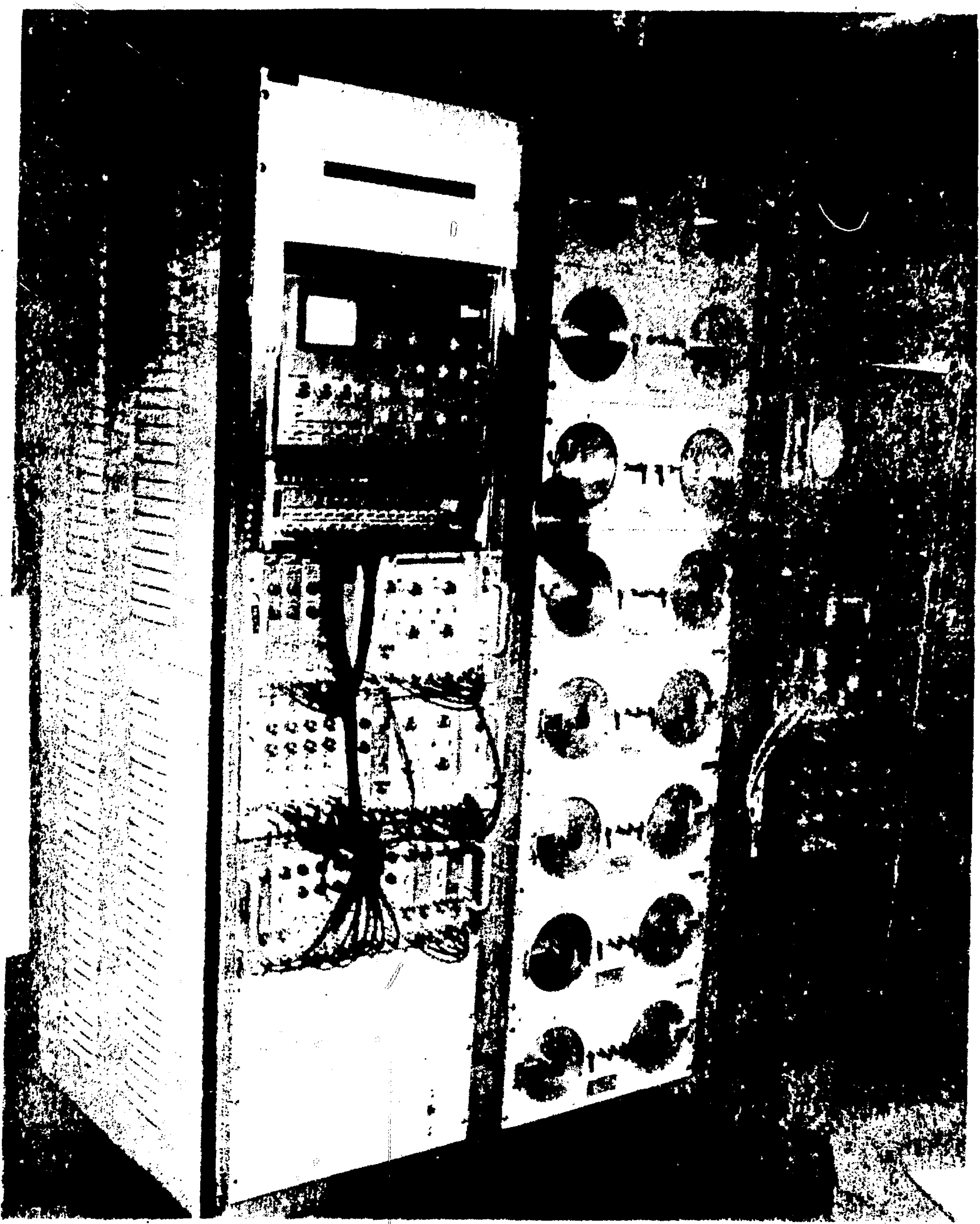


Figure 7.

Figure 3.

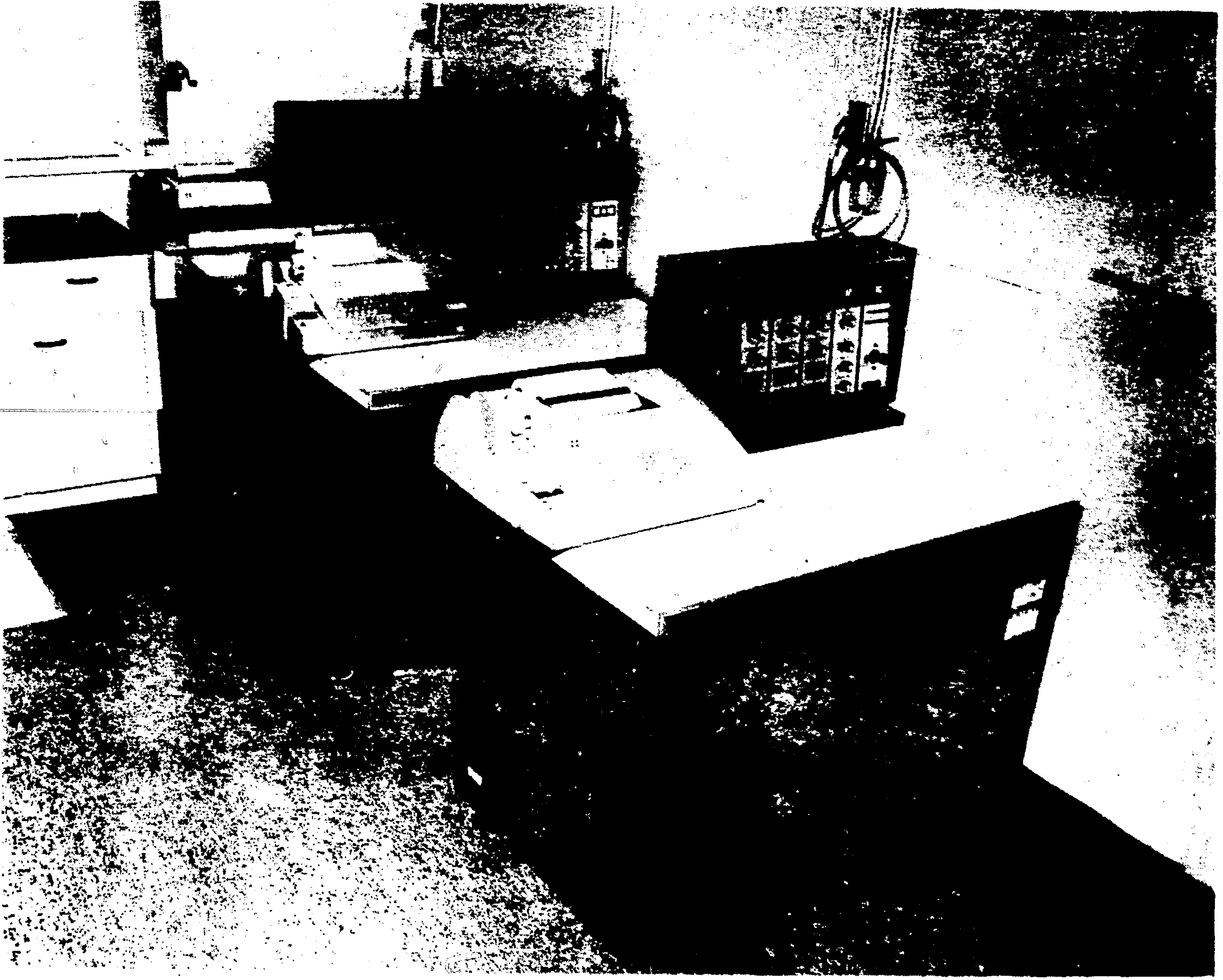


Figure 4.

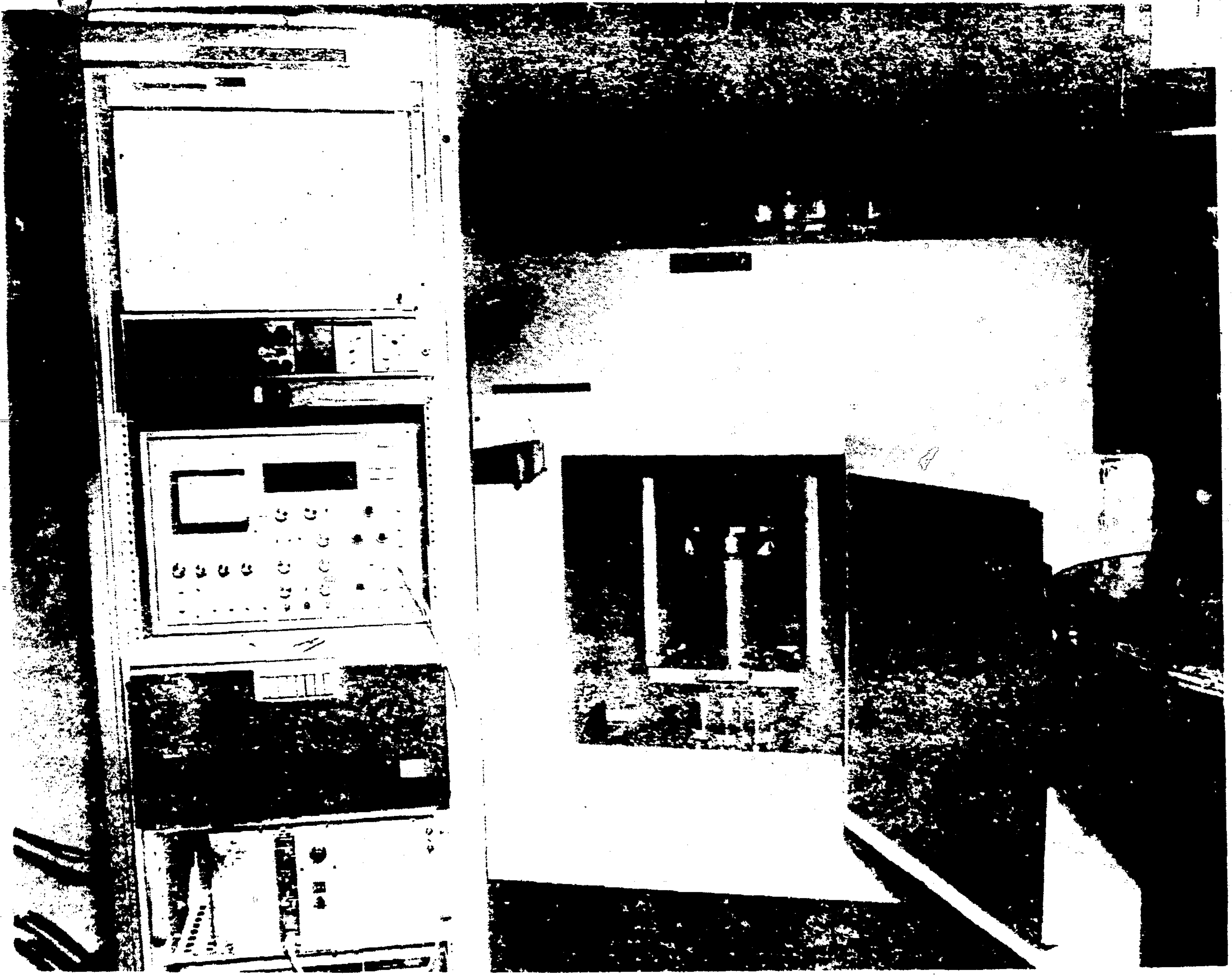


Figure 13.

Figure 11.

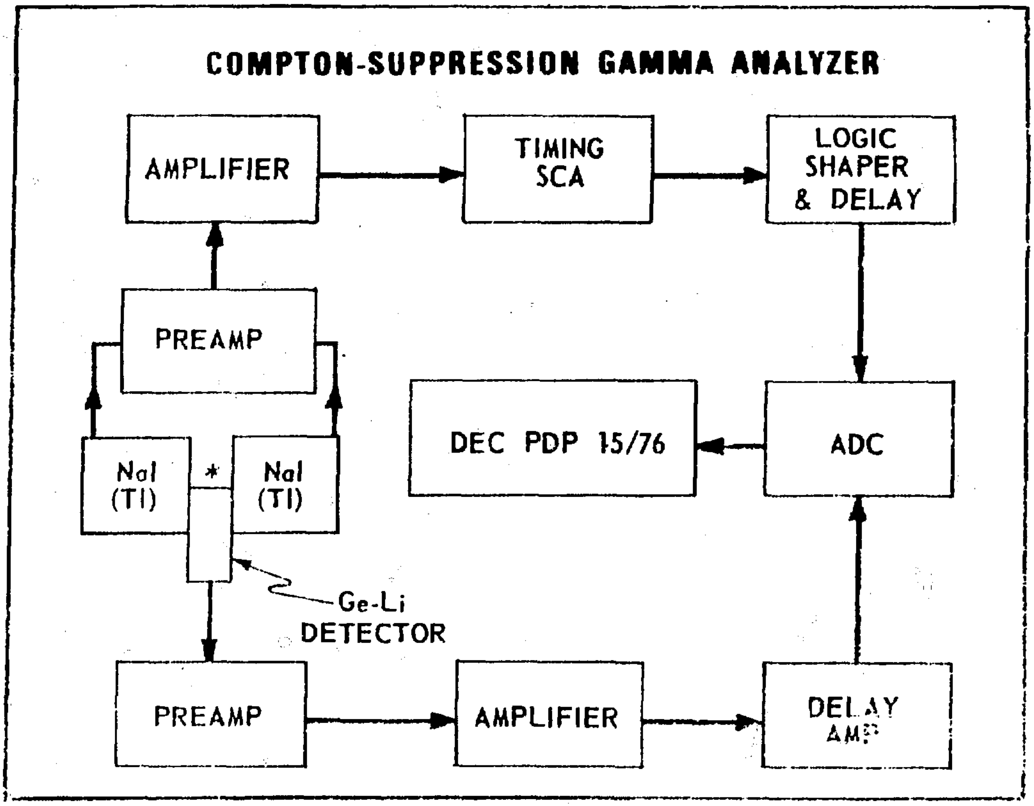
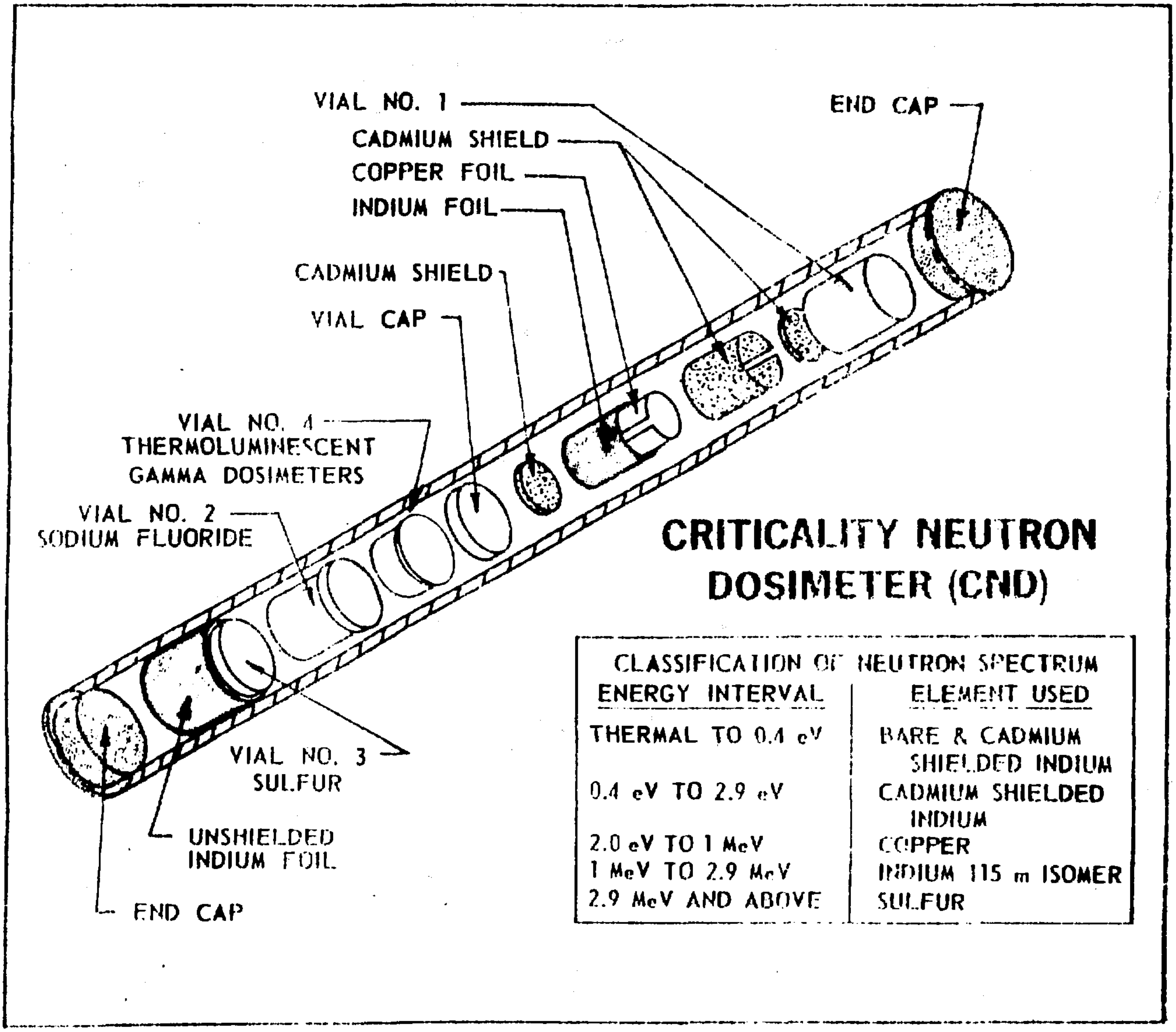

COMPTON-SUPPRESSION GAMMA ANALYZER
AMPLIFIER
TIMING SCA
LOGIC SHAPER & DELAY
PREAMP
DEC PDP 15/76
ADC
NaI (Tl)
*
NaI (Tl)
Ge-Li DETECTOR
PREAMP
AMPLIFIER
DELAY AMP

Figure 12.

Figure 13.



CRITICALITY NEUTRON DOSIMETER (CND)

CLASSIFICATION OF NEUTRON SPECTRUM

ENERGY INTERVAL	ELEMENT USED
THERMAL TO 0.4 eV	BARE & CADMIUM SHIELDED INDIUM
0.4 eV TO 2.9 eV	CADMIUM SHIELDED INDIUM
2.0 eV TO 1 MeV	COPPER
1 MeV TO 2.9 MeV	INDIUM 115 m ISOMER
2.9 MeV AND ABOVE	SULFUR

Figure 14.

FRONT EXPOSURE

ENERGY RANGE	NEUTRON FLUX	FRONT DOSE
THERMAL	0.12E + 12	0.91
EPITHERMAL	0.54E + 09	0.04
2 EV TO 1 MeV	0.13E + 12	72.59
1 MeV TO 3 MeV	0.89E + 10	20.25
ABOVE 3 MeV	0.67E + 10	23.22
TOTAL	0.27E + 12	117.01
RATIO THERMAL/FAST		5.87

Figure 15.

Question:

I was wondering if the two computers that were talking to each other had a Southern accent?

No answer.

Question (Tom Oakes, ORNL):

Could you describe briefly your quality assurance program associated with those number of samples you had?

Answer (Lewis Johnson):

For the alpha samples and the zinc sulfide detectors, we count our samples for 20 minutes each. In the tower of samples that we load, there are five background planchettes at the first of the load, a check source, samples, another check course, and five background counts at the end. The computer sums the backgrounds and divides by the number of planchettes in this case ten. Each of the counts is four 20 minutes. Our background is on the order of a tenth of a count per minute. This is done every night and amounts to about 3 hours of work for each sample. The beta counters also have five background planchettes front and rear, and also check sources. The data output from the IBM 360 shows what the background is, it also shows what the source count is and for that date, for the previous four days, the year-to-date average, the maximum and minimum. The sodium iodide equipment is source checked daily, the efficiency and fwhn are checked, the background is also checked daily, and the background stored in the computer is updated monthly with an overnight count of 1000 minutes. With the samples we spike internally approximately one in eight, of our samples to allow for individual checks. We participate in EML interlaboratory comparison. Our results have been very good. The Geli system we recalibrate yearly with NBS trace standards. All in all, approximately one in ten of our analyses involves some sort of quality assurance.

Question (Rod Melgart, Environmental Laboratories):

Three simple questions: I've tried ruggedized detectors for alpha spectrometry before and not found the resolution to be very good. Could you comment on that? Two, does your compton suppression just reduce background or does it have a positive effect on peak to total ratios? And three, how much of this is commercially available vs. how much internal engineering did you have to do on it?

Answer:

95% of it is commercially available. The interfaces that are being built right now are being done outselves, so most of it is commercially available.

Question:

If somebody went out and bought the other 95%, would it be any value to him without custom engineering, electronics, mechanical, computer programmers, and does the 5% that's custom cost as much as the 95% that's commercial?

Answer:

The manufacturer's cost is the majority of the cost. The PDP 15 76 computer system we bought the spectrum stripping program at the time that we bought system, there was very little in house done with this. We did build the lead shield ourselves. Most of our cost is manufacturer's cost, very little of it is ourself. We've done very little specification of the material, so I guess what I'm saying is somebody could walk out and buy the same thing from a manufacturer for almost what we paid for it. The ruggedized detectors, ground loop has been our major problem as far as resolution. We are gradually licking that problem by one fashion or another by installing the vacuum chambers and fiber glass panels rather than the aluminum panels, this type of situation. We write specifications when we buy the ruggedized detectors that they meet certain resolution requirements and also certain background requirements. These are also commercially available, and we will go out and spend additional that someone else could go out and spend. I'm sorry, your third question slips me right now.

Question:

That was whether or not the compton suppression did much for a prime detector or just reduces background?

Answer:

It mainly reduces background. We see decreases in some peaks due to summing which occur in the cobalt 60 and Cs-134. The way we solve this in our concentration calculation is to give it a bogus efficiency in the library that it looks up when its calculating the concentration. To do this we have to file Cs 134 standards, cobalt 60 standards, count them in the suppressed and unsuppressed mode because our system is unique and we cannot find values like that in the literature.

Question:

Anybody who still uses tracer lab gadgets must be very cost conscious.

CONCLUDING REMARKS AND CLOSING

Alan Moghissi

ADJOURNMENT

Janet Trunzo

Symposium Chair

Alan Moghissi Closing

I think all of us owe a great deal of appreciation to the Southern California Chapter of the Health Physics Society for organizing this meeting. I personally think that John Handoser and his crew, Frank Bold and his crew, and particularly Ralph Lundberg, that magician who organized the exhibits, deserve a great deal of thanks. Anybody who has organized a meeting knows how much work goes in it. The fact that we do not notice how complicated it is in fact is a sign that everyone did his homework so everything ran smoothly. So if I may ask for a round of applause for all of these good people.

I am going to ask Janet to come forward and close the meeting please.

Thank you.

Janet Trunzo, General Chair

I also would like to give thanks to all of you who have attended the meeting and helped to make it so successful. Also, I'd like to thank the authors of all the papers that were presented. They were very fine. A special thanks to our projectionist who did a fine job for the last two and one-half days.

I'd think the time constraints were well adhered to by both the presentors and session chairman.

Extra copies of the proceedings may be obtained by writing Bill Wadman whose address is in the front of the program.

At this time I would like to close the meeting. Thank you.