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UC-48

# FOURTH AEC WORKSHOP ON PERSONNEL NEUTRON DOSIMETRY

Battelle
Pacific Northwest Laboratories
Richland, Washington 99352

JUNE 14-15, 1973

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# FOURTH AEC WORKSHOP ON PERSONNEL NEUTRON DOSIMETRY

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Richland, Washington

June 14-15, 1973
Miami Beach, Florida

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FOURTH AEC WORKSHOP ON PERSONNEL NEUTRON DOSIMETRY

## FOREWORD

The first AEC Workshop on Personnel Neutron Dosimetry was held September 23-24, 1969 in Bethesda, Maryland and reported as BNWL-1340. The second Workshop was held July 8-9, 1971 in New York City and reported as BNWL-1616. A third Workshop was held March 16-17, 1972 in Savannah River Plant, Georgia, but no documented summary of the third Workshop was prepared. The fourth Workshop was held June 14-15, 1973 in Miami Beach, Florida immediately preceding the annual Health Physics Society meeting.

This report presents a summary of the information presented and discussed at the fourth Workshop. The editor has prepared the Summary of Discussions and presents it as his review of the discussions - not the papers presented. It was not reviewed by the participants. Readers of this report are urged to read the participants' summaries for specific information on personnel neutron dosimeter developments and to contact the participants directly for the best current status of any particular program. Note also that all graphic materials were reproduced directly from the copy provided to the editor and vary widely in reproduction quality.

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## SUMMARY OF DISCUSSIONS

A wide variety of discussions and commentary was presented during the two day workshop session. The informal nature of these workshops was beneficial in that the participants spoke freely of their work in progress, their successes, and their problems or difficulties. The exchange was fruitful and a continuation of these workshops on an annual to 18 month frequency is judged appropriate. It was particularly pointed out at this fourth workshop that many AEC contractors are doing related development work and through the workshop media, early information exchange and transfer of ideas is accomplished. This transfer of ideas is beneficial, both in expanding R&D relating to personnel neutron dosimetry and minimizing development costs by identifying unproductive areas of study. In the following text, summaries of the presentations are included as presented by the participants. Not all participants provided summaries for their reports and, unfortunately, the contributions from those who failed to provide summaries cannot be included.

Improved neutron dosimeters are becoming available in many countries. A thermal neutron monitoring system based on a TLD powder is available commercially in England and, surprisingly, for about $2.25 per unit. Switzerland is using a thorium foil fission fragment detector with a spark counter system for evaluation. Thorium thicknesses of 0.0005 inches are used with no shielding provided around the thorium. Some have used a uranium-235 alloy and a thorium foil in fission fragment detector type neutron dosimeters. The Karlsruhe facility is studying the wearing of neutron dosimeters on both the front and back of their workers and is planning to report these results in the IAEA neutron meeting proceedings. Germany is using a finger ring for neutron monitoring, utilizing a thorium foil and spark counting system. Some small dose to the finger results from the thorium foil during the wearing period. They feel the exposure is not significant and is received, of course, only by the few people wearing the ring. The English are also using a precision long counter with a TLD material replacing the $BF_3$ tube for neutron dose measurement.

Carl -O- Widell from Sweden reported on the use of diodes of a few mm in dimensions for monitoring fast neutrons. Some fading problems with this device are evident although heating in boiling water improved consistency of the response. The use of a 0.4 second pulse reading technique avoided diode readout heating problems. The diodes are basically used as an emergency fast neutron dosimeter with a lower detection range of about 0.6 rads if individually calibrated. Cost is about $10 per unit and three to four suppliers are available.

At the current state of development, spark counting still appears the best method for rapid evaluation of fission track detector foils. There may be some interest in using very low energy electrons instead of spark counting for detection of fission fragment tracks.

There are line scanning counters now available and probably not a lot of improvement can be made in their performance. The cost is high - running $20,000 to $70,000 per unit - and many have experienced high maintenance costs and considerable down time with these scanners.

Some are investigating the use of a laser light scattering such as biologists use on blood cell work for direct counting of fission fragment tracks.

Considerable discussion again took place on the "merits" (limits) of NTA film. Also the need to provide moisture-proof packaging of NTA film were reviewed. There was some disappointment expressed that Kodak continues to use a plastic packaging material for their film while other industry packagers use an aluminized plastic. The French and Swiss have reported no problems from humidity even under water with their aluminum foil sealing for NTA film. Sweden reported using NTA film in a non-water proof package in humidity up to 60 percent with fading factors no greater than two. Most surprising is the continuing evidence that many groups are using NTA film improperly and have failed to recognize its limitations.

The question was asked, "How many plant managers really want correct or much improved neutron dosimetry with the accompanying risk of finding higher neutron doses?" The activities at most AEC contractor facilities

would demonstrate a sincere desire by the Health Physicists at these facilities to accurately evaluate worker neutron doses. Contractors have gone to great lengths to develop new dosimeters and have even resorted to time and motion neutron dose study estimates when dosimeters were not available.

With regard to Albedo-neutron dosimeters, the energy response characteristics were again reviewed in depth. Most agreed the problem is one of calibration and some knowledge of the neutron spectrum in the work areas. The current best operating philosophy is to calibrate albedo neutron dosimeters in a neutron spectrum closely matching that to which they will be exposed. With proper handling and appropriate conditions of use, these dosimeters appear to be better than many others currently available; however, when they are used in widely varying neutron spectrums where a calibration cannot be devised to duplicate exposure conditions, their accuracy will decrease substantially.

Experimental results for completely cadmium encasing albedo TDL dosimeters were reviewed. The basic advantages appear to be that a completely encased dosimeter does not need to be in "near contact" with the body and does not have a front and back side. These advantages should be considered with the disadvantage of a loss of sensitivity by about a factor of 10.

Overall, the actual performance of each albedo dosimeter used needs to be evaluated. General rules for performance are difficult to develop. It appeared that the totally covered cadmium system tended to over-estimate the dose as the energy approached thermal while the partially covered cadmium system provided better response for the near thermal region.

Working around accelerators, the neutron to gamma exposure ratios change rapidly. The large proton component may contribute 90 percent of the total flux. For this type of facility, the accelerator experimenters may receive exposure primarily from neutrons while the maintenance crews will receive exposure primarily from gamma radiation. All-in-all, one probably needs several types of neutron dosimeters for work around different accelerators. It would appear too much to ask for a single device to cover all neutron energy ranges.

Russian data seems to indicate that NTA film provides an overevaluation of dose around high energy accelerators. The use of thick film emulsions for energies above 50 MeV should not be overlooked. Many believe that NTA film will detect protons only up to 20 or maybe 50 MeV. The NTA film will, of course, see a cascade of protons originating from a very high energy proton interaction. The use of a very thick L-4 emulsion can be of value although one needs some experience in scanning such an emulsion to appreciate its use. Fogging is not normally as serious as many believe. Even 100 mrads of gamma dose will not excessively fog an L-4 emulsion. Proper use of such thick emulsions requires study, equipment, and a knowledge of their properties the same as the proper use of any other neutron dosimeter system. For example, it will take about one week to properly process such a thick emulsion.

One can use Bonner spheres with gold foils for the detector in the spheres and get good results for neutron does as low as 1 mrem. The fission fragment method used with Bonner spheres may be good to 0.1 mrem.

Although not commonly practiced, some advocate that one should consider etching the fission fragment materials with low temperatures to avoid annealing. Even 60° centigrade is too high a temperature for processing.

Personnel neutron monitoring programs are directing considerable attention to calibration and standards. There is a large current interest in neutrons in the 1 keV to higher energy ranges. The U.S. Bureau of Standards is studying calibrations in these energy ranges but is not yet in a position to provide such services. The use of beam splitters at reactors such as the iron 25 keV neutron window and the scandium filter for 2 keV neutrons are being considered. The Bureau of Standards may provide some of these calibrations with small beam sizes late in FY-74 at the earliest. They may also investigate a silicon filter for 144 keV neutrons. Some discussion of americium-241 grow-in in Pu neutron sources serves to remind those using Pu neutron sources that neutron source emission rates need to be evaluated from time-to-time.

Growth in the number of medical facilities doing neutron work indicates that some five facilities are currently involved in this activity in the United States.

Discussion of the stability and reproducibility of TLD chips indicated there was substantial improvement in matching batches, but that differences between batches were still a problem. Early evidence indicates batches may match quite well initially but will build up substantial differences with use and repeated annealing. All should be alert to recheck batch matches at various times. An initial single check upon receipt of order is not adequate.

Questions were raised concerning the uniformity of polyethylene density. Particularly, it was mentioned that density variations between various spheres in a set of Bonner spheres should be checked carefully.

Canada reported monitoring about 100 people with neptunium-237 in personnel dosimeters. They are using about 600 micrograms per dosimeter which may give rise to a dose of about 35 mrads per year. The neptunium sources are made as thin as possible with a surface area of about three square centimeters.

In summary, lots of new work was reported. All investigators in the personnel neutron dosimetry field are encouraged to publish their finds as soon as possible. Many facets of the work in this field are not well understood and are not well documented. Definitive studies are still needed. Progress is being made - but slowly. Personnel neutron dosimeter development work is difficult, but a real need for improved dosimeters exist and continuing work leading toward a better full-neutron energy spectra dosimeter or family of dosimeters is encouraged.

# THE CALCULATED RESPONSE OF ALBEDO-NEUTRON DOSIMETERS TO NEUTRONS WITH ENERGIES $\leq$400 MeV*

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The calculated response of several albedo-neutron dosimeters that use $^{6}LiF$ as the neutron sensing element was presented for neutron energies $\leq$400 MeV. For each of the dosimeter geometries considered, results were presented for both monoenergetic and continuous incident neutron spectra and for both normally and isotropically incident neutron fluences. Data were given to indicate the effect on the response of an air gap between the dosimeter and the tissue phantom.

In general, it is found that the relative response of the dosimeters considered, i.e., the ratio of the neutron-absorption reactions in the $^{6}LiF$ to the dose equivalent in the tissue phantom, is not at all constant as a function of incident neutron energy as it would be for an ideal dosimeter.

The results of these studies are reported in ORNL-TM-3984 (Dec. 1972) which bears the same title and has the same authors as this summary.

* Research sponsored by the U. S. Atomic Energy Commission under contract with the Union Carbide Corporation.

# SUMMARY OF SOLID-STATE NEUTRON PERSONNEL DOSIMETRY RESEARCH AT ORNL*

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## 1. Photographic Film

There are still laboratories which use photographic films in personnel neutron dosimetry (either conventional emulsions for "thermal neutron dosimetry," or track emulsions for fast neutrons; with the old idea of a thick hydrogenous cover for establishing recoil proton equilibrium up to higher energies being revived at one laboratory). It will, therefore, be of interest to review some recent laboratory and field test data obtained by our own as well as by other groups concerning the latent image stability (fading) in both types of emulsions at higher humidities and temperatures.[1]

Careful sealing of the films into polymers or polymer-metal combinations delays, but does not prevent the penetration of humidity which is largely responsible for the fading. Mechanisms and kinetics of physical fading (thermal dissociation at the sublatent development center) and the more important chemical fading (catalytic $H_2O_2$ production, resulting in the oxidation of $Ag_{4-6}$ aggregates) are briefly discussed. We believe that the use of NTA films in their present form produces grossly misleading results in many, if not most locations during at least part of the year. Substantial fading errors may even occur in the conventional emulsion for X- and $\gamma$-radiation dosimetry such as Kodak PM Type 2.

Unfortunately, the search for a replacement of the NTA film in fast neutron personnel dosimetry has not yet produced a completely satisfactory result, but encouraging progress has been made in several areas.

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* Research sponsored by the U. S. Atomic Energy Commission under contract with the Union Carbide Corporation.

2. TLD

Various investigators have attempted to increase the low inherent fast neutron sensitivity of inorganic thermoluminescent (TL) materials by mixing them with liquid or solid organic recoil proton radiators, but all the systems described so far require the separation of the constituents prior to evaluation, because most hydrogen-rich organic compounds evaporate and/or disintegrate at temperatures well below those which are required for the read-out and annealing of the more stable TLD materials. Covering of TL phosphor layers with radiators results in a strong neutron energy dependence. Organic TL phosphors fade rapidly and/or are very insensitive.

In our recent studies,[2] finely powdered (<4 μm), highly sensitive and stable TL phosphors such as $CaSO_4$:Dy, $CaSO_4$:Tm or $Mg_2SiO_4$:Tb have been embedded into high-melting organics, for instance p-quaterphenyl or p-sexaphenyl (M.P. ∿450°C), for fast neutron dosimetry via recoil proton registration. If exposed to the mixed radiation field of the HPRR, such detectors exhibit a response three times higher than Teflon-embedded phosphors having the same gamma radiation sensitivity, the response difference being due to the fission neutrons. This indicates about 50 percent efficiency of the recoil protons in producing a TL signal in the phosphor. For 14 MeV neutrons from the (d,T) reaction, the response ratio is 10. The detectors are prepared by hot-pressing the constituents into re-usable pellets, followed by out-gasing in a vacuum furnace prior to first use. An increase in the visible light sensitivity was observed only in the p-sexaphenyl embedded (not in the Teflon-embedded) material.

The widely used LiF:Mg,Ti (TLD-100, 600, and 700) should not be assumed to be perfectly stable. In field tests, up to 23 percent fading has been observed during 3 months of storage in tropical climates. Other phosphors such as $CaSO_4$:Dy and $Mg_2SiO_4$:Tb are much more stable.[1] Other phosphors are being developed at ORNL.

3. TSEE

It has been reported previously[3,4] that the difference in the response of pairs of ceramic BeO discs can be used for the integrating measurement of fast neutrons between at least 0.1 and 20 MeV (the face of one of the discs is covered with a hydrogenous radiator such as polyethylene, and of the other with a non-hydrogenous low-Z material such as graphite or Teflon). In the 1972 ORNL Dosimetry Intercomparison Study, the TSEE results for neutron and gamma radiation measurements at the HPRR reactor were encouraging.[5] In further studies on this method, the directional response of simple and more complex multi-detector units has been investigated.

For a single radiator-covered detector, the response drops rapidly, as expected, once the angle of neutron incidence approaches or exceeds ∿90°. The directional response becomes less pronounced with increasing neutron energy. Another arrangement consists of placing one radiator between two BeO disks. With the two radiator-detector interfaces being read out, a much less directional dependent response is observed. In addition, information on the direction of neutron incidence can be obtained from the evaluation of such a dosimeter.[6] An even more accurate localization of the direction of neutron incidence is possible with a cubical unit containing six detectors.[5] Also, information on the average energy and energy distribution of fast neutrons up to high energies can be obtained by either measuring the neutron response as a function of radiator thickness, or by placing absorbers of different thickness between the (thick) radiator and the TSEE detector. Some implications of these results for personnel and area monitoring were discussed. Attempts to design a neutron personnel dosimeter covering a wide neutron energy range was mentioned. It is based on a combination of albedo techniques for low and intermediate neutron energies and recoil proton registration for higher neutron energies, but would require only a single dosimeter read-out.

4. Track Etching

A new spark counter has been designed for investigating the effect of ambient temperature, humidity, and air pressure on the sparking characteristics. Preliminary results were reported.

References

1. K. Becker, Long-term Stability of Film, TLD, and Other Integrating Detectors in Warm and Humid Climates, ORNL-TM, in press (1973).

2. K. Becker, T. D. Tham and F. F. Haywood, Paper, Third IRPA Congress, Washington, D. C. 1973.

3. K. Becker, Proceed, Sec. Internat. Conf. Luminescence Dosimetry (J. A. Auxier, K. Becker and E. M. Robinson, Ed.), CONF-680920, p. 205 (1968); and U. S. Patent 3.484.601 (filed 1968).

4. K. Becker and K. W. Crase, Nucl. Instr. Meth. 82, 297 (1970).

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6. K. Becker and M. Abd-el Razek, Nucl. Instr. Meth., to be published (1973).

# PERSONNEL NEUTRON DOSIMETRY AT CRNL

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## 1. Monte Carlo Calculations of Fast Neutron Spectra

Because no neutron detector measures dose accurately at all energies, it is important to know as much as possible about the neutron spectra in which personnel may be exposed. Spectra from a variety of sources, transmitted through and backscattered from various moderating materials, have been calculated.[1-4] This was done for two reasons: (a) to provide an atlas of spectra likely to be met in criticality accidents or around shielded accelerators and reactors, (b) to provide pseudo-experimental data against which simplified theories of neutron spectral variations could be checked. Monte Carlo calculations, using the ORNL 05R code, were used because they provide greater flexibility in geometry and multiple media, with fewer assumptions than other methods. Between 8,000 and 800,000 neutrons were followed, as required to give adequate statistics. These calculations are continuing.

Among the spectra calculated so far are:

1. Fission neutron sources attenuated by various thicknesses of $H_2O$, $D_2O$, "ordinary" concrete and Fe.
2. Escape spectra from $H_2O$ and $D_2O$ spheres of various radii containing uniform distributions of fission neutrons.
3. 14-MeV source neutrons attenuated by various thicknesses of $H_2O$, concrete and Fe.
4. Monoenergetic source neutrons of several energies attenuated by $H_2O$.
5. Neutrons from $H_2O$ and $D_2O$-moderated reactors, attenuated by concrete and Fe.

6. Neutrons from fission and 14-MeV sources back-scattered from $H_2O$ and concrete.

Examples of spectra of neutrons passing through concrete are shown in Figure 1.

The resulting spectra were used to calculate average cross sections for threshold reactions in Rh, In, S, Np and Th detectors, as well as average kerma and dose per $n/cm^2$. The variations of ratios of kerma (or dose) to cross section show the errors of these "dosimeters" in various spectra if no correction factor is used, or the correction factors required. Some of these ratios are shown in Figure 2, for 14-MeV neutrons passing through water. Results have also been applied to the design of collimator shielding for therapy with 14-MeV neutrons.

## 2. Containment of Np in Personal Damage Track Dosimeters

The use of $^{237}Np$ in fission-fragment, damage-track fast neutron dosimeters to be worn by people has been inhibited by the possible contamination and gamma radiation hazards. Two methods of containing Np in such a way as to ensure low contamination levels have been investigated. One is a Np-Al alloy, cast in a billet and cold-rolled to a thickness between 2 and 3 $mg/cm^2$. This is close to the optimum thickness for minimizing gamma radiation without reducing detector sensitivity to neutrons. The surface is protected by a thick evaporated layer of $MgF_2$. In the second method, a sintered mixture of $NpO_2$ and Au is sealed by heat and pressure between a silver backing and a cover of pure Au. After rolling, the sintered layer (containing 5 percent $^{237}Np$) is 2 $mg/cm^2$ thick while the Au cover is 0.5 $mg/cm^2$ thick. Both methods gave surfaces from which about 2 $dpm/cm^2$ of alpha activity could be removed by rubbing. This is several times smaller than the surface contamination from Th or natural U.

The gamma dose rate from $^{237}Np$ (and its daughter) was calculated from their spectra. For the amount of Np proposed for a dosimeter badge - 600 μg - the dose at 5 cm would be 35 mrem over a 2,000-hour working year. Whether or not this is acceptable for selected personnel depends on considerations specific to individual sites - e.g., the importance of doses received

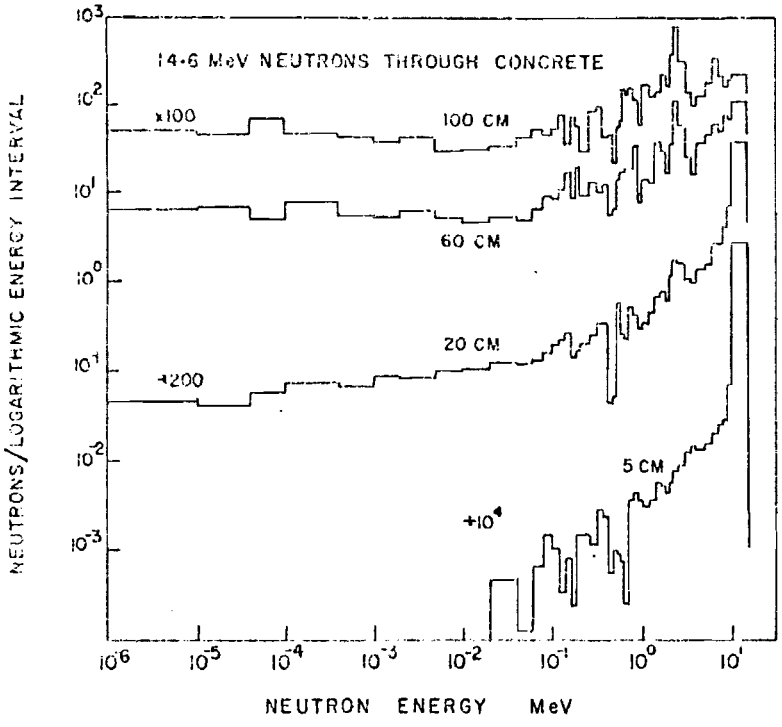


FIGURE 1. Calculated Spectra Produced When 14.6-MeV Neutrons Pass Through Various Thicknesses of Concrete

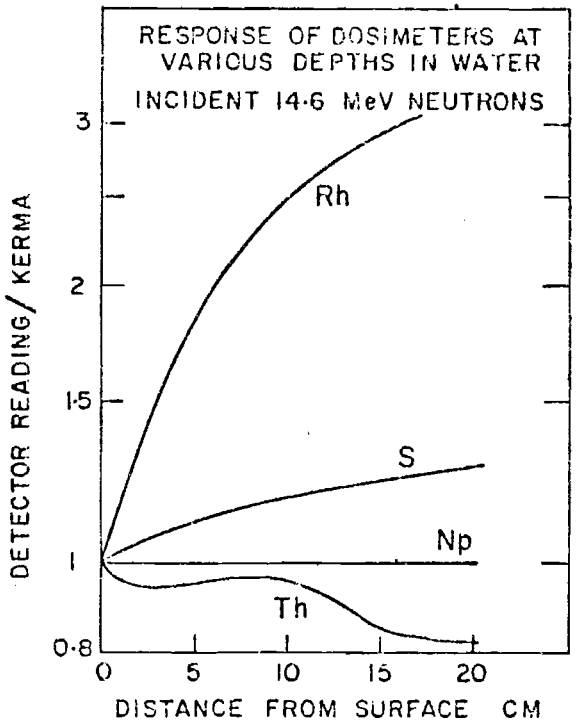


FIGURE 2. Variations of the Ratios of Dosimeter Reading to Kerma, Caused by Spectral Variations as 14.6-MeV Neutrons Pass Through Water. The detectors are calibrated to read kerma correctly for the incident neutrons.

from neutrons between 0.5 and 1.5 MeV, whether or not badges are likely to be taken home, etc.

For unmoderated fission neutrons, the sensitivity of the Au-Np radiator described is about half that of a "thick" Th radiator of the same area. This sensitivity is 30 percent smaller than would be obtained if the same amount of Np were undiluted and uncovered. It is planned to double this sensitivity by using Au-Np on both sides of the damage track foil.

Until recently, the Np available had an unacceptably high sensitivity to thermal neutrons, because of impurities. Np can now be obtained with a thermal sensitivity that is negligibly small for practical fast neutron dose measurements.

3. Use of Rh detectors for the Dosimetry of Criticality Accidents

The following papers on this topic will appear shortly:

"A Criticality Neutron Dosimeter using the $^{103}Rh(n,n')^{103m}Rh$ Reaction," H. Ing and W. G. Cross, Health Physics Vol. 25, 1973.

"Absolute Counting of K X-rays from $^{103m}Rh$ in Thick Foils," H. Ing and W. G. Cross, Int. J. of Applied Rad. and Isotopes, Vol. 24, 1973.

4. Developments in the Spark Counting Technique

The following papers on spark counting have been presented since the last workshop on neutron dosimetry:

"Improvements in the Spark Counting Technique for Damage Track Dosimeters," W. G. Cross and L. Tommasino, Proceedings of First Symposium on Neutron Dosimetry in Biology & Medicine, Neuherberg/München, May 1972, p. 283.

"Factors affecting the Accuracy of Spark Counting of Fission Fragment Damage Track Detectors," W. G. Cross and L. Tommasino, Eighth International Conf. on Nuclear Photography and Solid State Track Detectors, Bucharest, July 1972.

References

1. "Neutron Spectra in Criticality Accidents," W. G. Cross and H. Ing, Health Physics Society Annual Meeting, 1971.

2. "Modification of Fast Neutron Spectra by Thin Moderators," H. Ing and W. G. Cross, Health Physics Society Annual Meeting, 1972.

3. "Prediction of Fast Neutron Spectra in Criticality Accidents," W. G. Cross and H. Ing, IAEA Symposium on Fast Neutron Monitoring for Radiation Protection Purposes, Vienna, December 1972, paper SM 167/45.

4. "Spectral Variations and Dosimetry of 14-MeV Neutrons in Hydrogenous Media," W. G. Cross and H. Ing, Canadian Association of Physicists Annual Congress, June 1973.

# SUMMARY OF PERSONNEL NEUTRON DOSIMETRY DEVELOPMENT AT BROOKHAVEN NATIONAL LABORATORY*

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## INTRODUCTION

The Health Physics and Safety Division has developed a combination thermoluminescent-thorium damage track personnel neutron dosimeter for potential use by the staff and users of the Alternating Gradient Synchrotron.

Personnel neutron monitoring can be improved by applying damage track techniques. The motivating improvements expected are avoidance of information fading, easy readout and known energy response. The thorium detector radiator component of the damage track system allows personnel neutron monitoring at any site characterized by neutron radiation fields that extend above 1 to 2 MeV. This would include, but not be limited to, accelerator application. A recent report** describes the study that was the basis of the present thermoluminescent-thorium damage track personnel neutron dosimeter badge.

## SUMMARY

This description deals with developments subsequent to the above report. The earlier work is considered to be of laboratory or test tube level. The work herein described reflects pilot plant or semi-production scaling. Presently 30 badges are in service. Daily processing capability is about 100 badges.

Some comment will be made on the propriety of personnel neutron dosimetry with radiothorium detection elements.

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* Research carried out at Brookhaven National Laboratory under contract with the U. S. Atomic Energy Commission.

** BNL 17452, "Developmental Study of Personnel Neutron Dosimetry at the AGS."

## DOSIMETER BADGE

The Brookhaven Neutron Dosimeter (BND) is of two piece construction, a main body and a back (Figure 1). The front and main body consists of a micarta block. Four holes are machined into the rear face, three small ones for TLD's and one larger stepped hole. The latter contains the thorium fission radiator ring assembly, Kimfol detector foils, split retaining ring and pressure plug. The damage track assembly is oriented from the rear surface inward, as follows:

1) Micarta pressure plug
2) Outer glass laminate ring and Kimfol assembly
3) Brass split radiator retaining ring
4) Thorium radiator and brass ring assembly
5) Inner glass laminate ring and Kimfol assembly
6) Unmounted "piggy-back" accident exposure Kimfol

The six part assembly is held in place by a stainless steel back plate. The back plate extends along the top edge of the front micarta block, back and bottom edge. The two micarta pins that extend from the bottom edge of the main body are engaged by the bottom of the stainless steel back. A spring lock, consisting of a wrist watch spring bar, is housed in the top of the main body. A corresponding hole in the top lip of the stainless steel back is engaged by the spring lock, preventing accidental disassembly.

The TLD holes, located in the upper portion of the main body, afford about 10 $mg/cm^2$ filtering for one, and about 180 $mg/cm^2$ for the remaining two. The TLD 700 loaded into the thin window hole monitors the surface exposure. A TLD 600 and 700 pair in the remaining holes detect the penetrating gamma (TDL 700) and soft neutron exposure (TLD 600 less TLD 700).

## DETECTOR RING ASSEMBLY

Glass laminate rings and 10 micron thick Kimfol comprise the detector assembly. The glass laminate material is 1/16 inch thick printed circuit board "waste." The board is outside-inside punched, trimmed in a lathe, decoppered with $HNO_3$, and neutralized with waste KOH. Carter's rubber cement,

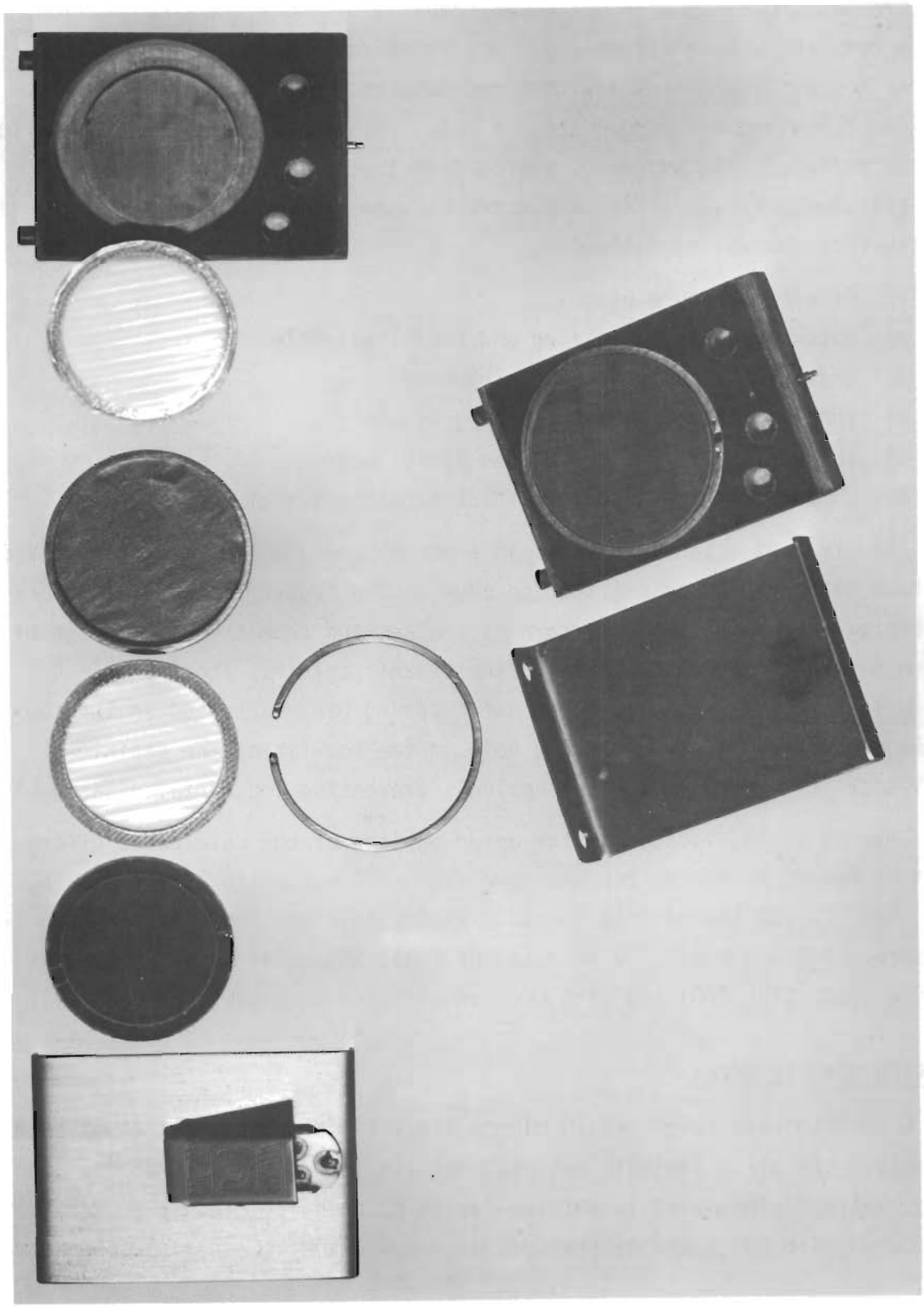

FIGURE 1

a mixture of pale crape natural rubber and naphtha is applied to the glass ring (Figure 2). The cementing tool orients the ring around the spindle. The tool top is dipped in cement and spindle guided to the ring top. The coated ring is removed and placed cement face down on a strip of Kimfol, dull side up. After a 10 minute rest, the ring can be trimmed with a knife, scissors, or hot knife (Figure 2). The hot knife requires temperature control for acceptable results.

Etching with 6N KOH for 75 minutes at 65°C has no adverse effect on the cement bond. After spark readout the Kimfol is mechanically removed and the ring cleaned and reused.

## DETECTOR RING PROCESS SEQUENCE

After wearing, the irradiated rings are removed from the badge assembly and mounted in one of ten numbered etch rack rods (Figure 2). Notation is made of the rod number and position on the rod for each mounted ring. The rings remain mounted throughout etching, washing and drying. They are dismounted individually for spark readout to preserve ring identity.

## ETCHING SYSTEM

The etch rack (Figure 3) consists of 10 removable lucite rods. The rods are slotted to accept the edge of the detector rings. A traverse nylon screw provides a simple and effective means of clamping the rings. The rods are individually inserted into the etch rack tray and keyed to radially orient the detector rings. This allows maximum loading flexibility and detector separation. Approximately 100 hours of etching have had no visible effect on the rack materials.

The detector loaded etch rack is heated in a 65°C oven for 5 minutes and then lowered into the 4 liter stainless steel etching vessel (Figure 4). The vessel is contained in a 4 gallon ultrasonic stirrer. A Lauda K4 water bath heater-circulator supplies 65 ± 0.1°C distilled water to the 4 gallon ultrasonic tank. The Lauda bath includes a small heat exchanger. Water, about 15°C, is circulated through the exchanger to offset the heat input

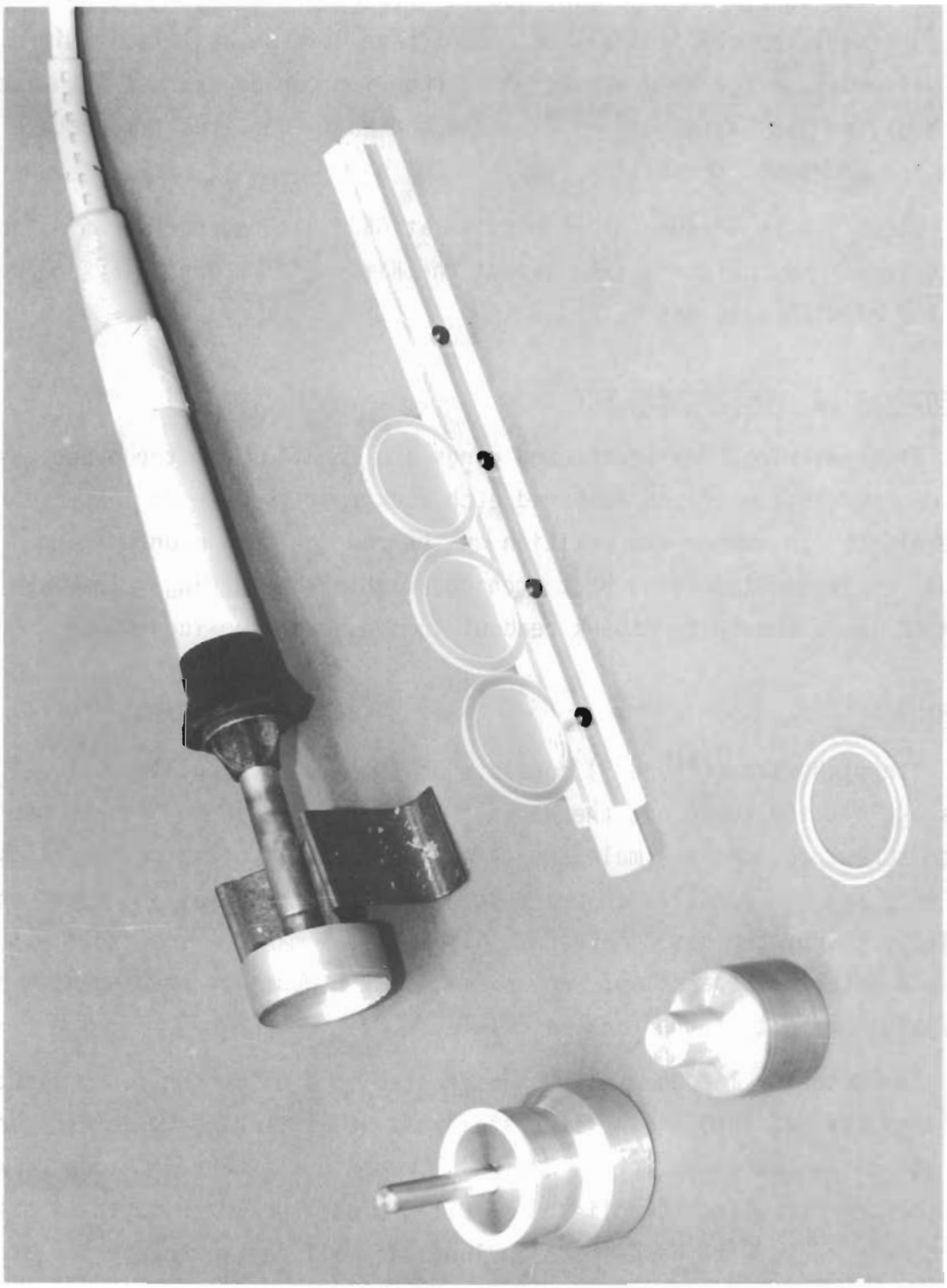

FIGURE 2

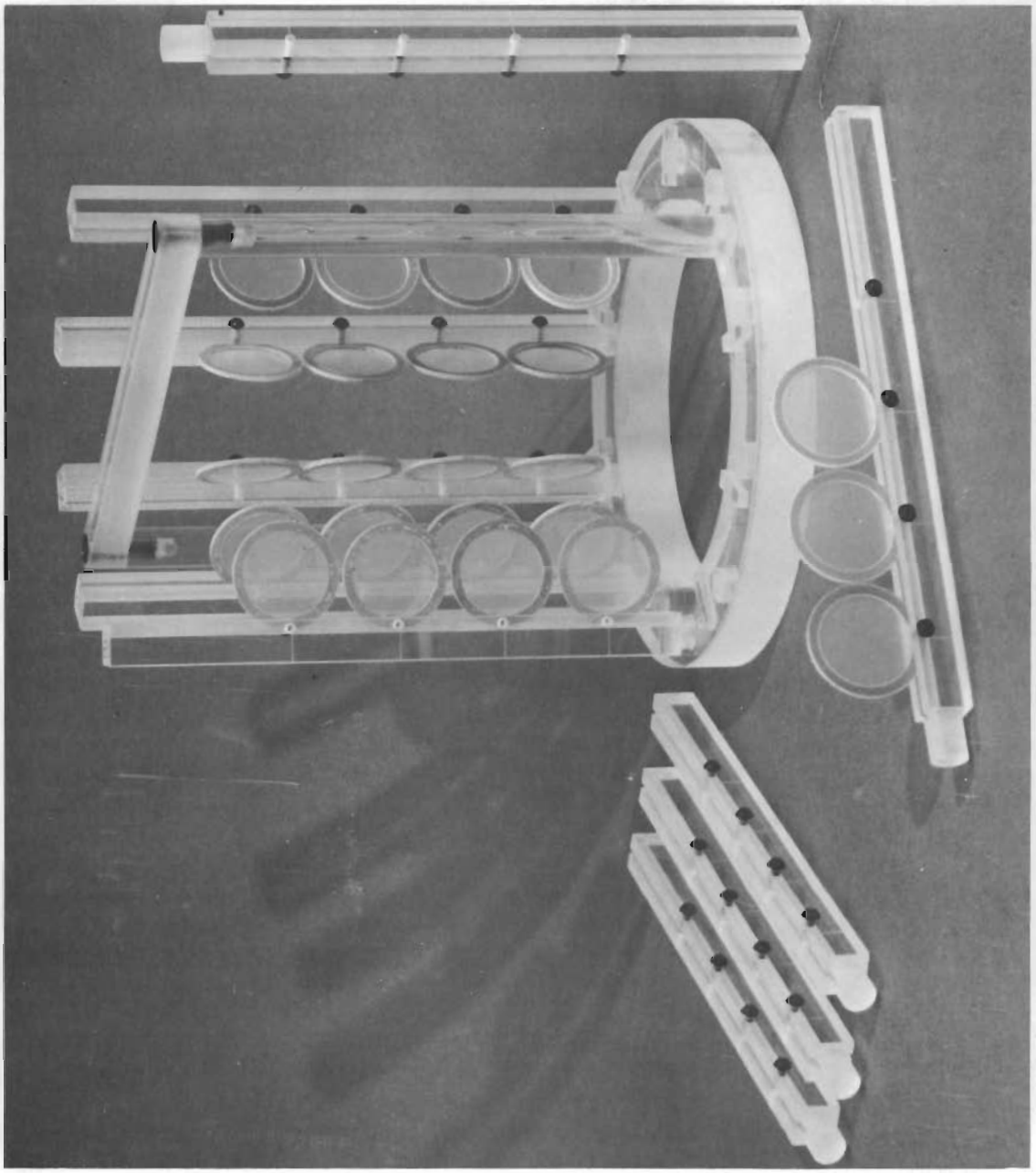

FIGURE 3

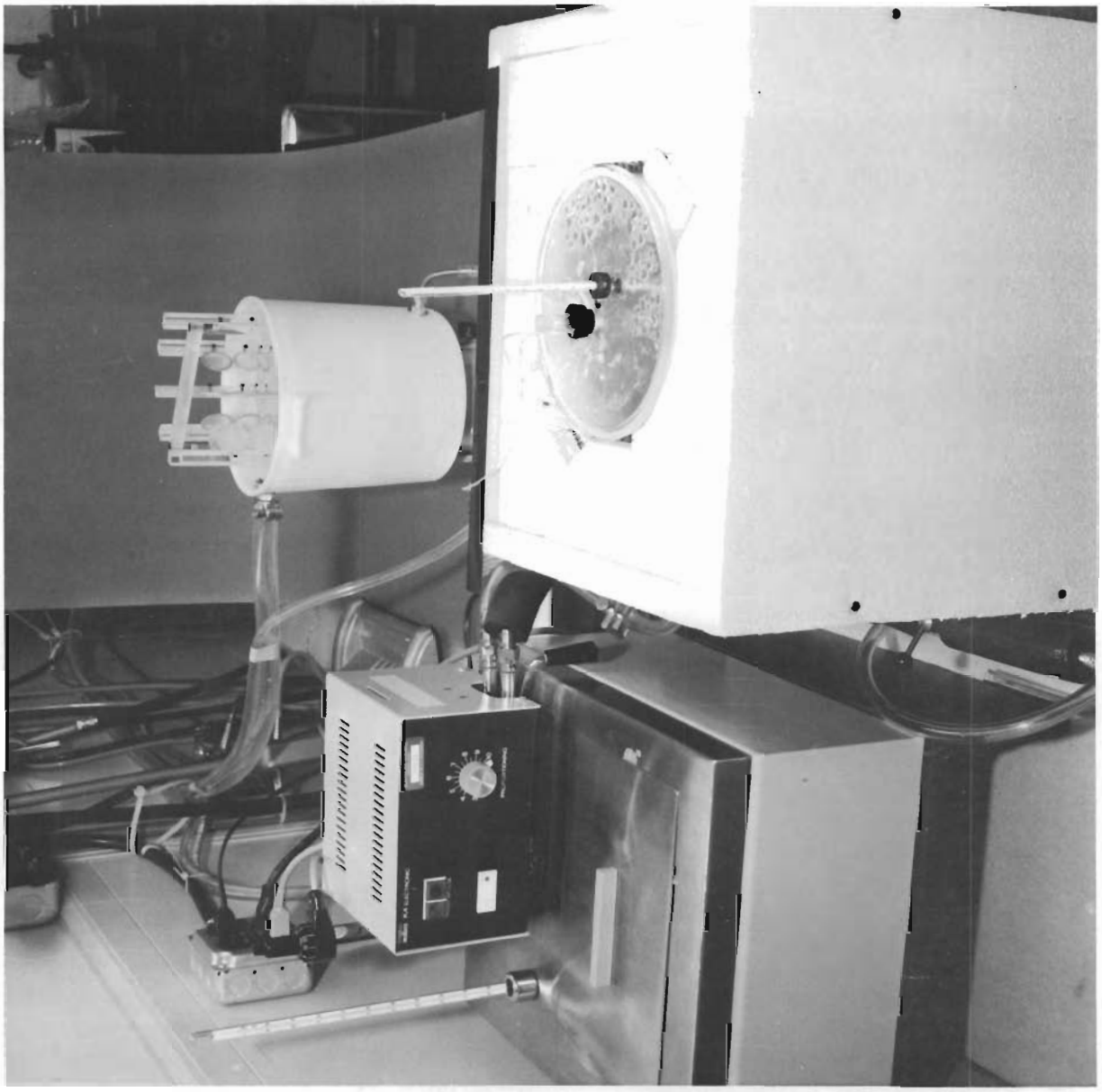

FIGURE 4

from the ultrasonic stirrer. Heat transfer to the KOH is helped by the high heat conductivity of the stainless steel etching vessel. Convection transfer within the KOH media is improved by circulating the etchent with an argon gas lift pump. The pump is patterned after the aquarium air lift method. Argon is used to avoid chemical interaction with the KOH. A flow rate of 1 CFH reflects a daily argon cost of about $0.30.

A study was made of the etching rate dependence with prior etching history or burnup of KOH. No effect was observed for 5.0 g/l of Kimfol dissolved in KOH; this reflects more than $10^3$ rings per liter.

A 15 minute washing cycle follows etching. The 4 liter polyethylene washing vessel is supplied with 0.5 gal/min of tap water (Figure 4). After washing, the etch rods are removed from the etching rack and placed in a 65°C drying oven for about 10 minutes.

## SPARK READER

The spark reader (Figure 5) and method is identical to that described in BNL 17452.

## ADMINISTRATIVE ASPECTS OF RADIOTHORIUM USE

The present badge contains less than 50 milligrams (less than 0.0055 microcuries) of natural thorium metal foil. The thorium foil is captive in a rigid plastic housing reinforced with a stainless steel backing. The level of external radiation due to the thorium foil is less than 0.06 mrad per hour measured 1 centimeter from any surface. This is less than the wrist or pocket watch radiation level authorized under Part 30.15, Title 10, of the Atomic Energy Act of 1954. Additionally, the wrist and pocket watch allowance is applied to the whole U. S. population, whereas the thorium-bearing personnel neutron dosimeters would be used by those occupationally associated with radiation, much less than 0.1 percent of the whole population.

Part 40.22 of Title 10 issues a general license for use and transfer of not more than 15 pounds of thorium at any one time, nor more than 150 pounds per year. The general license is issued for commercial, industrial,

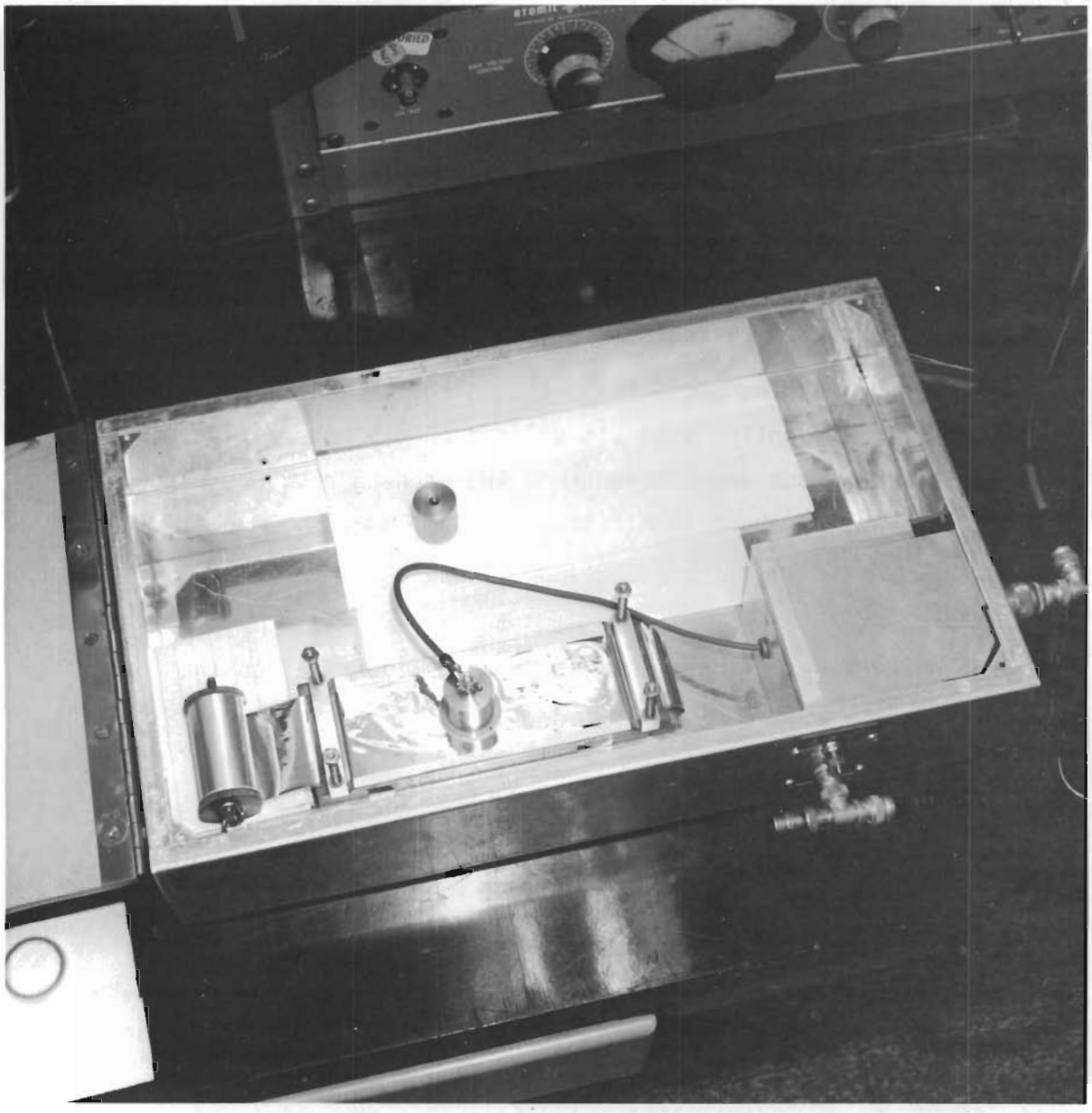

FIGURE 5

research, developmental, medical and educational purposes. Part 40.22 affords sufficient latitude for Brookhaven Laboratory, and others, to develop and produce damage track neutron dosimeters containing the specified amount of thorium. Issuance of thorium-containing badges to plant or laboratory users, staff members and visitors would be on a loan basis. Generally, laboratory and plant regulations require that dosimeter badges not be taken to the employees' homes. This requirement is not universal and where applied, not perfectly enforced. It must be assumed that a fraction of thorium-containing badges would be taken to the employee's homes during non-working hours.

Part 40.13, "Unimportant Quantities of Source Materials," authorizes all licensing exemptions for named commercial applications of thorium. Paragraph C (1) allows unrestricted manufacture, sale and use of lamps containing less than 50 milligrams of thorium each. The thorium-containing personnel neutron badge satisfies the above quantity restrictions, but unfortunately personnel neutron dosimetry is not one of the specified uses. A rule change recognizing dosimeter application is indicated. This step may be avoided by interpreting ownership of the dosimeter badge as belonging to the lending agency which has a general license. Since the badge must be returned for readout, and since personnel monitoring information is generated to satisfy legal operating requirements of the lending firm, agency or institution, dosimeter badge ownership should be interpreted as belonging to the lending agency.

# NEUTRON DOSIMETRY USING TLD AT ROCKY FLATS*

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The neutron dosimetry program using TLD at Rocky Flats is essentially the same as was reported at the Second AEC Workshop on Personnel Neutron Dosimetry, July 8-9, 1971. For a detailed description of this TLD neutron system, one is referred to the report of that workshop (BNWL-1616) and to the report "A Personnel Neutron Dosimeter Using Lithium Fluoride Thermoluminescent Dosimeters" (RFP-1581). This presentation will consist of a brief summary of the system and some comments and observations on the design and usefulness of TLD albedo neutron dosimetry systems.

At Rocky Flats we are mainly concerned with neutrons from ($\alpha$, n) reactions on light elements and from spontaneous fission of plutonium contained in glove boxes. The boxes may be shielded by from 0 to 4 inches of lucite. The shield may be voided by glove ports, especially when a person is working in the gloves at the box. The neutron spectra are mainly mixtures of unmoderated, moderated, and scattered neutrons.

We have found that a simple albedo-type TLD system gives a quite accurate indication of the neutron dose equivalent for these conditions. The original system was designed to be approximately the size of an NTA film packet and could fit into the cavity in our dosimetry badge which had been used for the NTA film. The current system has the same dimensions as the original but is incorporated as a permanent part of the dosimetry badge. The system consists of two pairs of TLD, each pair being one $^{6}$LiF and one $^{7}$LiF TLD. One pair is shielded from the front by 1/64 inch thick cadmium, and the other is shielded from the back by the same thickness of cadmium. The system is secured on the chest of the person to utilize the neutron scatter (albedo) from the person's body.

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* Research sponsored by the U.S. Atomic Energy Commission under contract with the Dow Chemical Company.

The neutron dose equivalent DE is obtained from the equation

$$DE = C \times (A - fxB) \qquad (1)$$

where A is the net neutron response of the TLD pair shielded from the front by the cadmium

B is the net neutron response of the TLD pair shielded from the back by the cadmium

f is a factor determined by calibration to optimize the system response

C is a calibration factor in terms of mrem per count.

In some cases it is useful to consider C as a function of the ratio B/A. The functional relationship is obtained experimentally by calibrating to the range of spectra which would normally be encountered in the working environment. It may also be useful to place a constraint on the value of the ratio to limit possible values of either A or B.

To demonstrate how the system functions and to illustrate some advantages of this system, consider the net response of each of the TLD pairs as a function of shield thickness for a $PuF_4$ neutron source. Data in Figure 1 are obtained from work done by Dale Hankins and presented in his report "Factors Affecting the Design of Albedo-Neutron Dosimeters Containing Lithium Fluoride Thermoluminescent Dosimeters" (LA-4832) from his Figure 16, page 11 for a badge containing 1/8 inch polyethylene. His response curve for "position 8" corresponds to values for A and the response curve for "position 1" corresponds to values for B in equation 1. It is seen that the relative sensitivity of each dosimeter pair varies considerably as a function of the shielding of the source. If a person desired to use crystal pair A alone as a neutron dosimeter system, the response would vary by a factor of 4 over this range of shielding.

If a two-pair system were used, the response of the system could be improved by several methods, illustrated in Figure 2. In one method the system response would be given by

$$\text{System Response} = A - 0.25xB \qquad (2)$$

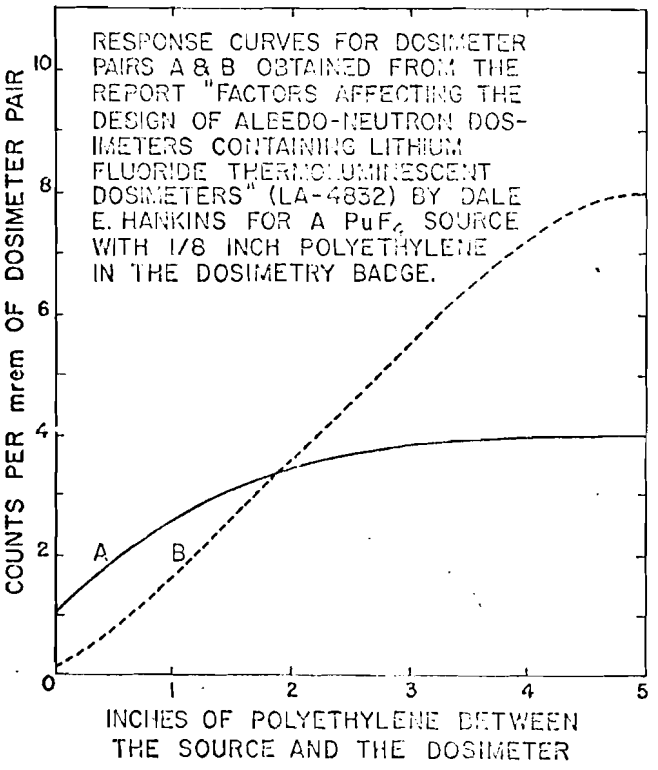


FIGURE 1

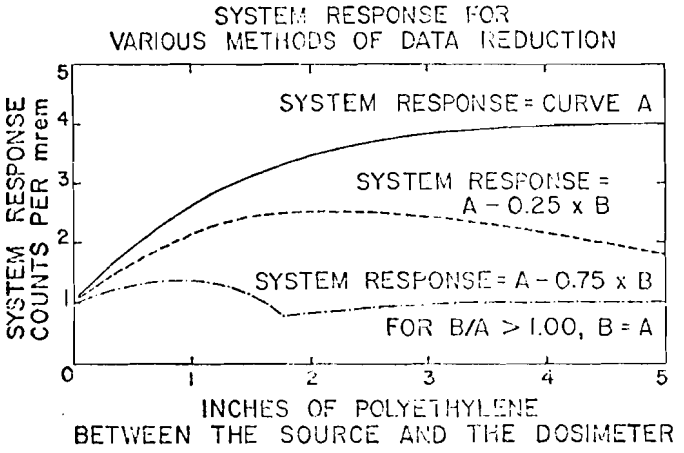


FIGURE 2

With this method the variation of response over the given range of shielding would be reduced from a factor of 4 to a factor of 2.5 and would be flat within ±25 percent for a range of shielding from 0.5 to 5 inches.

If one added a further constraint that for a ratio B/A greater than 1.00, the value of B would be set equal to the value of A, and used the equation

$$\text{System Response} = A - 0.75 \times B \qquad (3)$$

the system response would vary by +40 percent to -20 percent over the entire range of shielding thickness.

One more refinement is possible, which is to obtain a functional relationship between the ratio B/A and the calibration factor C. If we let

$$C = 1.25e^{-1.7\ B/A} + 0.8\ (B/A)^2 \qquad \text{for } B/A < 1.00$$

and

$$C = 1.05 \qquad \text{for } B/A \geq 1.000$$

and used the method of equation (3), we obtain a system response which would vary by +5 percent and -12 percent over the given range of shielding. It should be noted that although the last method results in the smallest variation in the system as a function of shielding thickness, statistical errors are magnified when the B/A ratio is used extensively as a correction factor, especially when there is also a significant photon DE.

Several conclusions are possible.

1. The use of two pairs of $^{6}LiF$ and $^{7}LiF$ can result in a system response which is significantly less sensitive to differences in neutron spectra than for one pair.

2. The method of data analysis is very important in obtaining an optimum response from a TLD albedo neutron system. This aspect is perhaps even more important than the physical design of the system.

3. Even with an optimized two-pair system, it is still necessary to calibrate to the source and range of spectra which would be encountered. The calibration for one source may not be applicable for other sources.

4. In designing a system considerations of automation should not take precedence over considerations of optimum response characteristics.

We have found that the TLD neutron system being used at Rocky Flats is very satisfactory as a personnel neutron dosimeter for the range of neutron spectra associated with plutonium processing in glove box operations.

# NEUTRON DOSIMETRY STUDIES AT LLL*

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The major emphasis in Neutron Dosimetry Development at LLL over recent months has been devoted to calibration facility improvement. We found that research was being hampered by the inadequacy of existing neutron source intensities, source spectra, and monitoring instrumentation. Therefore, our efforts have been devoted to upgrading source inventory, fabricating moderators for use with new sources, and developing a capability for low-level, neutron-spectral measurements. The following is a status summary of efforts at LLL:

## Source Inventory Upgrading

Most of the calibration work is being done in our low-scatter facility,[1] which is a shielded room 40 x 30 x 24 feet high. An aluminum grating, supported by an I-beam, 9 feet high, serves as a false floor. Neutron and gamma sources are introduced into the center of the room through two pneumatic systems, thus providing the potential for varying the neutron-to-gamma ratio incident on a detector.

The current source inventory includes a $^{252}Cf$ source that produces $8 \times 10^8$ neutrons/sec, a $^{238}PuBe$ source at $8 \times 10^7$ neutrons/sec, and a $^{238}PuLi$ source at $1.6 \times 10^6$ neutrons/sec. We are in the process of obtaining a $^{252}Cf$ source that will yield $5 \times 10^9$ neutrons/sec to replace the old Californium source. We are also having a sealed source of antimony made that can be activated in the Laboratory pool-type reactor. The $^{124}Sb$ can then be placed in the pheumatic system and fired into a 830-g sleeve of beryllium placed over the end of the system head. We expect this to give us a source of 25 keV (gamma, n) neutrons with a strength of about $1 \times 10^8$ neutrons/sec.

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* Work Performed Under Auspices of the U.S. Atomic Energy Commission

In order to design moderators that can be placed over the end of our pneumatic system for use with any of these neutron sources, we made a set of Monte Carlo calculations to determine the transmitted spectra for $^{252}Cf$ and $^{124}SbBe$ neutrons through 5-, 10-, 20-, and 25-cm radius spheres of $D_2O$, graphite, water, polyethylene, iron and aluminum. As a result of these calculations, we are going to fabricate 5-, 10-, 15- and 25-cm radius steel shields for $D_2O$ as well as 2-, 5-, and 10-cm radius spheres of polyethylene and a 20-cm sphere of aluminum. Each of these moderators can be used with all of our sources. We found that the spectra from graphite and $D_2O$ were very similar as were the spectra for water and polyethylene, and iron and aluminum so that we will not need graphite, water, or iron spheres.

Neutron Spectrometry

We feel that it is necessary to have a measurement of the neutron spectrum in order to adequately characterize the source and source-moderator fields used in our calibration facility. Therefore, work is being done to assemble a spectrometry system that will give us spectral information in the range from 10 keV or below to 10 MeV using proton recoil proportional counters. The system is now being assembled and no performance data is available.

The use of multisphere spectrometry is also being studied since it provides us with a simpler system that yields spectral information over the full range of neutron energies and is useful for spectrometry in the working areas. Multisphere spectral resolution is poor but adequate for radiation protection purposes.

The major concerns in multisphere spectrometry are the availability of good quality response functions and the ability to unfold the multisphere response data. Sanna[2] has recently calculated response functions for polyethylene and water spheres with $^{6}LiI$ scintillators or gold foils as detectors. We have started using this

data and have found it helpful. In fact, we plan to augment the data with calculations for $^{235}U$ track foil detectors to provide us with a sensitive, passive "field," spectrometry system.

We feel that our search for an unfolding technique has ended for the near future with LOUHI, a computer code developed by LBL.[3] This code seems to be highly satisfactory for spectrum unfolding, not only for multisphere detectors, but also for other systems such as activation foils.

Dosimeter Development

Our work in personnel dosimetry has been devoted to development of an albedo dosimeter for use in the near future and to research on track registration as a long term solution to the dosimetry problem. The albedo dosimeter design involves use of two pairs of TLD 600 and 700 chips on either side of a cadmium or borated plastic shield. The final shield selection will be made based on studies to be performed following completion of calibration facility improvements.

The track registration work has been confined to detection of light particles, such as alpha and recoil nuclei.[4] We have avoided fission foil dosimeters due to the potential hazard involved with such materials.

Using cellulose nitrate (CN) as registration material, we find that the proton tracks appear as pits with a mean apparent diameter of about 1.5 microns when measured with a TV type particle analysis system. The tracks are revealed only with low temperature etching, either after about 4 hours in 6N KOH at 22°C or 1/2 hour in the same solution at 40°C. Use of 6N NaOH appears to increase the necessary etch time by a factor of two. Using the particle analysis system, we found that an average track density of two CN pieces on either side of a polyethylene wafer was about 700 tracks/$cm^{-2}$/$rem^{-1}$ with a 13 percent spread for the three neutron spectra used; i.e., bare $^{252}Cf$, $^{252}Cf$ through 10-cm steel, and $^{252}Cf$ through 20-cm $H_2O$. However, the

artifact background using optical counting was equivalent to a dose of about 20 rem. We find the results encouraging and plan to conduct further studies.

References


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2. R. S. Sanna, Thirty-One Group Response Matrices for the Multisphere Neutron Spectrometer over the Energy Range Thermal to 400 MeV, Health and Safety Laboratory, USAEC Rept HASL-267, March 1973.

3. R. V. Griffith, "Progress in Multisphere Neutron Spectrometry," Hazards Control Progress Report No. 45, Lawrence Livermore Laboratory Rept UCRL-50007-73-1, to be published.

4. R. V. Griffith, "Developments in Light Particle Track Registration for Personnel Neutron Dosimetry," Hazards Control Progress Report No. 44, Lawrence Livermore Laboratory Rept. UCRL-50007-72-3, March 29, 1973.

# NEUTRON DOSIMETRY STUDIES AT LASL*

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Since the last workshop meeting, we have continued to investigate the possibility of replacing our existing film badge with a TLD system. We would like to have a TLD system which would have an automated read out capability and include an albedo neutron dosimeter. To do this the small albedo-neutron dosimeter described at the last meeting and in LA-5261 has been modified by cutting a slit across the top of the dosimeter to allow a card containing TLD's to be inserted into the dosimeter. This requires (1) the removal of the center 1/16-inch thick piece of polyethylene and (2) a lip of cadmium has to be placed around the slit to decrease the thermal-neutron leakage.

Removing the center piece of polyethylene from the dosimeter reduces the fast-neutron sensitivity of the dosimeter by ∿11 percent. The thermal-neutron leakage into the slit increases the thermal-neutron sensitivity to ∿50 percent higher than desired to match the fast-neutron sensitivity. This may be satisfactory, however, since the final model would have a plastic case which is expected to reduce the thermal-neutron leakage through the slit. At present, the thickness and type of plastic case has not been established and we have not determined the thermal-neutron sensitivity of the final dosimeter.

We conducted a 6-month comparison study using a similated TLD system and the film badge. The albedo-neutron dosimeters were taped to the bottom of the film badges to assure us that both were similarly exposed. In the actual dosimeter one pair of TLD's would be used but in this study, eight TLD's were placed in the albedo-neutron dosimeter and the average reading of the four $^{6}Li$ and four $^{7}Li$ TLD's was used. Additional TLD's (four compared to two that one would normally use) were placed inside the film-badge holder to measure the gamma exposure and thermal-neutrons. The extra TLD's were used to study statistical variation one would obtain from the use of fewer TLD's.

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* Research sponsored by the U.S. Atomic Energy Commission under contract with the Los Alamos Scientific Laboratory.

The results of the study were rather complex but generally indicated that for the neutron component of the dose, reasonable agreement was found in the $^{238}Pu$ areas. In areas where the dose rates were low (<1 mrem/h) the film reported only a small part of the total neutron dose. For the total neutron and gamma exposures, we found that an overresponse of the film to low-energy gamma rays gave a net reported dose that was approximately correct except for one area of the plant where it was high.

We conducted a study of $^{10}B$-loaded plastic to use in place of cadmium to surround our dosimeters. The $^{10}B$-loaded plastic is on loan from Dick Griffith at LLL. The amount of $^{10}B$ in the plastic was varied until the thermal-neutron leakage was approximately equal to the fast-neutron response. Unfortunately at this loading, the fast-neutron response of the dosimeter was decreased to approximately one-half the sensitivity of a 30 Mil cadmium covered dosimeter. This reduced sensitivity was felt to be too low for a practical dosimeter. Further work has been discontinued on $^{10}B$-loaded plastics. It is unfortunate that $^{10}B$ plastics could not be used, since they would have been lighter, easier to manufacture, and have no thermal neutron capture gamma problem. The dosimeters would probably have been more expensive because of the high cost of $^{10}B$. To reduce the cost, we also studied normal boron and found the thickness of plastic required was too large for a practical dosimeter.

At the last meeting I discussed a technique using a 9 inch sphere remmeter and 3 inch sphere with 12 Mil Cd cover to determine the calibration factor for the albedo-neutron dosimeter in field applications. The 3 inch sphere with the 12 Mil cadmium cover has a response to fast neutron very similar to that of the albedo-neutron dosimeter. Unfortunately the spheres are not sensitive to the direction of the neutrons and would, in cases where the neutron are coming from the rear of a phantom, give a calibration factor different than one obtained from dosimeters placed on the front of the phantom. A technique to correct the sphere response to agree with the phantom results has been developed. This consists of placing the bare probe of the remmeter at the equator of the sphere and by moving this probe around the sphere one can determine the approximate direction of

incidence of the neutrons. This procedure is described in LA-5261. Although it is only approximate, it gave good agreement in an experimental field test.

The size of the phantom for neutron studies of albedo-neutron dosimeters is important. We found that the response on the upper and lower chest of a phantom varies by about 30 percent for a phantom having lung cavities. A gallon jug gives a dosimeter response ∿10 percent higher than one obtains from the upper chest of this phantom with lung cavities. We also investigated the penetration of neutrons through the phantom and found this varied greatly with neutron energy (see LA 5261).

Albedo neutron dosimeters are very energy dependent. A small decrease in the average energy of the neutron spectrum causes a significant increase in the dosimeter response. To determine the effect of these spectral changes on the accuracy of the dosimeters used at LASL, we made a study at our reactor and our Plutonium facility, using the 9 and 3 inch sphere technique. The results have been plotted in Figures 1 through 4. One can determine that an accuracy of ±40 percent could be obtained in all areas except in the $^{238}Pu$ facility (Figure 2) where a larger spread of +60 and -40 percent would be required. The relative TLD sensitivity varies for the different facilities and the calibration factor varies proportionally. Therefore we must know where the exposure occurred if we are to accurately evaluate the exposure. This is the major disadvantage of an albedo-neutron dosimeter system.

I will give a report of the English albedo-neutron dosimeter and the Swiss spark-counter system. Both of these dosimeter systems were discussed with the developers during my recent trip to Europe.

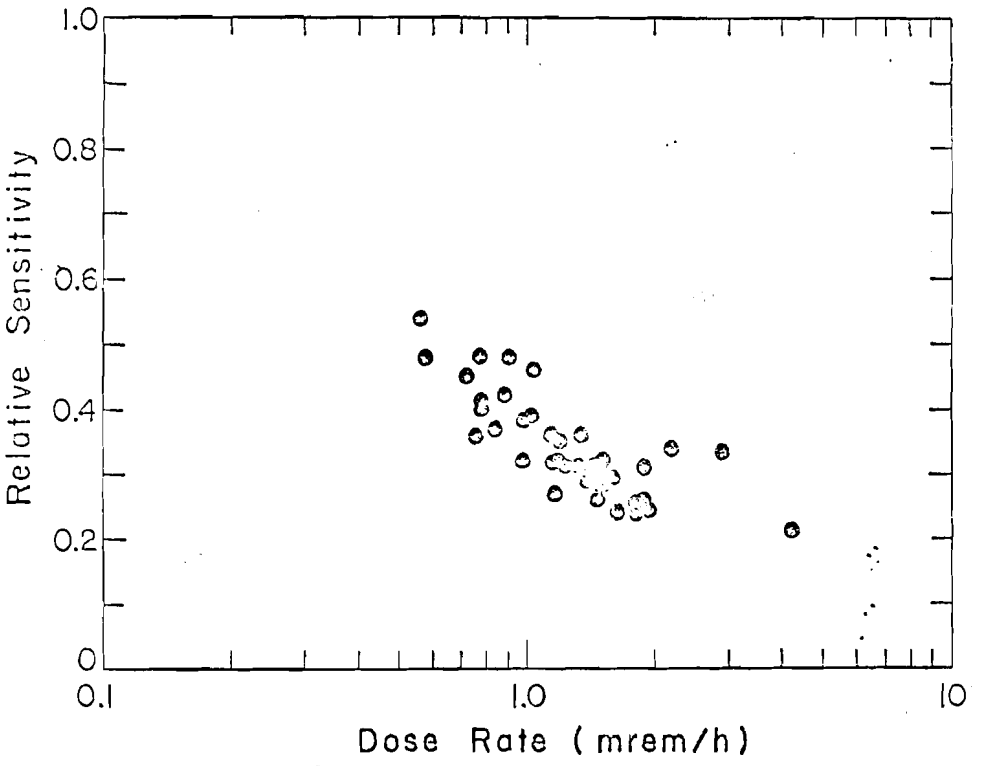


FIGURE 1. Results of a Survey at a Plutonium-Metal Fabrication Facility Taken with the 9- and 3-Inch Spheres Showing the Variations in the Relative Sensitivity of the Dosimeter as a Function of Dose Rate

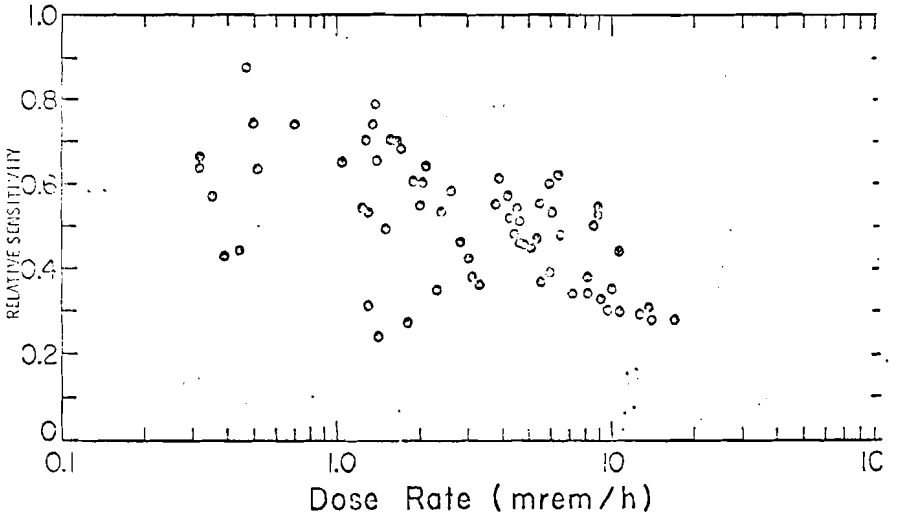


FIGURE 2. Results of a Survey at a Plutonium 238 Facility Taken with the 9- and 3-Inch Spheres Showing the Variations in the Relative Sensitivity of the Dosimeter as a Function of Dose Rate.

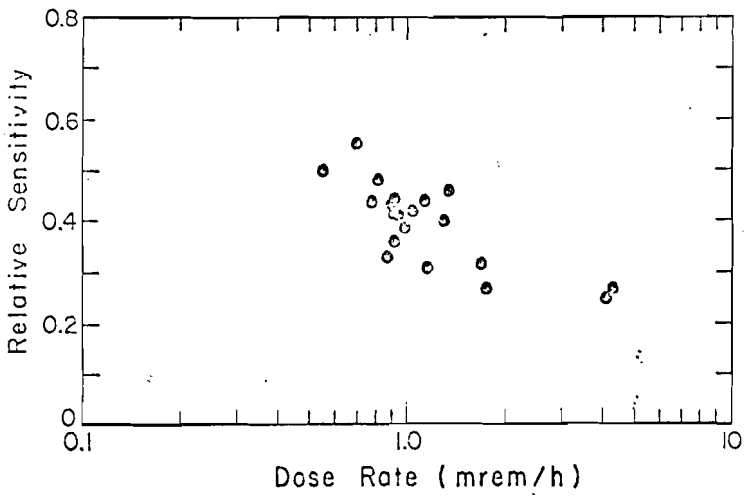


FIGURE 3. Results of a Survey at a Plutonium Recovery Facility Taken with the 9- and 3-Inch Spheres Showing the Variations in the Relative Sensitivity of the Dosimeter as a Function of Dose Rate

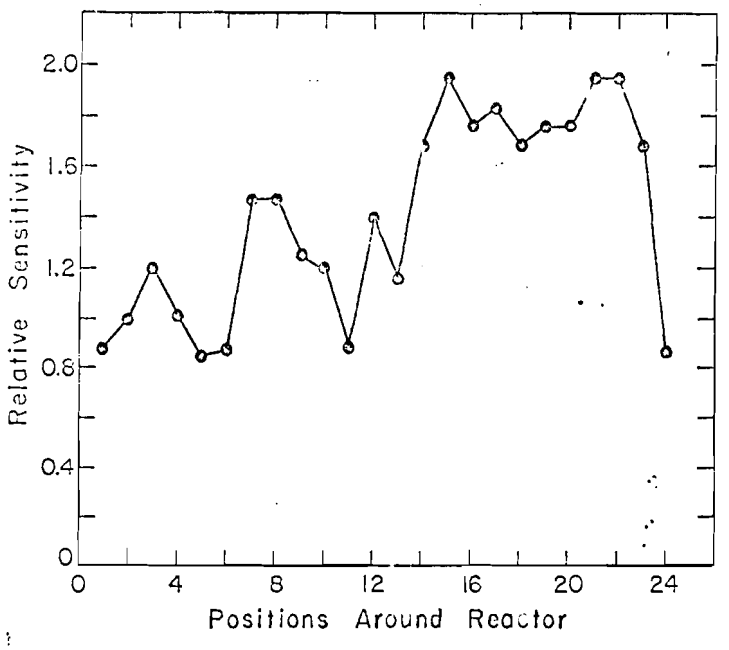


FIGURE 4. Results of a Survey at the OWR Reactor Taken with the 9- and 3-Inch Spheres Showing the Variations in the Relative Sensitivity of the Dosimeter

# PERSONNEL NEUTRON DOSIMETRY AT SLAC*

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At SLAC more than 90 percent of the recorded personnel doses are caused by photon radiation. Consequently neutron exposures are a minor problem. We use a $Li^6F/Li^7F$ personnel dosimeter for both photons and neutrons. The neutron doses are evaluated by a method to be discussed shortly. In 1972 there were 25 neutron exposures in excess of 100 mrem out of a total of 1763 people monitored. The highest neutron dose was 300 mrem.

The method of evaluating neutron doses was arrived at through a number of studies which determined the TLD response as a function of fast to thermal neutron flux ratio and the flux ratio in the area where personnel are exposed. Using a Pu Be source we found that when the fast to thermal flux ratio is less than 10, the dosimeter, placed on a rotating phantom, essentially is responding to the thermal neutrons from the source. Consequently, the calibration in terms of light output as a function of fast neutron dose varies with the fast to thermal flux ratio.

Measurements made in the SLAC research areas show that the mean fast to thermal flux ratio is 3:1 with a variation from 5:1 to 1.2:1. We use the mean flux ratio to evaluate the personnel neutron doses. This means that our neutron dose assignments may be high by as much as a factor of 3 if the ratio is really 1.2:1, or low by 60 percent if the ratio is really 5:1.

More studies are being done to evaluate the TLD response as a function of fast to thermal flux ratio using neutron sources of different energies.

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* Work performed under auspices of the U.S. Atomic Energy Commission.

# THE HANFORD THERMOLUMINESCENT MULTIPURPOSE DOSIMETER*

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This report briefly covers some of the problems that were encountered during the first year of operating experience with the Hanford Multipurpose Dosimeter.

The total number of dosimeters read during 1972 includes over 25,000 multipurpose with five TLD's each providing dose estimates for both neutron and beta-gamma exposures, and 7,400 basic dosimeter cards with one TLD each. These figures do not include experimental, quality control and calibration dosimeters that were also processed. Total operating time on the automatic reader to date is over 1,000 hours. As expected, most dosimeter results were zero; however, there were 1,500 fast neutron dose interpretations above 50 mrem and 2,100 penetrating gamma interpretations above 300 mrem for the multipurpose dosimeters. The basic dosimeter results include 110 interpretations greater than 300 mrem.

The fast neutron calibration procedure was modified to more closely match the field exposure conditions. This was done by an intercomparison between the tissue equivalent proportional counter (TEPC) and the multipurpose dosimeter. The TEPC and dosimeter were exposed to fast neutrons at several locations around Hanford. Using this data, the laboratory fast neutron calibration procedure was modified to match the relative sensitivity of the dosimeter when it is exposed under field conditions.

An additional study was made to evaluate the fast neutron response of the dosimeter as a function of position on the body. The initial study was done in the laboratory using a phantom.[1] This study was done in the field using volunteer employees. Tight fitting tee-shirts having three internal pockets were used to position three dosimeters over the stomach,

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* Work sponsored by the U. S. Atomic Energy Commission under contract with Battelle-Northwest.

right chest and the sternum of each employee. The tee-shirt kept the dosimeters in close contact with the body. A fourth dosimeter was worn clipped to the left chest pocket of a loose fitting coverall or lab coat. All dosimeters were worn for a 1-month period while the employee performed his normal work. Data from this study indicated the dosimeter worn on the loose fitting coverall averaged 70 percent of the average fast neutron response of the other three dosimeters. There was no statistical difference in the fast neutron response of the dosimeters at the other three locations. This 70 percent factor was incorporated in the fast neutron calibration procedure for the dosimeter so that it is automatically factored into the dose equivalent interpretation.

For fast neutron measurements the multipurpose dosimeter has two $^{6}LiF$ blocks that are 0.32 x 0.32 x 0.089 cm. One block has a cadmium-tin filter; the other has a tin filter. The filters are adjusted in thickness to be equivalent for photon radiation. The difference in the readings of the two blocks can be related to incident thermal neutrons since the cadmium will absorb essentially all thermal neutrons which impinge on it.

For fast neutron measurements the $^{6}LiF$ block behind the tin filter responds to incident thermal, backscattered fast neutrons and backscattered thermal neutrons. Fast neutron dose equivalent is derived by subtracting out incident thermal neutrons and correcting for backscattered thermal neutrons as determined from calibration exposures. Response due to photon interactions in the TLD's is corrected for by using a $^{7}LiF$ block and a tin filter similar to that in front of the $^{6}LiF$ blocks.

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# PERSONNEL NEUTRON DOSIMETRY STUDIES AT THE LAWRENCE BERKELEY LABORATORY*

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## 1. Introduction

It is generally recognized that personal neutron dosimetry is a difficult task, but it is perhaps around high-energy accelerators that the problem is most severe. Firstly, a high-energy accelerator is a potent source of radiation--for example, Distenfeld[1] has reported that the Alternating Gradient Synchrotron (AGS) contributes 80 percent - 90 percent of both the gamma and neutron exposures to Brookhaven personnel. Secondly, the variety of radiation fields is great. At proton accelerators neutrons usually dominate the radiation field outside thick shields, but large differences in neutron spectrum are observed[2] which can make the interpretation of measurements made with a single dosimeter difficult. Furthermore, under poorly shielded conditions (for example adjacent to primary particle beams) the gamma:neutron ratio may change by more than two orders of magnitude.[3] At many high energy electron accelerators, too, neutrons are being identified as an important component of the radiation environment.[4]

## 2. Personal Dosimetry at Accelerators

This author's experience at many high-energy laboratories with several different types of accelerators have led him to the following conclusions concerning individual dosimetry for accelerator personnel.

a. The "Universal" Dosimeter - which is dose-equivalent responding under all the varied radiation conditions existing around a high energy accelerator does not exist at the present time. It probably is a physical impossibility given the present administrative definitions of dose-equivalent.

* Work sponsored by the U.S. Atomic Energy Commission under contract with the Lawrence Berkeley Laboratory.

b. Neutron Dosimetry - Necessary. In most working environments around proton accelerators (and at many electron accelerators too) neutrons are dominant. It is therefore nearly always necessary to monitor the individual neutron exposure. (For example, it makes little sense to attempt to estimate 90 percent of the dose equivalent contribution due to neutrons from a measure of 10 percent of the dose equivalent with a gamma dosimeter.)

c. More than One Dosimeter Necessary. At least two personal dosimeters are necessary--one for photon exposures, at least one for neutrons. If personnel are exposed to a wide range of neutron energies two or more neutron dosimeters may be necessary.

d. The Appropriate Dosimeter? Intermediate energy neutrons rarely present a major problem at accelerators, although they may do so under certain shielding configurations, e.g. steel shielding,[5] at the entrance to labyrinths penetrating shielding.[6] For these unusual circumstances albedo-type dosimeters would be most helpful. At most proton accelerators the neutrons making the dominant contribution to the dose equivalent lie in the energy range 0.1 MeV-20 MeV.[2] This energy range is fairly well handled by nuclear track film--although this dosimeter is anathema to some it is still widely used by those who have a need to monitor neutrons. Fission foil track detector combinations have also been used to some degree in this energy region at accelerators.[1]

Under certain conditions (e.g. shields with a high water content) the contribution to the dose equivalent due to neutrons greater than 20 MeV in energy may be more than 50 percent[3]--even for accelerators in the GeV energy region. Additionally, at accelerator energies in the hundreds of GeV range the energy sensitivity of neutron personal dosimeters must extend well beyond 20 MeV. It is with this extension of neutron personal dosimetry to higher energies that we have been concerned with at Berkeley during the past year.

## 3. Personal Dosimetry Studies at Berkeley

Neutron track film has proved to be invaluable in assessing personal neutron exposures. We do not find ourselves in agreement with those who find it of no value in routine use. The most serious technical deficiency

claimed for neutron track film--track fading--is not serious at the relatively low humidities which obtain at Berkeley. Many studies now show that at moderate temperature and RH errors in DE estimation due to track fading are less than 20 percent.[7] Furthermore, if track fading is significant it may be adequately controlled by film packaging.[7] As an example of routine use packaged nuclear track film has been successfully used at the Rutherford Laboratory, in conjunction with a crude $^{6}LiF$ albedo type dosimeter to obtain good estimates of neutron dose equivalent in a variety of radiation environments.[8]

a. Thick Emulsions

At neutron energies above 20 MeV thick nuclear emulsions may be utilized as personal dosimeters. At Berkeley 600 μ thick Ilford L4 emulsions and NTA films were exposed to neutrons of nominal energy 4.5, 14, 25, and 225 MeV. In addition, emulsion sensitivity was compared with fission foil detectors. These exposures have been described in detail elsewhere.[9] In summary, it was found that the normalized sensitivity of L4 emulsion was greater than that of NTA film at all energies because of the greater emulsion thickness and the greater sensitivity of L4 emulsion to higher energy proton recoil tracks. The sensitivity of the technique is adequate: for example, at 225 MeV a neutron fluence of $2 \times 10^{5}$ n $cm^{-2}$ (about 10 mrem) may be measured in a scanning time of less than 30 minutes. An interesting observation was that NTA film showed a significant response to the 225 MeV neutron beam which cannot be explained in terms of low energy contamination. The response of NTA film to high energy neutron spectra is worth serious study because there may be some significant response from protons in equilibrium with the neutron component of the nuclear cascade.

b. Electronic Personnel Dosimeter

Preliminary studies lead us to believe that it is now technically feasible to produce a small personal neutron monitor (about the size of a Hewlett Packard pocket computer). A typical instrument might utilize a 0.5-inch x 4.6 inch long $BF_3$ counter (20 cm Hg filling, 96 percent $^{10}B$) moderated by 0.36 inch polyethylene surrounded by cadmium. Such a counter worn on the body would generate 350 counts/mrem for 252-Cf neutrons--easily

detectable above background.[10] The basic electronic technology is now available and after some basic design work it would appear feasible to produce these dosimeters for less than $1000. (Consultation with manufacturers would almost certainly lower this figure.)

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# THE SOLID-STATE TRACK-RECORDER FAST-NEUTRON DOSIMETER EMPLOYED AT ARGONNE NATIONAL LABORATORY*

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Before I describe our particular fast-neutron dosimetry system, let me just say that the material will be presented in detail in a forthcoming issue of Health Physics under the title "Solid-State Track Recorder Personnel Neutron Dosimetry at Plutonium-Fueled Zero Power Fast Reactor Facilities (ZPR-6 and ZPR-9)," by Raymond Gold, Roland J. Armani, Gordon K. Rusch and myself. Because the material will appear rather shortly, I would rather concentrate on the basics of the system, than get too involved in the details. My approach is also somewhat different than the one followed in the paper. If there existed an ideal neutron dosimeter, that is, one with the proper energy response, good sensitivity, made from non-toxic materials, and not a source of radiation itself, then much of what I have to say today would be of little practical interest. First, let me say how we got involved in the problem, since none of us are health physicists. Back in 1969 plutonium was introduced as a fuel in the reactors at Argonne National Laboratory in Illinois. The use of this fuel creates a dose hazard because of the presence of fast neutrons from spontaneous fission decay and subsequent multiplication of these neutrons in the subcritical assembly. Personnel working around the reactors have to be monitored. At the time there was no satisfactory fast-neutron personnel dosimeter available. The neutron sensitive film badges almost always showed little or no dose, even though we knew the doses were several hundred mrem in the reporting period. One thing should be immediately emphasized--the type of neutron source we are dealing with. The source is a degraded fission source with the peak of the distribution appearing at several hundred keV; also, more than 95 percent of the dose arises in our environment from fast neutrons.

We choose as a system to monitor the dose a card containing two solid-state track recorders with U-235 sources--one recorder and source is contained

* Work sponsored by U.S. Atomic Energy Commission under contract with the Argonne National Laboratory.

in an aluminum package, the other is cadmium covered. The dose is determined from the number of fissions in the two recorders. The fission tracks are counted manually with an optical microscope (~200X). A skilled reader can determine the number of tracks on both recorders in about 20 minutes. Figure 1 shows a solid-state track recorder. Figure 2 shows a photomicrograph of a recorder. The mica is pre-etched to enable the reader to distinguish tracks arising from spontaneous fission of the uranium impurities. The large diamond is such a track; the smaller ones are from fission in the source.

The obvious question is how can the fission rates in these two recorders be used to determine the dose. Neither recorder has the proper energy response. They both have a high sensitivity to intermediate-energy neutrons and a non-rem response to fast neutrons. The answer to the question is that all we can determine is an upper limit on the dose received by the wearer. The conceptual approach of the dosimeter was that the cadmium covered recorder could be used to determine an upper limit by calibrating it in the hardest spectrum--the spectrum in which the number of tracks per mrem would be lowest. The cadmium ratio (the response of the aluminum covered recorder divided by that of the cadmium covered one) would be used to correct for different relative contributions from intermediate-energy neutrons. One may construct a simple model of the neutron spectrum in our environment to show why this might be the case. The model is explained in the paper referred to at the beginning of the talk; however, the experimental determination of the neutron environment showed that several of its assumptions were not correct.

The manner in which the dose is related to the cadmium-covered track density $T_{cd}$ and the cadmium ratio CR (the only two independent variables in the system) is obvious if we demand that doses received under the same conditions in the consecutive time intervals $t_1$ and $t_2$ be additive, that is

$$D_{SSTR}(t_1 + t_2) = D_{SSTR}(t_1) + D_{SSTR}(t_2), \qquad (1)$$

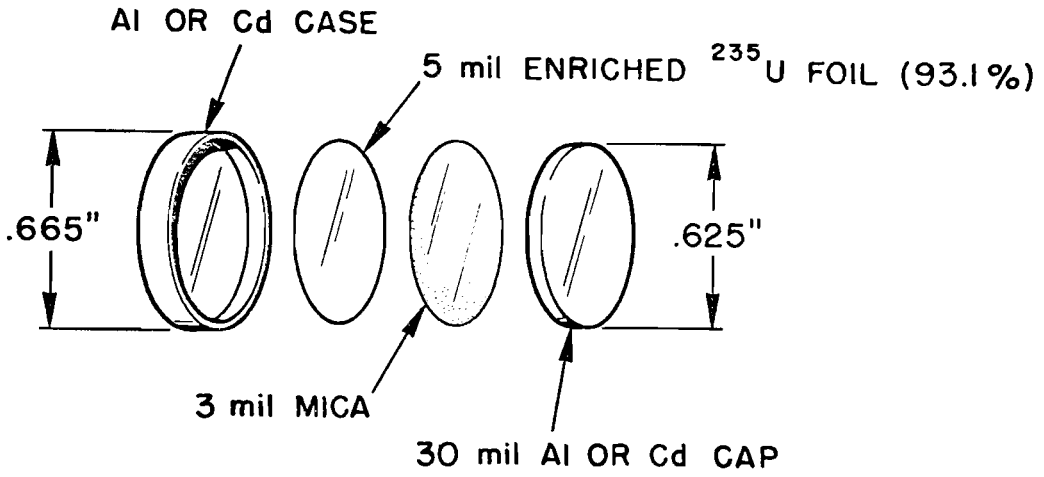


FIGURE 1. Solid State Track Recorders

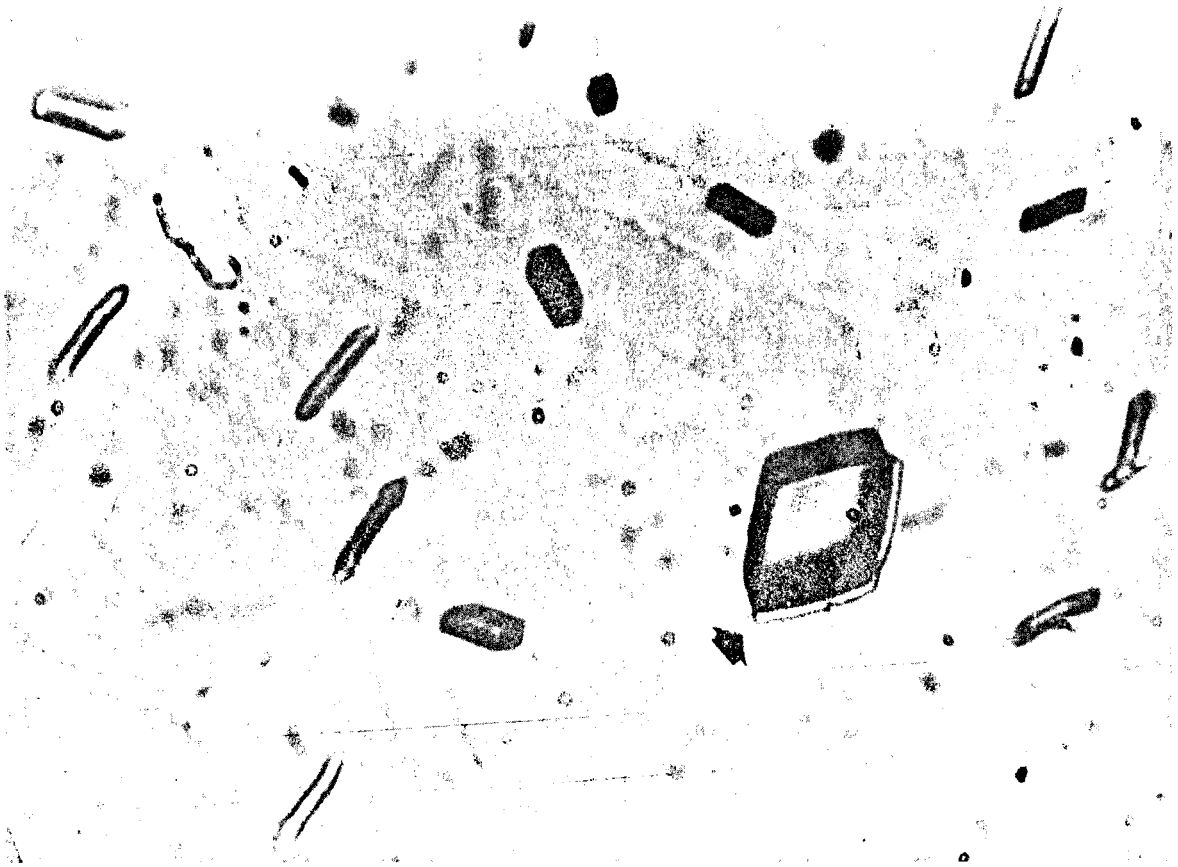

FIGURE 2. Photomicrograph Record

where $D_{SSTR}$ denotes doses determined from the track recorders. Since the track density is proportional to the exposure time, $D_{SSTR}$ must be a linear function of the track densities. Thus, the form we must use is

$$D_{SSTR} = [a - b(CR)] \quad T_{cd}, \tag{2}$$

where a and b are constants.

Now, to calibrate the dosimeter it is necessary to determine the constants a and b in such a way that we always obtain a $D_{SSTR}$ that is equal to or greater than the true dose. A minimum set of measurements require the measurement of doses and track densities at various locations in the area where personnel will be present. Of course, one wants to give some consideration to where exposures are made; it is necessary to have a measurement in the hardest spectrum and one in the softest spectrum. Thus, if we are dealing with a fission source, we need to have one measurement as close to the source as possible and one for which the source is in its most fully shielded configuration.

Having presented this background information, I would like to show what we found for our particular environment. Figure 3 shows one of the ZPR reactors. You can see the control rods and the matrix tubes, which contain drawers which contain the reactor fuel and other materials. Personnel must go between the halves of the reactor to load it. Figure 4 shows a top view of the cell containing the reactor. The cross-hatched rectangles represent personnel shields, which are used to decrease the neutron and gamma-ray doses. They are moveable and in three sections. The most important constituent of the shield is a hardboard-type material. The x's indicate positions at which dosimetry studies were carried out; the circled x's are locations at which more detailed studies were undertaken and constitute a representative set. These positions will be the only ones I will discuss.

In addition to measuring the doses at the various locations with a portable rem meter, we made other measurements to characterize the neutron environment. Measurements were made with bare and cadmium-covered $BF_3$

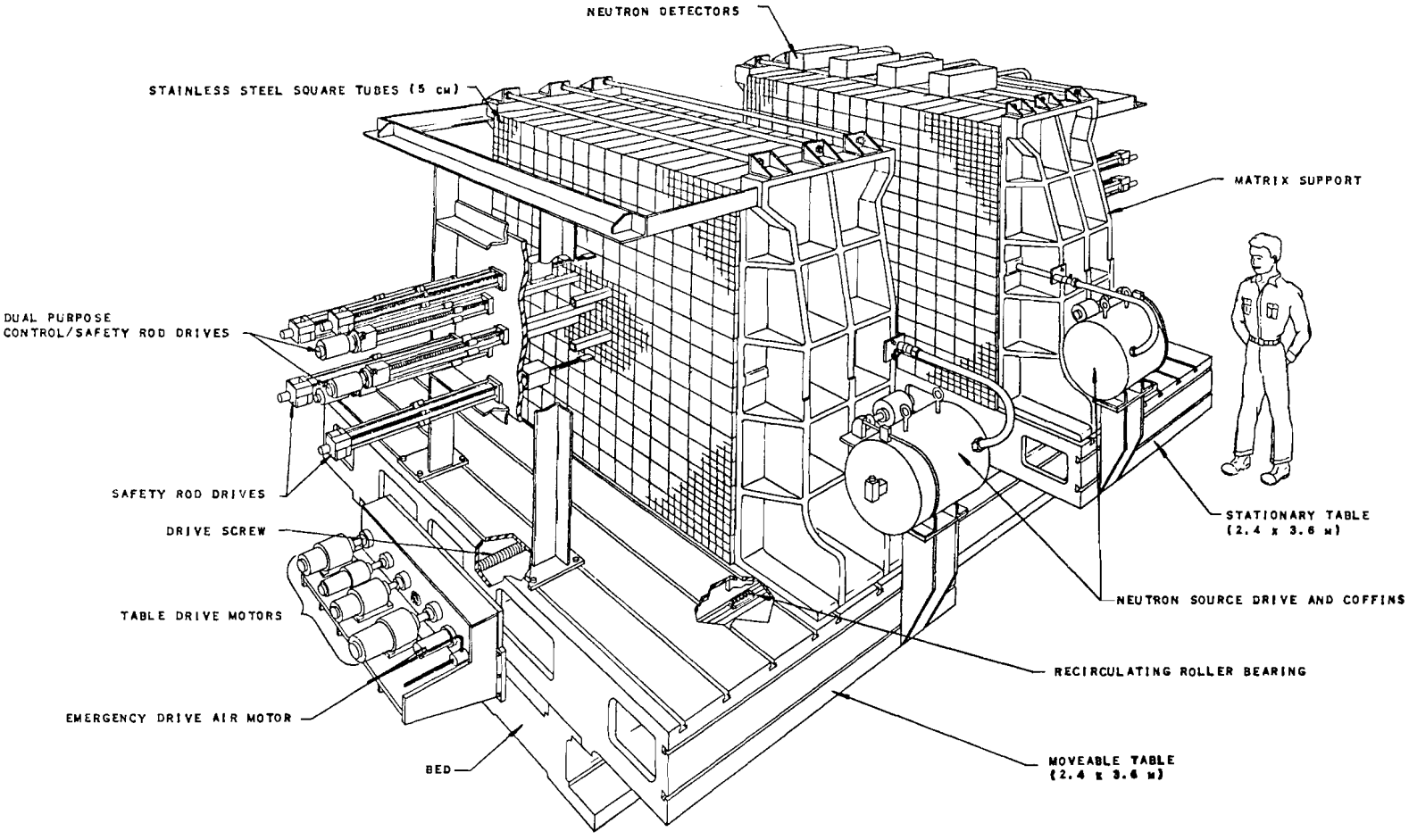


FIGURE 3. ZPR Reactor

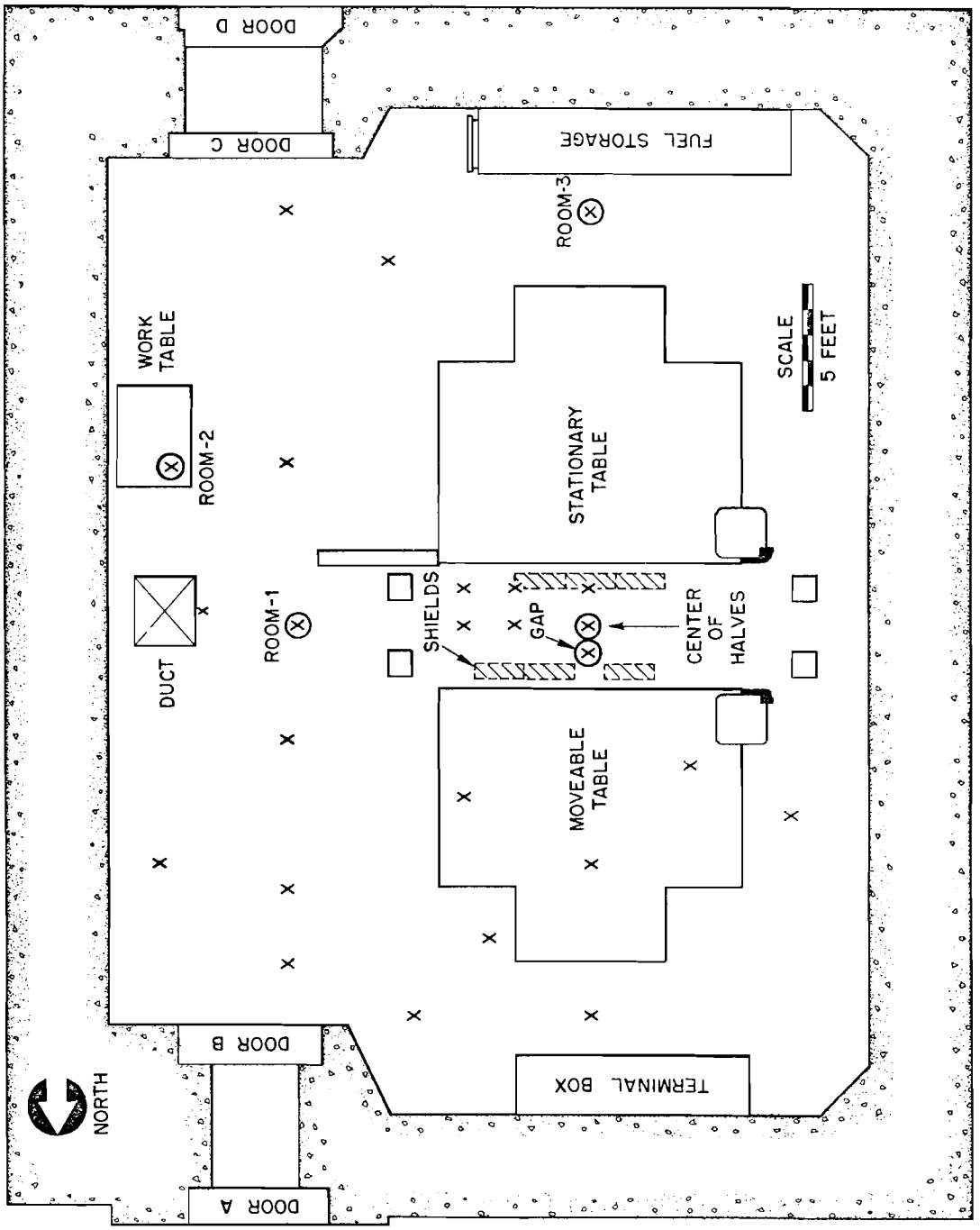


FIGURE 4. ZPR VI Cell

counters. These measurements can be used to determine the thermal flux and an equivalent intermediate-energy flux. Measurements were also made with a portable fast-neutron counter to determine the fast-neutron flux. These measurements alone give a considerable amount of information about the environment and for an unknown degraded fission source environment would probably allow one to interpret the dosimeter response. In addition to these measurements we made some spectrum measurements with proton-recoil proportional counters. The technique is rather sophisticated, and in this case the results are like frosting on the cake. However, they do indicate in a very graphic way the neutron environment. Figure 5 shows the spectrum between the halves. The ordinate is in units of flux per unit lethargy or E times of (E). We notice that the spectrum peaks at several hundred keV. Also shown on the slide is the energy dependent flux-to-dose conversion curve. We see that the spectrum extends over an energy region of rapid variation in the dose conversion curve. We actually found that for the various locations the dose per fast neutron varied over a factor of four. Figure 6 shows the spectrum between the halves when the shields are in place. The fast-neutron intensity is substantially reduced; the peak has also moved out to a higher energy. Figure 7 shows a spectrum measured in the room away from the reactor. We still notice that almost all the dose comes from high-energy neutrons. Figure 8 shows the effect of a human torso on the spectrum. The spectra were measured in front of a gap in the shields with and without a phantom present. The dose is only slightly changed by the presence of the phantom. However, scattering from the phantom substantially increases the number of low-energy neutrons. The track recorders, which are sensitive to low-energy neutrons, must be exposed on a phantom during calibration.

Table 1 shows the dose rate, thermal flux, equivalent 1/E flux, and fast flux observed at the circled positions. The dose rate varies from several hundred mrem/hour between the halves to less than one mrem/hour away from the reactor with the shields in place. We found that between the halves the presence of the phantom increases the thermal flux by a factor of twenty and the intermediate-energy flux by a factor of four. The number

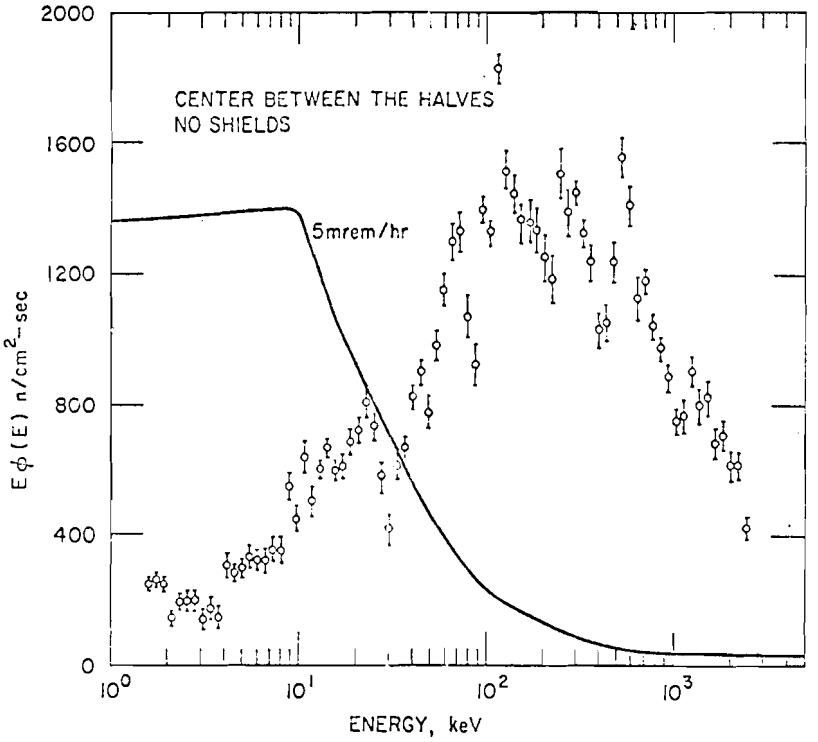


FIGURE 5

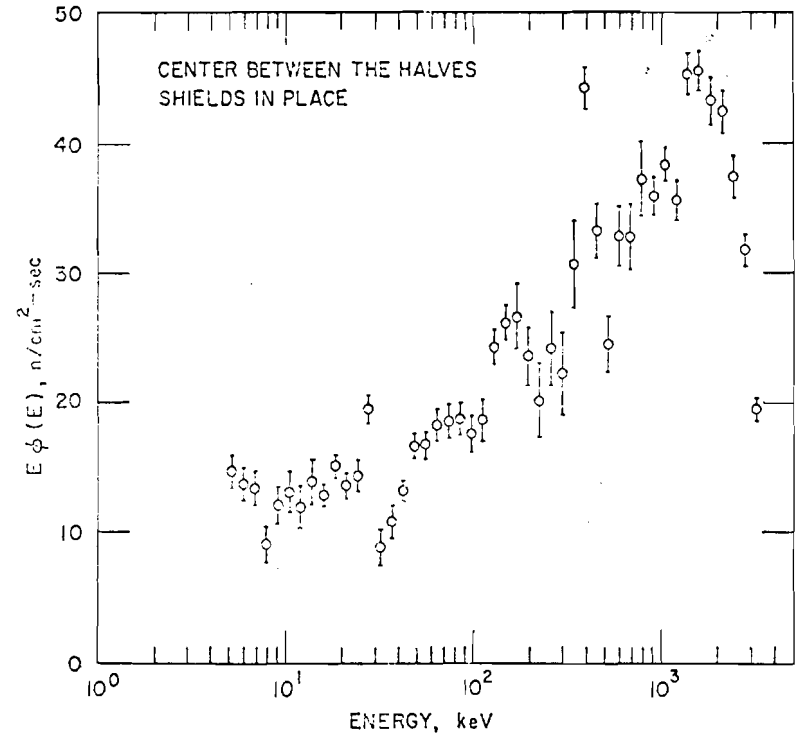


FIGURE 6

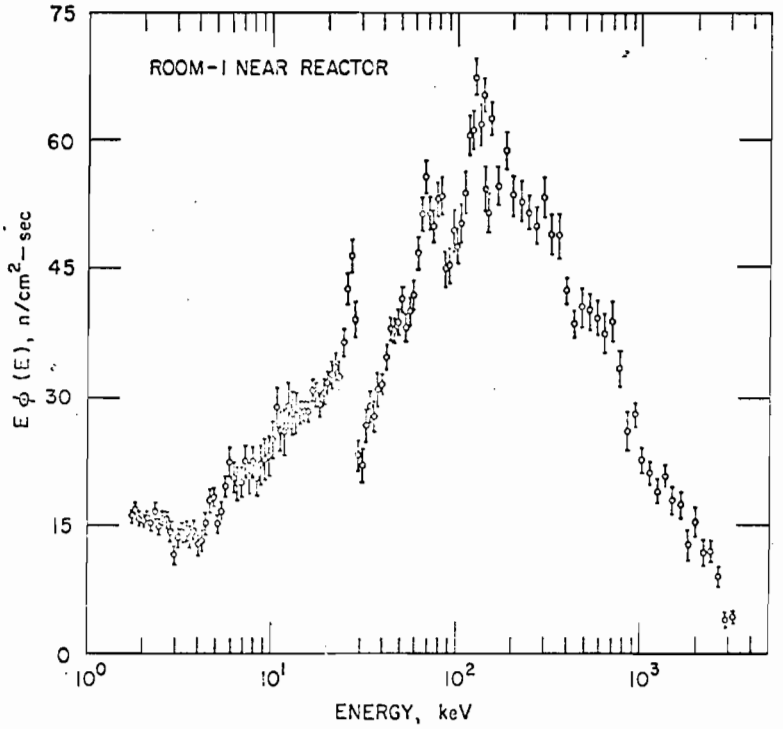


FIGURE 7

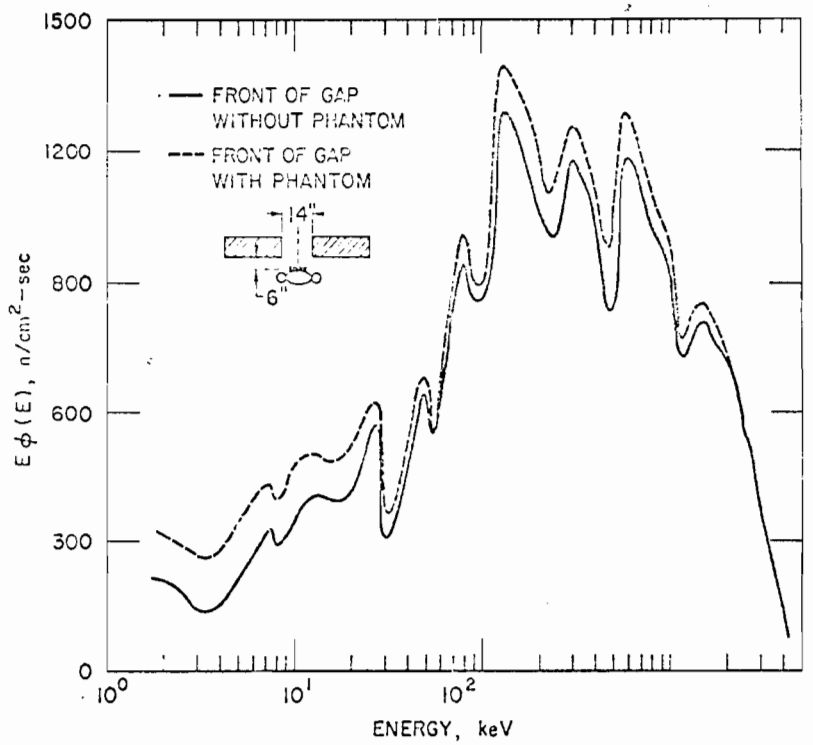


FIGURE 8

TABLE 1. Neutron Dosimetry Studies with a Phantom

| Location | Dose Rate mrem/hr | Thermal Flux $n/cm^2$-sec | I/E Flux $n/cm^2$-sec per unit lethargy | Fast Flux[(a)] $n/cm^2$-sec | Dose Rate / Thermal Flux | Dose Rate / 1/E Flux | Dose Rate / Fast Flux |
|---|---|---|---|---|---|---|---|
| Center | | | | | | | |
| Phantom ⊥ to axis | 307 | 360 | 70 | 3170 | 0.85 | 4.4 | 0.097 |
| Phantom // to axis | 431 | 250 | 55 | 3260 | 1.7 | 7.8 | 0.130 |
| Phantom ⊥ to axis shields in place | 7.4 | 104 | 5.7 | 93 | 0.07 | 1.3 | 0.080 |
| Front of gap | 340 | 560 | 100 | 3880 | 0.61 | 3.4 | 0.088 |
| Room No. 1 | 11.1 | 40 | 8.1 | 200 | 0.28 | 1.4 | 0.055 |
| Room No. 2 | | | | | | | |
| No shields | 7.4 | 34 | 6.4 | 130 | 0.22 | 1.2 | 0.057 |
| Shields in place | 0.77 | 11 | 1.4 | 25 | 0.07 | 0.6 | 0.031 |
| Room No. 3 | 3.2 | 24 | 5.1 | 75 | 0.13 | 0.6 | 0.043 |

(a) Neutrons with energy >100 keV.

of fast neutrons is more or less independent of the presence of the phantom except for shielding effects.

The last three columns are rather interesting. They show the dose rate divided by the thermal flux, 1/E flux, and fast-neutron flux. The largest variation occurs in the ratio of the dose rate to the thermal flux--a factor of twenty-five. There is less variation in the ratio of dose rate to 1/E flux, and the least in the ratio of dose rate to fast neutron flux. However, there is still a factor of four. This dramatically points out the need for a dosimeter with the proper energy response if doses are to be accurately determined.

Table 2 contains a summary of the track-recorder exposure data at these same locations. The first three columns contain track data obtained from these exposures. The next column gives the dose rate. The next two columns show the ratio of the dose rate divided by the aluminum and cadmium-covered track density rates. The variation in the aluminum-covered one is about the same as the ratio of dose rate to thermal flux. The ratio of dose rate to cadmium-covered one varies from 0.59 to 0.07, or about a factor of eight. This is consistent with the observation that the dose per fast neutron varies by a factor of four and that the dose per intermediate-energy neutron varies by a factor of thirteen. The last column shows the ratio of the dose rate inferred from the track recorders to the true dose rate. We notice that in the softer spectrum out in the room the dosimeter may overestimate the dose rate by a factor of six.

Figure 9 shows these results graphically, where we display the ratio of dose to cadmium covered track density as a function of the cadmium ratio. If we are to overestimate the dose, then the straight line must pass through the center no-shields point, which is the location at which the spectrum is the hardest, and the center shields point, where it is softest.

Let me now briefly state some conclusions. First, the limitations of this simple dosimeter have been clearly shown. We have also provided what I think is valuable spectral information to people who are considering various types of dosimeters which do not have an ideal energy response for

TABLE 2. SSTR Studies with a Phantom

| Location | $T_b/t$ tracks/ $cm^2$-hr | $T_{cd}/t$ tracks/ $cm^2$-hr | CR | $\dot{D}$ (a) mrem/hr | $\frac{\dot{D}}{(T_b/t)}$ | $\frac{\dot{D}}{(T_{cd}/t)}$ | $\frac{\dot{D}_{SSTR}}{\dot{D}}$ (b) |
|---|---|---|---|---|---|---|---|
| Center | | | | | | | |
| Phantom ⊥ to axis | 5107 | 820 | 6.2 | 307 | 0.060 | 0.37 | 1.2 |
| Phantom // to axis | 3159 | 725 | 4.4 | 431 | 0.136 | 0.59 | 0.82 |
| Phantom ⊥ to axis shields in place | 1424 | 44 | 33 | 7.4 | 0.005 | 0.17 | 1.0 |
| Front of gap | 7584 | 1251 | 6.1 | 340 | 0.045 | 0.27 | 1.7 |
| Room No. 1 | 561 | 76 | 7.4 | 11.1 | 0.020 | 0.15 | 3.1 |
| Room No. 2 | | | | | | | |
| No shield | 430 | 53 | 8.1 | 7.4 | 0.017 | 0.14 | 3.2 |
| Shields in place | 121 | 11 | 11 | 0.77 | 0.006 | 0.07 | 5.9 |
| Room No. 3 | 348 | 30 | 12 | 3.2 | 0.009 | 0.11 | 3.8 |

(a) Total dose rate, $\dot{D}$, is that given in Table 1.
(b) $\dot{D}_{SSTR} = [0.54 - 0.011\ (CR)]\ (T_{cd}/t)$

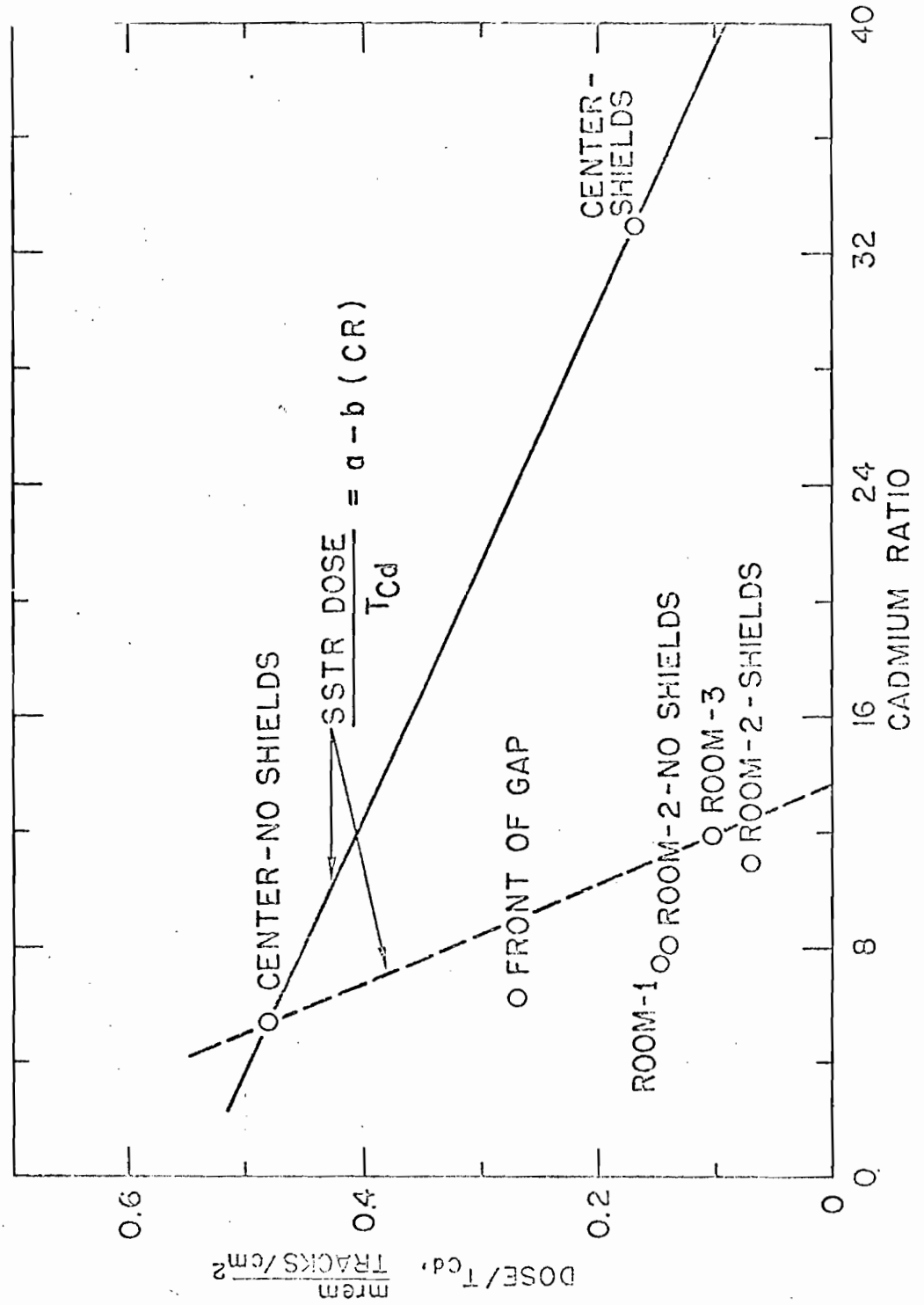
0.6
0.4
0.2
0
DOSE/$T_{Cd}$, $\frac{mrem}{TRACKS/cm^2}$
0
8
16
24
32
40
CADMIUM RATIO
CENTER-NO SHIELDS
$\frac{SSTR\ DOSE}{T_{Cd}} = a - b\,(CR)$
CENTER-SHIELDS
FRONT OF GAP
ROOM-1
ROOM-2-NO SHIELDS
ROOM-3
ROOM-2-SHIELDS

FIGURE 9

use in a degraded fission source environment. We have also established that we are adequately, although conservatively, monitoring the personnel at our facility. Depending on various factors, this type of monitoring system may be useful at other facilities.

Of course, we can ask the question if we were starting out all over again or if we had sufficient incentive to do more accurate dosimetry, what approach would we take? This is really the type of question we are trying to answer at this workshop. It seems first that we should really try to give film dosimeters another try using suitable precautions, since in principle it has adequate sensitivity and a rather good energy response. Next, we could try to decrease the low-energy sensitivity of the present dosimeter by the use of various moderators and absorbers, such as B-10. I am sure the dosimeter could be improved this way. There is also the possibility of using a threshold source. Np-237 is probably the most satisfactory. Becker at Oak Ridge has developed such a dosimeter. For our spectra one might get a response of about 2 mrem/track/$cm^2$, compared to the value of about 0.3 for the cadmium-covered U-235 recorder. Calculations indicate that the ratio of the dose inferred from such a dosimeter to the true dose would vary by a factor of two, rather than by a factor of six for the U-235 dosimeter. Of course, one must weigh the operational difficulties: the gamma dose from the foils, the production of the foils, and what I believe to be the most serious, the chance of alpha contamination.

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