PERSONNEL RADIATION DOSIMETRY SYMPOSIUM

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PROGRAM AND ABSTRACTS

Hyatt Regency Hotel
Knoxville, Tennessee
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FOREWORD

As radiation dosimetrists are well aware, national and international regulatory agencies are modifying requirements for personnel gamma, beta, and neutron monitoring. The most significant modifications will occur in areas affecting performance qualification programs, recommended dosimetry systems, dose equivalent estimation, and calibration. Considering the large amount of published and unpublished information concerning regulations, advances, and experience in these areas, it is very difficult for most dosimetrists to evaluate and adjust their programs to conform to the latest requirements and technologies.

This symposium has been designed with the objective of providing applied and research dosimetrists with sufficient information to evaluate the status and direction of their programs relative to the latest guidelines and techniques. To partially fulfill this objective, a technical program consisting of more than 60 oral presentations concerning experience, requirements, and advances in gamma, beta, and neutron personnel dosimetry has been prepared. Speakers represent a variety of international organizations including utilities, regulatory agencies, hospitals, universities, military services, vendors, and laboratories. Two special sessions will also be held to afford attendees the opportunity to make short presentations of recent work or to ask questions of general interest. A summary of the technical program and a collection of abstracts of presentations are included in this publication.

In addition to the formal presentations, the Technical Program Committee, which consists of internationally recognized dosimetrists with experience in a variety of areas, will be available for consultation throughout the conference. Vendors will display the latest commercially available monitoring systems and survey instruments in the lobby adjacent to the meeting rooms. Sufficient refreshment and lunch breaks have been planned to allow viewing of the vendor displays or consultation with symposium participants.

We believe that the conference program is informative and comprehensive and will prove valuable to all attendees. We are pleased to have you take advantage of this opportunity to determine the status and directions of personnel monitoring in general and to evaluate your program in particular.

Charles S. Sims
General Chairman
Oak Ridge National Laboratory

Richard E. Swaja
Technical Program Chairman
Oak Ridge National Laboratory

TABLE OF CONTENTS

TECHNICAL PROGRAM

PERSONNEL RADIATION DOSIMETRY SYMPOSIUM
Knoxville, Tennessee
October 15–18, 1984
TECHNICAL PROGRAM

All technical sessions will be held in the Regency Ballroom on the lower level of the Hyatt Regency Hotel. Technical presentations will last 15-20 minutes, refreshment breaks will last about 30 minutes, and about 1-2 hours will be available for lunch.

Monday, October 15

8:00 AM Registration

9:00 AM Welcome and Introduction to the Symposium
General Chairman - C. S. Sims (Oak Ridge National Laboratory)

9:20 AM Session I: Overview of Personnel Radiation Dosimetry
Chairman - W. M. Lowder (U.S. Department of Energy)

The Status of Personnel Radiation Dosimetry
D. E. Jones (Pacific Gas and Electric)

Personnel Dosimetry at Universities and Hospitals
C. M. Knight (Duke University Medical Center)

Personnel Radiation Dosimetry at Military Facilities
R. C. Nelson (U.S. Air Force)

Perspective from a Commercial Supplier of Personnel Radiation Dosimetry Services
R. C. Yoder (R. S. Landauer, Jr. and Co.)

Regulatory Actions for Improved Personnel Radiation Dosimetry
M. V. Federline (U.S. Nuclear Regulatory Commission)

IAEA Program in Occupational Personnel Monitoring
V. E. Aleinikov (International Atomic Energy Agency)

Lunch

1:30 PM Session II: Personnel Dosimetry Programs and Experience
Chairman - D. E. Jones (Pacific Gas and Electric)

The NRPB's Comprehensive Personal Monitoring Service: Experience and Future Prospects
T. O. Marshall (National Radiological Protection Board, UK)

System of the Operational Radiation Protection in the Federal Republic of Germany
H. Eckerl and G. Drexler (Gesellschaft fur Strahlen-und Umweltforschung, FRG)
G. Weimer (Federal Ministry of the Interior, FRG)

Technical Achievements and Trends of the Italian Group of Experts in Personnel Dosimetry
A. Cavallini (ENEA, Italy)
Italian Personnel Dosimetry Specialists

Berkeley Nuclear Laboratories Personal Monitoring Service
M. J. Hill (Central Electricity Generating Board, UK)

Refreshment Break

Collective Effective Dose Equivalent, Population Doses, and Risk Estimates from Occupational Exposure in Japan, 1983
T. Maruyama, Y. Kumamoto and Y. Noda (National Institute of Sciences, Japan)
N. Jutoh and S. Matsumoto (Japan Safety Appliances Association, Japan)
K. Nishizawa (Kyorin University, Japan)

Update - The NVLAP Dosimetry Processors LAP
R. L. Gladhill and J. Horlick (National Bureau of Standards)

Department of Energy Laboratory Accreditation Program in Personnel Dosimetry
T. F. Gesell (U.S. Department of Energy)
P. L. Roberson (Battelle Pacific Northwest Laboratory)
E. J. Vallario (U.S. Department of Energy)

7:00 PM Reception (William Blount Room)

Tuesday, October 16

8:30 AM Session III: Gamma and Beta Dosimetry
Chairman - J. P. Cusimano (U.S. Department of Energy)

A Survey of Gamma Personnel Radiation Dosimetry Programs at International Agencies
R. E. Swaja (Oak Ridge National Laboratory)

Evaluation of Beta Radiation Dose at Nuclear Power Plant Facilities
F. F. McWilliams and G. E. Chabot (University of Lowell)

Are Commercial Personnel Radiation Dosimeters Adequate for Monitoring Dose Equivalents of Personnel Working Around High Energy LINACS?
G. P Glasgow (Washington University School of Medicine)

Design and Function of Direct Reading Gamma and X-ray Dosimeters
L. L. Brackebush (Baird Corporation)

An Automated Data Recording System for Direct Reading Dosimeters
D. Riley (Dosimeter Corporation of America)
C. Clemmons (Pro-Tem, Inc.)

Refreshment Break

A Micro-Processor Controlled TLD Reader System
R. P. Bradley, D. Grogan, G. R. Symonds, R. J. St. Arnaud and L. Kelemen (Radiation Protection Bureau, Canada)
D. Johnston (National Research Council, Canada)

Dose Equivalent Evaluation for the Hanford Site Personnel Dosimetry Program
J. J. Fix, C. J. Card and M. K. Winegardner (Battelle Pacific Northwest Laboratory)

A Computer-Based Quality Assurance Trends Analysis Program for Dosimetry Processors
M. J. Sivertsen (Northeast Utilities Service Company)

Quality Control Features for Hanford Site Personnel Dosimetry System
J. J. Fix, C. J. Card and M. K. Winegardner (Battelle Pacific Northwest Laboratory)
G. R. Rao (U.S. Testing Company)

Special Case Exposure Monitoring at ORGDP
J. M. Mahathy (Oak Ridge Gaseous Diffusion Plant)

Lunch

1:30 PM Session IV: Gamma and Beta Dosimetry - Special Session
Chairman - T. F. Gesell (U.S. Department of Energy)

NBS Facilities for the Study of Radiation Protection Instruments
M. Ehrlich (National Bureau of Standards)

Studies of Dose Distribution in Radiation Fields Around Phantoms Using Personnel Dosimeters
M. Hofert and D. Wittekind (CERN, Switzerland)

Geometry Perturbation Corrections for the Measurements of Widely Distributed Beta Sources Using Extrapolation Chambers
E. L. Darois (Yankee Atomic Electric Company)
M. T. Scannell (University of Lowell)

Personnel Radiation Dosimetry in India Using TLD
S. J. Supe and R. K. Kher (Bhabha Atomic Research Centre, India)

Refreshment Break

An Assessment of a Commercial Individual Dosemeter Suitable for Low Penetrating Radiation
J. R. Harvey, J. R. W. Bates and B. Macfarlane (Central Electricity Generating Board, UK)

A Bidirectional Dosimeter for Measuring Gamma Radiation
R. I. Scherpelz (Battelle Pacific Northwest Laboratory)
F. Y. Tsang (EG&G Idaho, Inc.)

Two Energy-Independent Gamma-Ray Dosimetry Systems
C. E. Moss, E. J. Dowdy, M. E. Hamm and M. C. Lucas (Los Alamos National Laboratory)

Field Experience with a Dose Equivalent Rate Meter
J. L. Alvarez, B. L. Rich, L. O. Johnson, S. H. Daniel and D. E. Martz (EG&G Idaho, Inc.)

Wednesday, October 17

8:30 AM Session V: Neutron Dosimetry
Chairman - H. Ing (Atomic Energy of Canada, Ltd.)

The Status of International Neutron Personnel Radiation Dosimetry Programs
R. E. Swaja (Oak Ridge National Laboratory)

Neutron Personnel Dosimetry Calibration
R. B. Schwartz (National Bureau of Standards)

Neutron Fields for the Calibration of Neutron Dosimeters at the PTB
H. Lesiecki and M. Cosack (Physikalish-Technische Bundesanstalt, FRG)

Microdosimetric Measurements at Different Positions in the Primary Beam in the Irradiation Facilities at DREO
A. Taymaz, T Cousins and F. Szabo (Defense Research Establishment, Canada)

Neutron Calibration at the SEFOR Calibration Center
L. West and B. Brandon (University of Arkansas)

Refreshment Break

Calibration of 400M Exposure Facility
C. R. Heimbach (U.S. Army)

The Effect of Cadmium Covers on Bonner Sphere Responses
N. E. Hertel (University of Texas)
J. W. Davidson (Los Alamos National Laboratory)

Evaluation of Hand Exposure in Shielded Glove Boxes Using Wrist Albedo Neutron Dosimeters
D. E. Hankins (Lawrence Livermore National Laboratory)

Neutron Energy, Dose Rate, and Directional Response of a Commercial Remmeter
R. T. Greene (General Electric Company)
E. G. Bailiff (Oak Ridge National Laboratory)

Lunch

1:30 PM Session VI: Neutron Dosimetry - Special Session
Chairman - D. E. Hankins (Lawrence Livermore National Laboratory)

Neutron Dose Equivalent Estimation
C. S. Sims (Oak Ridge National Laboratory)

Fading of Tracks from High-Energy Hadrons in the NTA Emulsion
M. Hofert and F. Lehmann (CERN, Switzerland)

A Comparison of Fast Neutron Personnel Dosimeters Around an Electron Accelerator
A. Esposito and M. Pelliccioni (INFN Laboratori Nazionali di Frascati, Italy)

Thermoluminescence Albedo Neutron Dosimetry
M. S. Salem, J. W. McKlveen, and G. W. Klingler (Arizona State University)

Refreshment Break

Results, Developments, and Anomalies of a Panasonic TLD System
M. W. Lantz (Arizona Public Service)

On TL Residuals in LiF Above 300°C: Accumulation Effects and Their Minimization
C. W. King (Harshaw/Filtrol Partnership)
R. W. Pollock (Siemens Gammasonics)

Residual Neutron Signals in TLD-600 Ribbons Used in Automated Personnel Dosimetry Systems
C. J. Card, J. J. Fix and M. K. Winegardner (Battelle Pacific Northwest Laboratory)

Thursday, October 18

8:30 AM Session VII: Research and Advances in Personnel Radiation Dosimetry
Chairman - J. M. Aldrich (Rockwell International - Rocky Flats Plant)

The Assessment of Beta and Gamma Radiation Fields Using a New Multi-Element Personnel Dosimeter
F. S. Tsakeres and J. W. Poston (Georgia Institute of Technology)

A Compact Portable Radiac System for Dose Measurement and Documentation
P. T. Randtke, W. Wong, T. L. Young and L. A. Devigilli (GA Technologies, Inc)

The Development of an Intelligent Beta Survey Instrument
R. O. Murphy and J. W. Poston (Georgia Institute of Technology)

Experimental Studies of Solid State Detectors for Use in Discriminating Individual Dosemeters
P. Ambrosi, J. Bohm and U. Henkel (Physikalisch - Technische Bundesanstalt, FRG)
K. Ritzenhoff (Staatliches Materialprüfungsamt, FRG)

Fast Neutron Personnel Dosimeters Using Proton Recoil Proportional Counters
T. J. Yule and E. F. Bennett (Argonne National Laboratory)

Refreshment Break

Spectrometry for Neutron Dosimetry
A. E. Evans and A. A. Robba (Los Alamos National Laboratory)

Studies of a Digital Approach to Neutron Dosimetry and Microdosimetry
J. E. Turner, R. N. Hamm, H. A. Wright, G. S. Hurst and M. M. Chiles (Oak Ridge National Laboratory
R. T. Greene (General Electric Company)

Progress in the Design of Neutron Dosimeters Using Superheated Drops
R. E. Apfel and Y. C. Lo (Yale University)

Properties of Bubble-Damage Polymer Detectors
H. Ing and H. C. Birnboim (Atomic Energy of Canada, Canada)

Improvements in the Etching of CR-39 for Large-Scale Personnel Neutron Dosimetry
D. E. Hankins and S. G. Homann (Lawrence Livermore National Laboratory)

Lunch

1:30 PM Session VIII: Personnel Dosimetry - Needs and Directions
Chairman - R. E. Swaja (Oak Ridge National Laboratory)

Personnel Neutron and Gamma-Ray Dosimeter Intercomparison Studies at Oak Ridge National Laboratory
C. S. Sims (Oak Ridge National Laboratory)

Basic Requirements of Dosemeter Systems for Individual Monitoring
J. Bohm and P. Ambrosi (Physikalisch-Technische Bundesanstalt, FRG)

The Importance of Radiation Dosimetry in Radiation Compensation Litigation
B. P. Colby (American Nuclear Insurers)

Closing Statements
Symposium Technical Program Committee

ABSTRACTS OF PRESENTATIONS

Monday, October 15

9:20 AM

SESSION I: OVERVIEW OF PERSONNEL RADIATION DOSIMETRY

Session Chairman - Mr. Wayne M. Lowder
United States Department of Energy
Environmental Measurements Laboratory
New York, New York

Biographical Sketch

A graduate of Harvard University and the International School of Nuclear Science and Engineering, Mr. Lowder has been a physicist at the DOE Environmental Measurements Laboratory for about 30 years. He has been co-organizer of the three international symposia on the national radiation environment and a member of NCRP scientific committees on environmental radiation measurement and national background radiation in the United States. He is currently a member of the ICRP Task Group on External Radiation Dosimetry and is a program manager for radiation instrumentation and dosimetry for DOE's Office of Health and Environmental Research.

THE STATUS OF PERSONNEL RADIATION DOSIMETRY

D. E. Jones (Pacific Gas and Electric Company, Avila Beach, California 93424)

A historical perspective of personnel radiation dosimetry is presented along with a survey of recent regulatory requirements and monitoring techniques. Dosimetry practices are critiqued, and changes to existing standards are proposed. Perspectives presented in this talk are based on more than 20 years experience in the research, development, and application of personnel dosimetry systems for a variety of environments.

PERSONNEL DOSIMETRY AT UNIVERSITIES AND HOSPITALS

C. M. Knight (Duke University Medical Center, Durham, North Carolina 27710)

Few installations offer the diversity of personnel exposure situations as those typically found within the university/medical center environment. The inventory of ionizing radiation sources used at Duke University include: research accelerators; medical accelerators; radionuclide research laboratories, teletherapy sources; diagnostic X-Ray machines; brachytherapy sources; radiopharmaceuticals; X-Ray diffraction units; and sealed source irradiators. Each radiation source requires its own dosimetric considerations to include type of radiation (primarily beta, X and gamma, and neutron) and exposure conditions. A description of these considerations will be presented.

The Duke monitoring program was initiated in 1961 and currently includes approximately 2000 individuals. A review of badge readings over the past 5 years for selected high potential exposure groups (Radiology, Nuclear Medicine, Cardiac Catherization, Radiopharmacy, Radiation Oncology) and as related to the maximum permissible whole body dose reveals noteworthy decreases in personnel exposures. In 1983, more than 95% of the badges assigned to the high exposure groups exhibited readings of less than 10% of the MPD and greater than 80% of the badges showed exposures less than 5% of the MPD with the largest single reading being 11.8% MPD. Five years ago, the readings for these same groups were: 87% below 10% MPD and 72% below 5% MPD and the largest exposure was 55% MPD.

PERSONNEL RADIATION DOSIMETRY AT MILITARY FACILITIES

R. C. Nelson (USAF Occupational and Environmental Health Laboratory, Brooks AFB, Texas 78235)

Department of Defense related activities requiring personnel monitoring in accordance with applicable federal and service guidelines are carefully controlled. Within the Department of Defense, over 100,000 workers are provided personnel dosimetry annually which accounts for approximately one-third of all dosimetry provided by U.S. Federal agencies and one-tenth of the overall personnel dosimetry services provided within the United States.

Dosimetry for the military services is provided mainly by the Naval Dosimetry Center, Bethesda, Maryland for Navy activities, Lexington Bluegrass Army Depot, Lexington, Kentucky for Army activities and the USAF Occupational and Environmental Health Laboratory, Brooks AFB, Tx for Air Force activities. Each dosimetry center has participated regularly in national and international intercomparison studies and is presently engaged in certification efforts under the NVLAP umbrella.

Dosimetry services vary from film dosimetry to computer controlled automated thermoluminescent dosimetry systems with administrative record support varying from hand-scribed paper records through realtime computer data bases. Personnel supported by these programs are from all possible radiation specialties in the medical and industrial disciplines requiring capabilities in all forms of radiation dosimetry. Creation of new programs and public demand for technological advancements place the military services at the forefront of personnel dosimetry developments.

PERSPECTIVE FROM A COMMERCIAL SUPPLIER OF PERSONNEL RADIATION DOSIMETRY SERVICES

R. C. Yoder (R. S. Landauer, Jr. & Company, Glenwood, Illinois 60425)

The technological advances in electronic communications and information processing are allowing faster and easier assimilation of dosimetry data. The result is a reduction in the administrative and clerical burdens of radiation dosimetry programs and an increase in the amount of information that is easily available for the health physicist. Computers of commercial suppliers are directly accessible to users in remote locations so that information can be electronically sent and received. Program changes can be accurately implemented faster and problems of radiation exposure can be more quickly and easily recognized. The increased ability to process and transmit information will enable dosimeters to be developed that will provide more radiological information for the health physicist.

Other significant developments are the dosimetry performance testing and accreditation programs administered by the National Bureau of Standards and the Department of Energy. These programs have been beneficial by standardizing calibrations, testing protocols and quality requirements. However, the degree of standardization could be improved.

REGULATORY ACTIONS FOR IMPROVED PERSONNEL RADIATION DOSIMETRY

M. V. Federline (U.S. Nuclear Regulatory Commission, Occupational Radiation Protection Branch, Washington, D.C. 20555)

As part of a continuing program to improve health physics measurements, the Nuclear Regulatory Commission (NRC) is proposing amendments to 10 CFR 20 that would require its licensees to utilize the services of commercial or in-house dosimetry processors accredited by the National Bureau of Standards under the National Voluntary Laboratory Accreditation Program (49FR1205). Although data from the processing of personnel dosimeters are the basis for licensee and NRC records of external doses to workers, current regulations do not address the competency of dosimetry processors.

A brief discussion will be presented of the need for the proposed action which stems from concerns about inaccuracies in personnel dosimetry demonstrated during early performance tests and failure of voluntary programs to achieve corrective action through wide participation and acceptance by the dosimetry processing industry. Development of a consensus standard ANSI N13.11, "Criteria for Testing Personnel Dosimetry Performance," and results of a pilot study of the draft standard to evaluate the suitability of the criteria as a basis for regulatory action will be reviewed. Progress of interagency cooperation to coordinate correction of the dosimetry processor performance problem will be reported. The status of the NRC staff analysis of public comments on the proposed rule and staff recommendations regarding promulgation of a final rule will be reviewed.

Monday, October 15

9:00 AM

WELCOME AND INTRODUCTION TO THE SYMPOSIUM

Conference Chairman - Dr. Charles S. Sims
Health and Safety Research Division
Oak Ridge National Laboratory
Oak Ridge, Tennessee

Biographical Sketch

C. S. Sims received the B.S. degree in engineering physics and the M.S. and Ph.D. degrees in nuclear engineering from the University of Oklahoma. He is a member of the American Nuclear Society and has served on the Advisory Board of the Oak Ridge/Knoxville Section. Dr. Sims has had extensive experience in space radiation dosimetry, nuclear safety analysis, and nuclear reactor engineering. His current position is group leader of the Dosimetry Applications Research Group in the Health and Safety Research Division at Oak Ridge National Laboratory. Management and technical responsibilities include the Health Physics Research Reactor, neutron dosimetry research, and international dosimeter intercomparison studies.

IAEA PROGRAM IN OCCUPATIONAL PERSONNEL MONITORING

V. E. Aleinikov (International Atomic Energy Agency, Vienna, Austria)

A review of the IAEA program in occupational personnel monitoring for 1985-86 is presented. Following the issuing of new ICRP recommendations on occupational monitoring, work in this area will be expanded with the preparation of a guide on basic principles for personnel monitoring. Recommendations in support of this guide will be drawn up on the assessment of occupational exposure to external irradiation and on internal dose assessment. In addition Safety Series procedures and data will be issued for neutron spectra for application in radiation protection, the application of the dose-equivalent index quantity and the intercomparison of personnel dosimeters.

Monday, October 15

1:30 PM

SESSION II: PERSONNEL DOSIMETRY PROGRAMS AND EXPERIENCE

Session Chairman - Mr. Donald E. Jones
Pacific Gas and Electric Company
Avila Beach, California

Biographical Sketch

Don Jones began his work in dosimetry at the University of California Lawrence Livermore Laboratory in 1962 after spending two years in the Physics Department of Kansas State University doing research and teaching. He holds an M.S. in Physics from Kansas State University. At LLL, he led the development of the first automatic TLD dosimetry system and holds patents in TLD phosphor development and reading systems design. He has authored numerous publications in applied dosimetry and has managed personnel dosimetry programs at Lawrence Livermore Laboratory and the U.S. Department of Energy Idaho National Engineering Laboratory. He is one of the group of technical experts chosen by the NBS for its NVLAP Dosimetry Processors Accreditation Program. He is currently employed by Pacific Gas and Electric Company where he is setting up a personnel dosimetry program and manages part of the off-site emergency response program for the Diablo Canyon Nuclear Power Plant.

THE NRPB'S COMPREHENSIVE PERSONAL MONITORING SERVICE: EXPERIENCE AND FUTURE PROSPECTS

T. O. Marshall (National Radiological Protection Board, United Kingdom)

The NRPB operates a large scale personal monitoring service issuing over 700,000 dosemeters per year to persons in about 4,000 establishments in industry, research, teaching and the health service. The types of dosemeter available are designed to cope with all types of commonly produced ionising radiations from external sources.

Two options are available for monitoring the trunk for x-, β- and γ-rays-either a two-element thermoluminescent dosemeter or a photographic film dosemeter. The former provides information on dose only but the latter can provide additional information on the circumstances of the exposure.

The nuclear emulsion film is supplied for fast neutron monitoring. This type of dosemeter is suitable for energies greater than 0.5 MeV provided accompanying γ-radiations are not too intense. In some important situations this cannot be guaranteed so that a new type of dosemeter based on the observation of damage tracks in CR-39 plastic has been developed. An experimental service is now in operation in a nuclear fuel processing plant.

Extremity monitoring for x-, β-, and γ-rays is carried out with a thermoluminescent dosemeter in powder form. A new dosemeter developed by Vinten Instruments Ltd. has been evaluated by NRPB and will be put into service provided results of the present field trials are satisfactory. No dosemeter preparation is required and although at this stage it has to be manually placed in the reader, number recognition, processing and data handling is automatic, thus reducing the effort required and the possibility of errors. The β-ray energy threshold is also reduced from 0.5 MeV to about 0.2 MeV.

A computerised dose record keeping service is available which, if the customer requires, can meet his legal obligations for record keeping. The service has been extremely successful in reducing the clerical work of the employer, in improving accuracy and data acquisition and in making transfer records and the like readily available.

Until recently, personal monitoring had been carried out entirely by Government-sponsored organizations. Commercial interest has been slight and as a result customers have been offered economic but rather standard and rigid services. More commercial interest is now evident and services will be required to accommodate more customer requirements. Technical improvements will be required and financial and administrative arrangements will have to be more versatile. The demand for personal monitoring is expected to continue since, as well as satisfying legislation, it also serves to reassure the workforce and provides useful protection for the employer.

SYSTEM OF THE OPERATIONAL RADIATION PROTECTION MONITORING IN THE FEDERAL REPUBLIC OF GERMANY

H. Eckerl and G. Drexler (Gesellschaft für Strahlen- und Umweltforschung, Neuherberg, FRG)

G. Weimer (Federal Ministery of the Interior, Bonn, FRG)

The legal basis of radiation protection in the FRG relies on several laws and regulations. To comply with regulations in everyday work, a system of operational radiation protection exists which covers all radiation protection aspects from the very first stages of planning to routine monitoring procedures. In particular, all aspects of the physical radiation protection control are outlined in a code of practice.

As an example of the system, a description is given of the operational, dosimetric and administrative aspects of personnel monitoring performed by the operators as well as of a centralized governmental personnel dosimetry system of about 200,000 monitored persons.

Emphasis is given to quality assurance and quality control measures. For the governmental system, the performance standards not only cover technical aspects of the measuring devices but also the reporting to users and health authorities. Quality control is made by intercomparison measurements with a primary standards laboratory. For the personnel monitoring done by the operators where a variety of measuring devices can be used, quality assurance is basically ruled by a law on metrology and verification. According to this law, any instrument used in radiation protection has to fulfil clearly defined requirements.

Special emphasis is further given to the interpretation of the monitoring results. The dose limits for occupationally exposed persons are given as effective dose equivalent or organ doses. In most cases, the registered personnel doses are well below the limits and an investigation level is defined above which the measured personnel doses have to be converted into the limiting quantities. Since the monitoring results also shall be used as a tool to improve the standard of radiation protection in the sense of the ALARA-principle, statistical evaluations are made. This is done both by the monitoring services and the supervising authority. Some results of these statistical evaluations will be shown.

TECHNICAL ACHIEVEMENTS AND TRENDS OF THE ITALIAN GROUP OF EXPERTS IN PERSONNEL DOSIMETRY

A. Cavallini (ENEA, Bologna, Italy)
Italian Personnel Dosimetry Specialists

Since the early 1960's, specialists of the larger personnel dosimetry services in Italy have met at regular intervals under the auspices of ENEA (formerly CNEN). The aim was to exchange views and set up intercalibration programs directed at spotting and possibly solving technical and management problems of the services, then mostly based on photographic films.

In particular the following items have been examined in detail:

1) Preparation of the calibration set,
2) Interpretation and meaning of the individual dosemeter data,
3) File structure and record keeping, and
4) Approval.

Under the heading "Approval", the Group means the establishment of verification procedures of photographic dosimetry methods aimed at controlling the reliability of dosimetric services at a nationwide level and the identification of the best techniques to suggest to the national services. With "Record keeping", the expert group intends to set up a national data base of dosimetric results for epidemiological studies. The "Preparation of the calibration set" is directed at the standardization of the calibration procedures among the services. Under the heading "Interpretation and meaning of the individual dosimetric data" are included the discussions on the unification of the format used by dosimetry services to communicate the data to the users.

This work describes in detail the activities and the conclusions reached by the Group.

BERKELEY NUCLEAR LABORATORIES PERSONAL DOSIMETRY SERVICE

M. J. Hill (CEGB, Berkeley Nuclear Laboratories, United Kingdom)

This presentation describes the Personal Dosimetry Service operated by the Berkeley Nuclear Laboratories of the CEGB. The Service, which is 'approved' by the Nuclear Installations Inspectorate (NII), caters to about 40 establishments mainly within the CEGB but including some outside organizations.

The personal dosemeters used are film badges, fast neutron badges and thermoluminescent dosimetry (TLD) devices employing lithium fluoride in sachets for body extremity measurements and 'albedos' for whole-body neutron dose measurements. The methods adopted for personal dosimetry using these dosemeters are detailed. The report also describes the computer-based Radiation Records System.

COLLECTIVE EFFECTIVE DOSE EQUIVALENT, POPULATION DOSES AND RISK ESTIMATES FROM OCCUPATIONAL EXPOSURES IN JAPAN, 1983

T. Maruyama, Y. Kumamoto and Y. Noda (National Institute of Sciences, Chiba-shi, Japan)

N. Jutoh and S. Matsumoto (Japan Safety Appliances Association, Japan)

K. Nishizawa (Kyorin University, Japan)

In Japan, most personnel monitoring is still done by means of film badges with some thermoluminescent dosimeters are also being used. In order to use the historical results from this monitoring for population dose and risk estimates, organ or tissue doses were experimentally measured with a phantom using 30, 80 and 140 keV X-rays, and ^{60}Co and ^{137}Cs gamma sources. Individual effective dose equivalents of workers were calculated on the basis of the concept of effective dose equivalent recommended by the ICRP using the organ or tissue doses experimentally determined and the film badge readings. The genetically significant doses to the population from occupational exposures in Japan were estimated based on a nationwide survey. The risks of stochastic effects to individuals and the Japanese population from occupational exposures were estimated using the data on the annual collective dose equivalent of individuals and the population doses, respectively. The present status of personnel monitoring in Japan will also be discussed.

UPDATE - THE NVLAP RADIATION DOSIMETRY PROCESSORS LAP

R. L. Gladhill and J. Horlick (National Bureau of Standards,
Washington, DC 20234)

On January 1, 1984, the National Voluntary Laboratory Accreditation Program (NVLAP) officially began the evaluation of radiation dosimetry processors leading to NVLAP accreditation. The accreditation of individual processors is based on the satisfaction of predetermined published criteria. The first group of processors (17 eligibles) to successfully complete the evaluation process will be granted accreditation in the Fall of 1984. Additional applications will continue to be received.

The NVLAP evaluation process consists of two key elements; proficiency testing of dosimeters in accordance with ANSI N13.11 and on-site assessment in accordance with NVLAP criteria. Results from the early rounds of proficiency testing will be presented using the N13.11 pass/fail criteria and also using interlaboratory comparison techniques. Findings of the on-site assessments will be discussed.

The regulatory impact of the accreditation program continues to develop. To date, the Nuclear Regulatory Commission (NRC) has not published a final ruling on its proposal to require NRC licensees to use accredited dosimeter processors. Pending regulatory actions by other agencies will be discussed.

DEPARTMENT OF ENERGY LABORATORY ACCREDITATION PROGRAM IN PERSONNEL DOSIMETRY

T. F. Gesell (U.S. Department of Energy, Idaho National Engineering Laboratory, Idaho Falls, Idaho 83401)

P. L. Roberson (Battelle Pacific Northwest Laboratory, Richland, WA 99352)

E. J. Vallario (U.S. Department of Energy, Washington, D.C. 20545)

An accreditation program for personnel dosimetry is being established by the U.S. Department of Energy. This program is designed exclusively for DOE and DOE contractor facilities but is similar in concept to the National Voluntary Laboratory Accrediation Program (NVLAP) in dosimetry. Like NVLAP the DOE program will be based on a standard, in this case the "DOE Standard for the Performance Testing of Personnel Dosimetry Systems." The accreditation process will include performance testing of dosimeters and on-site inspections of processing facilities. The program differs from NVLAP in several significant ways: 1) More stringent performance requirements are imposed. 2) The photon tests will include mono-energetic photons in addition to bremsstrahlung spectra in order to simulate plutonium environments. 3) The beta test will include thallium-204 and uranium slab sources in addition to the strontium- yttrium-90 source used by NVLAP. 4) The neutron test will include unmoderated californium-252 in addition to the deuterium oxide moderated californium source used by NVLAP. 5) Information derived from the testing program will be used to guide the DOE research program in dosimetry.

The program will be directed by DOE Headquarters. The DOE's Radiological and Environmental Sciences Laboratory located at the Idaho National Engineering Laboratory will administer the program and serve as the lead performance testing laboratory. Pacific Northwest Laboratory, which was instrumental in initiating the program and preparing the documentation, will make some of the performance test exposures until such time as the Idaho facility is fully equipped.

Tuesday, October 16

8:30 AM

SESSION III: GAMMA AND BETA DOSIMETRY

Session Chairman - Mr. John P. (Jan) Cusimano
United States Department of Energy
Idaho National Engineering Laboratory
Idaho Falls, Idaho

Biographical Sketch

After being discharged from the US Air Force in 1955, Jan worked at the Rome Air Development Center in radar electronic research while attending Syracuse University. Upon receiving a degree in Engineering Physics in 1961, he joined the AEC Health and Safety Laboratory in Idaho Falls where he performed research in film dosimetry and supervised an HP portable instrument section. In 1964, Cusimano concentrated his efforts in thermoluminescence (TL) dosimetry research and was instrumental in implementing the first TL personnel dosimetry system in the US at the Idaho National Engineering Laboratory (INEL) in 1966. Since that time, he has performed research in albedo, environmental, high level and nuclear accident dosimetry. He has been responsible for over 20 publications and presentations in the dosimetry area. Cusimano holds memberships in the American Nuclear Society, the Health Physics Society and has chaired or served on numerous committees. He is presently the senior physicist in the Dosimetry Branch of the Radiological and Environmental Sciences Laboratory at the INEL. He serves as a dosimetry technical expert for the National Voluntary Laboratory Accreditation Program administered by the National Bureau of Standards.

A SURVEY OF GAMMA PERSONNEL RADIATION DOSIMETRY PROGRAMS AT INTERNATIONAL AGENCIES*

R. E. Swaja (Oak Ridge National Laboratory, Oak Ridge, Tennessee 37831)

In September of 1983, a mail survey was conducted to determine the status of external gamma personnel dosimetry at international agencies. A total of 130 agencies (102 from the United States and 28 from other countries) including military, regulatory, university, hospital, laboratory and utility facilities participated in this study. All responding agencies are or will be conducting personnel gamma monitoring in programs which involve about 500,000 workers. The survey requested information concerning basic dosimeter types, calibration sources, calibration phantoms, corrections to basic dosimeter responses, evaluating agencies, monitoring periods and ranges of typical or expected dose equivalents. Results of this survey are presented and discussed.

*Research sponsored by the Division of Pollutant Characterization and Safety Research, U.S. Department of Energy under contract DE-AC05-84021400 with Martin Marietta Energy Systems, Inc.

EVALUATION OF BETA RADIATION DOSE IN NUCLEAR POWER PLANT FACILITIES*

F. F. McWilliams and G. E. Chabot (University of Lowell, Radiation Laboratory, Lowell, Massachusetts 01854)

The University of Lowell, under contract with Yankee Atomic Electric Company, is evaluating the beta radiation doses from various environments within operating nuclear power facilities. The objectives are ultimately to provide recommendations and methodologies to the licensee for the protection of workers exposed to beta radiation fields.

Evaluation of beta radiation doses to a worker involves full characterization of beta radiation fields at distances that would constitute normal exposure conditions for an individual. This characterization involves beta spectroscopy, gamma spectroscopy, and extrapolation chamber measurements at those distances through various degrees of protective clothing.

It is also the intent of this work to evaluate the response of various beta radiation portable health physics instruments to these characterized beta radiation fields and to investigate the possibility of using samples of reactor coolant suspended solids (crud) as calibration sources in conjunction with comparison measurements made with the large area sources of beta emitting radionuclides available at the University of Lowell. From this, we intend to provide simple instrumental techniques to qualitatively determine the spectral quality of the beta radiation fields.

The results of this work are presented with conclusions and recommendations for the protection of workers exposed to beta radiation fields from various environments within operating nuclear power facilities.

*This work is supported by Yankee Atomic Electric Company.

ARE COMMERCIAL PERSONNEL RADIATION DOSIMETERS ADEQUATE FOR MONITORING DOSE EQUIVALENTS OF PERSONNEL WORKING AROUND HIGH ENERGY LINACS?

G. P. Glasgow, (Mallinckrodt Institute of Radiology,
Washington University School of Medicine, St. Louis, Missouri 63110)

Our therapy facility consists of a cobalt unit, and Clinac 4, 6, 18 and 35 MeV linacs. From 1978 to 1980, we monitored personnel using conventional commercial monthly film badge dosimeters. For 53 employees the average annual whole body dose equivalent (DE) was 1.0 mSv. In 1981, we changed to thermoluminescent (TL) dosimeters, which, for most employees, were collected quarterly. For 67 employees monitored in 1981 and 1982, the average annual whole body dose equivalent was 1.9 mSv, almost double the prior value. An increase was noted in almost all employee groups. In 9 quarters prior to changing to TL dosimeters only 2 of 123 quarterly whole body dose equivalents of technician operators exceeded 1.25 mSv. In the 9 quarters after the change to TL monitors 30 of 163 quarterly whole body dose equivalents of technician operators exceeded 1.25 mSv. The number of patient treatments was relatively constant from '78 to '82. Were there new sources of radiation in our clinic or is it the result of changing dosimeters?

To better understand the radiation environment in our clinic, area monitoring was performed in 1982 and 1983. Envelopes containing film and TL dosimeters were placed at selected locations throughout the clinic and left for a quarter. Near the low energy linacs, the DE from films and TLs were identical within the statistical uncertainties of the two systems. But near the high energy linacs, the DE measured with the TL badges are about twice as high as those measured with the film. A closer review of personnel DE indicated a clear pattern when technicians were assigned to work on our Clinac 20 and Clinac 35 they received a higher DE when wearing TL monitors than when at these same locations wearing film badges.

Now, how should one interpret these data? Were the film badges systematically under responding around the higher energy linacs? Are the TL badges over responding around the high energy linacs resulting in artifically higher dose equivalents? The question is unresolved; it appears that near our high energy linacs the type of personnel monitor will determine whether or not technicians *appear* to comply with or exceed an ALARA I level of 1.25 mSv. Possible explanations for these data will be presented along with additional monthly and quarterly area and personnel dosimetry dose equivalents currently being obtained with both monthly and quarterly film and TL monitors.

DESIGN AND FUNCTION OF DIRECT READING GAMMA AND X RAY DOSIMETERS

L. L. Brackebusch (Baird Corporation, Rockford, Maine 01730)

This paper deals with the general purpose and requirements of direct reading quartz fiber gamma and X-ray pocket dosimeters. It emphasizes the need for upcoming and inservice inspections for the user to conform with applicable ANSI standards.

This is followed by a general description and function of the various components of the dosimeter and how they interact with one another. The need for very precise manufacturing procedures of minute parts with extremely low capacitance values are discussed. The theory of linear electrical discharge and how it relates to dose is explained. The effects of external influences such as temperature, humidity and shock will be covered.

There are variants in dosimeters for specific applications. Examples of polarity and effects of wall thickness will be given. The paper concludes with comments applications and various ranges that are commercially available.

AN AUTOMATED DATA RECORDING SYSTEM FOR DIRECT READING DOSIMETERS

D. Riley (Dosimeter Corporation, Cincinnati, Ohio 45242)

C. Clemmons (Pro-Tem Inc., Seabrook, Texas 77566)

A *C*omputer *I*nterfaced *D*osimeter (CID) Reader has been built, tested, and put in use. CID is capable of both reading and charging direct reading dosimeters. It utilizes a mini computer and data from this unit can be directly interfaced to the central health physics main frame computer. CID's use in a calibration facility is also discussed.

A MICRO-PROCESSOR CONTROLLED TLD READER SYSTEM

R. P. Bradley, D. Grogan, G. R. Symonds, R. J. St. Arnaud and L. Kelemen
(Radiation Protection Bureau, Department of National Health
and Welfare, Ottawa, Canada)

D. Johnston (National Research Council, Ottawa, Canada)

As part of the development of a dosimetry system for the monitoring of exposures to external gamma sources received by personnel working in the uranium mine and milling environments, it was decided that the Occupational Radiation Hazards Division of the Radiation Protection Bureau would undertake to design and construct its own thermoluminescence dosimeter reader. Since 1976, the Division has used three readers of the type designed by the Chalk River Nuclear Laboratories to evaluate the dosimeters used in its national monitoring programme. Using the basic hardware configuration of this system, a microprocessor-based electronics package was designed and incorporated into the new instrumentation. The microprocessor used was the model LC-1 manufactured by Cybernex Limited of Ottawa, Canada. Besides allowing for more extensive monitoring of reader functions during operation and a reduced need for direct technical supervision, the system now provides for the acquisition and storage of digitized thermoluminescence glow curves. This data will be used in direct analysis and verification of the exposure results. Details of the new instrument design as well as performance testing results are presented.

DOSE EQUIVALENT EVALUATION FOR THE HANFORD SITE PERSONNEL DOSIMETRY PROGRAM

J. J. Fix, C. J. Card and M. K. Winegardner (Battelle Pacific Northwest Laboratory, Richland, Washington 99352)*

Dose equivalent evaluation for the Hanford Site Personnel Dosimetry Program is based on processing procedures to control or minimize the influence of known sources of error. Dosimeter responses to selected calibration sources, normalization of the reader sensitivity to the mean response of selected control dosimeters and an energy compensating algorithm are used to interpret dose equivalent in four dose components (nonpenetrating, penetrating, fast neutron and slow neutron). Chip sensitivity factors have been determined from replicate exposures for each dosimeter. These factors are used to adjust the observed chip results during processing. The sensitivity of the reader is controlled at all times to within ± 5 percent. This is critically necessary to correctly relate the chip sensitivity factors to the dosimeter results during processing. The algorithm compensates for any variation in reader response by chip position during processing. Calibration sources are chosen which are directly traceable to NBS and site calibration factors are used where necessary. Acceptability of dosimeter results is determined from comparison of calculated doses with known given doses using an extensive audit procedure.

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*Pacific Northwest Laboratory is operated for the U.S. Department of Energy by Battelle Memorial Institute.

A COMPUTER-BASED QUALITY ASSURANCE TRENDS ANALYSIS PROGRAM FOR DOSIMETRY PROCESSORS

M. J. Sivertsen (Northeast Utilities Service Company,
Hartford, Connecticut 06141-0270)

Northeast Utilities Service Company performs in-house processing of thermoluminescent dosimeters (TLD's) worn by radiation workers at the company's three nuclear power stations. The Quality Assurance (QA) Trends Analysis Program tracks and analyzes QA dosimeter data and various calibration factors generated by the Northeast Utilities Personnel Dosimetry Laboratory. These trends are not readily observable on a day-to-day basis with the laboratory's operational quality control criteria.

The Trends Analysis Program provides and documents in a concise, easily understandable format the following: 1) TLD reader performance and stability over extended time intervals, 2) the impact of reader calibrations and laboratory operations on reported doses, 3) a periodic check on laboratory quality control criteria, and 4) information which will aid in the fulfillment of National Voluntary Laboratory Accreditation Program criteria.

The collection of QA dosimeter data and subsequent analysis have been computerized. A graphical and statistical analysis is performed with a program written by the author using the Statistical Analysis System (SAS) language. The SAS program is run quarterly on an IBM 370 mainframe computer here at Northeast Utilities. The graphics output for the past three months provides a detailed visual identification of trends which have recently developed for a particular TLD reader. A graphical and statistical summary analysis for the same TLD reader over the past twelve months is also performed. Correlation and regression analyses contained in the SAS program aid in determining whether recent trends are random in nature or due to a developing systematic error in TLD reader performance and/or laboratory operations.

Isolated operational and QA areas for improvement, not previously noticed, have been detected with the Trends Analysis Program. The overall effect has been to emphasize the importance of individual efforts to the continued quality and credibility of Dosimetry Laboratory results. TLD reader calibrations and other laboratory operations have come under closer scrutiny due to the documented impact of such operations on reported doses. The Dosimetry Laboratory QA program now provides both day-to-day and long-term documentation of our processing performance.

QUALITY CONTROL FEATURES FOR THE HANFORD SITE PERSONNEL DOSIMETRY SYSTEM

J. J. Fix, C. J. Card and M. K. Winegardner (Battelle Pacific Northwest Laboratory, Richland, Washington 99352)*

G. R. Rao (U.S. Testing Company, Richland, Washington 99352)

Several quality control features of the Hanford Site Dosimetry System will be presented. These include 1) the parameters monitored during the automated microprocessor-controlled readout system, 2) the dosimeter assignment, use and dose history, and 3) statistical evaluations of reader and dosimeter parameters for bias and precision. Control values are preset in the microprocessor to stop routine processing whenever the monitored results exceed these values. Control values are established for several characteristics of the reader such as high voltage and dark current as well as expected results for background and 1-R exposed dosimeters. The assignment of all dosimeters at Hanford is recorded in a dosimeter history file. This file includes the chip sensitivity factors, the data used to determine these factors, each assignment of the dosimeters and the dose results for each dosimeter readout. Statistical evaluations are conducted for each processing to determine the acceptability of the dosimeter processing, the distribution about zero dose for background control dosimeters and the distribution of doses by dose component for each Hanford contractor. Details for each of these areas will be presented.

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*Pacific Northwest Laboratory is operated for the U.S. Department of Energy by Battelle Memorial Institute.

Tuesday, October 16

1:30 PM

SESSION IV: GAMMA AND BETA DOSIMETRY - SPECIAL SESSION

Session Chairman - Dr. Thomas F. Gesell
United States Department of Energy
Idaho National Engineering Laboratory
Idaho Falls, Idaho

Biographical Sketch

For the past 3½ years, Tom has been employed by the U.S. Department of Energy as Chief of the Dosimetry Branch at the Idaho Falls National Engineering Laboratory. He spent 10 years teaching and conducting health physics research, including dosimetry, at the University of Texas and held a simultaneous adjunct appointment at Rice University. He holds a PhD in Physics and belongs to the Health Physics Society. He is the author of numerous papers and articles and has done standards committee work for HPS, NCRP and ANSI.

NBS FACILITIES FOR THE STUDY OF RADIATION PROTECTION INSTRUMENTS

M. Ehrlich (Center for Radiation Research, National Bureau of Standards, Gaithersburg, Maryland 20899)

Historically, radiation instrument calibration at NBS first was directed mainly toward the needs of the medical community. But even as long as three decades ago there was considerable envolvement in the study of both survey meters and pocket-ionization chambers for use in bremsstrahlung beams produced at exciting potentials up to ~1.4 MV. Standardization on certain fixed bremsstrahlung beams including narrow spectra suited for obtaining instrument response functions evolved gradually, and cesium-137 and cobalt-60 gamma-ray beams were added. Extension to other radiation fields for instrument calibration is quite recent at NBS and was carried out in response to needs expressed mainly by other U.S. Government agencies.

An account will be given of the NBS radiation facilities currently available for the study of radiation protection instruments. Covered in addition to the customary gamma-ray beams and the recently updated bremsstrahlung beams will be the new ~6 MeV essentially monoenergetic photon beam produced by the $^{19}F(p,\alpha\gamma)^{16}O$ reaction in the positive-ion Van de Graaf accelerator; the beta-particle beams (prometheum-137, thallium-204, and strontium-90/yttrium-90); the essentially monoenergetic electron beams covering the energy range from ~0.2 to 2.5 MeV produced in the electrostatic accelerator and the electron Van de Graaff accelerator; and the facility for instrument studies in clouds of radioactive gases. The characterization of the facilities in terms of absorbed dose rate to water at the point of instrument calibration and in terms of spectral distributions will be described. Examples will be given of the results of studies of instruments in these radiation facilities.

STUDIES OF DOSE DISTRIBUTION IN RADIATION FIELDS AROUND PHANTOMS USING PERSONNEL DOSIMETERS

M. Höfert and D. Wittekind (CERN, Geneva, Switzerland)

Personnel monitoring in radiation fields arising from induced radioactivity around the CERN proton accelerators is presently performed with film badges. Their calibration has so far been based on a depth dose equivalent as measured with a thick-walled tissue equivalent ionization chamber while in other European laboratories personnel dosimeters are generally calibrated in exposure free in air.

In the United States, an ANSI standard aims to approach the practical situation by introducing a calibration on a phantom and at the same time the notions of depth and surface doses.

At CERN, results of routine personnel monitoring were compared with calibrations both in free air and on a phantom performed in radiation fields from induced radioactivity. The experiments gave a better appreciation of calibration philosophies and variations of dose with depth. At the same time, the distribution of the radiation fields around the body in typical working environments was studied. The paper will discuss the point that neither the free-in-air nor the phantom calibrations reflect the practical exposure situation. There is also a difference of up to a factor of two between surface and depth dose and strong gradient of attenuation within the body which is a pronounced function of the irradiation conditions. In the light of these results, an approach to effective dose equivalent will be discussed.

GEOMETRY PERTURBATION CORRECTIONS FOR THE MEASUREMENTS OF WIDELY DISTRIBUTED BETA SOURCES USING EXTRAPOLATION CHAMBERS

E. L. Darois (Yankee Atomic Electric Company, Framingham, Massachusetts 01701)

M. T. Scannell (University of Lowell Radiation Laboratory, Lowell, Massachusetts 01854)

The extrapolation chamber is considered a primary reference standard for the measurements of electron dose. Typically, a user will calibrate the chamber against a known "point" source with reference to the NBS or the PTB. Once calibrated, it is used universally to measure doses from any source without regard to its spatial distribution. This work presents the results of measurements made by the NBS extrapolation chamber and the Yankee Atomic Chamber to a 9-inch diameter Tl-204 slab source. These results show excellent agreement with the NBS collecting electrode diameter of 30.48 mm and the Yankee diameter of 60 mm. However, as the Yankee collecting electrode diameter is decreased, the measured dose rate increases by approximately 10% at "0" diameter. This effect is described as a geometry perturbation effect due to the viewing angle of the chamber to the source. Due to the geometrical design differences of the NBS and Yankee chambers, for a collecting electrode diameter of 60 mm (Yankee) the viewing angle of each chamber is nearly identical.

Given this hypothesis of a geometry perturbation, a mathematical model was developed based upon fundamental principles which allows for correction of extrapolation chamber measurement to distributed sources. The input to the model requires a detailed description of the source to detector geometry and an estimate of the apparent absorption coefficient. A comparison is made of the calculated surface dose rate from various energy slab sources to the measured dose rates with and without the perturbation correction. The magnitude of the correction is up to 50% depending upon the input conditions. It is shown that this correction is also pertient and necessary for depleted uranium slab measurements.

It is also shown that point source calibrations of extrapolation chambers are only useful for verification of system operability. If measurements are made of distributed sources without corrections for the perturbation effects then a significant underestimation of dose may occur.

PERSONNEL RADIATION DOSIMETRY IN INDIA USING TLD

S. J. Supe and R. K. Kher (Bhabha Atomic Research Centre, Bombay, India)

A TLD Personnel Monitoring (PM) badge basically designed as a beta-gamma badge using $CaSO_4$:Dy phosphor embedded in a teflon thermoluminescent dosimeter disc is being used by more than 6000 radiation workers. From detailed studies on various aspects of fabrication parameters and use of TL dosimeters, quality control procedures have been developed. Thus, large numbers of discs of uniform sensitivity could be fabricated and individual calibration of discs could be avoided. Appropriate procedures for (1) determining nature of radiation, (2) evaluating badges exposed to low energy photon radiation, and (3) evaluating beta dose in the presence of both low energy and high energy gamma radiation are discussed.

Results of intercomparison studies during a field experiment lasting for more than two years among the TLD badge, film badge and quartz fibre pocket dosimeter indicated good agreement.

The ICRP recommendation on the dose limits are in terms of effective dose equivalent (EDE) whereas personnel doses as recorded by a PM badge essentially give surface dose on the body. To estimate EDE, doses to individual tissues at risk were evaluated from TLD Dosimeters worn on various body parts by workers from different categories of radiation installations. The conversion factors to obtain the EDE from chest dose are 0.6, 0.77 and 0.86 for personnel in diagnostic X-ray departments, teletherapy departments and nuclear reactor facilities, respectively.

It is concluded that the TLD badge system including associated conversion factors is a reliable, economic and convenient system for personnel radiation dosimetry.

AN ASSESSMENT OF A COMMERCIAL INDIVIDUAL DOSEMETER SUITABLE FOR LOW PENETRATING RADIATION

J. R. Harvey and J. R. W. Bates (CEGB, Berkeley Nuclear Laboratories, United Kingdom)

B. Macfarlane (CEGB, Hartlepool Nuclear Power Station, United Kingdom)

An assessment has been made of a commercially available disposable extremity dosemeter. The sensitive element, which is located on a plastic strip, utilises crystals of thermoluminescent lithium fluoride adhering to a small piece of high-temperature self-adhesive tape. The dosemeters are read out on a reader which can be used for a number of other dosemeters. Experiments were undertaken to ascertain reproducibility, linearity, background, fading and beta energy response as well as operational aspects. It is concluded that the dosimetric performance is very good but that operational convenience could be improved.

A BIDIRECTIONAL DOSIMETER FOR MEASURING GAMMA RADIATION*

R. I. Scherpelz (Pacific Northwest Laboratory, Richland, Washington 99352)

Francis Y. Tsang (EG&G Idaho, Inc., Idaho Falls, Idaho 83415)

A new dosimeter has been developed to measure doses due to gamma radiation in a light-water reactor environment. This dosimeter was designed to determine two components of the dose corresponding to radiation emitted by sources in two opposite directions from the dosimeter. Thus, the dosimeter could be exposed on the outside surface of a tank, and it would measure components of the dose due to radiation emitted by material inside the tank and due to radiation emitted by sources outside the tank.

The dosimeter contains two cylindrical inserts, each holding three TLD-700s. These inserts are set into holes drilled in opposite faces of a lead separator. The inserts are constructed of layers of brass, aluminum, and polyethylene, designed to minimize interfering TLD responses due to secondary radiation produced in the lead separator.

The bidirectional gamma dosimeter was calibrated by performing a number of exposures using radioisotopic sources containing ^{60}Co and ^{137}Cs. The exposures were performed with the dosimeter in several different positions, providing different incident angles for the radiation striking the front face of the dosimeter. A computer code was then written to combine the results of the calibration exposures into a simulated isotropic source of radiation. This code determined calibration factors to adjust a TLD response for interference from radiation that penetrated the separator, and to convert the adjusted TLD response to a dose. Calibration studies were performed for an unshielded source, and for sources with various thicknesses of steel shielding between the source and dosimeter.

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*Work performed for the U.S. Department of Energy under Contract DE-AC06-76RLO 1830.

TWO ENERGY-INDEPENDENT GAMMA-RAY DOSIMETRY SYSTEMS

C. E. Moss, E. J. Dowdy, M. E. Hamm and M. C. Lucas (Los Alamos National Laboratory, Los Alamos, New Mexico 87545)

Spectral-dependent response is a serious problem in gamma-ray survey instruments especially below 150 keV and above 3 MeV. We describe two systems based on bismuth-germanate scintillators that avoid this problem. The first system, consisting of eight scintillators and a LeCroy 3500 data acquisition system, can measure spectra at eight positions simultaneously. The second, consisting of a single scintillator and a Canberra 10 analyzer, can only measure a single position but is much more portable. We calibrated and characterized the systems from 0.06 to 8.29 MeV by using gamma-ray spectra from a variety of radioactive sources and nuclear reactions. By fitting pulse-height distributions with a function containing 17 parameters, we determined theoretical response functions that we then used to obtain the gamma-ray flux spectra from a variety of radioactive objects of interest to nuclear safeguards. We used a flux-spectrum-to-dose-rate conversion curve to obtain dose-rate spectra. The integrals of these spectra give the total gamma-ray dose rates. For a composite source, consisting of several sources with accurately known strengths, the results of our procedure agreed with the expected value to within less than 10%. Our systems are especially well suited for measurements at low dose rates, where survey instruments lose precision because of the high efficiency of bismuth-germanate scintillators.

SPECIAL CASE EXPOSURE MONITORING AT ORGDP

J. M. Mahathy (Oak Ridge Gaseous Diffusion Plant,
Oak Ridge, Tennessee 37831)*

A special evaluation program has been undertaken at the Oak Ridge Gaseous Diffusion Plant (ORGDP). The Special Case Exposure Monitoring Program (SCEMP) involves the use of thermoluminescent dosimetry (TLD) to monitor radiation exposures to the extremities and external penetrating radiation to the gonads and bone marrow. Data on extremity exposure is needed to re-evaluate the relationship of extremity exposure to whole body exposure for low-energy beta and gamma emissions from uranium daughters. Several employees are being monitored for exposures to the hand and wrist. In addition, some of these employees are being monitored for whole body exposure. Ratios of wrist to hand exposure and whole body to wrist exposure will be derived for future use. Estimates of penetrating exposures to internal gonads and bone marrow will also be estimated.

Through SCEMP, a method is being tested to monitor technetium exposures to personnel. Technetium-99 is a low energy beta emitter and is a source of potential skin exposure in some ORGDP equipment. Results and findings of SCEMP will be provided.

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*Based on work performed at the Oak Ridge Gaseous Diffusion Plant operated by Martin Marietta Energy Systems, Inc., for the U.S. Department of Energy under Contract DE-AC05-84021400.

FIELD EXPERIENCE WITH A DOSE EQUIVALENT RATE METER

J. L. Alvarez, B. L. Rich, L. O. Johnson, S. H. Daniel and D. E. Martz
(EG&G Idaho, INEL, Idaho Falls, Idaho 83415)

A dose rate instrument has been developed that measures shallow and deep dose equivalent for beta and gamma radiations. The response is energy and dose rate independent. Several of these instruments have been used at various facilities with a broad range of radiation exposure fields. The instrument's use and response was evaluated under these differing conditions. It was compared to commercially available instruments used in the facilities as well as in their dosimetry programs. The fields at the facilities were characterized using beta and gamma spectrometry, and the dose rate measured with an extrapolation chamber. The results of the evaluations were very favorable and showed the short comings of many dosimetry programs. They also suggest methods of improvement.

Wednesday, October 17

8:30 AM

SESSION V: NEUTRON DOSIMETRY

Session Chairman - Dr. Harry Ing
Atomic Energy of Canada
Chalk River Laboratory
Chalk River, Ontario, Canada

Biographical Sketch

Dr. Ing has worked at the Chalk River Nuclear Laboratories as a research scientist since 1969 except for one year when he was at the Stanford Linear Accelerator Center as a visiting foreign scholar. His research is mainly in calculational and experimental neutron dosimetry. He has performed numerous calculations of neutron spectra pertinent to radiation protection many of which were compiled in an IAEA technical report "Compendium of Neutron Spectra in Criticality Accident Dosimetry". He was also responsible for calculating the dosimetric characteristics of the D_2O moderated ^{252}Cf assembly currently used for the calibration of personal neutron dosimeters in the United States and recommended by the International Standards Organization. He has worked with a variety of neutron spectrometers and dosimeters including gaseous and liquid scintillation counters, various activation detectors, damage track detectors and TLD's. Most recently, he has developed a new type of detector called the bubble damage polymer detector for neutron dosimetry which has attracted wide-spread interest. He is frequently consulted by national and international groups regarding neutron-related problems or activities, and lectures and writes often on this topic. He has served as an editor for the Health Physics Journal and is currently on the editorial board of Radiation Protection Dosimetry and on a scientific committee of the NCRP.

THE STATUS OF INTERNATIONAL NEUTRON PERSONNEL RADIATION DOSIMETRY PROGRAMS*

R. E. Swaja (Oak Ridge National Laboratory, Oak Ridge, Tennessee 37831)

A mail survey was conducted in September of 1983 to determine the status of external neutron personnel dosimetry at international agencies. A total of 105 agencies (80 from the United States and 25 from other countries) participated in this study including military, regulatory, university, hospital, laboratory, and utility facilities. All responding agencies are or will be conducting neutron personnel monitoring in programs that involve about 72,000 workers. Information concerning basic dosimeter types, calibration sources, calibration phantoms, corrections to dosimeter responses, evaluating agencies, dose equivalent reporting conventions, ranges of typical or expected neutron dose equivalents, and degree of satisfaction with existing systems was requested from the participants.

Results of this survey indicate that to provide the best possible occupational radiation monitoring programs and to improve dosimetry accuracy in performance studies, facility dosimetrists, regulatory and standards agencies, and research laboratories must act within their areas of responsibility to become familiar with their total radiation monitoring systems, establish common reporting guidelines and performance standards, and provide opportunities for dosimetry testing and evaluation. Other information obtained from this survey is presented and discussed.

*Research sponsored by the Division of Pollutant Characterization and Safety Research, U.S. Department of Energy under contract DE-AC05-84021400 with Martin Marietta Energy Systems, Inc.

NEUTRON PERSONNEL DOSIMETRY CALIBRATION

R. B. Schwartz (National Bureau of Standards, Gaithersburg, Maryland 20899)

In principle, neutron dosimetry calibration is a very straight-forward procedure: the device to be calibrated is placed at a convenient distance from a neutron source of known emission rate and irradiated for a known time. From the emission rate of the source, the distance, and the time, the neutron fluence at the device is calculated. Using a conventional fluence-to-dose-equivalent conversion factor, the dose equivalent to which the device has been exposed is then calculated. This calculated dose equivalent divided by the reading on the device may then be considered to be the calibration factor. Unfortunately, this simple procedure will generally give the wrong answer since the effects of room-scattered neutrons, source emission anisotropy, and even air scattering can have a profound effect on the instrument reading.

In this talk, we shall discuss procedures for correcting for these effects when testing and calibrating neutron dosimeters and remmeters with radioactive neutron sources. We will give explicit analytic expressions for correcting for room scatter, compare the results so obtained with those obtained using shadow cones, and discuss the strengths and weaknesses of different methods of making these corrections. Emphasis will be placed on the importance of adopting uniform procedures so that calibrations will be a function only of the instrument type and the source energy spectrum and not depend on such extraneous factors as the source-detector distance or the room size.

NEUTRON FIELDS FOR THE CALIBRATION OF NEUTRON DOSIMETERS AT THE PTB

H. Lesiecki and M. Cosack (Physikalisch-Technische Bundesanstalt, Braunschweig, FRG)

For several years, the Physikalisch-Technische Bundesanstalt, Fed. Rep. of Germany, has had facilities for the calibration of neutron dosimeters with monoenergetic neutrons produced by nuclear reactions with a Van de Graaff accelerator, reactor-filtered beams and radionuclide neutron sources. Monoenergetic neutrons between 1 keV and 20 MeV are obtained via the reactions:

$^{45}Sc(p,n)^{45}Ti$ (E_n: 1 keV - 40 keV);
$^{7}Li(p,n)^{7}Be$ (E_n: 140 keV - 600 keV);
$T(p,n)^{3}He$ (E_n: 1 MeV - 2.5 MeV);
$D(d,n)^{3}He$ (E_n: 3.2 MeV - 6 MeV);
$T(d,n)^{4}He$ (E_n: 14.8 MeV - 20 MeV).

The range of ISO-recommended energies and sources is essentially covered. Special care was taken to determine the properties of the neutron fields. The fluence, and from it the dose equivalent (using appropriate fluence to dose equivalent conversion factors), is determined with a proton recoil proportional counter below 2.5 MeV and with a proton recoil telescope above 2.5 MeV. The influence of neutron backscattering from the walls can be kept small as the experimental hall is large (30 m $\times$ 24 m $\times$ 14 m) and the target is arranged 6 m above the solid ground. The neutron spectra containing the 'monoenergetic line', additional neutrons from target contaminations, neutron scattering from the target construction, neutron scattering from the walls and other constructions were measured with the time-of-flight technique using a pulsed ion beam. The facility will be described and problems encountered in the calibration of dosimeters will be discussed.

The PTB as a national laboratory is responsible for maintaining, improving and passing on standards-in this case the neutron fluence. We therefore took part in several international fluence intercomparisons organized by the BIPM, Paris, in the years from 1975 - 1984. A European workshop on a neutron dosimeter intercomparison was organized at the PTB in 1977. We participated in irradiations for the joint US-EC personal dosimeter intercomparison EMNIP II in 1982. Besides intercomparisons, we perform irradiations for various European laboratories.

MICRODOSIMETRIC MEASUREMENTS AT DIFFERENT POSITIONS IN THE PRIMARY BEAM IN THE IRRADIATION FACILITIES AT DREO

A. Taymaz, T. Cousins and F. Szabo (Defense Research Establishment, Ottawa, Ontario, Canada)

Microdosimetric measurements have been taken at different positions in the irradiation facility at Defence Research Establishment-Ottawa to assess the contribution of scattered neutrons from the walls of the irradiation room to the total spectrum. Neutrons were produced by a Van de Graaff accelerator employing the $T(d,n)^4He$ and $D(d,n)^3He$ reactions. Microdosimetric event distributions for monoenergetic and polyenergetic neutrons have been determined from the measurements of ionization with a modified version of 4" Rossi counter. Total microdosimetric event distributions for given neutron energies have been obtained in a single measurement. To do this, an electronic system has been developed in this laboratory to handle different gains. Event spectra were taken at cavity sizes of 1, 2 and 5 μm. Microdosimetric distributions of $yf(y)$ and $y^2f(y)$ as a function of y are displayed for given cavity sizes and neutron energies.

NEUTRON CALIBRATIONS AT THE SEFOR CALIBRATION CENTER

L. West and B. Brandon (University of Arkansas, Fayetteville, Arkansas 72701)

The SEFOR Calibration Center of the University of Arkansas is being developed into a neutron radiation calibration facility. Primary emphasis has been on the calibration of neutron remmeters and personnel dosimeters used at nuclear power plants. Calibrations are performed using Cf-252 neutron sources which have been calibrated by the National Bureau of Standards. To simulate power reactor environments, a D_2O-moderated source configuration is utilized as per ANSI N13.11. Neutron irradiations are performed in two calibration rooms. A high dose rate facility has been constructed in the heavily shielded refueling cell of the decommissioned SEFOR fast reactor. Low and intermediate dose rate calibrations are performed in an adjacent building.

When neutron detector calibrations are performed in an enclosed room, the flux of neutrons reflected from the room surfaces has a significant effect on the response. The fraction of response due to the wall-return fluence rate must be removed from the total response if the calibration factor is to be dependent only on the characteristics of the detector and of the source. The wall-return response can be determined experimentally if it is assumed that the total detector response is composed of a direct response to source neutrons which varies as r^{-2}, a wall-return response which is constant throughout the room, and a small contribution from air-scattered neutrons which varies as r^{-1}. The results of wall-return response measurements for several types of detectors in the SEFOR facilities are presented and compared with calculated values. General calibration procedures are also reviewed.

CALIBRATION OF 400M EXPOSURE FACILITY

C. R. Heimbach (U.S. Army Aberdeen Proving Ground, Maryland 21005)

The Army Pulse Radiation Division (APRD) of Aberdeen Proving Ground is characterizing a simulated tactical nuclear environment suitable for both in-phantom dosimetry and shielding measurements of material. A position 400m from the APRD radiation source has been selected as giving a suitable compromise of spectrum, intensity, and angular distribution.

Both neutron and gamma-ray spectra have been measured at the 400m position using an NE-213 liquid scintillator. The low-energy cutoff in the measured neutron spectrum has been extrapolated below 600 keV by use of bare and Cadmium-covered BF3 counters. The neutron spectrum has also been measured with a set of Bonner spheres. The neutron spectra agree well. Both neutron and gamma spectra have reached equilibrium at this distance in that they match spectra measured at much larger distances.

The spectra have been integrated to give kerma values and compared with independent kerma measurements made with a tissue-equivalent ion chamber and a Geiger-Muller counter. These measurements agree to within 10% and give values of 3.4 mR/kw-hr for neutrons and 1.4 mR/kw-hr for gamma rays. This is sufficient intensity for performing health physics-type dosimetry in place as well as for performing transmission-factor measurements for vehicles.

The angular distributions of neutron and gamma-rays are currently being measured. Indications are that there is still a strong line-of-sight component coming directly from the source. The significance of this forward cone of higher intensity and higher energy neutron radiation is currently being investigated. Measurements are in progress to relate in-phantom dosimetry to free-field and surface phantom dose values, and to compare with calculations.

THE EFFECT OF CADMIUM COVERS ON BONNER SPHERE RESPONSES

N. E. Hertel (University of Texas, Austin, Texas 78712)

J. W. Davidson (Los Alamos National Laboratory, Los Alamos, New Mexico 87545)

Responses for Bonner spheres have recently been calculated at The University of Texas at Austin using the 171 neutron group DLC-41/VITAMIN-C cross-section library and the code ANISN. The responses were calculated in a manner similar to the approach employed by Sanna (HASL-267) using the adjoint technique of Hansen and Sandmeier (NSE *22*, 315). These calculations were performed for both 4 mm × 4 mm ϕ and 12.7 mm × 12.7 mm ϕ LiI detectors in the center of moderating polyethylene spheres. The detector configuration was modeled in one-dimensional spherical geometry based on the Ludlum LiI detector assembly design. Both the $^{6}Li(n,\alpha)$ and $^{7}Li(n,n'\alpha)$ cross-sections were included in the detector response and were weighted so that the collision probability for the equivalent spherical detector matched that of the actual cylindrical detector.

Responses were calculated for the bare detector and for the detector at the center of 5.08 cm, 7.62 cm, 12.7 cm, 20.32 cm, 25.40 cm, 30.48 cm, 38.10 cm, and 45.72 cm dia. moderating spheres of 0.95 g/cm^3 polyethylene. Responses were also calculated for the detector and the 5.08 cm, 7.62 cm, and the 12.7 cm dia. spheres surrounded by 30 mils of cadmium.

The differences between the cadmium-wrapped sphere responses have been compared to the responses obtained without cadmium wraps. As expected all the cadmium-wrapped sphere responses are drastically reduced below ~1.0 eV. Minor differences in the responses are also visible in the 100 eV to 1 keV region where cadmium cross-section resonances occur. For neutrons above 1 MeV, the sensitivity of the moderated detectors is increased by the cadmium wrap. A significant increase in sensitivity is observed above 9 MeV for the 5.08 cm and 7.62 cm spheres. At 10 MeV the sensitivity of the 5.08 cm dia. sphere is increased by ~25% while at 14 MeV it is increased by ~60%. This effect is negligible for the cadmium-wrapped detector, highest for the 5.08 cm dia. moderator, and fairly low for the 12.7 cm dia. moderator.

EVALUATION OF HAND EXPOSURE IN SHIELDED GLOVE BOXES USING WRIST ALBEDO NEUTRON DOSIMETERS*

D. E. Hankins (Lawrence Livermore National Laboratory, Livermore, California 94550)

We studied the response of wrist albedo neutron dosimeters inside simulated glove boxes. Three boxes measuring 3 × 3 × 5 ft. were made with ½, 1½, and 3¾ in.-thick polyethylene walls; port holes were cut into the sides and end of the box to permit the placement of arm phantoms inside the box. We placed a ^{252}Cf neutron source at four different locations in the box. The albedo dosimeters were placed on the arm phantoms which were inserted into the port holes. Thus, we measured the exposure at various angles around the arm. We measured the neutron dose rate using the 9-in. sphere remmeter and determined the 9/3 in. sphere ratio and the thermal neutron dose using the same instrument. Differences exist in the readings of the dosimeters for various source locations in the glove box, distances from the source, angles around the arm, and for each of the three polyethylene glove box thicknesses. We developed a procedure for evaluating the albedo readings and estimated the agreement obtainable between the readings of the 9-in. sphere dose and the albedo dosimeter readings for a real exposure situation.

*This work was performed under the auspices of the U.S. Department of Energy by the Lawrence Livermore National Laboratory under contract number W-7405-ENG-48.

NEUTRON ENERGY, DOSE RATE, AND DIRECTIONAL RESPONSE OF A COMMERCIAL REMMETER*

R. T. Greene (General Electric Company, St. Petersburg, Florida 33733)

E. G. Bailiff (Oak Ridge National Laboratory, Oak Ridge, Tennessee 37831)

Portable neutron remmeters are the most widely used monitoring instruments at nuclear facilities. Their popularity stems from the notion that these instruments have a response which is done equivalent over a wide range of neutron energies. Remmeters are often used to calibrate neutron dosimeters and in some cases the remmeter response (mrem/h) multiplied by worker stay time is used to assign the dose equivalent.

One such popular instrument is the Eberline Portable Neutron Rem Counter (Model PRS-1P/NRD and HP-280). The NRD is a 23 cm (9 in) diameter cadmium loaded polyethylene sphere and BF_3 detector which has a reported dose equivalent response from thermal to 10 MeV. The HP-280 is a 7.6 cm (3 in) diameter cadmium covered sphere which is used with the NRD to provide information about the neutron energy spectrum. The ratio of sphere responses (NRD/HP-280) has been correlated to the energy dependence of TLD albedo neutron dosimeters and is used to determine a calibration factor which is applied to the response of the albedo dosimeter.

Studies have been performed using the Health Physics Research Reactor and isotopic neutron sources to evaluate the energy, orientation, and dose rate response of this instrument. These studies indicate that the remmeter calibration factor (cpm per mrem/h) is very dependent on the incident neutron energy spectrum. These experiments also show that the ratio of sphere responses depends on the detector orientation and dose rate.

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*Research sponsored by the Division of Pollutant Characterization and Safety Research, U.S. Department of Energy under contract DE-AC05-84021400 with Martin Marietta Energy Systems, Inc.

Wednesday, October 17

1:30 PM

SESSION VI: NEUTRON DOSIMETRY - SPECIAL SESSION

Session Chairman - Mr. Dale E. Hankins
Special Projects Division
Lawrence Livermore National Laboratory
Livermore, California

Biographical Sketch

Dale Hankins has worked extensively on various aspects of neutron dose measurement for radiation protection and the design and development of personnel neutron dosimeters. He first became interested in neutron dosimetry while working in the Health Physics Special Projects Group of the Phillips Petroleum Company at the National Reactor Testing Station in Idaho from 1959 to 1961. He accepted a position as an Applied Health Physicist at the Los Alamos National Laboratory where he stayed until 1975. During this time, he continued working on neutron instrument development, criticality accident dosimetry and personnel neutron dosimetry. In 1975, he moved to the Lawrence Livermore National Laboratory as an Applied Health Physicist and in 1982 transferred to the Hazards Control Department, Special Projects Division as a Research Physicist. He is presently working on the field application of CR-39 as a personnel neutron dosimeter. He has served as the Chairman for the DOE Workshop on Personnel Neutron Dosimetry and on various ANSI and DOE advisory committees. He has about 70 papers and publications, most concerned with neutron monitoring and personnel dosimetry.

NEUTRON DOSE EQUIVALENT ESTIMATION*

C. S. Sims (Oak Ridge National Laboratory,
Oak Ridge, Tennessee 37831)

The concept of dose equivalent causes much confusion among neutron dosimetrists. This is partly because of the lack of uniformity in dose equivalent terminology and partly because of the availability of several sets of neutron fluence-to-dose equivalent conversion factors each developed using widely differing phantoms, different computer codes, and different basic physical data. Various methods of dose equivalent estimation are examined in detail and several sets of fluence-to-dose equivalent conversion factors are compared. Specific examples of the application of the conversion factors are presented.

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*Research sponsored by the Division of Pollutant Characterization and Safety Research, U.S. Department of Energy under contract DE-AC05-84021400 with Martin Marietta Energy Systems, Inc.

FADING OF TRACKS FROM HIGH-ENERGY HADRONS IN THE NTA EMULSION

M. Höfert and F. Lehmann (CERN, Geneva, Switzerland)

The Kodak NTA nuclear emulsion is still widely used in neutron personnel dosimetry. Two major drawbacks of this detector are its energy threshold and the loss of latent information (fading) with time.

In employing the NTA emulsion in the stray radiation field outside the shielding of high-energy proton accelerators, the first point becomes irrelevant while the second increases in importance. Drying the film package and subsequently sealing it in an aluminized plastic pouch under air or nitrogen prevents the fading of nuclear tracks to a large extent. Three different methods of sealing were employed and results of experiments will be reported.

While one method used avoided the loss of information for tracks from radioactive source neutrons, the fading of tracks from high-energy hadrons was still considerable. Different development techniques were also tried to control the loss of information.

Even under the optimal established conditions, the NTA emulsion cannot be considered as a dosimeter. Therefore, it is proposed to revalue its monitoring capabilities in view of the extremely satisfactory situation with respect to radiation levels and area monitoring around the CERN proton accelerators.

A COMPARISON OF FAST NEUTRON PERSONNEL DOSIMETERS AROUND AN ELECTRON ACCELERATOR

A. Esposito and M. Pelliccioni (INFN Laboratori Nazionali di Frascati, Italy)

The use of NTA-type personnel neutron films in the mixed radiation fields around electron accelerators is completely unsatisfactory. In fact, a large fraction of the neutron spectrum is in this case under the detection threshold of these instruments. Nevertheless, nuclear emulsions are still the most popular personnel neutron dosimeters. Recently, other kinds of dosimeters have been developed. The most promising are the CR-39 plastic or a combination of a CR-39 plastic with an albedo dosimeter. These two systems are also commercially available.

At the L.N.F., the responses of the three above-mentioned dosimeters were compared around a 350 MeV linear accelerator. The results of the comparison are shown.

THERMOLUMINESCENCE ALBEDO NEUTRON DOSIMETRY

M. S. Salem, J. W. McKlveen and G. W. Klingler (Arizona State University, Tempe, Arizona 85287)

The thermoluminescence albedo neutron dosimeter is considered the best personal neutron dosimeter available today. However, it does not satisfy all requirements for being satisfactory for universal application and usage in a wide variety of radiation environments.

A commercially available thermoluminescent neutron dosimeter (Panasonic UD-809AQ) was used for this albedo neutron research. The dosimeter is good for thermal and epithermal neutrons but very poor for fast neutrons due to the source energy dependent response. The uncertainty associated with the thermal neutron dose equivalent ranges from 2% to 24% and for the epithermal neutrons is 2% to 47%. The response is dependent on the beam direction.

Distance between the dosimeter and the body should be very small (0-1 cm). For dosimeters placed more than 1 cm from the body, the measured dose equivalent will be underestimated by as much as 24%. The dosimeter should be worn carefully because if it is reversed the dose equivalent will be underestimated by 65%. The fading rate at 24°C is 12% after 3 days, 37% after 3 weeks and increases with the temperature.

RESULTS, DEVELOPMENTS AND ANOMALIES OF A PANASONIC TLD SYSTEM

M. W. Lantz, (Arizona Public Service, Phoenix, Arizona 85036)

A TLD system can be made to perform at excellent levels of accuracy and precision or can be treated as a "black box" and provide generally acceptable measurements. Anomalous readouts can be termed significant problems or reconcilable events. This report, an evaluation of the Panasonic Model 710 TLD Reader and Model 812 and 809 Dosimeters, presents the results and calibrations of a series of interlaboratory intercomparisons in order to document improved accuracy and precision. It lists a summary of TLD system characteristics that have been or need to be resolved and also presents a set of good practices for the analysis of these dosimeters.

ON TL RESIDUALS IN LiF ABOVE 300°C: ACCUMULATION EFFECTS AND THEIR MINIMIZATION

C. W. King (Harshaw/Filtrol Partnership, Solon, Ohio 44139)

R. W. Pollock (Siemens Gammasonics, Inc., Des Plaines, Illinois 60018)

Encapsulating LiF dosimeters in thin Teflon film generally limits the measurement temperature to a maximum of 300°C. The residual TL whose characteristics lie above this temperature continues to accumulate with each exposure. At relatively high dose levels, migration from these higher temperature traps can contribute to the readings in subsequent reuses of the dosimeter.

These effects have been investigated for various dose levels from neutrons and gamma rays. The length of time between irradiation, measurement and re-read were controlled. The results of this investigation and procedures to minimize these effects are presented and discussed.

RESIDUAL NEUTRON SIGNALS IN TLD-600 RIBBONS USED IN AUTOMATED PERSONNEL DOSIMETRY SYSTEMS

C. J. Card, J. J. Fix, and M. K. Winegardner (Battelle Pacific Northwest Laboratories, Richland, Washington 99352)*

G. R. Rao (United States Testing Company, Richland, Washington 99352)

Complete readout of neutron induced signals in TLD-600 ribbons used in automated personnel dosimetry systems is difficult because of the presence of higher temperature traps and an upper temperature limit imposed because of the decomposition of Teflon at about 310°C. Residual thermoluminescent signals remaining on these ribbons after readout can significantly impact subsequent readouts. Several aspects of this problem have been investigated at Hanford including methods for determining the presence of residual neutron signals, preventing and removing residual signals, and minimizing their impact on personnel dose results. The results of these investigations will be reviewed. The effectiveness of different annealing, readout and dosimeter assignment procedures will also be discussed. Glow curves of TLD-600 ribbons will be used to illustrate the presence of residual neutron signals during primary and subsequent readouts. The relative magnitude of residual signals and their impact on personnel dose calculations will be demonstrated using dosimeter readout data. Much of the discussion will address the significance of residual neutron signals for large multiuser dosimeter systems and the practicality of various methods of resolving this problem.

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*Pacific Northwest Laboratory is operated for the U.S. Department of Energy by Battelle Memorial Institute.

Thursday, October 18

8:30 AM

SESSION VII: RESEARCH AND ADVANCES IN PERSONNEL RADIATION DOSIMETRY

Session Chairman - Mr. Joe M. Aldrich
Rockwell International Corporation
Rocky Flats Plant
Golden, Colorado

Biographical Sketch

For the past 3½ years, Joe has been employed by Rockwell International at Rocky Flats, Colorado as a Senior Research Health Physicist in HS&E Application Technology, a research and development group. For 4 years prior to joining Rocky Flats, he worked as a Research Scientist in Dosimetry Technology and as a Health Physics Technical Support Engineer for Battelle Northwest Laboratory in Washington. Earlier experience includes: 5½ years in the Nuclear Navy Submarine Program, and 1½ years as a Radiation Laboratory Technician. He holds a M.S. in Radiological Sciences, belongs to several professional societies, has authored publications relating to radiation measurement, and is currently serving as a Technical Assessor for the NVLAP program.

THE ASSESSMENT OF BETA AND GAMMA RADIATION FIELDS USING A NEW MULTI-ELEMENT PERSONNEL DOSIMETER

F. S. Tsakeres and J. W. Poston (Georgia Institute of Technology, Atlanta, Georgia 30332)

Accurate assessment of beta radiation absorbed dose is difficult to achieve with the personnel monitoring devices presently available. Many of the conventional dosimeters are designed to detect penetrating radiations and, in most cases, the assessment of skin dose is practically impossible. This difficulty can be attributed to many factors including the weakly penetrating nature of beta radiation, the spectral energy distribution and wide energy range of beta emitting radionuclides, the variability of beta radiation fields, the complexity of mixed field radiation dosimetry, and the lack of suitable calibration sources and techniques. Most attempts to assess this type of dose contribution have been performed using insensitive dosimeters covered with relatively thick absorbing materials. The resultant measurements of the weakly penetrating components of a radiation field have yielded large statistical errors for relatively significant accumulated absorbed doses (e.g., 100 to 500 mrad).

Because of the inadequacies of past dosimetry systems, a highly sensitive beta/gamma personnel dosimeter has been developed at Georgia Tech. The CHEMM (CaF_2:Dy Highly Efficient Multi-element Multi-filter) dosimeter is comprised of four large CaF_2:Dy (TLD-200) chips and a standard LiF (TLD-100) TLD. The weakly penetrating and penetrating components of a radiation field are separated by the CHEMM dosimeter with a series of filters. The light output obtained from the TLD/filter combinations are used to derive the effective energy of the weakly penetrating and penetrating components. The large TLD-200 chips are used to assess the weakly penetrating component while the LiF TLD is used to determine the penetrating dose contribution. The use of CaF_2:Dy in conjunction with LiF allows a significant increase in beta sensitivity, accuracy, and precision at relatively low absorbed doses (i.e., 100 mrad of Tl-204 beta radiation) without sacrificing a tissue equivalent photon response. The performance results of the CHEMM dosimeter in mixed beta and gamma radiation fields will be presented.

A COMPACT PORTABLE RADIAC SYSTEM FOR DOSE MEASUREMENT AND DOCUMENTATION

P. T. Randtke, W. Wong, T. L. Young and L. A. Devigili
(GA Technologies Inc., San Diego, California 92138)

A small rugged RADIAC system has been designed which incorporates the latest microcomputer technology and solid state radiation probes. The measurement units can be either hand-held or badge size, the latter incorporating a fixed sensor inside its housing. The hand-held version can be mated with a variety of radiological probes for alpha, beta, gamma and neutron assessment as well as solid-state gas sensors for toxic vapor sensing. All units are programmable by a single portable supervisor module which minimizes the operational tasks by untrained personnel yet maximizes the flexability of the system in regard to calibration, radiation type, dose, dose rate and staying time. This unit is capable of instant printout of accumulated dosimetry data from the measurement modules as well as storage on mini-mag tape. Low maintenance, light weight and long battery life make this an attractive unit for both military surveillance operations as well as industrial personnel dosimetry requirements.

THE DEVELOPMENT OF AN INTELLIGENT BETA SURVEY INSTRUMENT

R. O. Murphy and J. W. Poston (Georgia Institute of Technology,
Atlanta, Georgia 30332)

The nuclear power industry has assigned doses to personnel from beta-gamma fields for many years, but the techniques and instrumentation for beta dosimetry have not been thoroughly developed. Field measurements of the dose due to beta radiation are used to estimate the hazard to certain tissues, such as the skin, and this estimate is used as a basis for radiation protection. Maintenance tasks such as steam generator entry or BWR turbine repair require workers to be near to or inside of a very large beta-gamma source with little or no shielding. Under such conditions, controlling the exposure by the dose to the skin or to the lens of the eye might be necessary. Commercially available survey instruments are not totally adequate for use in beta or mixed beta-gamma fields.

An intelligent beta survey instrument for assessing the absorbed dose rate due to beta radiation has been developed at the Georgia Institute of Technology. The beta dose rate is calculated using the differential energy spectrum of the beta radiation in conjunction with a determination of the beta particle fluence. The algorithm is based on the concept that the mass stopping power of a particular energy beta particle multiplied by the fluence of beta particles with that energy is equivalent to the dose rate contribution from the particles of that energy. Since beta particles are emitted with a continuum of energies, the differential energy spectrum is used to calculate a spectrum weighted average mass stopping power. This factor is multiplied by the beta particle fluence and conversion factors for the units to give dose rate such as rads per hour.

Results of measurements using a standard MCA and a personal computer will be presented to demonstrate the validity of the technique. Design criteria for a beta survey instrument and the development of the intelligent survey instrument will be described. The internal firmware for calculating the beta dose rate and integral dose will be explained. Measurements of various beta sources, including sources at the National Bureau of Standards, will be presented with a discussion of the advantages and limitation of the instrument.

EXPERIMENTAL STUDIES OF SOLID STATE DETECTORS FOR USE IN DISCRIMINATING INDIVIDUAL DOSEMETERS

P. Ambrosi, J. Bohm and U. Henkel (Physikalisch-Technische Bundesanstalt, Braunschweig, FRG)

K. Ritzenhoff (Staatliches Materialprufungsamt, Dortmund, FRG)

Discriminating individual dosemeters not only serve to measure the individual dose but also other quantities relevant to the determination of the body dose such as the radiation quality and the angle of radiation incidence. Measurements were performed with different types of detectors to study their suitability for discriminating individual dosemeters. The detectors were irradiated in a universal badge containing a variety of filters used at present in individual dosemeters distributed by dosemeter services.

The following types of detectors were studied:

1) commercially available TL detectors of different material and thickness evaluated in a modified commercial TLD reader,
2) recently developed BeO thin film TSEE detectors evaluated in a commercial reader, and
3) commercially available films evaluated on the one hand by a recently developed semi-automatic system and on the other hand by an authorized dosemeter service.

The angular and spectral responses of the different types of detectors are compared. The results are interpreted bearing in mind that corrections of the evaluation procedures used at present by the dosemeter services are necessary to determine the new operational quantities for receptor-present conditions which, it is hoped, will soon be published by ICRU.

FAST NEUTRON PERSONNEL DOSIMETERS USING PROTON RECOIL PROPORTIONAL COUNTERS*

T. J. Yule and E. F. Bennett (Argonne National Laboratory, Argonne, Illinois 60439)

Many different devices have been proposed for use as fast neutron personnel dosimeters. However, at the present time there is no satisfactory dosimeter; i.e., no dosimeter combines both good sensitivity and the proper energy response. These shortcomings may be overcome by using a proton recoil proportional counter as the detecting element. Furthermore, such a dosimeter will have a real-time readout. Proton recoil counters have been used extensively for fast neutron spectroscopy. For spectroscopy, the proton recoil spectrum is collected and then processed to yield the neutron spectrum. We will show that for dosimetry the proton recoil spectrum can be used to directly determine the dose equivalent by using a suitable transformation. For the neutron energy range from 10 keV to 10 MeV almost all of the dose equivalent is delivered by recoil protons. It is precisely this distribution that is measured with a proton recoil counter. Below 10 keV the dose equivalent is dominated by protons from the $^{14}N(n,p)^{14}C$ reaction and gamma rays. Both of these effects can be accounted for in a suitably designed counter. A dosimeter using a proton recoil counter will have high sensitivity and a good energy response. We also compare this dosimeter with one based on an LET spectrometer and show that for degraded fission spectra the proton recoil dosimeter is significantly more sensitive.

*Work supported by the U.S. Department of Energy.

SPECTROMETRY FOR NEUTRON DOSIMETRY

A. E. Evans and A. A. Robba (Los Alamos National Laboratory,
Los Alamos, New Mexico 87545)

The response of neutron dosimeters is dependent on neutron energy. Measurement of neutron spectra is therefore necessary to determine correction factors for neutron doses measured with integral neutron dosimeters and dose-rate measuring instruments. For neutron energies from 0.6 to 20 MeV, proton-recoil spectrometry with liquid scintillators is the normally used technique. For lower energies, proton-recoil proportional counters are used. The ^{3}He ionization chamber operating in the energy range of from 20 keV to 2 MeV offers a better energy resolution (14 keV for low-energy neutrons, 30 keV at 1 MeV) than proton-recoil counters over most of its range and the pulse height distribution is usually easier to unfold. We have applied these three techniques to the measurement of neutron spectra of the intrinsic radiation (INRAD) emitted by nuclear weapons and from a source built to replicate the spectrum of radiation emitted by the Little Boy explosion at Hiroshima. INRAD spectral measurements are difficult because of the conditions under which the measurements must be made and because of the low source strengths (near or below background) which are sometimes encountered. The Little Boy neutron spectrum measurements have been repeated by researchers from several laboratories using various techniques, including the above and track-etch and emulsion foils, sulphur activation, Bonner spheres, threshold activation foils, thermoluminescent dosimetry of quartz in roof tiles, and even chromosomal changes in blood. Comparison of spectra determined from these measurements has revealed a gratifying degree of agreement among the participants.

STUDIES OF A DIGITAL APPROACH TO NEUTRON DOSIMETRY AND MICRODOSIMETRY*

J. E. Turner, R. N. Hamm, H. A. Wright, G. S. Hurst and M. M. Chiles
(Oak Ridge National Laboratory, Oak Ridge, Tennessee 37831)

R. T. Greene (General Electric Company, St. Petersburg, Florida 33733)

Preliminary theoretical work has been conducted to develop a new, digital approach to measuring neutron fields. The track of a charged recoil particle produced by a neutron in a gas can be characterized digitally by the numbers of ions that occur in given volume elements in the gas. In principle, knowing the initial position of every secondary electron produced would represent complete information about a track as far as an ionization detector is concerned. Such information would be of considerable importance for neutron monitoring and for microdosimetry. We are investigating the design and possible construction of an instrument based on time-projection chambers used in particle physics to obtain three-dimensional information about tracks. We will describe the digital approach to dosimetry and report on Monte Carlo calculations we have made of detailed proton and electron transport in several gases. Problems associated with construction of a practical instrument and its potential use for neutron monitoring will be briefly discussed.

*Research sponsored by the Office of Health and Environmental Research, U.S. Department of Energy, under contract DE-AC05-84OR21400 with Martin Marietta Energy Systems, Inc.

PROGRESS IN THE DESIGN OF NEUTRON DOSIMETERS USING SUPERHEATED DROPS*

R. E. Apfel and Y. C. Lo (Yale University, New Haven, Connecticut 06520)

We report on our progress in using superheated drops for neutron dosimetry. The four sensitive liquids used are: Freon-114 (CCl_2F_4, b.p. $-4°C$) Freon-142B ($C_2H_3ClF_2$, b.p. $-10°C$), isobutane (C_4H_8, b.p. $-10°C$), and Freon-12 (CCl_2F_2, b.p. $-30°C$). At room temperature and atmospheric pressure, each of these liquids, in drop form immersed in an aqueous gel, is sensitive to sufficiently energetic fast neutrons; Freon-12 detectors are also sensitive to thermal and intermediate neutrons. We have measured threshold energies for the first three detectors, and the absolute sensitivity of all the detectors when exposed to any of a number of neutron sources including D_2O moderated ^{252}Cf; thermal and intermediate energy neutron beams of 2 keV, 24 keV and 144 keV from a reactor source; reasonably monoenergetic neutrons of 570 keV, 890 keV, and 1100 keV using a (p,n) reaction; and 3.8 MeV neutrons from the d-d reaction. All but the last of these sources is at the National Bureau of Standards; the last is from Yale's 1 MeV Van de Graaff accelerator.

The most sensitive detector, Freon-12, has an energy dependence that follows the general form of the ICRP recommended dose equivalent curve, although the thermal sensitivity of the superheated drop detector (SDD) is lower relative to the fast response than "the ideal" dosimeter.

I shall present our laboratory's most recent work on these detectors, including our efforts of producing compositions of uniform superheated drops, and our recent construction of both a hand held detector and a larger five channel neutron spectrometer.

*This work supported by DOE under Contract No. DE-AC02-81EV10673.

PROPERTIES OF BUBBLE - DAMAGE POLYMER DETECTORS*

H. Ing and H. C. Birnboim (Chalk River Nuclear Laboratories,
Chalk River, Ontario, Canada)

The bubble damage polymer detector works on the principle of neutron-induced vaporization of tiny droplets (~20 μm diam) of superheated detector liquid uniformly dispersed in an elastic-solid polymer. Such detectors are insensitive to gamma radiation but are sensitive to very low doses of neutrons. Neutron radiation causes visible bubbles of gas (~1 mm) to form immediately throughout the polymer which can be counted to give a measure of the neutron dose.

Our current experiments emphasize application of these novel detectors to personal neutron dosimetry. These studies have been made possible by improvements in detector preparation methods which allow for "batch processing" of up to 200 detectors per day. Uniformity of detectors is reasonably good, giving neutron detection efficiencies which vary by less than $\pm 15\%$.

In their "inactive state", the detectors are insensitive to radiation and can be kept for months with no apparent change in their characteristics. When the detectors are to be used, they are "activated" by breaking the seal of the detector. In this condition, neutrons will cause bubbles to form in seconds. The detection sensitivity can be varied by minor changes in the detector preparation technique to measure neutron doses from below 0.1 mrem to above 1 rem. Most of our studies have involved detectors which give about 1 to 3 bubbles per mrem of Pu-Be neutrons.

The behaviour of bubbles formed by irradiation has been extensively studied. With a very firm polymer, the bubbles can cause local ruptures of the medium and leave permanent damage sites. For a soft polymer, it is possible to avoid medium damage. The bubbles, if allowed to fade gradually over a period of weeks, can be made to reappear. However, we have also found a method of eliminating bubble fading so that the detector will integrate neutron exposure over time.

Measurements are in progress to determine the effect of temperature on detector characteristics and of neutron energy on detection sensitivity.

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*This work was partially supported by a contract with the Atomic Energy Control Board.

Thursday, October 18

1:30 PM

SESSION VIII: PERSONNEL DOSIMETRY - NEEDS AND DIRECTIONS

Session Chairman - Dr. Richard E. Swaja
Health and Safety Research Division
Oak Ridge National Laboratory
Oak Ridge, Tennessee

Biographical Sketch

Richard Swaja received a Ph.D. degree in nuclear science from Carnegie-Mellon University in 1973. Between 1968 and 1980, he worked at the Westinghouse Bettis Atomic Power Laboratory where he was involved in areas concerning nuclear physics, radiation transport, reactor design and testing, radiation protection, and emergency planning. Since 1980, he has been a senior research staff member at Oak Ridge National Laboratory. Dr. Swaja has conducted research and development programs in external dosimetry, coordinated international dosimetry intercomparison studies, and developed training courses at Oak Ridge. He has also authored numerous technical reports and open literature publications in areas concerning radiation dosimetry, nuclear physics, reactor analysis, and emergency planning.

PERSONNEL NEUTRON AND GAMMA-RAY DOSIMETER INTERCOMPARISON STUDIES AT OAK RIDGE NATIONAL LABORATORY*

C. S. Sims (Oak Ridge National Laboratory, Oak Ridge, Tennessee 37831)

Ninety-five different organizations (64-USA, 31-other nations) have participated in ten annual personnel dosimetry intercomparison studies conducted at ORNL since 1974. The well-characterized Health Physics Research Reactor has been operated on 61 occasions during these studies to provide dose equivalents to about 7,000 dosimeters under a variety of shielding conditions. Five basic types of neutron dosimeters and two basic types of gamma-ray dosimeters have been used to make over 98% of the measurements. The performance of these dosimeters is compared with accepted standards and guidelines. Conclusions relative to the application of various types of personnel dosimeters are presented.

*Research sponsored by the Division of Pollutant Characterization Safety Research, U.S. Department of Energy under contract DE-AC05-84OR21400 with Martin Marietta Energy Systems, Inc.

BASIC REQUIREMENTS OF DOSEMETER SYSTEMS FOR INDIVIDUAL MONITORING

J. Bohm and P. Ambrosi (Physikalisch-Technische Bundesanstalt,
Braunschweig, FRG)

At irregular intervals, internationally acknowledged organizations such as IAEA and ICRP publish and revise reports dealing with general principles of individual monitoring. For a number of years, these praiseworthy activities have provided a general guideline. Nevertheless, this guideline did not lead to a diminution in the variety of dosemeter types and thus to a reduction of costs. Even the film badge, which has been declared dead for two decades by many colleagues, is still being used.

Again and again, newly developed promising dosemeter systems have been offered but have hardly found recognition. This is, at least in part, due to the fact that internationally accepted detailed requirements on dosemeter systems which are not specific to detectors cannot be found in the publications of organizations like IAEA or ICRP. Designers of new systems must therefore define their own specifications which may not be accepted by all potential users. For example, reference to the overall uncertainty of measurement - a quantity which is of great importance to dosemeter systems - is rather insufficient in ICRP publication 35 (1982) as it does not specify how to combine the inherent component uncertainties. Different methods of combination may result in the overall uncertainties differing by a factor of 3 or more.

This paper attempts to critically analyse the current situation. A plea is made for more detailed requirements which are not specific to detectors and which should at least be specified for individual dosemeters used in routine monitoring.

THE IMPORTANCE OF RADIATION DOSIMETRY IN RADIATION COMPENSATION LITIGATION

B. P. Colby (American Nuclear Insurers, Farmington, Connecticut 06032)

What is the probability that past radiation exposure caused a particular cancer? Courts, health physicist and physicians are searching for a credible answer to this question.

A method has been proposed to determine the probability of causation (PC) by the radiation dose in question. The arithmetic expression consists of the risk of cancer from the radiation dose to the organ or tissue of interest divided by the risk of cancer from the radiation dose plus all other sources. The application of radiation dose to this equation is not simple. Workers can be exposed to many different source geometrics, radiation types and energies, and other conditions which create uncertainty in a particular measurement. Another source of uncertainty is the error associated with the extrapolation from dosimetry measurement to organ or tissue dose.

The confidence a dosimetry processor and licensee places on a dose measurement will be highly scrutinized under a PC approach to radiation compensation. Practices important in defending dose measurements will be presented.

IMPROVEMENTS IN THE ETCHING OF CR-39 FOR LARGE-SCALE PERSONNEL NEUTRON DOSIMETRY*

D. E. Hankins and S. G. Homann (Lawrence Livermore National Laboratory, Livermore, California 94550)

We have made application of CR-39 as a personnel neutron dosimeter more practical by simplifying the etching procedure and by using a new etch chamber and power supply. The new etching procedure developed by Tommasino (Laboratory Dosimetria e Biofisicia-ENEA-Casaccia, Rome, Italy), which uses an elevated temperature (~60°C), eliminates the previously required 5-hour pre-etch cycle. Electrochemical etching is performed at 60 Hz rather than that at ~2 to 10 kHz. This greatly improves, or flattens, the neutron energy dependence of the CR-39. We developed new etch chambers capable of handling 25 or more foils simultaneously and having only one liquid electrode. To make the tracks larger and more uniform in size, we used a high-frequency power supply in the final etching step. Because the track sizes are uniform in size, the track-counting results are more reproducible when this step is used. The high-frequency electrical parameters are produced and controlled via a parallel-resonant RLC circuit in which the etch chamber represents the capacitor.

*This work performed under the auspices of U.S. Department of Energy by the Lawrence Livermore National Laboratory under contract number W-7405-ENG-48.

AUTHOR INDEX

AUTHOR INDEX

Henkel, U.	75	VII	Thursday morning
Hertel, N. E.	60	V	Wednesday morning
Hill, M. J.	27	II	Monday afternoon
Hofert, M.	45	IV	Tuesday afternoon
Hofert, M.	65	VI	Wednesday afternoon
Homann, S. G.	81	VII	Thursday morning
Horlick, J.	29	II	Monday afternoon
Hurst, G. S.	78	VII	Thursday morning
Ing, H.	80	VII	Thursday morning
Johnson, L. O.	51	IV	Tuesday afternoon
Johnston, D.	37	III	Tuesday morning
Jones, D. E.	16	I	Monday morning
Jutoh, N.	28	II	Monday afternoon
Kelemen, L.	37	III	Tuesday morning
Kher, R. K.	47	IV	Tuesday afternoon
King, C. W.	69	VI	Wednesday afternoon
Klingler, G. W.	67	VI	Wednesday afternoon
Knight, C. M.	17	I	Monday morning
Kumamoto, Y.	28	II	Monday afternoon
Lantz, M. W.	68	VI	Wednesday afternoon
Lehmann, F.	65	VI	Wednesday afternoon
Lesiecki, H.	56	V	Wednesday morning
Lo, Y. C.	79	VII	Thursday morning
Lucas, M. C.	50	IV	Tuesday afternoon
Macfarlane, B.	48	IV	Tuesday afternoon
Mahathy, J. M.	41	III	Tuesday morning
Marshall, T. O.	24	II	Monday afternoon
Martz, D. E.	51	IV	Tuesday afternoon
Maruyama, T.	28	II	Monday afternoon
Matsumoto, S.	28	II	Monday afternoon
McKlveen, J. W.	67	VI	Wednesday afternoon
McWilliams, F. F.	33	III	Tuesday morning
Moss, C. E.	50	IV	Tuesday afternoon
Murphy, R. O.	74	VII	Thursday morning
Nelson, R. C.	18	I	Monday morning
Nishizawa, K.	28	II	Monday afternoon
Noda, Y.	28	II	Monday afternoon
Pelliccioni, M.	66	VI	Wednesday afternoon
Pollock, R. W.	69	VI	Wednesday afternoon
Poston, J. W.	72	VII	Thursday morning
Poston, J. W.	74	VII	Thursday morning
Randtke, P. T.	73	VII	Thursday morning
Rao, G. R.	40	III	Tuesday morning
Rao, G. R.	70	VI	Wednesday afternoon
Rich, B. L.	51	IV	Tuesday afternoon
Riley, D.	36	III	Tuesday morning
Ritzenhoff, K.	75	VII	Thursday morning
Robba, A. A.	77	VII	Thursday morning
Roberson, P. L.	30	II	Monday afternoon
Salem, M. S.	67	VI	Wednesday afternoon
Scannell, M. T.	46	IV	Tuesday afternoon
Scherpelz, R. I.	49	IV	Tuesday afternoon
Schwartz, R. B.	55	V	Wednesday morning

Sims, C. S.	64	VI	Wednesday afternoon
Sims, C. S.	84	VIII	Thursday afternoon
Sivertsen, M. J.	39	III	Tuesday morning
St. Arnaud, R. J.	37	III	Tuesday morning
Supe, S. J.	47	IV	Tuesday afternoon
Swaja, R. E.	32	III	Tuesday morning
Swaja, R. E.	54	V	Wednesday morning
Symonds, G. R.	37	III	Tuesday morning
Szabo, F.	57	V	Wednesday morning
Taymaz, A.	57	V	Wednesday morning
Tsakeres, F. S.	72	VII	Thursday morning
Tsang, F. Y.	49	IV	Tuesday afternoon
Turner, J. E.	78	VII	Thursday morning
Vallario, E. J.	30	II	Monday afternoon
Weimer, G.	25	II	Monday afternoon
West, L.	58	V	Wednesday morning
Winegardner, M. K.	38	III	Tuesday morning
Winegardner, M. K.	40	III	Tuesday morning
Winegardner, M. K.	70	VI	Wednesday afternoon
Wittekind, D.	45	IV	Tuesday afternoon
Wong, W.	73	VII	Thursday morning
Wright, H. A.	78	VII	Thursday morning
Yoder, R. C.	19	I	Monday morning
Young, T. L.	73	VII	Thursday morning
Yule, T. J.	76	VII	Thursday morning