VOL. I)

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ABSTRACTS

OF THE LITERATURE EXAMINED BY THE RADIATION SHIELDING INFORMATION CENTER

RADIATION SHIELDING INFORMATION CENTER

A B S T R A C T S

OF THE LITERATURE EXAMINED BY THE RADIATION SHIELDING INFORMATION CENTER

The abstracts included in this loose-leaf binder cover all the documents that are both in the literature store of the Radiation Shielding Information Center and in the latest edition of the RSIC bibliography. At this writing (September 1966), the latest edition of the bibliography is ORNL-RSIC-5 (Rev. 1), Bibliography, Subject Index, and Author Index of the Literature Examined by the Radiation Shielding Information Center, May 1966, which is being issued along with these 1136 abstracts. Additional abstracts will be issued for inclusion in this loose-leaf binder when the next bibliography is ready for publication.

Although most of the abstracts included here were taken directly from the published documents, some were written by RSIC staff members. These were limited to documents which either did not contain abstracts or contained abstracts that were not sufficiently informative. Only unclassified abstracts are included; all are subject to revision or replacement. At the lefthand margin next to each abstract is an RSIC accession number for the document itself.

Indexes to Abstracts

Indexes to the abstracts both by subject matter and by authors can be found in the latest edition of the bibliography.

Availability of Documents Abstracted

The availability of documents abstracted here is given along with the bibliographic information preceding each abstract. The abbreviations used are the same as those listed in Nuclear Science Abstracts, a bi-monthly publication by the U. S. Atomic Energy Commission covering international literature on nuclear science and technology. For convenience, those pages of NSA which define the abbreviations used in the availability column and identify the depositories from which documents may be obtained are reproduced here as they appeared in the August 1966 issue.

REPORT NUMBER INDEX

Reports, conference papers and proceedings, numbered translations, and patents abstracted during the period are included in this index. Each item is listed by report number; this is followed by the abstract number and a brief notation regarding public availability. The [illegible] entries show which reports are on file in the d[illegible] libraries, which reports have been publish[illegible] open-literature journals (journal citation given) [illegible] such reports can be purchased (source and price) [illegible] er a report is identified by more than one [illegible] cross references from secondary to primary [illegible] numbers are cited.

The report num[illegible] es apply to AEC reports unless designated othe[illegible] Non-AEC reports are identified by parenthetic[illegible] after the report codes; e.g., (non-AEC), (British), [illegible] nan), (French). The note (Misc.) identifies those codes used for both USAEC and non-AEC reports. Patents are listed under the heading "PATENTS"; the country of origin is included in parentheses.

A comprehensive listing of both AEC and non-AEC report codes is contained in TID-8515(5th Rev.), *Report Number Series Used by the Division of Technical Information in Cataloging Reports*. This information is also included in the Cumulative Report Number Index (issued annually).

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U. S. Reports. Many reports of U. S. Government agencies are available to the public from the Clearinghouse for Federal Scientific and Technical Information (CFSTI). This availability is given when known. Such reports should be ordered by AD number or NASA-N number. If availability is not shown, inquiries may be directed to CFSTI. In the case of NP numbers, inquiries should be by title and author.

Non-U. S. Reports. Many reports of atomic energy establishments of other countries are deposited in the libraries listed on the inside covers of each issue of *NSA*.

Additionally, the following have made their reports available as indicated.

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Australian Atomic Energy Commission Research Establishment. Available upon request from the Library, Australian Atomic Research Establishment, Private Mail Bag, Sutherland, New South Wales, Australia.

BELGIUM

Centre d'Étude de l'Énergie Nucléaire. Available upon request to individuals and organizations, from the Centre d'Étude de l'Énergie Nucléaire (CEN), 31, Rue Belliard, Bruxelles, Belgium.

CANADA

Atomic Energy of Canada Limited. For sale by the Technical Information Officer, Atomic Energy of Canada Limited, Chalk River, Ontario, Canada. Copies are also deposited in libraries in the United Kingdom.

National Research Council of Canada. Available from the National Research Council of Canada, Ottawa 2, Ontario, Canada.

DENMARK

Atomic Energy Commission Research Establishment, Risø. Available on exchange from the Library, Danish Atomic Energy Commission, Risø, Roskilde, Denmark. Also for sale by Jul. Gjellerup, Booksellers, 87 Sölvgade, København K., Denmark.

Danatom. Available on loan from Research Establishment, Risø, Roskilde, Denmark. Copies are available from Danatom, Aurehøjvej 2, Hellerup, Denmark.

EURATOM

Reports from the Commission of the European Atomic Energy Community are available from: Presses academiques europeennes, 98, chaussee de Charleroi, Brussels 6, Belgium; and Office central de vente des publications des Communautes eurodeennes 2, place de Metz, Luxemburg.

FRANCE

Commissariat à l'Énergie Atomique. Inquiries for reports numbered 1-2199 should be addressed to Commissariat à l'Énergie Atomique, Service Central de Documentation, Centre d'Études Nucléaires de Saclay, Boite Postale No. 2, Gif-sur-Yvette (S. et O.), France. Reports numbered 2200 and above are available from La Documentation Francaise, 16 Rue Lord-Byron, Paris VIII, France.

Service Central de Protection contre les Rayonnements Ionisants. Copies are available from Ministère de la Santé Publique, Service Central de Protection contre les Rayonnements Ionisants, B. P. n° 35–LE VESINET (S. et O.), France.

GERMANY

Deutsches Elektronen-Synchrotron DESY. Copies are available from Deutsches Elektronen-Synchrotron Desy, Dokumentation, 2000 Hamburg-Gr. Flottbek 1, Notkestieg 1, Germany.

Gesellschaft für Kernforschung mbH (KFK). Copies are available from Gesellschaft für Kernforschung mbH, 75 Karlsruhe, Postfach 947, Germany. Copies are also deposited in many national nuclear centers in the world.

Kernforschungsanlage. Copies may be ordered from Zentralbibliothek der Kernforschungsanlage Jülich, Jülich, Bundesrepublik Deutschland.

Hahn-Meitner-Institute für Kernforschung. Available without charge from Hahn-Meitner-Institute für Kernforschung Berlin, Bibliothek, Glienicker Strasse 100, 1 Berlin

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39, Germany. Copies are deposited in many national nuclear centers in the world.

Johann-Wolfgang-Goethe Universität, Frankfurt Am Main, Institut für Kernphysik, (IKF). Copies are available from Gmelin Institut, 40-42 Varrentrappstrasse, Frankfurt (Main), West Germany.

ISRAEL

Israel Atomic Energy Commission. Available without charge from Ministry of Defence/Atomic Energy Commission, Soreq Research Establishment, Yavne, Israel. Copies are deposited in libraries in Israel and in other countries.

ITALY

Centro Informazioni Studi Esperienze (CISE). For sale at CISE Press Office, Casella Postale 3986, Milano, Italy. Copies are deposited in libraries in Italy.

Comitato Nazionale per l'Energia Nucleare. Copies are available from Comitato Nazionale per l'Energia Nucleare, Divisione Affari Internazionali e Studi Economici, Ufficio Pubblicazioni, Via Belisario, 15, Roma, Italy.

JAPAN

Japan Atomic Energy Research Institute (JAERI). Available from the Staff Supply Cooperative of Japan Atomic Energy Research Institute (JAERI), 1-1 Tamura-Cho, Shiba, Minato-Ku, Tokyo, Japan. Copies are deposited in depositories in Tokyo, Nagoya, and Osaka.

Nagoya University. Copies are available from Research Information Center, Institute of Plasma Physics, Nagoya University, Nagoya, Japan.

University of Tokyo. Available free of charge from Institute for Nuclear Study, University of Tokyo, Tanashi-Machi, Kitatama-gun, Tokyo, Japan.

NETHERLANDS

Reactor Centrum Nederland. For sale by Reactor Centrum Nederland, 112 Scheveningseweg, 's-Gravenhage, The Netherlands. Copies are deposited in university libraries in Holland and in many national nuclear centers in the world.

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NORWAY

Forsvarets Forskningsinstitutt. Copies available from Norwegian Defence Research Establishment, Library, P. O. Box 25, Kjeller, Norway.

Institutt for Atomenergi. "KR" reports are available on an exchange basis from the Library, Institutt for Atomenergi, Box 40, Kjeller, Norway. Other series are not generally available to the public.

Norwegian Reactor School. Copies are available from The Library, Institutt for Atomenergi, P.O. Box 40, Kjeller, Norway.

PAKISTAN

Atomic Energy Center. Copies are available from Atomic Energy Center, P. O. Box 658, Lahore, (West Pakistan).

POLAND

Reports of the Polish nuclear institutes are published by the Nuclear Energy Information Center of the Government Commissioner for Use of Nuclear Energy. Copies are available from Ośrodek Informacji o Energii Jądrowej, Warszawa, PKiN, Poland.

SPAIN

Junta de Energia Nuclear. "JEN" reports are available without charge from Servicio de Documentation, Junta de Energia Nuclear, Ciudad Universitaria, Madrid (3), Spain.

SWEDEN

Aktiebolaget Atomenergi. "AE" reports are for sale from Aktiebolaget Atomenergi, Library Section, Box 43041, Stockholm 43, Sweden. Bibliographies and literature surveys of the VDDIT- series may be obtained from Aktiebolaget Atomenergi, Information Office, Scientific and Technical Information, Box 43041, Stockholm 43, Sweden. Other series are not generally available to the public. Organizations in the atomic energy field may obtain copies from Aktiebolaget Atomenergi. Microcopy of AE reports is available from the International Documentation Centre, Tumba, Sweden.

SWITZERLAND

CERN European Organization for Nuclear Research. Available without charge from CERN European Organization for Nuclear Research, Geneva 23, Switzerland. Copies are deposited in libraries in 13 countries.

Eidgenössisches Institut für Reaktorforschung. "EIR" reports are available from Eidgenössiches Institut für Reaktorforschung, c/o Bibliothek, Würenlingen/Ag, Switzerland.

TURKEY

Turkish Atomic Energy Commission. Copies are available free of charge from T.A.E.C., Çekmece Nuclear Research Center, P. O. Box 1, Airport, Istanbul, Turkey.

UNITED KINGDOM

United Kingdom Atomic Energy Authority. Some reports are for sale in the United States by the British Information Service (BIS), 845 Third Avenue, New York, N. Y. 10022. United Kingdom reports generally may be purchased from Her Majesty's Stationery Office, London. Copies not for sale by Her Majesty's Stationery Office may be purchased in microcopy form from Micro Methods Limited, East Ardsley, Wakefield, Yorks, England. Report copies are deposited in a large number of libraries in England, depositories of the USAEC in the United States and abroad, and in most national nuclear centers throughout the world.

Conferences

Published conference proceedings as well as individual unpublished papers from conferences are included in this index. A CONF number is assigned to the conference if published proceedings are received and also if only a single paper is received. The base number used is year-month-sequence. Thus, the number CONF-650101 would indicate that the conference was held in 1965, in January, and that this was the first January 1965 conference so received. This number would be assigned to complete published proceedings, or if individual papers are received, the base number would be extended to include a paper number, i.e., CONF-650101-1. As additional papers are received, they will be assigned consecutive numbers in the series, CONF-650101-1, 2, 3, 4, etc. This numbering system ensures that all papers received from a conference will be associated with that conference in this index.

If published proceedings are received after the receipt of individual papers, the complete proceedings will be abstracted and the abstract number will appear in this index with the base CONF number.

Abbreviations Used in Availability Column

AECL	For sale by Atomic Energy of Canada Ltd., Chalk River, Ontario, Canada.
BIS	For sale by British Information Service, 845 Third Avenue, New York, N. Y. 10022
CFSTI	For sale by the Clearinghouse for Federal Scientific and Technical Information, U.S. Department of Commerce, Spring-

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EM — Blueprints and microcopy of AEC engineering drawings are available from Clearinghouse for Federal Scientific and Technical Information, U. S. Department of Commerce, Springfield, Virginia 22151. Current prices of the CAPE- packages may be obtained from this address. TID-4100, available from the Division of Technical Information Extension, P. O. Box 62, Oak Ridge, Tennessee 37831, contains information on all the CAPE- packages for sale.

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HMSO — Her Majesty's Stationery Office, London.

IAEA — Available from the International Atomic Energy Agency, Vienna, Austria, and from its sales agencies in various countries throughout the world. A list of these agencies is found in each STI/DOC and STI/PUB publication.

JCL — For sale by the Special Libraries Association Translation Center, John Crerar Library, 35 West 33rd Street, Chicago, Illinois 60616.

LC — For sale by the Library of Congress, Photoduplication Service, Washington, D. C. 20540.

NAS — National Academy of Sciences, National Research Council, Washington, D. C. 20418.

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UN — United Nations Bookshop, United Nations Headquarters, New York, New York.

USGS — Available free of charge from the U. S. Geological Survey, Washington, D. C. 20242.

[The symbols "ph," "mf," and "fs" are used to designate that copy is available only in photostat, microfilm, or facsimile form. The symbol "mn" is used to designate that copy is available in micronegative (microfiche) form. The symbol "cy" is used to designate full size or photographically enlarged copy (two-thirds of original size).]

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Collections of USAEC reports are maintained in the libraries of the organizations listed below for reference use by the general public. Facilities for making photocopies are available. The USAEC also makes available through these libraries atomic energy reports of countries other than the U. S. that are abstracted in *Nuclear Science Abstracts*.

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0. O. Witte, Radiation Protection Doors of Steel, AEC-tr-6181 (November 1962). Availability: $1.10 (fs), $0.80 (mf) JCL.

The problems of fabricating radiation shielding doors are described. The emphasis is on doors constructed of steel or encased in steel.

1. S. Orecher and M. Oberhofer, Shielding of Radiations from Reactor-Beam Tubes, AEC-tr-6185 (November 1962). Availability: $1.10 (fs), $0.80 (mf) JCL.

A survey of the materials which might be used in the shielding of reactor beam tubes is given and a very brief description of the construction of the shields is presented.

2. F. Kohl, Nomographic Determination of Permissible Distances and Protective Screens Against Gamma Radiation, AEC-tr-6101 (June 1962). Availability: $1.60 (fs), $0.80 (mf) JCL.

Connecting nomographs were constructed to describe the attenuation of gamma rays from various radioisotopes through concrete, iron, and lead, assuming that the source is a point source. The source strength ranges from 0.1 μc to 10^4 C. The safe distances from the shield range are from 1 cm to 100 m. The dose-rate ranges from 10 μr/hr to 100 mr/hr.

3. A. Nykänen and S. E. Pihlajavaara, Finnish Recommendations for Radiation Shielding Concrete Structures, AEC-tr-6083 (1962). Availability: $1.60 (fs), $0.80 (mf) JCL.

Recommendations are given for the preparation of concrete for shielding of radiations. These include legal aspects, structural considerations, and shielding properties. Questions of density, uniformity, water content, and testing are reviewed.

4. Walter Lowen, A Buildup Corrected Single Collision Model for Gamma-Ray Backscattering, EIR-53 (June 1963). Availability: Free of charge from CERN European Organization for Nuclear Research, Geneva 23, Switzerland.

A buildup corrected single-collision kernel is developed to calculate the backscattered energy flux at a specified location resulting from a monoenergetic point monodirectional gamma beam incident with a given angle of incidence on a scatterer of specified thickness. The formulation is supported by phenomenological analysis and by a comparison of total energy albedos, as calculated by a single-collision kernel, to Monte Carlo data. The angular distribution of reflected energy is computed by means of a buildup corrected single-collision kernel for 1-MeV gammas backscattered from concrete. The results are shown to be in good agreement with Monte Carlo data.

A scattering factor, derived from the buildup corrected kernel was evaluated for 1-MeV gammas incident on concrete by means of a computer program for the IBM 709. Application of the scattering factor to the solution of reflection problems in realistic geometry is illustrated by a sample problem. Extended applications to slant incidence transmission problems are indicated.

5. B. T. Price, C. C. Horton, and K. T. Spinney, Radiation Shielding, Pergamon Press, 1957.

This book is a general reference on radiation shielding. Two unique features are: (1) a section describing how to design a reactor shield on page 230 and (2) a discussion of energy-dependent removal cross sections in relation to neutron attenuation in thick shields.

6. R. Aronson, M. J. Berger, R. E. Beissner et al., Reactor Handbook, Second Ed., Vol. 3, Part B (Shielding) (E. P. Blizard and L. S. Abbott, eds.), Interscience Publishers, 1962.

This book is a general reference on radiation shielding. Many tables and graphs of pertinent shielding data are included.

7. H. Goldstein, Fundamental Aspects of Reactor Shielding, Addison-Wesley Publishing Co., Inc., 1959.

This book is a general reference on radiation shielding. The emphasis is, however, on calculational techniques and the basic physics involved in shielding.

8. J. B. Whittaker, "The Dose Rate from a Spherical Source with Self-Absorption," Phys. Med. Biol., Vol. 8, No. 3, pp. 309-318 (September 1963)

The dose rate at any point outside a spherical homogeneous radioactive source is calculated allowing for absorption both in the surrounding medium and in the source. The absorption is considered to be exponential, no account being taken of secondary interactions which might produce a "buildup" of dose rate in the medium.

The dose rate for arbitrary absorption coefficients inside and outside the source is derived initially in the form of an integral, from which the dose rate on the surface of the sphere is obtained explicitly. The special cases of the source in a nonabsorbing medium and in a medium of absorption coefficient equal to that of the source are considered in some detail. The expressions for the dose rate in the two cases have been solved numerically for a range of typical values and the results are presented in graphical form. The dose rate in the three limiting cases of large distance from the source, and high and low self-absorption have been derived from expansions of the expression for the dose rate in a non-absorbing medium. The errors involved in considering only the first one or two terms of each series are given. The expression for the dose rate due to a source of absorption coefficient equal to that of its environment has been similarly expanded to give the dose rate at large distances, and also for low absorption, the errors involved in the first one or two terms again being given.

An estimation of the effect of scattered radiation by means of "buildup" factors is discussed.

9. P. A. Ombrellaro and F. D. Federighi, A Variational Procedure for Calculating Fast Few-Group Constants, KAPL-2220 (August 10, 1962). Availability: Dep.; $1.25 OTS.

A variational procedure for calculating fast-energy few-group constants is described. For a given medium, the method permits one to express the flux and current solutions of the Boltzmann equation, treated according to P-1 slowing-down theory in each group of a few-group scheme, as a linear combination of two base flux spectra, to obtain the group flux, and as a linear combination of two base current spectra, to obtain the group current. The coefficients for combining the base spectra are provided by the theory and depend only on the concentrations of the component elements of the medium. Once the flux and current spectra in each group are calculated, the group constants for the medium can be easily calculated from base-flux-spectra weighted library microscopic cross sections. Group constants calculated in this manner agree well with those obtained from the MUFT 5 program.

10. W. R. Faust, Some Exact Solutions to the Three-Dimensional Equation of Radiative Transport, NRL-5838 (September 26, 1962). Availability: $1.10 (fs) OTS.

Fourier techniques are applied to obtain an exact solution of the three-dimensional integro-differential equation of radiative transport for a point isotropic source imbedded in an infinite medium. Results of these calculations are then extended to obtain the solution for a point source radiating along the z axis (beam of radiation).

11. D. Meneghetti, Table of Trim Discrete Quadrature Parameters for Thin Binary Slab Cells, ANL-6613 (September 1962). Availability: $0.50 OTS.

Discrete quadrature weights and angles for use in discrete ordinate methods of solutions for thin cells are tabulated for a series of regional cell thicknesses of applicability. Parameters are given for 4, 6, 8, 10, and 12 angle approximations. The quadrature assignments are based upon a simplified integral transport solution to the spatial average of the angular distribution of uncollided flux, assuming flat sources. The quadrature assignments are designated TRIM, for thin-region integral method, and are most useful with thin cells having either or both regions very thin in mean-free-path units. The IBM-704 Fortran program used to compute the parameters is described.

12. M. L. Wohl, Monte Carlo Calculations of Neutron Number Spectra and Buildup Factors in Infinite Conical Configurations, NASA TN D-1434 (September 1962). Availability: $0.50 OTS.

A Monte Carlo code simulating neutron transport in infinite cones of water and water-equivalent hydrogen was prepared for an IBM 704 computer. The code was essentially a modification of the point-source, infinite-medium code used in NASA TN D-850.

Studies were made of differential neutron number spectra and associated buildup factors for infinite cones having apex half-angles of 15°, 30°, 45°, and 60°. The buildup factors obtained were compared with those for the appropriate infinite medium, which allowed an examination of the effect of solid angle subtended by material on the transport of 6-MeV source neutrons emanating from the cone apex.

The variation of number buildup factor with distance for the various cones shows that neutron scattering out of the cones is predominant in the first 30 to 40 centimeters of material, and that transport beyond this distance is of a similar nature in all the cones.

13. A. H. Redmond and M. B. Salamon, A Standardization of the Hurst Radsan Neutron Detector, USNRDL-TR-598 (November 19, 1962). Availability: $1.60 OTS.

Dose measurements have been made with five Radsan detectors at the same operating conditions using a 100-channel analyzer and computing dose from the pulse-height distribution. The detectors responded to an 80-gram plutonium beryllium neutron source at 50 centimeters with a sensitivity about 21 percent higher than the first collision dose computed from the spectral distribution. An explanation is offered. Responses to a 2-gram similar source mounted on the detectors were correlated with the counting rate on a scaler, giving a figure for a quick check of detector condition. Both measurements gave similar ratios of dose-to-counts in a 10-minute period. A shift of spectrum threshold with detector temperature rise is described.

14. K. F. Hansen, Multi-Group Diffusion Methods, NYO-10206 (April 1962). Availability: Dep.; $1.75 OTS.

The purpose of this report is to provide an introduction to the multi-group diffusion methods of reactor analysis. The report is divided into two sections: the first concerning the derivation of the multi-group equations from the age-diffusion equation, while the second portion pertains to numerical methods of solving the equations.

No new techniques of analysis are contained in the report. The mathematics has been kept simple in order that the approximations at each step are clearly apparent. The adjoint function is introduced and the reasons for considering the adjoint are repeatedly stressed.

The selection of numerical methods considered in the second section is by no means exhaustive. Simple illustrations of various iterative methods are included. Relations between the rates of convergence of various methods may be found in the references, along with proofs of various results quoted in the report.

15. H. Goldstein and J. E. Wilkins, Jr., Calculations of the Penetration of Gamma Rays, NYO-3075 (June 30, 1954). Availability: Dep.; $30.30 (ph), $8.70 (mf) OTS.

An extensive series of calculations on the penetration of gamma rays in infinite homogeneous media has been made using the "moments method" introduced by Spencer and Fano. Spectra of scattered photons, due to monoenergetic sources ranging in energy from 0.5 MeV to 10 MeV, were obtained at distances up to 20 mean-free-path lengths in eight materials whose atomic numbers varied from 0 to 92. Source geometries included point isotropic and plane monodirectional. The computations were performed with the aid of a high speed automatic computer, the SEAC. The choice of microscopic absorption coefficients for the calculation is discussed at length, and the data used are reproduced. Definitions are given for the quantities describing the scattered photons, and the fundamental equation governing the photon transport is derived. The "moments method" is described in detail, both as to the derivation and integration of the equations for the spatial moments of the flux, and as to the procedures for reconstructing the flux from the moments. Scattered flux spectra and buildup factors are reproduced for all the calculated problems in the form of some 140 tables and 75 graphs. Methods for extending the results to problems not calculated are described. Possible sources of error are discussed. It is concluded on the basis of internal checks, comparison with other theoretical predictions, and examination of published experiments, that the accuracy of the calculations is more than adequate for almost all practical applications.

16. J. Moteff, Miscellaneous Data for Shielding Calculations, APEX-176 (December 1, 1954). Availability: Dep.; $9.60 (fs), $3.59 (mf) OTS.

This report represents an effort to consolidate under one cover the data of importance to the shielding groups. The major topics considered are as follows:

Gamma Ray Attenuation Coefficients
Buildup Factors
Fission Product Gamma Spectrum
Geometric Considerations
Flux to Dose Conversion
Attenuation Fission Product Gamma Rays Through Lead, Iron, and Water.

17. R. T. McGinnies, X-Ray Attenuation Coefficients from 10 keV to 100 MeV, Suppl. to NBS Circular 583 (October 30, 1959). Availability: $0.35 GPO.

A revision is given of the X-ray attenuation coefficients presented in National Bureau of Standards Circular 583. Table 4 of that publication is eliminated and a new table is given for each material for photon energies less than 100 keV. The uncertainties in the estimates of attenuation coefficients at low energies are from 3 to 5 percent, which is the same as was previously given at higher energies. The cross sections for scattering are unchanged. Two values are listed for the photoelectric cross section, one calculated from the Sauter-Stobbe formulas and the other derived from new experimental evidence. The procedures for smoothing experimental data are described and are generally the same as were used in Circular 583. In addition to the systematic coverage of the region from 10 keV to 100 MeV, some data are included for a number of elements based on experimental measurements below 10 keV and above 100 MeV. A comparison is made between calculated and experimental total attenuation coefficients at energies above 10 MeV.

18. G. W. Grodstein, X-Ray Attenuation Coefficients from 10 keV to 100 MeV, NBS Circular 583 (April 30, 1957). Availability: $0.35 GPO.

A tabulation of attenuation coefficients of X-rays and gamma rays from 0.01 to 100 MeV for 29 materials is presented. A summary of information on the probability of the basic interaction processes of photons with matter and a detailed analysis of experimental and theoretical evidence are included. Present information on the basic processes is adequate for many applications; however, improved theory and additional experimental data are needed in certain areas. A comparison of calculated and experimental coefficients points up this need.

19. T. Rockwell, Reactor Shielding Design Manual, D. Van Nostrand Co., Inc. 1956. Availability: $2.10 OTS.

This book is a general reference, but is more oriented toward designing a shield for a water-cooled power reactor. There are many useful tables and graphs presented.

20. E. I. Nowstrup, L. A. Beach, and W. R. Faust, Shadow Shields, NRL Report 4275 (November 16, 1953). Availability: $1.50 (ph), $1.50 (fs) OTS.

Studies have been made of the effectiveness of shadow shields placed between a source and detector. Experiments were performed by observing counting rates with a Geiger tube at various distances in water from a source hidden behind conical lead shadow shields of various dimensions.

21. J. F. Batter, A. L. Kaplan, and E. T. Clarke, An Experimental Evaluation of the Radiation Protection Afforded by a Large Modern Concrete Office Building, CEX-59.1 (January 22, 1960). Availability: Dep. (mc); $0.60 OTS.

An experimental study was made to determine the effective shielding provided by a modern reinforced-concrete office building (AEC Headquarters building) from nuclear fallout. Pocket ionization chambers were used for measurement of the radiation-field strength. Fallout was simulated with distributed and point-source configurations of Co^{60} and Ir^{192} sources.

Four typical sections were selected for study, and experiments were performed on each. These included an external wing with exposed basement walls and an external wing with a buried basement. Roof studies were made on an internal wing with a full basement and on the east end of wing A, which has a thin-roof construction. The thick-roof construction of 8 in. of concrete and 2 in. of rigid insulation covers all the building except the east end of Wing A, which has 4 in. of concrete and 2 in. of insulation.

22. G. S. Hurst and R. H. Ritchie, "Fast Neutron Dosimetry," Radiology Vol. 60, No. 6, 864-868 (June 1953).

Recent work on fast-neutron dosimetry done by the Health Physics Division of the Oak Ridge National Laboratory is summarized. The use of count-rate dosimeters and the pulse-integration method are discussed in detail.

23. G. S. Hurst, "An Absolute Tissue Dosimeter for Fast Neutrons," Brit. J. Radiol 27, No. 318, 353-357 (June 1954).

It is shown that the Bragg-Gray principle can be applied to proportional counter designs in such a manner that the absolute tissue dose (and hence the flux if the energy spectrum is known) due to fast neutrons can be measured. The method has the combined advantages of high sensitivity to fast neutrons and very low sensitivity to gamma radiation. Two practical designs are described.

24. F. Akerhielm, Transmission of Thermal Neutrons Through Boral, AE-24 (August 1960). Availability: Dep. (mc).

Transmission measurements have been performed using Maxwellian distributed neutrons from the R1 reactor perpendicularly incident upon a boral absorption plate. American, English, German, Swedish, and Swiss samples have been investigated and the results are compared to calculated values. The influence of the absorber grain size is discussed.

25. A. W. Starbird, J. D. Velletri, R. L. MacNeil, and J. F. Batter, The Effect of Interior Partitions on the Dose Rate in a Multistory Windowless Building, TO-B 63-6 (January 31, 1963). Availability: $6.60 OTS.

This report evaluates the effects of three common types of interior partitions within multistory structures on the dose rates from infinite uniform fields of fallout contamination. Comparisons are made between experimentally determined steel model results utilizing cobalt-60 gamma radiation and those obtained through use of the OCD engineering manual entitled "Design and Review of Structures from Fallout Gamma Radiation." A comparison is also made between partition results from limited fields of contamination and previous experimentally determined limited-field results for similar structures without partitions.

Agreement is excellent between experimentally measured and computed infinite-field dose rates for the three partition geometries compared -- typical box, corridor, and compartment types.

26. A. Shambon, W. M. Lowder, and W. J. Condon, Ionization Chambers for Environmental Radiation Measurements, HASL-108 (February 1963). Availability: $1.00 OTS.

The development of ionization chambers for precise dosimetric measurements of environmental gamma and cosmic radiation in the field is described. The design, calibration, and evaluation of 20-liter air ion chambers and high-pressure argon and nitrogen chambers are discussed in terms of the well-known difficulties associated with field surveys of environmental radiation.

27. R. Dennis, S. N. Purohit, and L. E. Brownell, Procedures for Shielding Calculations, AECU-3510 (January 1957). Availability: Dep.; $9.10 (fs), $3.50 (mf) OTS.

This report deals with the shielding of nuclear radiation facilities. It is divided into two sections. The first part of the report deals with the gamma-radiation shielding. The concept of the "buildup factor" is discussed in detail. As an illustration, the analysis of the heterogeneous gamma-radiation spectrum from the MTR fuel element from the point of view of shielding has been given. The analytical expression for the buildup factor, as obtained by Taylor, has been used in the above analysis. The second part of the report discusses the problems involved in shielding a nuclear reactor. A sample calculation of the shielding of a nuclear reactor is given. The report contains graphs for determining the standard integrals involved in calculating the radiation flux for standard geometries and also an extensive bibliography.

28. H. H. Van Tuyl, Fission Product Radiation and Shielding Calculations, HW-69533 (May 1961). Availability: Dep. (mc); $6.00 (fs), $2.06 (mf) OTS.

Calculations relating to radioactive decay by beta and gamma emission are discussed, with special emphasis on processes which are often ignored, but which become important when dealing with purified fission products. These include beta spectra analyses to permit accurate evaluation of average beta energies, bremsstrahlung production in beta decay, and internal conversion in gamma decay. Calculations of fission product production and dose rates are also described.

The attenuation of both primary and secondary gamma rays is discussed, and data are presented which permit the accurate estimation of gamma dose rates from point sources, including both primary attenuation and dose build-up factors. Formulas for extending these calculations to other simple geometries are given.

29. H. S. Davis, "Concrete for Shielding Nuclear Radiations," J. Struct. Div. Am. Soc. Civil Engrs. Vol. 88, No. ST1, Paper 3062, 123-134 (February 1962).

Various types of concrete are described with reference to their physical and shielding properties. Heavy aggregates, hydrous ore, cements, and water content, are also described and their affect on the shielding properties of concrete is evaluated. Current development programs (as of January 1962) are briefly described as well as areas needing further investigation.

It is deduced that concrete can be used at normal or at high temperatures for shielding nuclear radiations if structural requirements associated with high-temperature operation can be satisfied and allowances are made for changes in shielding properties caused by loss of water.

30. H. S. Davis and A. M. Asce, "Thermal Considerations in the Design of Concrete Shields," J. Struct. Div. Am. Soc. of Civil Engrs, paper 1755 (September 1958).

The behavior and properties of concrete structures for shielding atomic power plants are discussed briefly. Methods are presented for estimating thermal effects associated with linear and non-linear distributions of temperature, nuclear heating, and shield cooling. These data and procedures are subsequently used for analyzing several design problems and thermal criteria.

31. A. D. Krumbein, Summary of NDA Unclassified Results of Moments Calculations for the Penetration of Neutrons Through Various Materials, NDA-92 (Rev.) (April 1958). Availability: Dep. $1.25 OTS.

This report summarizes the unclassified results presently available at NDA for the penetration of neutrons through hydrogen, water, carbon, beryllium, and several hydrocarbons. The moments method was used throughout. Point isotropic sources were obtained and used in all the calculations except for one carbon problem. Most of the results are for fission sources but monoenergetic sources were included for several of the materials.

Results are given in the form of both graphs and tables for the differential number spectra as functions of penetration depths. In addition, doses are given where they have been calculated. Where available, the fitting parameters, C_i and B_i or $\sqrt{B_i}$, are included in the results.

32. B. J. Henderson, Conversion of Neutron or Gamma-Ray Flux to Absorbed Dose Rate, XDC 59-8-179 (August 14, 1959). Availability: $2.60 OTS.

This report contains conversion factors for the conversion of gamma-ray photon flux and gamma-ray energy flux to absorbed dose rates in air, carbon, and tissue. First-collision conversion factors for the conversion of neutron flux to absorbed dose rates are given for ethylene and tissue. The maximum biological effectiveness (RBE) for neutrons is plotted as a function of energy as is the RBE dose rate in rem per hour. Results are presented in the form of both graphs and tables.

33. H. H. Rossi, E. P. Blizard, R. S. Caswell et al., Protection Against Neutron Radiation up to 30 Million Electron Volts, NBS Handbook 63 (November 22, 1957). Availability: $0.40 GPO.

This handbook is a compilation of the biological hazards of neutron radiation up to 30 MeV as legislated by the National Committee on Radiation Protection and Measurement as of the Data of issuance.

34. J. O. Blomeke and M. F. Todd, Uranium-235 Fission-Production as a Function of Thermal-Neutron Flux, Irradiation Time, and Decay Time. I. Atomic Concentrations and Gross Totals, ORNL-2137, Part I (December 15, 1958). Availability: Dep.; $22.25 (fs), $13.07 (mf) OTS.

Levels of fission products resulting from thermal fission of U^{235} in reactor fuels were computed over a wide range of reactor operating conditions and decay times. Values for approximately 300 fission products are presented in graphical form together with gross totals of their activities, radiation powers, and thermal-neutron poisoning. The gamma spectrum is further broken into four groups of specified energy ranges, which are suitable for use in shielding design. Tabulations of these properties are arranged as to chain or mass number, element, and as to the rare-gas and rare-earth groups. The calculations assume constant replenishment of U^{235}, constant reactor power operation, and no fission-product separations during irradiation.

35. E. Storm, E. Gilbert, and H. Israel, Gamma-Ray Absorption Coefficients for Elements 1 Through 100 Derived from the Theoretical Values of the National Bureau of Standards, LA-2237 (November 18, 1958). Availability: Dep.; $37.80 (ph), $11.10 (mf) OTS.

A compilation of gamma-ray absorption coefficients for elements 1 through 100 derived from the theoretical values of the National Bureau of Standards is presented.

36. R. C. Hawkings, W. J. Edwards, and E. M. McLeod, Tables of Gamma Rays from the Decay of Radionuclides, CRDC-1007 (March 1961). Availability: Dep.; $8.00 AECL.

The energies and abundances of gamma rays from the decay of those radionuclides having a half life $\geq$ 1 sec are listed. The radionuclide decay modes, half lives, and production modes are also listed.

37. D. Meneghetti, "Discrete Ordinate Quadratures for Thin Slab Cells," Nucl. Sci. Eng., 14, 3, 295-303 (November 1962).

Fluxes of thin slab cells calculated by discrete SN method, DSN (N = 2, 4, 8, 16), single-spherical harmonics, PN (N = 1, 3, ..., 11, 13) and double-spherical harmonics, DPN (N = 1, 2, 3, 4, 5), are compared. The anisotropic flux component occurs predominantly in the region about $\mu = 0$. The spatially dependent part of the spatial flux is largely determined by a few nearest-neighbor source regions. The anisotropic flux component originates largely from the isotropic effective sources arising from the non-spatially dependent part of the spatial flux.

Two quadrature forms, for use in discrete ordinate methods of solutions for thin cells, are described which emphasize the small $|\mu|$ directions. One is based on a simplified integral transport approximation. The other is a modified form of the single Gaussian quadrature. Methods of choosing the quadratures and representative results are discussed.

38. M. J. Barrett, "Radiation of an Extended Source," Nucl. Sci. Eng., 14, 2, 186-191 (October 1962).

The gamma current intensity (flux) and the gamma emission (leakage) of a homogeneous, spherical medium are derived by inserting a source spectrum in the Boltzmann transport equation. In the range of photon energies from 0.5 MeV to 10.5 MeV, Compton scattering by electrons of the medium dominates the energy degradation of photons, so that one may use Klein-Nishina cross sections for the transfer kernel. Lumping the flux into energy groups permits an approximation of the transport equation as a matrix equation. Numerical solutions for the flux and leakage spectra, found by inverting the matrix equation, agree well with the results of previous theoretical studies.

39. W. R. Faust and A. D. Anderson, "Penetration and Diffusion of a Beam of Radiation," Nucl. Sci. Eng., 14, 1, 17-21 (September 1962).

An approximate solution is obtained to the three-dimensional equation of transport for a point source radiating along the Z-axis. The radiation is separated into scattered and unscattered components, and the transport equation for the scattered radiation is derived. The appropriate solution for the scattered radiation is obtained by expanding the intensity in tesseral harmonics and solving the resulting equations for the amplitude of each harmonic. Some numerical results are presented to illustrate the behavior of the solution at great depths of penetration.

40. R. Kladnik and I. Kuscer, "Velocity Dependent Milnes Problem," Nucl. Sci. Eng., 13, 2, 149-152 (June 1962).

Infinite medium solutions of the transport equation are used to obtain two integrals of this equation, to express the asymptotic approximation to the solution of Milne's problem, and to construct a variational trial solution. This solution is improved by one iteration. The linear extrapolation distance is expressed by the stationary value of the variational functional.

41. G. R. Dalton, "Interaction of Adjacent Neutron Detectors," Nucl. Sci. Eng., 13, 2, 190-196 (June 1962).

The time and speed independent neutron transport equation in its integral form is applied to the calculation of the disturbed neutron flux in the vicinity of neutron detectors. The method is also applied to the calculation of the average flux in a two-detector system where the detectors are close enough to interact with each other. Hand methods of solving the integral equations are developed and shown to give results in good agreement with more accurate numerical methods of solving the integral equations.

42. Z. Weiss, "The Slowing Down of Neutrons as a Stochastic Process," Nucl. Sci. Eng., 12, 4, 464-468 (April 1962).

The solution of the neutron slowing down equation in an infinite, homogeneous moderator (with nuclei of mass M) has been obtained by means of direct summation of transition probabilities between the initial and final energy states. It has been possible to obtain an exact formula for the distribution of neutrons in lethargy space after N collisions with moderator nuclei. The asymptotic expansion of the results for large and for very large N is in agreement with the Dancoff's formulas. The accuracy of the asymptotic expansion has been estimated.

43. M. Tobias and T. B. Fowler, "The Equipoise Method - A Simple Procedure for Group Diffusion Calculations in Two and Three Dimensions," Nucl. Sci. Eng., 12, 4, 513-518 (April 1962).

An extremely simple iterative procedure is described for performing group-diffusion calculations in two and three dimensions. No proofs have been found to guarantee its convergence, but successful experience with a wide variety of problems, some realistic, others with specially introduced difficulties, shows the method to be rapid and reliable. Three large computer programs have been devised embodying the principle: EQUIPOISE-3, TWENTY GRAND, and WHIRLAWAY. The first two are two-dimensional programs, while the third is three-dimensional.

If desired, the programs can be used to compute adjoint fluxes and those integrals necessary for perturbation calculations automatically. Possible further applications of the method are suggested.

44. L. Rosen and L. Stewart, The Neutron-Induced Disintegration of Li-6 and Li-7 by 5- to 14-MeV Incident Neutrons, LA-2643 (October 22, 1961). Availability: Dep.; $2.50 OTS.

A detailed study of the $Li^6(n,dn)He^4$ and $Li^7(n,tn)He^4$ reactions has been made over the energy region from 5 to 14 MeV. The reaction cross sections, as well as the spatial and spectral distributions of all reaction products were determined for eleven energies for Li^7 and twelve energies for Li^6. The high reaction cross sections have important implications for the design of mobile reactor shields.

45. H. Lustig, Differential Elastic Neutron Scattering Cross Section for Oxygen, NDA-2111-3, Vol. A (November 1, 1959). Availability: $3.30 (fs), $2.40 (mf) OTS as PB-146777.

A set of elastic scattering angular distributions is presented for neutron energies up to 18 MeV. Calculations undertaken for the determination of resonance parameters in the incident neutron energy region from 2.5 to 4.0 MeV are described.

46. E. S. Troubetzkoy, Fast-Neutron Cross Sections of Iron, Silicon, Aluminum, and Oxygen, NDA 2111-3, Vol. C (November 1, 1959). Availability: $9.30 (fs), $3.60 (mf) OTS as PB-146779.

A set of neutron cross sections for iron, silicon, aluminum, and oxygen has been prepared for energies up to 18 MeV. The cross sections tabulated include σ_T, σ_{nn}, $\sigma_{nn'}$, σ_{nx}, $\sigma_{n\gamma}$, $\sigma_n(\theta)$, $\sigma_{nn'}(E,E')$, $\sigma_{nn'\gamma}(E,E_\gamma)$, $\sigma_{n,2n}$, and cross sections for charged-particle emission.

47. E. S. Troubetzkoy, M. H. Kalos, H. Lustig et al., Fast-Neutron Cross Sections of Manganese, Calcium, Sulfur, Sodium, NDA-2133-4 (January 31, 1961). Availability: $7.60 OTS as AD-255536.

A set of neutron cross sections for manganese, calcium, sulfur, and sodium has been prepared for energies up to 18 MeV. The cross sections tabulated include σ_T, σ_{nn}, $\sigma_{nn'}$, σ_{nx}, $\sigma_{n\gamma}$, σ_{np}, $\sigma_{n\alpha}$, $\sigma_{n,2n}$, $\sigma_n(\theta)$, $\sigma_{nn'}(E,E')$, and $\sigma_{nx\nu}(E,E_\nu)$.

48. A. D. Krumbein and H. Goldstein, Neutron Cross Sections for Beryllium, UNC-5014, Vol. B (May 31, 1962). Availability: $2.60 OTS as NDL-TR-36, Vol. B (AD-276088).

Neutron cross sections for energies up to 18 MeV are given for beryllium, along with a brief description of each cross-section table. Revisions made in previous compilations are noted and the uncertainties in the existing data are discussed.

49. N. Tralli, M. H. Kalos, A. J. Longano et al., Neutron Cross Sections for Titanium, Potassium, Magnesium, Nitrogen, Aluminum, Silicon, Sodium, Oxygen, and Manganese, UNC-5002 (January 31, 1962). Availability: $10.10 OTS as AD 272 809.

A set of neutron cross sections for titanium, potassium, magnesium, and nitrogen has been prepared for neutron energies from 0.0253 eV to 18 MeV. The cross sections tabulated include σ_T, $\sigma_{n,n}$, $\sigma_{n,\gamma}$, $\sigma_n(\theta)$, $\sigma_{n,n'}(E,E')$, $\sigma_{n,n'\gamma}(E,E_\gamma)$, σ_{n2n}, and cross sections for charged-particle emission.

A revision of some of the neutron cross sections for Al, Si, Na, and O given in NDA 2111-3, Vol. C, Nov. 1, 1959, and NDA-2133-4, Jan. 31, 1961, has been made based on experimental results which have appeared in the literature since the publication of these reports. In addition, some typographical errors in the tabulation of $\sigma_{n,n}$ for Mn in NDA 2133-4 are corrected.

The results of Hauser-Feshbach calculations of reactions of O^{16} are reported.

50. P. W. Schreiber and D. T. James, Flux Measurements in the Battelle Memorial Institute Fast-Neutron Activation Facility, XDC 61-1-103 (January 1961). Availability: Dep. (mc); $4.80 (ph), $2.70 (mf) OTS.

Fast-neutron foil activation traverses were run in the No. 4 6-in. beam hole of the Battelle Memorial Institute reactor by R. Jung.

Foil activation and integral neutron fluxes are reported for the foil shield tank filled with water and isopropyl biphenyl. These liquids provided sufficient neutron spectrum hardening to indicate the relative activation threshold energies for the various materials.

51. G. D. Joanou, A. J. Goodjohn, and N. F. Wikner, Legendre Expansion Coefficients for the Angular Distribution of Elastically Scattered Neutrons and Fast-Neutron Cross Sections for D, Be, C, O, Zr, Pb, and Bi, GA-2156 (May 15, 1961). Availability: Dep.; $2.00 OTS.

An analysis was made of experimentally determined angular distributions for elastically scattered neutrons in terms of Legendre polynomial expansion coefficients. The first six coefficients of the expansion were calculated as a function of energy from 0.1 to 14.0 MeV. In addition to the tabulated results, the procedures used in obtaining the coefficients and the cross-section data for each element are discussed.

52. A. Prince, The Status of Neutron Cross Sections and Related Data, XDC-60-11-164 (November 30, 1960). Availability: Dep. (mc); $15.30 (ph), $5.40 (mf) OTS.

The current status of experimentally defined neutron cross sections for materials pertinent to shield analysis is presented. In addition, the experimental and theoretical techniques employed in obtaining accurate data are outlined. Materials included are:

Hydrogen	Oxygen	Yttrium
Lithium	Aluminum	Zirconium
Beryllium	Chromium	Molybdenum
Boron	Iron	Tungsten
Carbon	Cobalt	Lead
Nitrogen	Nickel	Uranium

53. G. T. Chapman and C. L. Storrs, Effective Neutron Removal Cross Sections for Shielding, AECD-3978 (September 19, 1955). Availability: Dep.; $0.75 OTS.

The effective removal cross-section concept as applied in shield calculations is discussed and a method of determining the numerical value of this cross section from LTSF experimental data is presented. Radiation intensity measurements in water beyond some 20 elements and compounds and the corresponding removal cross-section values are reported. These values are useful in determining the relative shielding effectiveness of the various materials, but can be used with complete assurance for shielding calculations only when the geometry under consideration closely resembles that employed at the LTSF. Specifically there must be many relaxation lengths of hydrogenous shield following the material.

A graphical comparison of the effective removal cross sections and the total cross sections at a high neutron energy (8 MeV) is also made and should be applicable to shield calculations. In addition, calculations are presented of correction factors for various values of the water thickness, sample thickness, and attenuation lengths in order to facilitate the calculation of the removal cross section from future measurements.

54. D. K. Trubey and G. T. Chapman, Effective Neutron Removal Cross Sections for Carbon and Oxygen in Continuous Mediums, ORNL-2197 (September 3, 1958). Availability: Dep.; $3.60 (fs), $1.10 (mf) OTS.

Thermal-neutron flux measurements at the Lid Tank Shielding Facility in water, oil, and a sugar-water solution have been used to calculate effective neutron removal cross-section values of carbon and oxygen distributed in continuous mediums. The values are 0.72 ± 0.05 barn for carbon and 0.92 ± 0.05 barn for oxygen. These values are somewhat lower than previously reported values determined by another method, but the differences may be attributable to experimental errors.

55. L. V. Spencer, Structure Shielding Against Fallout Radiation from Nuclear Weapons, NBS Monograph 42 (June 1, 1962). Availability: $0.75 GPO.

The theory of structure shielding from fallout gamma radiation is developed to the point of applications to elementary structure types. Examples discussed in the text include the density interface, foxhole, shielded foxhole or basement, light superstructure, vertical wall, blockhouse, vents, compartmentalization effects, and mazes. A large number of engineering charts and graphs are presented for engineering calculations, including many obtained from angular distributions of the exposure dose. Results are given for a fission spectrum, and for Co^{60} and Cs^{137} sources. This information has been obtained almost completely by machine calculations utilizing basic cross-section data. A number of sources of experimental data are mentioned, but detailed comparisons with experiment are not included.

56. R. J. Howerton, Tabulated Differential Neutron Cross Sections, Vol. I, Part III, UCRL-5573 (January 1961). Availability: Dep.; $5.00 OTS.

Tables are presented of experimental differential neutron cross sections for the elastic scattering of neutrons by nuclei in the energy range of 0 to 15 MeV. Nuclear reactions induced by neutrons are also included, particularly those that are significant for reactor-type calculations. The tables include nuclei from H to Pu.

57. N. G. Sjöstrand and A. L. Story, Neutron Cross Sections and Fission Parameters of U-233, U-235, and Pu-239 at 2200 M/sec, AEEW-M 125 (April 1961). (United Kingdom Atomic Energy Authority.)

This memorandum is to be published as one chapter (4) of a book entitled "Nuclear Data for Reactor Design, Vol. I" by N. G. Sjöstrand and J. S. Story (pub. Pergammon Press), and is a critical assessment of experimental data relating to the 2200 m/sec cross sections of the three main fissile materials.

58. R. Sher and J. Felberbaum, Least Squares Analysis of the 2200 M/sec Parameters of U-233, U-235, and Pu-239, BNL-722 (June 1962). Availability: Dep.; $0.75 OTS.

An analysis similar to that of Du Mond and Cohen on the atomic constants has been applied to measurements of the 2200 m/sec cross sections of U^{233}, U^{235}, and Pu^{239} considered as a single set. The following values are obtained:

U^{233}: $(\sigma_T = 591 \pm 3)$, $\sigma_a = 576.1 \pm 2.7$, $\sigma_f = 527.5 \pm 2.4$, $\nu = 2.503 \pm 0.010$

U^{235}: $(\sigma_T = 694 \pm 3)$, $\sigma_a = 682.0 \pm 2.6$, $\sigma_f = 582.2 \pm 2.2$, $\nu = 2.430 \pm 0.009$

Pu^{239}: $(\sigma_T = 1040 \pm 10)$, $\sigma_a = 1030.1 \pm 7.4$, $\sigma_f = 748.2 \pm 4.9$, $\nu = 2.882 \pm 0.016$

From these values, the following values are derived:

U^{233}: $\eta = 2.292 \pm 0.008$, $\alpha = 0.092 \pm 0.003$

U^{235}: $\eta = 2.074 \pm 0.006$, $\alpha = 0.171 \pm 0.003$

Pu^{239}: $\eta = 2.093 \pm 0.014$, $\alpha = 0.377 \pm 0.011$

The values in parentheses, the σ_T's, are obtained by adding the estimated scattering cross section to the least squares value of σ_a.

59. R. E. Lynch, J. W. Benoit, W. P. Johnson et al., A Monte Carlo Calculation of Air Scattered Gamma Rays, ORNL-2292 (September 10, 1958). Availability: Dep.; $7.60 (fs), $2.30 (mf) OTS.

A calculation is presented of air-scattered gamma rays using the Monte Carlo method of computation. For simplicity, the problem was idealized to have a monoenergetic, monodirectional point source of gamma rays in an infinite body of air. The results of the calculation included the angular distributions, energy spectra, and tissue dose rates at source-detector separation distances of from 5 to 100 ft. A complete parameter survey was completed for source energies from 0.6 to 12 MeV and for beam angles with respect to the source-detector axis of from 0 to 180 deg. The report is divided into five volumes: Volume 1 gives a complete description of the mathematical procedure used in the calculation, while volumes 2 and 3 present tables of data representing the energy spectra and angular distributions obtained from the calculation. Volumes 4 and 5 present information similar to that in volumes 2 and 3, except that the data have been converted to tissue dose rates.

60. M. A. Capo, Gamma-Ray Absorption Coefficients for Elements and Mixtures, APEX-628 (August 1961). Availability: Dep.; $0.75 OTS.

Mass gamma-ray total and energy absorption coefficients are tabulated as a function of energy from 0.2 MeV to 10 MeV for various elements, mixtures, and compounds. The data presented here are based on the latest published results of Dr. G. White Grodstein, National Bureau of Standards. Polynomial expressions are given for approximation of absorption coefficients for elements or energy levels not tabulated by Dr. Grodstein.

61. L. B. Engle and P. C. Fisher, Energy and Time Dependence of Delayed Gammas from Fission, LAMS-2642 (July 7, 1962). Availability: Dep.; $1.00 OTS.

This report contains some results of an experiment designed to measure the energy and time dependence of the gammas emitted after effectively instantaneous neutron-induced fission of Th^{232}, U^{233}, U^{235}, U^{238}, and Pu^{239}. Seventeen-bin distributions of the absolute number of photons/fission-sec-MeV and intercomparisons and integrations over energy and time of these distributions are given. Gammas having energies between 0.12 and 6.5 MeV were detected during five short time intervals between 0.2 sec and 45 sec after fission occurred.

62. H. E. Hungerford, "The Nuclear, Physical, and Mechanical Properties of Shielding Materials," Reactor Handbook, 2nd Ed., Vol. I, Chapt. 51 (Materials) (C. R. Tipton, Jr., ed.), 1960. (Interscience Publishers)

Many tables of physical and nuclear properties of importance in shielding applications are presented for many materials.

63. W. J. Stapp and M. Tetenbaum, "Radiation Damage," Reactor Handbook, 2nd Ed., Vol. I, Chapt. 52 (Materials) (C. R. Tipton, Jr., ed.), 1960. (Interscience Publishers).

A summary of radiation effects is presented for materials which may be used in a shielding application.

64. R. D. Schamberger, F. C. Maienschein et al., "Bulk Shielding Data," Reactor Handbook, 2nd Ed., Vol. I, Chapt. 53 (Materials) (C. R. Tipton, Jr., ed.), 1960. (Interscience Publishers)

Extensive graphs and tables of data obtained from rather basic experiments at several shielding facilities are presented. The facilities are the Bulk Shielding Reactor, the Lid Tank Shielding Facility, the Ground Test Reactor, the Aircraft Shield Test Reactor, and the Brookhaven Shielding Facility.

65. J. G. Carver, U-235 Fission Gamma-Ray Source Data, XDC 60-11-63 (November 9, 1960). Availability: Dep. (mc); $6.30 (ph), $3.00 (mf) OTS.

Presented in this report are the latest data available concerning gamma rays associated with the fission of U^{235} both during fission and for long times afterward. The data is tabulated in a convenient manner for use in shielding calculations. The gamma rays were categorized as follows: (1) prompt, (2) short half-life, (3) intermediate life, and (4) delayed gamma rays. The data contained here is not intended for use in predicting so-called shutdown dose rates, but rather describing completely the radiation coming from a reactor during operation, after that operation has continued for an appreciable time.

66. E. S. Troubetzkoy and H. Goldstein, A Compilation of Information on Gamma-Ray Spectra Resulting from Thermal-Neutron Capture, ORNL-2904 (May 17, 1960). Availability: Dep.; $2.25 OTS.

This report consists of a compilation of thermal-neutron capture gamma-ray spectra of various nuclei which are useful for reactor shielding applications. The bulk of the data is taken from two groups: a Chalk River group, originally led by B. B. Kinsey and later by G. A. Bartholomew, and a group headed by L. V. Groshev at the Institute of Atomic Energy of the Academy of Sciences USSR. All the data are presented in tabular form for seven energy intervals from 0 to above 9 MeV. In addition, figures illustrating the appearance of the spectra are given, including continuous differential spectra for 50 nuclei and line spectra for 15 nuclei. The line spectra consist of vertical lines having heights proportional to intensities.

67. M. O. Burrell, Nuclear Radiation Transfer and Heat Deposition Rates in Liquid Hydrogen, NASA TN D-1115 (August 1962). (National Aeronautics and Space Administration.) Availability: $2.25 OTS.

Stochastic methods are used to calculate the radiation transport and energy deposition of neutrons and gamma rays in liquid hydrogen slabs and cylinders. The sources are treated as monoenergetic and either point isotropic for the cylinder or plane parallel rays for the slabs. A description of the methods used and a rather extensive compilation of results are given. The results include heat rate deposition as a function of depth, albedo factors, slow-neutron spatial distributions, and transmitted angular distributions of gamma rays.

68. R. L. Walker and M. Grotenhuis, A Summary of Shielding Constants for Concrete, ANL-6443 (November 1961). Availability: Dep.; $2.00 OTS

This report is intended to fulfill the need to consolidate, classify, and revise the present knowledge of the shielding constants of concretes. The densities, elemental compositions, and mixes, where available, are given for a wide range of concretes. From these data, various constants useful for shielding calculations have been computed. These constants include the removal cross sections, total cross sections, average thermal-neutron absorption cross sections, thermal-neutron diffusion coefficients, reciprocal thermal-neutron diffusion lengths, total gamma-ray linear attenuation coefficients, gamma-ray energy absorption linear attenuation coefficients, the effective atomic numbers for the determination of buildup factors, and the capture gamma-ray spectra. Experimental results are included where available.

69. M. Leimdorfer, "The Backscattering of Gamma Radiation from Spherical Concrete Walls," Nucl. Sci. Eng., 17, 3, 345-351 (November 1963).

The Monte Carlo technique has been applied to investigate the effect of wall curvature on the backscattering properties of concrete. The wall was considered infinitely thick and the source radiation was normally incident. Monte Carlo calculations were only performed at 1-MeV source energy but an analytical formula was derived to facilitate extrapolation to other energies as well as materials. The results show that for practical purposes the plane wall albedo is a sufficient, and conservative, approximation, 90% of its value being reached at a concrete wall radius of about 100 cm for source energies up to 10 MeV.

70. M. Leimdorfer, "The Backscattering of Radiation from Plane Concrete Walls," Nucl. Sci. Eng. 17: 352-356 (November 1963).

Monte Carlo calculations have been performed for source energies from 1 to 10 MeV, and normally incident radiation, showing that 90% of the infinite-barrier energy flux albedo is reached with a 10-cm concrete wall. The spectrum of backscattered energy flux is presented for the above sources and wall thicknesses ranging from 5 to 50 cm. An analytical expression, based on a single-scattering approximation, is shown to produce good fits to the Monte Carlo results.

71. M. Leimdorfer, "Multiple Scattering of Gamma Radiation in a Spherical Concrete Wall Room," Nucl. Sci. Eng. 17: 357-364 (November 1963).

The Monte Carlo method has been applied to the calculation of the energy flux of scattered gamma radiation in a spherical room surrounded by an infinitely thick spherical wall and with a point source at the center. Source energies were 1, 2, 4, 6, and 10 MeV. The main investigation was carried out at a room radius of 500 cm but, for the 1-MeV source, the influence of varying the room radius down to 1 cm was analyzed. The results contain energy distributions of the first four successive reflection components at the center of the room and at the wall surface, as well as spatial distribution of the successive energy flux components. The neglect of reflection contributions of order five and higher was estimated to introduce an error of less than 0.2% of the total scattered energy flux.

An analytical approximation is shown to produce a useful and easily applicable method of predicting the amount of scattered radiation in a spherical room.

72. L. A. Bowman and D. K. Trubey, Deposition of Gamma-Ray Heating in Stratified Lead and Water Slabs, ORNL CF 58-7-99 (July 28, 1958). Availability: Dep. (mc), $4.80 (ph), $2.70 (mf) OTS.

This memorandum presents typical results from a calculation of the deposition of heat in stratified lead and water slabs caused by a monodirectional, monoenergetic beam of gamma rays incident on the slabs. A total of 512 cases were calculated for infinite slabs with finite thicknesses of 1, 2, 4, and 6 mean free paths; source energies of 1, 3, 6, and 10 MeV, and source angles of incidence which were chosen to give slant slab thicknesses of 1, 2, 3, and 4 times the normal thickness. The results were fitted to an empirical formula, which can be simplified for special cases. While for the cases examined, the fit was usually good to within 5%, it is to be emphasized that the formula has been compared only with the results from a very limited number of parameters.

73. D. M. Peterson, Radiation Heating Methods Report, NARF-60-5T (MR-N-249) (June 30, 1960). (Convair, Fort Worth.) Not for sale to public.

This report describes methods for evaluating the volumetric heat source in a material, due to neutron and gamma-ray reactions within the material. Methods are set forth for solving the resulting heat-transfer and thermal-stress problems in cylindrical or slab geometries.

74. O. H. Hill and E. E. Kerlin, Experimental Investigation of Radiation Heating in Materials, NARF-61-38T (MR-N-290) (December 12, 1961). (Convair, Fort Worth.) Not for sale to public.

An experimental study has been performed to measure the nuclear-reactor radiation-induced heating rates in lead, 1018 steel, Marlex Type 2 Polyethylene, and an 80% lithium hydride-polyethylene matrix material. The data obtained from this study will be compared with data predicted by methods employing computer techniques.

Thermal transducers of the material types were fabricated to provide for the measurement of rate of temperature rise at some nine to twelve isolated planes in the material. The transducers were embedded in slabs of material which approximated an infinite half-plane configuration exposed to the reactor. Measurements are given of heating rates observed on the reactor centerline and at various off-centerline positions.

75. G. E. Miller, Analysis of Radiation Heating Experiment, FZK-9-169 (NARF-61-40T) (December 29, 1961). Availability: OTS.

Heat deposition due to nuclear radiation is a problem of great concern in the design of high-power reactor systems. To aid in the solution of this problem, two high-speed digital-computer codes were developed for calculating heat generation rates. These codes are based on the differential-energy spectra data obtained by a moments-method solution of the Boltzmann transport equation and yield the modified spectrum at the point of interest from a given source point after penetration of a multilayer shield. A summation of the spectra from the individual source points gives the total spectra. Application of energy-absorption coefficients and an integration of the spectrum over energy yields the heat generation rate. To provide a verfication of the results obtained from the codes, four materials - lead, iron, polyethylene, and lithiated polyethylene - were exposed to radiation from the Ground Test Reactor. The calculated results agree within 40% of the measured heating rates.

76. G. T. Western, Energy and Angular Distribution Experiment, Vol. I, Angular Distribution of Reactor Radiation from Slabs and of Emergent Secondary Gamma Rays, NARF-62-16T (FZK-9-183-1) (December 31, 1962). Availability: $6.60, OTS as AD-295718.

Radiation from the Aerospace Systems Test Reactor positioned within the Outside ASTR Tank was directed, with a narrow-beam slab geometry, onto test materials of 3% borated polyethylene and of lead. Measurements for polyethylene and lead, except as noted, were made of (1) the radiation source term; (2) the flux distribution of thermal and fast neutrons within the test materials; (3) the angular distribution of thermal- and epithermal-neutron flux, fast-neutron dose rate, and secondary gamma rays from the test materials; (4) the reflection of neutrons from borated polyethylene and from steel; (5) the angular distribution of fast-neutron and gamma-ray number-energy flux; and (6) the angular distribution of gamma-ray dose rate resulting from primary gamma rays scattering in and from the test materials. Data for items 1, 2, 3, and 4 are presented in Volume I of this report; data for items 5 and 6 will be published in Volume II.

77. M. J. Berger and J. W. Cooper, "Reflection of Fast Neutrons from Water," J. Res. Nat. Bur. Std., A, 63A, 2, 101 (September 1959).

The backscattering of fast (0.3, 1, 3, 6, 9, and 14 MeV) neutrons from a semi-infinite water medium has been calculated by the Monte Carlo method. The information obtained includes the joint angular and spectral distribution of the reflected neutrons, the dependence of the number albedo and energy albedo on the source energy and obliquity, and the contributions to the albedo of successive orders of scattering. The spectra were calculated down to epithermal energies (~ 0.5 eV). The results for each case are based on the analysis of 3,000 neutron histories, generated with the use of an IBM-704 computer. In the random sampling, elastic scattering from hydrogen and oxygen, inelastic scattering from oxygen and absorption due to n-α and n-p processes were taken into account. The cross sections for some of these interactions are not well known. Parallel calculations with different assumptions about the cross sections were made in order to estimate how sensitively the albedo depends on the cross sections. The paper includes a self-contained description of the Monte Carlo method, its application to the calculation of radiation diffusion and in particular to the neutron albedo problem. Emphasis is placed on the technique of correlated sampling which makes possible an accurate estimate of albedo differences resulting from different assumptions about the cross sections. The random sampling computations were supplemented by analytical calculations of the single-scattering albedo. This was useful for an understanding of the Monte Carlo results because a considerable fraction of the reflected neutrons return after only one collision.

78. C. M. Huddleston. Z. G. Burson, R. M. Kinkaid et al., Ground Roughness Effects on Energy and Angular Distribution of Gamma Radiation from Fallout, CEP 62.81, EG+G L-586 (October 1962). Availability: OTS, $1.25.

The effect of ground roughness on the radiation field above ground which has been contaminated with fallout from explosion in the atmosphere of a nuclear device was investigated for three types of Nevada terrain: (1) a flat dry lake bed, (2) semi-rough terrain or land which has a known and controlled degree of roughness, and (3) typical wild desert. A modified Navy 5-ton 6 x 6 truck was used as a mobile laboratory. A collimator and a scintillation crystal with a 512-channel analyzer were used to measure gamma-ray spectra at various polar angles. Dose-vs-height measurements were also made up to a height of 40 feet.

The project was undertaken to determine the importance of ground roughness as a factor in radiation shielding from fallout.

79. R. H. Peterson and J. A. Williams, A Complete Set of Functions for Describing the Angular Distribution of the Neutron Flux from a Point Source in Air, BRL-1432 (September 1962). Availability: $4.60 (fs) as AD-291045.

Based on the properties of Jacobi polynomials, a set of functions $g_n^m(\cos\theta)$ cos $m\phi$ which is complete over a unit sphere are developed. They differ from the spherical harmonics $P_n^m(\cos\theta)$ cos $m\phi$ in that the zonal terms $g_n^o(\cos\theta)$ contain singularities of order $(1 - \cos\theta)^{-1/2}$ at $\theta = 0^o$ and are finite elsewhere thus providing a rapidly converging series for the observed peakedness of the angular distributions of neutron fluxes from a poing source in air. Tables of the function $g_n^m(\cos\theta)$ as well as its integral and its adjoint for $0 \leq m \leq n$ and $0 \leq n \leq 4$ are included.

80. H. Hurwitz, Jr., Note on a Theory of Minimum Weight Shields, KAPL-1441 (TID-4500) (January 1957). Availability: Dep.; $1.60 (fs), $0.80 (mf) OTS.

A mathematical solution of the minimum weight problem is obtained under the assumption that the neutron distribution in the shield is determined in advance and does not depend on the distribution of heavy material.

81. R. R. Coveyou, "Serial Correlation in the Generation of Pseudo Random Numbers," J. Assn. Comput. Mach. 7, 1, 72 (January 1960).

The serial correlation between a pseudo-random number and its immediate successes for the class of pseudo-random numbers generated by the method of congruence is estimated as well as the method of obtaining the correlation between two members of the sequence separated by p-1 other members.

82. A. J. Weiss, W. D. Tucker, and L. G. Stang, Li-6D as a Source of 14-MeV Neutrons, BNL-5366 (March 1961). Availability: Dep. (mc), $1.60 (fs), $0.80 (mf) OTS.

Lithium-6 deutride was prepared and irradiated with thermal neutrons to produce a flux of 14-MeV neutrons, by the $Li^6(n,t)He^4$ and $D(t,n)He^4$ reactions. The flux produced was determined by measuring the Sc^{42} activity induced in a vanadium monitor. The best ratio of fast-to-thermal flux obtained was 7×10^{-5}. The fast-neutron flux obtainable with Li^6D compared favorably with that obtained by other methods.

83. H. B. Ethridge, P. R. Flusset, and R. H. Ritchie, Proton Recoil Energy Loss Distribution in Plane Geometry, ORNL-3089 (June 1962). Availability: Dep.; $0.75 OTS.

Computation of proton recoil energy losses from neutron irradiation of hydrogenous systems possessing plane symmetry have been made for the purpose of fast-neutron counter design. A digital computer code has been written to carry out the calculations for two irradiation geometries: (1) the neutron beam is assumed to be normally incident upon the system, and (2) the system is exposed to an isotropic flux of neutrons. Numerical results are presented for neutrons of various energies incident on plane radiators of different thicknesses for these two irradiation geometries. These results have been used in the conceptual design of fast-neutron dosimeters employing a generalized dosimetry principle. The applications are described more fully elsewhere.

84. R. D. Schamberger, F. J. Shore, and H. P. Sleeper, Jr., Transmission of Neutrons and Gamma Rays Through Air Slots. Part VI. Effects of Multiple Offsets on Neutron Transmission of Air Slots, BNL-2024 (September 1954). Availability: Dep.; $3.30 (ph), $2.40 (mf) OTS.

This report is part of the series of BNL reports 2019 through 2028. See accession number 95, which is BNL-2019, for the description of the experimental set up.

85. H. J. Kouts, Theory of Flux Perturbation by Voids in Shields, BNL-1357 (September 1952). Availability: Dep.; $4.80 (ph), $2.70 (mf) OTS.

The one-velocity integral transport equation is derived from first principles assuming, at first, a source distribution and a convex scattering medium over which the cross sections are constant. This restricted scattering geometry is later modified to include the presence of shield voids. The one-group theory is then applied to spherical voids in water. The two-group calculations are developed more empirically on the assumption that in the absence of voids the flux can be divided into two-energy groups which are in equilibrium. One group consists of a relatively high-energy component with a strong forward angular distribution, and the second group consists of an isotropic secondary component which must be visualized as originating from multiple scattering of the primaries.

86. M. Leimdorfer, Gamma Radiation from Neutron Induced Ground Activity for a Given Bomb Composition, Explosion Altitude, and Ground Analysis, FOA 4 Report C 4135-454 (March 1962). Availability: Dep.; (mc), $1.10 (fs), $0.80 (mf) JCL.

A method for determining neutron-induced activity in the ground is described. Results from a test are included along with the description of a bomb composition from which the neutrons are assumed to originate.

87. R. D. Schamberger, F. J. Shore, and H. P. Sleeper, Jr., Transmission of Neutrons and Gamma Rays Through Air Slots. Part IV. Effect of an Offset in the Transmission of Neutrons Through an Air Slot in Water, BNL-2022 (September 1954). Availability: Dep.; $1.60 (fs), $0.80 (mf) OTS.

This report is part of the series of BNL reports 2019 through 2028. See accession number 95, which is BNL-2019, for the description of the experimental set up.

88. R. D. Schamberger, F. J. Shore, and H. P. Sleeper, Jr., Transmission of Neutrons and Gamma Rays Through Air Slots. Part X. Angular Distribution of Neutrons Emerging from an Air Slot, BNL-2028 (September 1954). Availability: Dep.; $3.30 (ph), $2.40 (mf) OTS.

This report is part of the series of BNL reports 2019 through 2028. See accession number 95, which is BNL-2019, for the description of the experimental set up.

89. R. D. Schamberger, F. J. Shore, and H. P. Sleeper, Jr., Transmission of Neutrons and Gamma Rays Through Air Slots. Part IX. Thermal and Epithermal Neutron Distributions Through an Air Slot in Water, BNL-2027 (September 1954). Availability: Dep.; $1.10 (fs), $0.80 (mf) OTS.

This report is part of the series of BNL reports 2019 through 2028. See accession number 95, which is BNL-2019, for the description of the experimental set up.

90. R. D. Schamberger, F. J. Shore, and H. P. Sleeper, Jr., Transmission of Neutrons and Gamma Rays Through Air Slots. Part VIII. Effect of the Source Size on the Neutron Transmission of an Air Slot, BNL-2026 (September 1954). Availability: Dep.; $1.60 (fs), $0.80 (mf) OTS.

This report is part of the series of BNL reports 2019 through 2028. See accession number 95, which is BNL-2019, for the description of the experimental set up.

91. R. D. Schamberger, F. J. Shore, and H. P. Sleeper, Jr., Transmission of Neutrons and Gamma Rays Through Air Slots. Part VII. Effect on Neutron Transmission of Changing the Wall Material of an Air Slot, BNL-2025 (September 1954). Availability: Dep.; $1.80 (ph), $1.80 (mf) OTS.

This report is part of the series of BNL reports 2019 through 2028. See accession number 95, which is BNL 2019, for the description of the experimental set up.

92. R. D. Schamberger, F. J. Shore, and H. P. Sleeper, Jr., Transmission of Neutrons and Gamma Rays Through Air Slots. Part V. Effect of the Vertical Position of a Single Offset on the Neutron Transmission of an Air Slot, BNL-2023 (September 1954). Availability: Dep.; $1.80 (ph), $1.80 (mf) OTS.

This report is part of the series of BNL reports 2019 through 2028. See accession number 95, which is BNL-2019, for the description of the experimental set up.

93. R. D. Schamberger, F. J. Shore, and H. P. Sleeper, Jr., Transmission of Neutrons and Gamma Rays Through Air Slots. Part III. Transmission of Gamma Rays Through Straight Air Slots in Water, BNL-2021 (September 1954). Availability: Dep.; $3.30 (ph), $2.40 (mf) OTS.

This report is part of the series of BNL reports 2019 through 2028. See accession number 95, which is BNL-2019, for the description of the experimental set up.

94. R. D. Schamberger, F. J. Shore, and H. P. Sleeper, Jr., Transmission of Neutrons and Gamma Rays Through Air Slots. Part II. Transmission of Neutrons Through Straight Slots in Water, BNL-2020 (September 1954). Availability: Dep.; $2.60 (fs), $1.13 (mf) OTS.

This report is part of the series of BNL reports 2019 through 2028. See accession number 95, which is BNL-2919, for the description of the experimental set up.

95. R. D. Schamberger, F. J. Shore, and H. P. Sleeper, Jr., Transmission of Neutrons and Gamma Rays Through Air Slots. Part I. Mechanical Construction of the Slots and Representation Background Data, BNL-2019 (September 1954). Availability: Dep.; $3.30 (ph), $2.40 (mf) OTS.

This report is the first of a series of ten reports describing experimental investigations of the streaming of neutrons and gamma rays through air slots. The air slots are surrounded by water in the BNL shield Tank. Multiple offsets as well as straight slots were investigated. The other reports in the series are BNL-2020 through 2028.

96. W. W. Pratt and H. J. Kouts, Leakage of Gamma Radiation Through Spherical Voids, BNL-158 (March 1952). Availability: Dep.; $3.30 (ph), $2.40 (mf) OTS.

The measurements reported here were made in the BNL water tank shielding facility. The measurements were made with a brass-walled G-M counter, shielded on all sides by 2-1/2 in. of Pb. The experimental arrangement is illustrated. The tank was filled with water to a depth of 143 in. An outlet is provided at the top surface of the tank, so that the water level remains constant when a void is placed in the tank. The counter was mounted 1 in. above the water surface, and measurements were made at various points in the horizontal plane. A natural U source plate 39-5/8 in. x 39-5/8 in. x 5/8 in. is mounted below the tank to provide a fission source of gamma rays and neutrons. The top surface of the source plate is shielded with boral from thermal neutrons from the water tank itself. A plate of Bi 3-5/8-in. thick is mounted below the source plate to suppress gamma-ray leakage from the pile reflector. Even with the Bi shield in place, the fraction of the gamma-ray flux at the surface of the water which is attributed to the source is only about 5% of the total flux. Thus, the gamma-ray measurements are primarily characteristic of the pile gamma rays and those produced by neutron capture in the material surrounding the water tank facility.

97. J. C. LeDoux and A. B. Chilton, Attenuation of Gamma Radiation Through Two Legged Rectangular Ducts and Shelter Entranceways - An Analytical Approach, USNCEL TN 383 (January 1961). Availability: $5.60 OTS as PB 158290.

The use of the albedo approach for the analytical prediction of attenuation characteristics of two-legged rectangular ducts transmitting gamma rays is extended to cases involving the asymmetrical location of source and/or detector. Illustrative problems are presented and solved.

98. A. B. Chilton, Further Analysis of Gamma-Ray Attenuation in Two Legged Rectangular Ducts, USNCEL TN 412 (May 1961). Availability: $5.60 OTS.

An analytical approach is developed to permit determination of gamma radiation attenuation as it passes through two-legged rectangular ducts and shelter entranceways. The approach used employs the albedo concept for wall scattering and includes correction terms necessary to account for the "corner lip effect." With appropriate simplifying assumptions, moderately simple engineering formulas are obtained. Actual use of the formulas requires better knowledge of differential angular albedo than is presently available; however, by assuming isotropic distribution of the albedo function, a very good comparison of experimental information with results calculated by this technique is obtained.

99. M. G. Wing, An Introduction to Transport Theory, John Wiley and Sons, Inc., 1962.

The relatively new idea of invariant imbedding is introduced against a background of classical transport theory through the use of simple mathematical models.

100. REIC, Radiation Effects State of the Art 1961-1962, REIC Report No. 24 (June 30, 1962). (Radiation Effects Information Center) Availability: $8.60 (fs) OTS.

No abstract necessary.

101. H. E. Strickler, H. E. Gilbert, and J. A. Auxier, "Fast Neutron Scattering from Thick Slabs," Nucl. Sci. Eng., 3, 1, 11-18 (January 1958).

An experimental evaluation has been made of the Po-Be fast-neutron absorbed dose albedos for concrete, water, and aluminum. By assuming neutrons are scattered isotropically in the center of mass system, a good fit to Plesset's albedo theory was obtained. The values for the albedos were found to be 0.32 for aluminum and concrete, and 0.092 for water.

102. C. E. Lee, The Discrete Sn Approximation to Transport Theory, LA-2595 (March 9, 1962). Availability: Dep.; $3.00 OTS.

The generalized discrete S_n Method for solution of the multigroup, multidimensional transport equation is investigated. The transport and diffusion equations are derived in difference and differential form based upon a conservation law. The discrete direction representation of the angular integration variable is derived on the basis of a rotation-reflection symmetry condition. In addition, if the reasonable assumption that the mechanical quadrature point weights are each taken proportional to an area surrounding each point on the unit sphere, then the usual moment conditions are shown valid in an asymptotic sense. In particular, the P_n, $DP(n/2)-1$, and Wich-Chandrasekhar methods of solving the transport equation appear now as special cases. The full anisotropic scattering formulation, elementary perturbation theory, and group collapsing techniques are discussed with reference to the proposed equations. The general methods for solving the transport and diffusion difference equations are reviewed briefly. Finally, simple one-dimensional comparisons are made to indicate the treatment of the general method in comparison with other specialized approximation techniques.

103. A. F. Avery, D. E. Bendall, J. Butler, and K. T. Spinney, Methods of Calculation for Use in the Design of Shields for Power Reactors, AERE-R-3216 (1960). Availability: Dep.; $3.60 BIS.

Recommendations are given on methods of calculation for use in the design of shields for the gas cooled, graphite moderated, natural uranium reactors of the U.K. Civil Power Programme. Topics dealt with include: primary shielding of neutron and gamma radiation, nuclear heating, coolant activation, fission product energy decay rates and radiation streaming in ducts. A new method of calculating neutron penetration is introduced and compared with experimental measurements in a Calder Hall reactor shield.

104. W. D. Allen, Neutron Detection, George Newnes Limited, 1960.

This is a general reference for detection of neutron radiation.

105. J. K. Basson and J. A. Dennis, "Measurement of the Energy Response of a B-10 Shielded U-235 Fission Chamber," Nucl. Instr. Methods 16: 321-5 (1962).

A U^{235} fission chamber enclosed in absorbing shields of B^{10} is used as a threshold detector for intermediate energy neutrons.

The energy response has been measured by using time-of-flight techniques in conjunction with the Neutron Booster at Harwell.

The effective fission cross section of U^{235} when shielded by 1 gm cm^{-2} of B^{10} is found to be in good agreement with that calculated by the application of the sphere transmission theory of Bethe, Beyster, and Carter. The agreement for a shield of 2 gm cm^{-2} is reasonable.

106. A. F. Avery, The Prediction of Neutron Attenuation in Iron-Water Shields, AEEW-R 125 (April 1962). Availability: Dep.; $1.26 BIS.

The multigroup method which was previously used to predict neutron attenuation in water (SWP/P 60) is applied to iron-water systems. Calculations for several shield configurations are presented and the results are compared with the experimental measurements made in the Panel C facility of the LIDO reactor. The accuracy of predictions of reaction rates due to sub-cadmium neutrons is generally to within 50% for penetrations up to 160 cm from the core surface.

107. M. B. Wells, Monte Carlo Calculations of Fast-Neutron Scattering in Air (Two Volumes), NARF-60-8T (FZK-9-147) (May 13, 1960). Availability: $8.10 OTS (PB-156649).

The results of a Monte Carlo calculation of the scattering of fast neutrons in an infinite homogeneous medium of air are presented. The results of these calculations, which include the effect of inelastic scattering, are presented in terms of angular distributions, energy spectra, and tissue dose rates at source-detector separation distances of from 10 to 100 feet. A parameter study was completed for source energies of 0.33, 1.1, 2.7, 4, 6, 8, 10.9, and 14 MeV and source angles with respect to the source-detector axis from 5° to 180°.

This report, the first of two volumes, gives a description of the Monte Carlo problem and the mathematical procedures used in the calculations. The first volume also contains graphs giving the distribution of the scattered flux and dose rate as a function of source energy, source angle, and source-detector separation distance. It also contains tables giving the energy distribution of the air-scattered flux. Volume II contains tables presenting the angular distributions of the air-scattered neutron energy spectra for all source energies, source angles, and source-detector separation distances included in this parameter study.

108. C. A. Diffey, Monte Carlo Air-Scattering Data for Monoenergetic Fast Neutrons from Point Isotropic Sources, NARF-61-29T (MR-N-284) (August 15, 1961). (Convair, Fort Worth.) Not for sale to public.

The Monte Carlo fast-neutron air-scattering data presented in FZK-9-147, Volumes I and II, have been integrated to obtain the angular distributions and energy spectra for a point isotropic source emitting one neutron per second at a given energy E_0. These calculations were performed for source-detector separations of 10, 35, 64, and 100 feet and for initial neutron energies of 0.33, 1.1, 2.7, 4.0, 6.0, 8.0, 10.9, and 14.0 MeV.

109. M. B. Wells, Studies in Shielding-III, Monte Carlo Calculations of Neutron Scattering in Air, NARF-57-49T (FZK-9-120A) (December 2, 1957). (Convair, Fort Worth) Not for sale to public.

This report describes a Monte Carlo method of calculating the angular distribution and energy spectrum of air-scattered neutrons. The Monte Carlo method, which is outlined in detail, was used to solve the integrals that represent the contribution to the total-scattered flux from each order of scattering. Results of the Monte Carlo calculations giving the scattered flux from a 3.25-MeV point monodirectional source are presented in both graphical and tabular form for various source angles and separation distances at 10,000-ft density altitude.

110. M. B. Wells, Air and Concrete Scattering of Neutrons, Monte Carlo Calculations, NARF-58-15T (MR-N-197) (April 2, 1958). Availability: $4.60 OTS as PB-162860.

This report presents results from Monte Carlo calculations of the angular distribution of neutrons scattered from air and concrete. The Monte Carlo method was used to evaluate the integrals that represent the contribution to the total scattered flux from the first five orders of scattering. The scattered neutron flux at various separation distances was computed for a 3.25-MeV point monodirectional source located at a height of 12.5 feet above the surface of a concrete slab. Differential scattering cross sections based on a complex square well model and differential scattering cross sections assuming isotropic scattering in the center-of-mass system were used in the Monte Carlo calculations. The method used in computing the variance of the scattered flux is outlined. The standard deviation computed was found to be generally not more than 10% of the scattered flux.

This report supplements the information presented in Convair-Fort Worth report FZK-9-120A.

111. J. R. Barcus, Transmission of Neutrons by Cylindrical Ducts Penetrating Radiation Shields, SCTM 21-59 (16) (March 1959). Availability: Dep.; $1.60 (fs), $0.80 (mf) OTS.

After a brief introduction to elementary neutron transport theory, the problem of the transmission of neutrons by straight and bent cylindrical ducts penetrating a radiation shield is considered. The direct scattered component is treated to a first approximation using the diffusion approximation applied to the distribution of effective anisotropic sources along the duct wall. The geometric approximations are such that the method is applicable mainly to ducts with diameter-to-length ratios small compared to unity.

112. M. H. Kalos, Methods in Monte Carlo Solution of Radiation Transport Equation, NDL-TR-36 (UNC-5014) (Vol. A) (May 31, 1962). (Army Chemical Corps Nuclear Defense Lab, Md.) Not for sale to public.

A discussion is given of certain methods of importance sampling and scoring in the Monte Carlo solution of the radiation transport equation.

113. C. S. Cook, "Calculation of the Penetration of Gamma Rays Through the Edges of a 1/2 in. Diameter Lead Collimator, Nucl. Instr. and Methods 4: 103-6 (1959).

Gamma radiation from a point radioactive source located on the axis of a cylindrically-shaped collimator is not sharply defined by the geometrical aperture of the collimator. Instead some of the radiation penetrates the edges of the collimator. Calculations of the magnitude of this penetration have been made for a 1/2-inch diameter cylindrical aperture in a lead collimator, from point sources 8, 9, 10-1/2, 13, 16, 32, and 48 inches from the detector end of the collimator. The results of these calculations are reported here.

114. C. Eisenhauer, Analysis of Experiments on Light Residential Structure with Distributed Co-60 Sources, NBS-6539 (October 1959). (National Bureau of Standards.) Not for sale to public.

The shielding potentialities for gamma radiation were analyzed for light residential structures of the following type: two story wood frame house, one story wood rambler, two story brick house, and a precast concrete house. A phantom house, consisting of Al tubing fastened by clamps, was erected to measure dose rates with no intervening attenuating material. The dependence of shielding on source energy and on reversal of source and detector is discussed. The results of the investigation are compared with various calculational schemes for evaluating radiation protection in structures.

115. J. A. Auxier, J. O. Buchanan, C. Eisenhauer, and E. Meuker, Experimental Evaluation of the Radiation Protection Afforded by Residential Structures Against Distributed Sources, CEX 58.1 (January 1959). Availability: Dep.; $10.50 (fs), $4.19 (mf) OTS.

A study was made to obtain information that could be used to evaluate the protection afforded by residences against radiation due to fallout. The sources used were Co^{60} and Cs^{137}, and the radiation dosimeters used were pocket-type ionization chambers. Measurements were made for distributed sources (400 Co^{60} and 20 Cs^{137}) and for single sources located inside the structures (one each, 2-curie Co^{60} and Cs^{137}).

Attenuation measurements were made for five houses of typical domestic design and construction. Several modifications were made to the houses, and the attenuation measurements were repeated. The houses, located at the Nevada Test Site, included single- and two-story houses with and without basements and with light and heavy walls. For comparison with the house data the dose-rate distribution above an extended plane source was measured in a "phantom" house, i.e., air measurements with the instrument fixed on a framework of thin-walled aluminum tubing. Although the complete analysis of the data is not yet available, some typical analyses are presented, and the effectiveness of some of the modifications is illustrated.

116. S. Tomoeda, M. B. Hastings, and W. G. Miller, Gamma-Ray Penetration Experiments for a Light Aircraft Carrier Using Distant Sources and Sources Simulating Contamination of the Hull, USNRDL TR 533 (October 1961). (Naval Radiological Defense Lab, San Francisco.) Not for sale to public.

The penetration of gamma rays into compartments of an aircraft carrier from two different radioisotopes was measured in two different experiments.

In one experiment, cobalt-60 and cesium-137 point isotropic sources were exposed near the hull of the ship to simulate hull contamination. Measurements of dose were made in the forward part of the ship at four different deck levels. Dose distribution data are presented for the two sources used and for the different source points considered.

In the second experiment, a nominal 500-curie collimated cobalt-60 source was exposed at a distance of roughly 100 feet from designated points to irradiate two general regions of the ship. Dose distributions are presented for each of 18 different source positions used. Five decks were instrumented in each of the two general regions considered.

Diagrams indicating source positions, detector positions, and hull-plating thicknesses are also presented.

117. F. L. Keller, C. D. Zerby, and J. Hilgeman, Monte Carlo Calculations of Fluxes and Dose Rates Resulting from Neutrons Multiply Scattered in Air, ORNL-2375 (December 15, 1958). Availability: Dep.; $2.60 (fs), $0.95 (mf) OTS.

A Monte Carlo code previously developed for the Oracle was used for a parameter study to calculate the fluxes and dose rates resulting from neutrons multiply scattered in air. Both the source and the detector were assumed to be suspended in an infinite body of air (ρ = 0.00125 g/cm^3), and the source was assumed to be a point source which emitted a monoenergetic line-beam of neutrons at various angles with respect to the source-detector axis. Calculations were performed for separation distances, g, of 5, 10, 20, 40, 64, and 100 ft; for source energies, E_o, of 5, 3, 2, 1.2, and 0.55 MeV; and for beam angles, θ_o, of 2, 15, 30, 60, 90, 135, and 180 deg. The scattering was assumed to be isotropic in the center-of-mass system. The results of the flux calculations were found to fit the expression $[\Sigma_s(E_o)/4\pi g] \cdot [\pi - \theta_o/\sin\theta_o]$, where $\Sigma_s(E_o)$ is the total macroscopic scattering cross section of air at the initial energy of the neutron. The results of the dose-rate calculations were found to fit the expression $[1/1.2] \cdot [C(E_o)\ \Sigma_s(E_o)/4\pi g] \cdot [\pi - \theta_o/\sin\theta_o]$, where $C(E_o)$ is the flux-to-dose rate conversion factor.

118. M. B. Wells, Gamma-Rays from Inelastic Scattering of Neutrons in Air, NARF-60-22T (FZK-9-150) (August 12, 1960). (Convair, Fort Worth.) Not for sale to public.

This report describes a Monte Carlo calculation of the gamma rays from neutron inelastic scattering in air. The results of these calculations are given in the form of energy and angular distributions of the detected gamma-ray fluxes and angular distributions of the detected gamma-ray dose rates for point monodirectional neutron sources of 6, 8, 10.9, and 14 MeV at source-detector separation distances of 10, 35, 64, and 100 feet.

119. M. O. Burrell and D. L. Cribbs, A Monte Carlo Calculation of Gamma-Ray Penetrations Through Iron Slabs, NR-82, Vol. 2 (January 1960). Availability: Not for sale to public.

A Monte Carlo sampling is used to calculate the transmitted number and energy current, number and energy flux, and dose of gamma rays penetrating an iron slab. The results are presented for 6 thicknesses of iron, 5 source energies, and 5 incident angles. Scattered dose and energy flux transmission factors are determined for 120 equal solid angles. The energy flux transmission factors are divided into four energy groups. Also, a weighted average energy is determined for each solid angle. The total output represents some 25 hours of computing time on the IBM-704. Most of the data have been preserved on IBM cards.

120. M. O. Burrell and D. L. Cribbs, A Monte Carlo Calculation of Neutron Penetrations Through Iron Slabs, NR-82, Vol. 3 (May 1960). Availability: Not for sale to public.

A stochastic technique is used to calculate the transmitted and reflected number current, flux, and dose rates of monoenergetic, monodirectional beams of neutrons impinging upon iron slabs. The results are presented for five thicknesses of iron, five source energies, and five incident angles. Scattered dose transmission factors and dose reflection factors are determined for 120 equal solid angles. Also, a weighted average energy of the transmitted and reflected neutrons is determined for each solid angle. The total computer output presented in the appendix represents about 20 hours of computing time on the IBM-704. All of the dose data have been preserved on IBM cards.

121. F. J. Shore and R. D. Schamberger, The Transmission of Neutrons Through Ducts in Water, BNL-390 (T-74) (March 1, 1956). Availability: Dep.; $4.60 (fs), $1.73 (mf) OTS.

The Brookhaven Shielding Facility has been utilized to study the transmission of fission neutrons through air ducts in a water shield. The program included the study of the transmission properties of ducts of two different basic geometries, plane slots, and right circular cylinders. The following report represents an effort to systemize and interrelate the various results of the investigation.

122. G. deSaussure, Calculation of the Photoneutron Flux in Water Near the Bulk Shielding Reactor, ORNL-2545 (July 9, 1958). Availability: Dep.; $1.10 (fs), $0.80 (mf) OTS.

The measurement of the gamma-ray dose rate at a distance 267 cm from the Bulk Shielding Reactor has been used to estimate the photoneutron flux at that point. The resulting calculated flux was 1.65×10^{-5} neutron$\cdot w^{-1} \cdot sec^{-1} \cdot cm^{-2}$. This is to be compared with a measured value of the thermal-neutron flux of 1.58×10^{-5} neutron$\cdot w^{-1} \cdot sec^{-1} \cdot cm^{-2}$, all of which is assumed to consist almost entirely of photoneutrons.

123. D. K. Trubey, An Estimation of Photoneutrons from Carbon-13 in an Oil Shield, ORNL-2200 (August 13, 1958). Availability: Dep.; $1.60 (fs), $0.80 (mf) OTS.

Measurements of thermal-neutron flux in the presence of high-energy gamma rays in oil indicate that photoneutrons comprise a large fraction of the total thermal-neutron flux at large distances from the neutron source. This is attributed chiefly to the (γ,n) reaction in C^{13}. Measurements at the Lid Tank Shielding Facility in oil with and without a bismuth slab to reduce the gamma rays, and consequently the photoneutrons which they produced, agreed with calculated fluxes.

124. R. D. Albert and T. A. Welton, A Simplified Theory of Neutron Attenuation and Its Application to Reactor Shield Design, WAPD-15 (November 30, 1950). Availability: Dep.; $6.30 (ph), $3.00 (mf) OTS.

A theory has been developed for the attenuation of neutrons in mixtures of hydrogenous and heavy materials primarily for use in connection with shielding designs. The theory is a simplified one which assumes a neutron is essentially lost after a single collision; an equivalent cross section which includes contributions to scattering by heavy materials is left adjustable in the theory. The procedure outlined in this paper involves comparison of the theory with experimental bulk shielding results to determine the adjustable parameter. Excellent agreement -- for shielding purposes -- has been obtained in this way with experimentally obtained spatial flux distributions for water, iron and water, and two lead and water shields of differing composition. These results have been developed for design of the Mark I STR shield.

125. R. E. Beissner and D. R. Smith, Studies in Shielding II, Single Scattering and Capture Theory in Air and Ground, NARF 57-39T (FZK-9-117) (September 30, 1957). (Convair, Fort Worth.) Not for sale to public.

This report presents the derivation and evaluation of four of the more important equations used by Convair-Fort Worth in the prediction of gamma and neutron dose rates. The relations derived concern air and ground single scattering of neutrons and gamma rays and the radiative capture of thermal neutrons in the air and in the ground. A discussion of the equations and their importance in total dose rate prediction follows the derivations.

126. D. Spielberg, Penetration of Neutrons from a Point Fission Source in Air - Moments Method Calculation, WAL TR 551.1/2 (NDA 2106-10) (April 28, 1961). (Watertown Arsenal, Massachusetts.) Not for sale to public.

The spatial and energy distributions of neutrons from a point fission source in air were calculated by the moments method. The results were used to calculate the neutron and capture-gamma doses as functions of distance from the source.

The numerical results are given in Sections 5 and 6.

This work was performed under Technical Supervision of Watertown Arsenal Laboratories, Watertown 72, Massachusetts.

127. F. L. Keller, C. D. Zerby, and W. W. Dunn, Gamma-Ray Dose Rates Resulting from Neutron Captures in Air, ORNL-2462 (December 15, 1958). Availability: Dep.; $3.60 (fs), $1.19 (mf) OTS.

A Monte Carlo calculation of the gamma-ray fluxes and dose rates resulting from neutron captures in air was performed. For this calculation the source was assumed to be monoenergetic, and both the source and detector were assumed to be suspended in an infinite body of air. The neutron source was taken to be either a point isotropic or a line-beam source, and the dose rate and angular distribution of capture gamma rays at the detector were determined. For the point isotropic source and a 50-ft separation distance it was found that the dose rate varied with the neutron source energy approximately as $E^{-1/3}$. When curves for the various monoenergetic line-beam sources and a 65-ft separation distance were plotted as a function of the beam angle, they were relatively flat over the region from 30 to 180 deg, becoming flatter as the source energy increased.

128. M. J. Berger and D. J. Raso, "Monte Carlo Calculations of Gamma-Ray Backscattering," Radiation Res. 12, 20-37 (1960).

Monte Carlo calculations have been performed to determine the backscattering of gamma rays with energies between 0.02 MeV and 2.0 MeV from hydrogen, water, concrete, iron, tin, and lead. The radiation was assumed to be incident on a semi-infinite medium, either isotropically, or at angles of 0° (perpendicular incidence), 30°, 60°, or 90° (grazing incidence). Several thousand photon histories were sampled with the use of an IBM 704 computer to determine the albedo for each case. The information obtained includes (1) the number albedo and energy albedo, (2) the energy distribution of reflected radiation integrated over all directions, (3) the angular distribution integrated over all energies, (4) the angular distribution of the reflected energy, (5) the average energy of the photons reflected in various directions, and (6) the average obliquity of emergence of the reflected radiation as a function of the gamma-ray energy. Tables of albedo and illustrative energy spectra and angular distributions are given in this paper. More detailed results can be found in an accompanying informal publication (NBS Report 5982).

129. M. J. Berger and D. J. Raso, Backscattering of Gamma Rays, NBS Report 5982 (2nd ed.) (February 8, 1960). (National Bureau of Standards.) Not for sale to public.

Monte Carlo calculations have been performed to determine the backscattering of gamma rays with energies between 0.02 MeV and 2.0 MeV from hydrogen, water, concrete, iron, tin, and lead. The radiation was assumed to be incident on a semi-infinite medium, either isotropically, or at angles of 0° (perpendicular incidence), 30°, 60°, or 90° (grazing incidence). Several thousand photon histories were sampled with the use of an IBM 704 computer to determine the albedo for each case. The information obtained includes (1) the number albedo and energy albedo, (2) the energy distribution of reflected radiation integrated over all directions, (3) the angular distribution integrated over all energies, (4) the angular distribution of the reflected energy, (5) the average energy of the photons reflected in various directions, and (6) the average obliquity of emergence of the reflected radiation as a function of the gamma-ray energy. Tables of albedo and illustrative energy spectra and angular distributions are given in this paper. The description of the calculational method is found in Radiation Res. 12, 20-37 (1960).

130. H. Kahn, Applications of Monte Carlo, AECU-3259 (April 19, 1954). Availability: Dep.; $37.80 (ph), $11.10 (mf) OTS.

The general principles of making Monte Carlo calculations are outlined. Discussions are included on technique with random variables, transformations of random variables, rejection techniques for producing independent sample values of a distribution F(x), and manipulations of distributions. The various methods which are described are illustrated by numerous examples. Detailed analyses are also given of Monte Carlo methods of evaluating integrals and integral equations.

131. M. H. Kalos, H. Goldstein, and J. H. Ray, Revised Cross Sections for Neutron Interactions with Oxygen and Deuterium, NDL-TR-40 (UNC Project 5038) (August 31, 1962). Availability: $2.60 (fs) OTS as AD-284996.

A revision of the data for angular distributions for elastic neutron scattering in oxygen and deuterium is presented. The kinetics of the (n,2n) reaction in deuterium is discussed and a Monte Carlo scheme for generating the energy and direction of secondary neutrons is given. Complete sets of the cross sections for both materials are tabulated.

132. J. Griffin, Beta Decays and Delayed Gammas from Fission Fragments, LA-2811 (February 18, 1963). Availability: Dep.; $1.25 OTS.

A theoretical study is made of the gross behavior of beta decays following nuclear fission in times from 10^{-2} to 10 seconds. First a simple model is formulated to describe the situation in terms of a few parameters. Then the most uncertain of these parameters are chosen to fit the observed rate of delayed gamma emission (assumed proportional to the beta-decay rate) for the U^{235} (n,f) process. The description is extended to other isotopes by assuming that they differ only by small shifts of their initial fragment distributions away from that appropriate to neutron-induced fission of U^{235}. The result is a theoretical summary of the presently available data which can be used in predictive extrapolation of that data to situations not yet studied experimentally.

133. J. T. Madell, T. J. Thompson, A. E. Profio, and I. Kaplan, Spatial Distribution of the Neutron Flux on the Surface of a Graphite-Lined Cavity, NYO-9657 (April 1962). Availability: Dep.; $4.00 OTS.

A graphite-lined cavity was built at the outer end of the thermal column of the MIT Reactor. Seven variations of the material and geometrical arrangement of the cavity were made. The neutron flux was measured on the top surface and in the top wall of each variation and also on all surfaces and in the cavity itself of one of the variations. Absolute flux, cadmium ratio, and albedo measurements were made. The experimental results demonstrated that the cavity suitably modified the magnitude, distribution, and direction of the neutrons emerging from the thermal column.

A theoretical model was developed to calculate the magnitude and distribution of the neutron flux on the surfaces of a cavity. The values of the view factors and the albedos, which were necessary to calculate the flux in the cavities, were obtained from computer programs. The values of the flux were also calculated by using an electronic computer.

The calculated values of the flux are in excellent agreement with the measured values and it is felt that the use of the theoretical model is justified.

134. J. S. Church and J. K. Donahoe, Flexible Neutron Shielding Systems Based on Vinyl Plastisols, LAMS-2800 (February 28, 1963). Availability: Dep.; $0.75 OTS.

A method has been developed for fabrication of cast, flexible neutron shielding. The process is based on a filled polyvinyl chloride (PVC) plastisol system which can be fabricated by either slush casting or low pressure compression molding. Filler concentrations of 50 w/o boron or boron carbide are readily attained with retention of a fair degree of flexibility.

135. Z. G. Burson, Experimental Evaluation of the Fallout Radiation Protection Provided by Selected Structures in the Los Angeles Area, CEX-61.4 (February 26, 1963). Availability: Dep.; $2.25 OTS.

An experimental study designed to provide a basis for estimating protection against fallout radiation was conducted on four diversified structures in the Los Angeles, California area. This study was sponsored by the Civil Effects Test Operations (CETO), Division of Biology and Medicine, U.S. Atomic Energy Commission. The four buildings studied were (1) the Laboratory of Nuclear Medicine and Radiation Biology at the University of California at Los Angeles (UCLA); (2) a family fallout shelter; (3) the communications section of the Los Angeles Police Department building; and (4) a typical classroom located at North Hollywood High School.

A fallout radiation field was simulated by the Mobile Radiological Measuring Unit. The unit employed a single radioactive Co^{60} source, which was pumped at a uniform speed through a long length of tubing evenly distributed over the area of interest. Measurements of the radiation levels at selected points inside the structures were made with highly sensitive ionization-chamber detectors. Protection factors ranged from 10 to 2000 in the UCLA building, up to 10,000 in the family fallout shelter, from 50 to 150 in the communication section of the police building, and from less than 10 to approximately 20 in the high school classroom.

136. R. K. Osborn, "A Discussion of Theoretical Analyses of Probe-Induced Thermal-Flux Perturbations," Nucl. Sci. Eng. 15, 3, 245-258 (1963).

This review deals exclusively with the theoretical task of interpreting the measurement of a thermal-neutron absorption rate in terms of the neutron density that existed at the probe location prior to probe insertion. Successful accomplishment of this task is fairly obviously of considerable practical importance, since activation measurements of thermal fluxes are an important adjunct to many reactor experiments and are also frequently employed to obtain thermal flux maps in reactor cores and reflectors. Because of its importance, the problem has received extensive and varied (but usually piecemeal and semi-intuitive) theoretical attention over the past twenty years. Many experiments have been performed concurrently. The net result has been to perpetuate a lively interest in the problem, since comparisons of experiment with experiment, experiment with theory, and theory with theory have been ragged and controversial. It will be the purpose of this review to present an adequately precise and general statement of the problem and then attempt deductive comparisons of various theoretical attacks upon it. Attention will be focused primarily upon the task of obtaining operational descriptions of the diverse approximations explicit or implicit in the calculations considered.

Throughout the review attention is presistently directed to an aspect of the problem that has received less than adequate attention so far, i.e., that of the effect of "flux-hardening" on the absorption rate. An attempt to obtain a qualitative estimate of the importance of this effect is presented, but the only conclusion drawn so far is that the effect is not obviously an ignorable one.

137. W. D. Lanning, "Application of the Spherical Harmonics Technique to Problems in Gamma Transport," Nucl. Sci. Eng. 15, 3, 259-267 (1963).

The feasibility of applying the spherical harmonics technique to problems in gamma transport is investigated. The spherical harmonics procedure is described for one-dimensional slab geometry for plane monoenergetic sources. Energy spectra obtained by the P-3 approximation covering a wide range of source energies and materials are compared to moments results for penetrations up to fifteen mean free paths. The comparisons indicate that it is plausible to extend the method to multislab geometry for potential design applications.

138. J. V. Walker, J. D. Randall, and R. C. Stinson, Jr., "Thermal Neutron Flux Perturbation Due to Indium Foils in Water," Nucl. Sci. Eng. 15, 3, 309-313 (March 1963).

Flux perburbations due to indium disc foils exposed to an isotropic thermal-neutron flux in a water medium were experimentally determined. The unperturbed flux, or the "zero thickness" activity, was determined using foils which matched the moderating ratio of the medium, thereby causing no flux perturbation. Comparison of the results with several modifications of the Ritchie and Eldridge theory gives excellent agreement for foil thickness from 0 to 0.025 in.

139. G. C. Hanna, "The Neutron Flux Perturbation Due to an Absorbing Foil - A Comparison of Theories and Experiments," Nucl. Sci. Eng. 15, 3, 325-337 (March 1963).

Skyrme has given a theoretical treatment of the perturbation of the neutron flux in a diffusing medium by an absorbing foil. His theory is re-examined, with particular reference to the modification proposed by Ritchie and Eldridge, and the "edge correction" is evaluated. The accuracy of this modified Skyrme theory is tested by comparison with Dalton and Osborn's computer calculations for monokinetic neutrons; the agreement is generally better than 1%.

This theory is then extended to a Maxwellian neutron spectrum, for which computer calculations are not available, with the result

$$\frac{A}{A_0} = \frac{\bar{\alpha}}{2\bar{\tau}} \cdot \frac{1 + \bar{\epsilon}}{1 + g\bar{\alpha}/2}$$

A is the activity per unit mass of a foil of thickness τ (in units of the absorption mean free path), A_0 that of a zero thickness foil $\alpha = 1 - 2f_1^{\infty} x^{-3} \exp(-\tau x)\, dx$, and $\bar{\epsilon} \approx t/\pi R$, where t and R are the thickness and radius of the foil. The bars denote averages over the Maxwellian spectrum. The flux-depression parameter g is of the order of R/λ_{tr}, but its exact value depends on the velocity dependence of the transport mean free path, and on the thermalization properties of the medium.

This formula is used to obtain, from the available measurements of the dependence of A on foil thickness, "experimental" values of g. For both graphite and hydrogenous media they are smaller than expected.

140. H. A. Steinberg, "Generalized Quota Sampling," Nucl. Sci. Eng. 15, 2 142-145 (February 1963).

A generalization of quota sampling is described and a quantitative evaluation of its validity is made.

141. T. Hyodo, "Backscattering of Gamma Rays," Nucl. Sci. Eng. 12, 2, 178-184 (February 1962).

Spectra of backscattered radiation from semi-infinite slabs of paraffin, aluminum, iron, tin, and lead were measured by means of a scintillation spectrometer, as a function of the measuring angle θ, which is the angle between the normal to the plane face of the slabs and the detector axis. The point gamma sources of Co^{60} and Cs^{137} were placed in close contact with the scattering slabs. By using this geometry the measurement was performed without perturbing effects produced by detector or source collimating shields. The energy and number albedos, the angular distributions of scattered energy and number of photons, and the energy distributions are given for each combination of gamma source and scatterer.

142. D. K. Trubey, "The Single-Scattering Approximation to the Solution of the Gamma-Ray Air-Scattering Problem," Nucl. Sci. Eng. 10, 2 102-116 (June 1961).

The results of a calculation using the single-scattering approximation, with no exponential attenuation or buildup, has been compared to results of a Monte Carlo calculation of the dose rate and number flux in air from a monoenergetic gamma-ray source. The comparison shows that the simple approximation is generally adequate for unshielded detectors.

143. W. B. Doub, "Particle Self-Shielding in Plates Loaded with Spherical Poison Particles," Nucl. Sci. Eng. 10, 4, 299-307 (August 1961).

An approximate heuristic expression for the particle self-shielding factor for a set of purely absorbing spheres of radius r and volume fraction V well mixed with another set of non-absorbing spheres has been derived.

The resulting expression has been experimentally verified using transmission data at several incident neutron energies for a plate-type sample containing a mixture of aluminum and boron-carbide spheres with nominal diameters $85 \pm 15\mu$. The boron-carbide spheres occupied about 37% of the sample volume. The transmission was measured at energies ranging from 0.03 to 1.2 eV using a crystal neutron spectrometer.

Since, however, the sample contained boron-carbide spheres with a distribution of diameters the experimental self-shielding factors are "average" values. It is shown, using an approximate model, that a plausible theoretical self-shielding factor is a volume weighted average of the self-shielding factors for the spheres of diameters, d_1, d_2, d_3.

The particle self-shielding factors derived by several other authors have also been compared with the present experimental results. The Hurwitz-Zweifel expression gives quite bad agreement, though this is expected because of the high volume fraction of poison in the sample. The Burrus expression gives much better agreement though not as good as the expression derived in this paper.

144. R. Kladnik and I. Kuscer, "Milnes Problem for Thermal Neutrons in a Nonabsorbing Medium," Nucl. Sci. Eng. 11, 2, 116-120 (October 1961).

The velocity dependent transport equation for a non-absorbing semi-infinite medium with no sources is approximately solved by a variational method. A simple trial function, which takes into account the asymptotic behavior of the exact solution, is used to obtain an approximation for the extrapolation distance q_∞, and, by iteration, an approximation for the flux distribution.

Numerical results are given for a monatomic gaseous medium with the atom mass equal to the neutron mass. The value $q_\infty = 0.9345\ l\ (\infty)$ is obtained. The velocity distribution of the merging neutron shows a hardening effect, corresponding, in the average, to a 14% increase in the neutron temperature.

145. J. R. Mika, "Neutron Transport with Anisotropic Scattering," Nucl. Sci. Eng. 11, 4, 415-427 (December 1961).

The general procedure of solving the one-velocity Boltzmann equation in the case of plane geometry is presented. It is assumed that the scattering function can be expanded into the finite series of Legendre polynomials.

The complete set of eigenfunctions of the Boltzmann equation is found. The orthogonality and completeness of the eigenfunctions are proved.

By way of illustration, solutions of some basic problems of neutron diffusion are given.

146. J. C. LeDoux and A. B. Chilton, "Gamma-Ray Streaming Through Two-Legged Rectangular Ducts," Nucl. Sci. Eng. 11, 4, 362-368 (December 1961).

An analytical approach is developed to permit determination of gamma radiation attenuation as it passes through two-legged rectangular ducts and shelter entranceways. The approach used employs the albedo concept for wall scattering and includes correction terms necessary to account for the "corner lip effects." With appropriate simplifying assumptions, moderately simple engineering formulas are obtained. Actual use of the formulas requires better knowledge of differential angular albedo than is presently available; however, by assuming isotropic distribution of the albedo function, a very good comparison of experimental information with results calculated by this technique is obtained.

147. S. K. Penny and C. D. Zerby, "Examination of the Range of Applicability of Conditional Monte Carlo to Deep Penetration Problems," Nucl. Sci. Eng. 10, 1, 75-82 (May 1961).

The conditional Monte Carlo method of sampling has been applied to the spatial part of the gamma-ray transport problem in an infinite medium for the purpose of evaluating its general usefulness and its applicability to deep penetration problems. A simplified derivation of the application is presented, and the results of calculations for a water medium and a lead medium are shown. The calculations indicate that the conditional Monte Carlo method, as used in this application and without the aid of other special techniques, gives reasonably good results in a physical deep penetration problem out to approximately 10 mean free paths penetration distance independent of the absorbing properties of the material and can be carried out to 20 mean free paths if some inaccuracy can be tolerated.

148. D. F. Rhodes, R. A. Stallwood, and W. E. Mott, "Intensity of Unscattered Gamma Rays Inside Cylindrical Self-Absorbing Sources," Nucl. Sci. Eng. 9, 1, 41-46 (January 1961).

A general expression for the intensity of unscattered gamma rays at points on the axis of a cylindrical absorber, surrounded by a ring-shaped self-absorbing source of rectangular cross section, is derived in terms of two integrals that have been evaluated on the IBM 704 computer. The results are presented in the form of curves from which the intensity can be determined for a wide variety of cases. For points not on the axis of symmetry a solution is given for the special case of source with large peripheral dimensions.

149. G. R. Dalton and R. K. Osborn, "Flux Perturbations by Thermal Neutrons," Nucl. Sci. Eng. 9, 2, 198-210 (February 1961).

The transport equation which describes the thermal-neutron population in and around a neutron detector is converted to an iterative integral equation. This integral equation is then solved for a wide range of specific cases using a digital computer.

Using this method of calculation, the effects upon neutron density of nonisotropic scatter in the surrounding medium, of finite detector dimensions and of scatter by the detector are calculated to an accuracy of better than 1%. Detailed maps of the scalar neutron density in and around finite detectors are available from the calculations. The problem of nonisotropic, nonuniform initial neutron density is formulated using the integral method.

150. D. W. Drawbaugh, "On the Solution of Transport Problems by Conditional Monte Carlo," Nucl. Sci. Eng. 9, 2, 185-197 (February 1961).

The application of a conditional Monte Carlo method for estimating the solution of transport problems is described. Statistical estimators for solving transport problems by the conditional Monte Carlo method are given for both homogeneous and heterogeneous problems. The analog and conditional Monte Carlo Solutions of a simple one-speed transport equation are compared with the exact deterministic solution. The application to deep penetration shielding problems is discussed.

151. R. H. Ritchie and H. B. Eldridge, "Thermal-Neutron Flux Depression by Absorbing Foils," Nucl. Sci. Eng. 8, 4, 300-311 (October 1960).

The perturbation of a thermal-neutron flux field by an absorbing foil is considered for the case of a foil of thickness t and of lateral dimensions $\gg$ L, where L is the diffusion length of thermal neutrons in the medium. The integral equation for "one-velocity" transport of neutrons in the medium containing the foil is solved by a variational method in which the "eigenvalue" is closely related to the foil activation. The results are compared with the predictions of the Bothe and Skyrme theories. The Bothe and Skyrme theories are compared for the case of the finite disk-shaped foil and are shown to differ primarily in the transport correction. This difference may be important in cases where L is not very large compared with the mean free path of neutrons in the medium. On the basis of these considerations, a new analytic approximation for the activation of a finite foil is proposed.

152. W. R. Conkie, "Velocity Dependent Neutron Transport Theory," Nucl. Sci. Eng., 7, 4, 295-303 (April 1960).

A method of extending the spherical harmonics method to calculation of velocity dependent neutron transport processes is described. Application of the method is made to a variation of the Milne problem with the model of an ideal gas of unit mass for the scattering medium. The results indicate that the method can give accurate analytical expressions for the neutron density for such velocity dependent problems.

153. W. Kofink, "Complete Spherical Harmonics Solution of the Boltzmann Equation for Neutron Transport in Homogeneous Media with Cylindrical Geometry," Nucl. Sci. Eng. 6, 6, 475-486 (December 1959).

The aim of this paper is to show that the treatment of the transport equation in cylindrical geometry does not involve essentially more tedious calculations than the treatment in plane geometry. A complete solution is given for homogeneous media including the complementary solutions. Every partial solution contains in its expansion of spherical harmonics some functions of a parameter with appropriate coefficients. It will be shown that these functions are Legendre polynomials and Legendre functions of the second kind as in the case of plane geometry for the "main" solution, and derivatives of these functions for the "complementary" solutions. They are solutions of the recursion relations for the expansion and yield a further recursion relation for the coefficients. Tables of these coefficients are given up to the eleventh spherical harmonic approximation and a general formula is derived for them. Two examples are worked out, a first based upon the supposition of a linearly anisotropic scattering law, and a second in which two higher terms of anisotropy are added to this law.

154. C. C. Horton, "The Shielding of Helical Ducts," Nucl. Sci. Eng. 6, 6, 525-529 (December 1959).

The problem of radiation penetration through a shield containing a helical duct is considered by representing the duct by successive straight ducts joined at a fixed angle. Expressions for the emergent radiation flux are derived and illustrated for a typical case. It is shown that, in a representative example, the contribution from radiation scattered along the duct is a small fraction of the general increase in radiation level due to the reduced effective density of the bulk shield in the vicinity of the helix.

155. H. E. Hungerford, R. F. Mantey, and L. P. Van Maele, "New Shielding Materials for High-Temperature Application," Nucl. Sci. Eng. 6, 5, 396-408 (November 1959).

Investigation and development of several new materials for high-temperature shields have yielded three reasonably cheap materials which are structurally stable and able to withstand high temperatures and high radiation fields. Calculations indicate good neutron attenuation properties. These materials have undergone extensive development and testing for both physical and radiation effect data. They are (1) serpentine rock, (2) calcium borate, and (3) borated diatomaceous earth.

Serpentine rock ($3\ MGO \cdot SiO_2 \cdot 2H_2O$), as asbestos mineral, retains its water of hydration to temperatures as high as 950°F. It can be used either dry-packed, or as the aggregate in concrete, with densities attainable of about 130 lb/cu ft. Structurally, the aggregate is not quite as good as concrete. Calcium borate is the commercial name applied to a number of borated calcium minerals pressed into an asbestos matrix to give a boron content of about 12 w/o, with a density of over 70 lb/cu ft. Although the composite is brittle, it can be fabricated into shapes rather easily. Tests indicate it will withstand temperatures up to 1800°F with less than 3% shrinkage, and can be exposed to a neutron irradiation of 2.4×10^{20} nvt without damage. Diatomaceous earth, a porous commercial refractory material, has been successfully borated to the extent of about 2 w/o boron. It can be used as an aggregate in portland or luminite concrete to give good strength properties and densities of 78-82 lb/cu ft.

156. W. R. Conkie, "Polynomial Approximations in Neutron Transport Theory," Nucl. Sci. Eng. 6, 4, 260-266 (October 1959).

The Milne problem, with absorption, of neutron transport theory is solved by expanding the angular distribution in Tchebycheff polynomials, rather than the more conventional Legendre polynomials. It is shown that the Tchebycheff approximation of order N, the T_N approximation, gives results for the extrapolated end point which are closer to the exact results over most of the range of absorption values considered than the corresponding P_N approximation.

157. W. R. Conkie, "An Iterative Method in Neutron Transport Theory," Nucl. Sci. Eng. 6, 4, 267-271 (October 1959).

An iterative method for solving problems in neutron transport theory is described. The method uses the result of a low-order spherical harmonics calculation as the initial function and attempts to improve this with one iteration. The results for the one-velocity Boltzmann equation with plane symmetry show that good accuracy is obtained for many practical cases.

158. D. G. Chappel, "Permissible Voids in Photon Shields," Nucl. Sci. Eng. 6, 2, 140-146 (August 1959).

Permissible void volume in photon shields is calculated in terms of shield thickness, photon energy, and permissible leakage factor. Several graphical aids for lead are presented. A simple method of void evaluation is included which may form the basis for an engineering acceptance test.

159. L. A. Beach, P. Shapiro, R. C. ORourke et al., "Solutions of the Constant Velocity Transport Problem," Nucl. Sci. Eng. 6, 1, 66-75 (July 1959).

By Fourier transform techniques the exact solutions of the one-velocity neutron diffusion problem for a uniform infinite medium with isotropic scattering have been derived for plane isotropic, plane parallel monodirectional, and plane parallel bidirectional source terms. These exact solutions in terms of Fourier inversion integrals were numerically evaluated upon the NAREC to give the angular distribution of the scattered intensity, the total scattered intensity, and the total intensity. By solving the integrals by contour integration in the complex plane, asymptotic solutions were obtained which are good approximate solutions for deep penetrations and problems with little absorption.

160. H. S. Wilf, "The Transmission of Neutrons in Multilayered Slab Geometry," Nucl. Sci. Eng. 5, 6, 306-319 (June 1959).

The use of the method of discrete ordinates as a numerical tool in reactor calculations is described. A scheme for the numerical integration of the discrete ordinate equations is developed, and the results compared with known exact solutions. The problem of reflection and transmission of neutron beams in stratified slab geometry is considered from the point of view of the method of discrete ordinates. A matrix formalism is derived which permits the calculation of transmitted and reflected distributions if the incident beam is given. Asymptotic expressions for the relevant matrices are given, and comparison is made with the work of Chandrasekhar. Tables of the matrix elements in the P_1 approximation are presented.

161. B. L. Byrum and J. A. Biggerstaff, "Nuclear Radiation Heating - Preliminary Design Considerations," Nucl. Sci. Eng. 5, 1, 28-31 (January 1959).

Quick engineering estimates of nuclear radiation heating effects exterior to reactors are frequently desired in preliminary flight vehicle design. A technique that is useful for such evaluations is presented. Analogies are suggested among the capture, inelastic scattering and neutron-induced charged particle processes; a means is provided for treating neutron interactions giving rise to gamma rays of various discrete energies; and a cosine distribution of the core gamma radiation is discussed.

162. G. Goertzel, "The Method of Discrete Ordinates," Nucl. Sci. Eng. 4, 4, 581-587 (October 1958).

In this paper the solution of the neutron transport (Boltzmann) equation in plane and spherical geometries is considered. It is demonstrated that the method of discrete ordinates differs from the spherical harmonic method merely by a linear transformation (with constant coefficients) of the dependent variables.

163. S. Ziering and D. Schiff, "Yvons Method for Slabs," Nucl. Sci. Eng. 3, 6, 635-647 (June 1958).

The method of half-range polynomials is applied to neutron transport theory. The specific applicability of this method to problems having discontinuities in the nuclear parameters at the boundaries or interfaces is discussed. Half-range polynomial expansions are used to obtain solutions for both finite and semi-infinite slabs, which consist of isotropically scattering media. The results indicate that the half-range approximations compare favorably with higher approximations obtained from the full-range spherical harmonic or several discrete ordinate methods. In particular, the poor convergence, found in the full-range methods in regions close to the discontinuity, is not present in the half-range method. The latter method is used to obtain a pair of second-order coupled differential equations, as in diffusion theory.

164. J. F. Perkins and R. W. King, "Energy Release from the Decay of Fission Products," Nucl. Sci. Eng. 3, 6, 726-746 (June 1958).

The total disintegration rates, rates of beta- and gamma-energy release, and gamma-ray energy spectrum, are calculated for fission products due to thermal-neutron fission of U^{235}. Information on decay schemes was largely obtained from the compilations of the Nuclear Data Group of the National Research Council to July, 1957. Total fission yields are from Katcoff and from Steinberg and Glendenin. Nuclear charge distributions are taken from Pappas' work, which includes the effect of closed shells. Reactor operating times of 1, 10, 100, and 1000 hours are treated, and the results plotted for decay times ranging from 10^2 to 10^8 seconds. In addition, results for instantaneous operation are compared to other calculations and measurements. The present results fall below Way and Wigner's predictions of both disintegration rate and total energy release over the entire range of decay times, though they agree satisfactorily with the Way-Wigner rule-of-thumb expressions. The present results are in very good agreement with experimental measurements. The gamma spectrum is found to vary considerably with decay time but to be only a weak function of reactor operating time. The total beta and antineutrinos energies per fission are found to be 7.6 ± 0.5 and 10.0 ± 0.7, respectively.

165. F. Titus, "Measurement of the Gamma-Ray Dose Near the Interface Between Two Media," Nucl. Sci. Eng. 3, 5, 609-619 (May 1958).

Measurements have been made of the distribution of exposure dose in the neighborhood of a plane boundary separating two media of very different density, steel-wool and steel. A point-isotropic Co^{60} source was located in the vicinity of the boundary. Control measurements were performed in a homogeneous medium of steel-wool. In this way the effect on gamma-ray propagation of an abrupt density change was established. The main result was a progressive decrease of exposure dose near the density-interface compared with the homogeneous medium situation, as the source-detector distance was increased. There is good agreement between the experimental results and corresponding Monte Carlo calculations.

166. R. S. Caswell, R. F. Gabbard, D. W. Padgett et al., "Attenuation of 14.1-MeV Neutrons in Water," Nucl. Sci. Eng. 2, 2, 143-159 (April 1957).

The attenuation of 14.1-MeV neutrons in water has been studied under conditions simulating an isotropic point source of monoenergetic neutrons in an infinite water medium. The migration area of thermal neutrons $[(r^2)th/6]$ was found to be 156 ± 6 cm^2. The "age" of indium resonance neutrons was measured to be 150 ± 6 cm^2. Fast-neutron dose measurements are in agreement with the theoretical calculations of Goldstein et al.

167. E. Fisher, "The Streaming of Neutrons in Shields," Nucl. Sci. Eng. 1 3, 222-238 (July 1956).

The streaming of neutrons in narrow channels through thick shields is treated theoretically, using one-velocity diffusion theory. Scattering by the channel walls is evaluated, and it is shown how the entrance and exit of a channel affects its streaming current. The cases examined are the straight duct, gap, and annulus, offsets for the gap and annulus, and bends for the duct and gap. In all cases simple, approximate formulas are obtained.

168. A. Simon and C. E. Clifford (USA), "The Attenuation of Neutrons by Air Ducts in Shields," Nucl. Sci. Eng. 1, 2, 156-166 (May 1956).

The attenuation of fast neutrons by long, thin air ducts in shields is considered from a simple phenomenological point of view. It is shown that the wall-scattered flux in a straight duct is small compared to the uncollided neutrons which travel directly in air from one end of the duct to the other. An expression for the attenuation due to a bend is derived and this is generalized to the case of several bends.

169. C. D. Zerby, Radiation Flux Transformation as a Function of Density of an Infinite Medium with Anisotropic Point Sources, ORNL-2100 (October 9, 1956). Availability: Dep.; $3.30 (ph), $2.40 (mf) OTS.

The derivation is presented of a general equation used to transform the radiation current, flux, or dose rate from an anisotropic point source in an infinite homogeneous medium at a given density to the values at any density of the medium. These equations are then used to describe the method of transforming the Tower Shielding Facility data to apply at any altitude.

170. E. P. Blizard, On the Disk to Point or Infinite Plane Shielding Transformations, ORNL-2882 (December 22, 1959). Availability: Dep.; $1.60 (fs), $0.80 (mf) OTS.

The detector readings to be expected from an infinite plane isotropic source in a uniform infinite medium is derived in terms of measurements at specific distances along the axis of a uniform isotropic disk source. Extrapolation of the higher terms in this series on the basis of exponential attenuation is described and approximations are given for the remainder of this series.

Another transformation is given which yields directly the point-to-point attenuation kernel in terms of a series of measurements on the axis of a plane isotropic disk source. Approximations based on the extrapolation of the data are also given to enable estimation of the remainder of the series.

171. V. Mangulis, Transformations of Solutions of Diffusion Equations in Two-Medium Geometries, WADC Technical Report 58-112, Astia Doc. No. 155771 (August 1958). Availability: $0.75 OTS as PB-151313.

In some cases the effectiveness of a shield for a certain geometry is predicted by comparison with a different geometry for which the necessary data have been measured; therefore transformations are necessary from one situation to another, as, for example, the well-known Hurwitz transformation, which gives the dose to be expected with an infinite plane source in an infinite homogeneous medium in terms of measurements on the axis of a finite disk source in the same medium. Since in realistic cases there are at least two media present, transformations were investigated for two semi-infinite slab media with a point,disk, infinite plane, or solid cylinder isotropic source inside one of the media and for a sphere of one material contained in an infinite medium of the second material, with a spherical shell or solid sphere isotropic source inside the sphere medium. The fluxes in all media were assumed to satisfy the one-velocity diffusion equation with constant coefficients.

172. A. V. H. Masket, R. L. Macklin, and H. W. Schmitt, Tables of Solid Angles and Activations. I. Solid Angle Subtended by a Circular Disc. II. Solid Angle Subtended by a Cylinder. III. Activation of a Cylinder by a Point Source, ORNL-2170 (November 1956). Availability: Dep.; $7.60 (fs), $2.51 (mf) OTS.

The solid angles subtended by discs and cylinders have been calculated for a large range of disc and cylinder dimensions and spacings and are presented in tabular form. The interaction of radiation from a point source on right cylindrical bodies is calculated for a large range of cylinder dimensions and spacing.

173. C. D. Zerby, Radiation Fluxes in a Cylindrical Cavity Resulting from an Idealized Entering Current, ORNL-2177 (February 12, 1957). Availability: Dep.; $6.30 (ph), $3.00 (mf) OTS.

The equations for the contribution to the radiation flux in a cylindrical cavity coming from the current entering through the curved surface are developed for two cases. In one case the cylinder is of finite length and the flux is obtained along the centerline. In the other case the cylinder extends to infinity in one direction from the plane in which the flux is determined along the radial from the center. In each case the entering current is idealized to have an angular distribution with an axis of symmetry which is not necessarily along the normal to the surface.

174. R. D. Sheffield, Shield System Optimization - Gradient Nonlinear Programming, NARF-57-62T (MR-N-186) (December 30, 1957). Availability: LC $3.00 (mi), $6.30 (ph) as PB-139349.

A procedure is developed for minimizing a nonlinear function of n variables subject to inequality constraints. Under suitable assumptions concerning convexity of the functions, a unique minimum is attainable by the method. The technique is applied to optimization of the total weight of a divided shield system subject to a prescribed dose rate level permitted in the crew compartment and to reactor leakage constraints. The procedure consists of two phases: a dose rate reduction phase followed by an isodose-rate weight reduction phase. Provision is made for satisfying the reactor leakage constraints which are expressed as linear inequalities. The process is iterative and provides a convergence criterion depending upon equality of certain weight-dose coefficient ratios.

175. W. H. Yanko, L. Yoder, M. E. Ginn et al., Summary Report for Development of a Lightweight Plastic Neutron Shielding Material Based Upon Boron Compounds with High Hydrogen Content, MRC-62-4501-7A (April 3, 1962). (Materials Research Corp., Yonkers, N.Y.) Not for sale to public.

All known boron compounds, composed of elements of atomic numbers 1 through 20, were evaluated for neutron shielding properties vs high density polyethylene, containing 6% amorphous boron, by comparison of calculated neutron removal cross sections and hydrogen densities.

All boron compounds, derived from a comprehensive literature survey up to March 1961, were tabulated along with their pertinent properties: density, hydrogen density, microscopic and macroscopic neutron removal cross sections, thermal-neutron capture gamma-ray energies, melting point, stability and toxicity. Classified compounds are reported separately in Part B, this report. Known densities were available for only 12% of the compounds. A method for calculating densities by use of atomic volume summation corrected by derived factors was devised. An IBM-704 computer program was developed to effect all calculations and a compact print out of all data.

A total of 129 unclassified compounds were found with hydrogen densities and neutron removal cross sections higher than polyethylene 6% boron. Of these, 7 were selected as neutron shielding candidates. Cost estimates for these and 3 selected classified compounds were made for 50 and 500 lb. samples and at production rates of 2 and 4 million lb./yr.

176. W. Arbiter, Neutron Shielding Materials for Light Weight Shields, NDA 2143-6 (April 21, 1961). (Nuclear Development Corp.) Not for sale to public.

One hundred and eighty-three elements and compounds are reviewed for suitability as components of light weight neutron shields. A new criterion of excellence relating neutron removal cross section and density is used. Materials of interest are H, NH_4BH_4, $LiBH_4$, $LiBH_4 \cdot 2NH_3$, C_8H_{18}, BeH_2, $(CH_3)_3NBH_3$, NH_3, LiH, $(CH_3)_4NBH_4$, $(CH_3)_x$, $(CH_2)_n$ $(C_2H_5)_4NBH_4$, $C_{30}H_{62}$, $NaBH_4$, $LiNH_2$, $(C_5H_8O_2)_n$, Be_2B, Be_2C, $NH_4H_2PO_2$, B, Be_3N_2, BeO, TiH_2, $(Ti,Mo)H_2$, $(Ti,Cb)H_2$, $(TiV)H_2$, CrB_2, VH, CoB, Mo_2B_5, Co, Ru, Rh, TaB_2, W_2B_5, WB_2, Re_2B_5, WB, Re, Pt, Ir, Os.

177. E. G. Peterson, Shielding Properties of Ordinary Concrete as a Function of Temperature, HW-65572 (August 2, 1960). Availability: Dep.; $1.00 OTS.

The objective of this work was to determine the effect of high temperature on neutron attenuation in ordinary concrete. It is part of a continuing research program to determine the neutron and gamma attenuation properties of concrete shields where the ambient temperature may reach 300 C. The data supplements existing work on iron-limonite, magnetite-limonite, magnetite, and ferrophosphorus concretes.

The fast-neutron attenuation is exponential and can be adequately predicted using removal theory. The calculated removal cross sections agree with the measured values to within three per cent.

Tests indicate that even after penetrating 47.5 inches of concrete the thermal and resonance neutron fluxes have not reached equilibrium with the fast-neutron flux.

Significant changes in neutron flux values were observed after the concrete had been baked at 100, 200, and 300 C. A combination of removal and age theories adequately predicts the increase in the fast, resonance, and thermal-neutron fluxes.

178. E. G. Peterson, Shielding Properties of Ferrophosphorus Concrete as a Function of Temperature, HW-64774 (July 15, 1960). Availability: Dep.; $2.00 OTS.

An investigation was conducted to determine the shielding properties of ferrophosphorus concrete at temperatures up to 320°C for neutrons and gamma radiations. Ferrophosphorus concrete was selected because of its high density and possible use in biological-shield construction. The neutron and gamma attenuation properties of ferrophosphorus concrete were determined experimentally. The thermal and resonance fluxes are based on gold-foil activation while the fast flux was determined using the $S^{32}(n,p)P^{32}$ threshold reaction. The gamma attenuation was measured with ionization chambers. The tests indicate that the ferrophosphorus concrete could be used as a biological shield. It should be pointed out that, under the conditions under which the ferrophosphorus concrete was tested, there are other concretes that could be used which would be more efficient as well as more economical.

179. W. J. Grantham, Jr., Barytes Concrete for Radiation Shielding - Mix Criteria and Attenuation Characteristics, ORNL-3130 (July 10, 1961). Availability: Dep.; $1.50 OTS.

Concrete mix design criteria, based on existing theories of proportioning and specifically oriented toward the solution of radiation shielding problems, were developed. Effects of aggregate gradation, cement-to-aggregate ratio, and water content were examined. A barytes concrete, designed according to these criteria, was thoroughly investigated in the Lid Tank Shielding Facility. Relative effectivenesses of dry aggregates, aggregates plus cement, and cured concrete were compared through thermal-neutron flux, fast-neutron dose, and gamma-ray dose measurements behind slab configurations. Attenuation was measured for the aggregate, the aggregate plus cement, and for the barytes concrete. Comparison with attenuations calculated on the basis of removal cross sections for the measured chemical compositions showed satisfactory agreement.

180. E. P. Blizard and J. M. Miller, Radiation Attenuation Characteristics of Structural Concrete, ORNL-2193 (August 13, 1958). Availability: Dep.; $5.60 (fs) $1.70 (mf) OTS.

An experiment was performed at the Lid Tank Shielding Facility to determine the dependence of the neutron attenuation of a structural concrete shield on the amount and placement of water hydrogen throughout the shield. It was found that a 7 wt% water content is adequate to ensure that intermediate-energy neutrons are quickly slowed down to thermal energy, at which energy they are easily captured. A greater water content improves the overall neutron attenuation according to the removal cross-section theory. In addition to the investigation of the neutron attenuation, the measured gamma-ray attenuation was compared with the predicted gamma-ray attenuation obtained in a calculation using buildup factors determined by the moments method. The two values were in good agreement.

181. T. V. Blosser, G. W. Bond, L. A. Lee et al., A Study of the Nuclear and Physical Properties of the ORNL Graphite Reactor Shield, ORNL-2195 (August 15, 1958). Availability: Dep.; $4.60 (fs), $1.40 (mf) OTS.

A study of the nuclear and physical properties of the concrete shield of the ORNL Graphite Reactor was performed both to determine the radiation attenuation characteristics of the shield and to discover any effects of a long-term (12-year) irradiation. In the experiment neutron and gamma radiation measurements were made in a 4-5/8-in.-diam hole in the shield as the hole was drilled in increments of 1 or 1/2 ft. Concrete plugs were always placed behind the detectors to simulate a homogeneous shield. The average relaxation length for gamma rays in the barytes-haydite concrete, of which most of the shield is made, was found to be approximately 13.6 cm, increasing gradually from 13.0 cm at a shield thickness of 2 ft to 14.6 cm at a shield thickness of 5 ft. The fast-neutron relaxation length in the barytes-haydite concrete varied from 10.0 cm to 10.6 cm, the average being approximately 10.2 cm. Measurements of the streaming of radiation through the hole were also made. The concrete dust collected from the drillings was used to determine the chemical composition, water content, density, compressive strength, and radioactivity of the shield at the various depths. The temperature gradient through the shield was also measured. This investigation showed that the chemical properties and density of the shield have not changed appreciably since a similar investigation in 1948, but its compressive strength is lower (~ 40% near the reflector-shield interface).

182. R. L. French, Reactor Shield Heating Calculations for Gamma Rays, FZM-1076 (June 1958). Availability: Not for sale to public.

The renormalization in configuration space of a slight generalization of the Lee model is discussed as an example of general procedures, which seem to be applicable also to more general cases than the present one. The replacement of ordinary with finite-part integrals removes consistently all ultraviolet infinities and permits writing the Lie equations of the renormalization group, the solution of which yields the familiar relations between unrenormalized and renormalized parameters.

183. J. H. Hubbell and R. L. Bach, Gamma-Ray Buildup Data for Use in Distributed Source Calculations, NBS Report 7501 (May 9, 1962). (National Bureau of Standards.) Not for sale to public.

The spatial distribution of primary radiation flux from a finite plane source may, for points less than a mean-free-path distance, be evaluated as the sum over an infinite series. This series is derived by expanding the exponential dependence $\exp(-\mu_o r)$ as a power series in $\mu_o r$ and integrating, numerically or analytically, each term over the source array. If the medium is effectively infinite and homogeneous, the scattering properties of this medium can be characterized by a point isotropic source buildup factor. Buildup factor data are often approximated by formulas having simple analytic dependences on $\mu_o r$ and numerical parameters independent of $\mu_o r$. A given set of such parameters can be used to construct a corresponding set of weight functions β_n' which can be used with the above primary-flux series solution to give, instead, a total-flux series solution. Tables of β_n''s (air-dose buildup only) derived from coefficients of cubic polynomials fitted to the Goldstein-Wilkins data (NYO-3075) are given for $0 \leq n \leq 13$ at primary photon energies of 0.5, 0.7, 1.0, 1.5, 2.0, 3.5, 5.5, 7.5, and 9.5 MeV in water, Al, Fe, Sn, Pb, W, and U. A comparison with experimental data from a rectangular food irradiator is included as a numerical example.

184. J. H. Hubbell, R. L. Bach, and R. J. Herbold, "Radiation Field from a Circular Disk Source," J. Res. Nat. Bur. Std., 65C, 4, 249-264 (October 1961).

A number of radiation shielding problems involve calculations of the response of an isotropic detector to radiation of arbitrary angular distribution from uniform finite plane sources. Series expansion methods previously used for treating the rectangular source are here extended to include the circular disk source with detector off-axis. These methods involve the family of integrals $\int_S(\cos\theta\ dS/r^2)\ P_l(\cos\theta)$ and the integral $\int_S(dS/r^2)\exp(-\mu r)$ where θ is obliquity with respect to the normal to the disk surface S, r is the distance from an element of source area, dS, to the detector, and μ is the attenuation coefficient. Tabulations of the first type of integral facilitate use of Legendre expansion representations of radiation directional distributions. The second integral relates primarily to exponentially attenuated radiation from a plane isotropic disk source, but the expansion coefficient solution can be readily adapted to take into account a point isotropic buildup factor of the polynomial form

$$\sum_{i=0}^{N} \beta_i(\mu r)^i.$$

This adaptation applies equally well to the corresponding expansion coefficients previously given for the rectangular source. Formulas and numerical results are presented.

185. J. H. Hubbell, R. L. Bach, and J. C. Lamkin, "Radiation Field from a Rectangular Source," J. Res. Nat. Bur. Std., 64C, 2, 121-138 (April 1960).

Many radiation shielding problems involve calculations of the response of an isotropic detector to radiation of arbitrary angular distribution from uniform rectangular sources. In calculations of this type the family of integrals $\int_S(\cos\theta\ dS/r^2)\ P_l(\cos\theta)$ and the integral $\int_S(dS/r^2)\ \exp(-\mu t/\cos\theta)$ are frequently encountered, where θ is obliquity with respect to an axis perpendicular to the plane containing the rectangular radiant surface, S, r is the distance from an element of source area, dS, to the detector, μ is the attenuation coefficient, and t is the barrier thickness. Solutions of the first type of integral facilitate use of Legendre expansion representations of radiation directional distributions, and may also have application in other radiant surface studies, such as illumination and heat exchange engineering. The second integral relates to exponentially attenuated radiation from a plane isotropic rectangular source separated from the detector by a layer of material of thickness t. Formulas, expansions, and numerical results are presented.

186. R. T. Ackroyd and E. D. Pendlebury, Survey of Theoretical Calculation Methods, DEG Report 362(R) (March 27, 1961). (United Kingdom Atomic Energy Authority. Development and Engineering Group, Risley, Lancs, ENGLAND.)

A survey is given for those methods of calculating critical sizes of isolated cores of fissile material which have been found useful by British workers in the criticality field. (A companion paper, "Methods of Calculation for Interacting Arrays with Special Reference to Storage Problems," deals with the problem of interaction between several cores of fissile material.) The methods range from the simplest, which are based on few-group diffusion theory, to advanced methods such as Monte Carlo, and those based on numerical solution of the Boltzmann equation of transport theory. Methods are given for all types of core and reflector spectra. The advanced methods, such as the Carlson S_n, are recommended for establishing accurate critical size data for the simplest shapes of cores, when experiments are not available, provided adequate cross-section data is at hand. For interpolating between accurate critical sizes, which may be either experimental or theoretical, experience has shown that simple methods based on a diffusion theory model are often adequate, but it cannot be taken as a rule. For critical size calculations for cores of complex shape, the use of diffusion theory is recommended if the physics is well understood. If the physics is not well understood but there is adequate cross-section data, the Monte Carlo method is suggested. The review of methods of calculation concludes with a sketch of a geometrical method of solving numerically the Boltzmann equation, which gives at each stage of computation the maximum error of the approximate solution. It is suggested that methods of this type may be found useful in cheapening advanced calculations, as they could be terminated as soon as the maximum error was acceptable instead of having to wait in the usual way for convergence to be attained.

To facilitate the wise use of the methods of calculation which are given, the review concludes with a note on cross-section data requirements.

187. G. T. Chapman, J. B. Dee, J. M. Miller, and W. J. McCool, Measurement of an Effective Neutron Removal Cross Section of Lithium at the Lid Tank Shielding Facility, ORNL CF 54-11-3 (November 2, 1954). Availability: Dep.; $3.30 (ph), $2.40 (mf) OTS.

An effective neutron removal cross section of 1.01 ± 0.04 barns was determined for lithium at the LTSF. This value was obtained from measurements made in a medium of oil behind a slab of metallic lithium which was placed against the source plate.

188. REIC, Transient Radiation Effects on Electronic Components and Semiconductor Devices, REIC Report No. 26 (April 19, 1963). Available from the Radiation Effects Information Center.

No abstract necessary.

189. D. S. Duncan and H. O. Whittum, Jr., Application of Fast-Neutron Removal Theory to the Calculation of Thermal-Neutron Flux Distributions in Reactor Shields, NAA-SR-2380 (July 1, 1958). Availability: Dep.; $4.60 (fs), $1.46 (mf) OTS.

A calculational method is presented which may be used to determine fast and thermal-neutron flux distributions at deep neutron penetrations in hydrogenous shields. The method is based primarily on fast-neutron removal theory, which is used in conjunction with a two-group diffusion model for calculating the thermal-neutron flux distribution. A comparison of calculated flux distributions with the results of experimental measurements made in the shield of the Bulk Shielding Reactor and in a portion of the Submarine Thermal Reactor shield mock-up indicates that the method is accurate in the hydrogenous shield regions at several diffusion lengths from nonhydrogenous boundaries. However, the method is only approximately correct in the remaining shield regions. Although it is recommended that other methods be investigated in future neutron penetration studies, it is concluded that the method presented is extremely useful for many reactor shield calculations because it is relatively simple to apply and yields excellent results in the hydrogenous portions of a composite shield.

190. B. Davison, Neutron Transport Theory, Oxford University Press (1958).

This is a general reference for transport theory methods.

191. E. Tochilin, Flux and Spectral Measurements of Primary and Moderated Neutron Sources, USNRDL-TR-587 (October 8, 1962). (U.S. Naval Radiological Laboratory, San Francisco.) Not for sale to public.

Radioactive (α,n) neutron sources maintained by the U.S. Naval Radiological Defense Laboratory (USNRDL) include Ra-Be, PuF_4, and four $PuBe_{13}$ sources. Comparative neutron emission rates for these sources were determined by two separate methods. These included: (1) the manganese sulfate bath with appropriate corrections made for fast-neutron escape or capture and for thermal-neutron capture by the source; and (2) long counter measurements corrected for energy response and source anisotropy. Agreement to within 1 percent was obtained by the two independent methods. A Ra-Be source calibrated against the National Bureau of Standards neutron standard was used as a primary reference.

Spectral measurements of $PuBe_{13}$ and PuF_4 neutron sources were made with nuclear emulsions. By the use of cylindrical plastic and lead absorbers, an effort was made to moderate the $PuBe_{13}$ spectrum and thereby obtain additional calibration points for instrument response studies. Spectral changes were initially studied with the double-moderator technique, followed by later measurements with nuclear emulsions. From flux and spectral measurements of the primary and moderated radioactive sources, the first collision tissue dose was calculated and, in turn, compared to experimental measurements of absorbed dose with proportional counters designed according to the Bragg-Gray principle.

192. A. W. Casper, Comparison of Bulk Shielding Reactor Centerline Measurements in Water with Predictions, APEX 504 (November 1958). Availability: Dep.; $2.60 (fs), $1.01 (mf) OTS.

Measurements of fast-neutron and gamma-ray dose rates in water along the centerline of the Bulk Shielding Reactor (BSR) are compared with predictions made by use of three available computer programs. It is found that the predictions are very dependent on the assumptions made concerning power distribution and that the more accurate descriptions of the source lead to predictions that agree with measurements to within 20 percent throughout thickness ranges as great as 100 centimeters of water.

193. R. E. Baker, Shadow Shields and Scattered Gamma Radiation, APEX-432 (June 1957). Availability: Dep.; $4.60 (fs), $1.49 (mf) OTS.

It is extremely difficult to predict the gamma-ray intensities arriving at a point which is completely shielded from the primary gamma rays traveling a line-of-sight path but relatively unshielded from the scattered gamma rays which arrive at the receiver point from various angles. In order to obtain data which will yield a better general understanding of the importance of the scattered component, a series of investigations were undertaken to show the effect of incremental changes in some of the variable quantities involved in a source-barrier-detector arrangement in a scattering medium.

194. A. W. Casper and J. G. Carver, Comparison of Computed Centerline Dose Rates from Different Areas of a Source Plate, APEX-507 (November 1958). Availability: Dep.; $2.60 (ps), $1.07 (mf) OTS.

A shield fixture that consists of a slab of iron with a cylindrical hole into which plugs of various materials can be inserted is described. Computations made to determine the effect of source-plate size on centerline dose rates are described, and incremental dose contributions from various annular sections of the source plate are presented. Plugs of iron, aluminum, and lead were used in the computations.

195. G. S. Stanford, Foil Data Thickness Corrections as Affected by Self-Shielding and Flux Depression, TID-16733 (June 19, 1962). Availability: Dep.; $1.60 (fs), $0.80 (mf) OTS.

A method is presented for obtaining matched-foil data from irradiations of unmatched foils, by making use of the first derivative of published flux-perturbation formulas. It is necessary to treat thermal and resonance activation separately. It is shown that non-uniformity in thickness within a given foil is not normally an appreciable source of error, although the area of the foil should be well known. For the special case of 25-micron gold foils, it is shown that, for moderate thickness deviations from foil to foil, it is a good approximation to neglect the thermal flux perturbations and to assume that the foils are very thick to resonance neutrons, resulting in a rather simple weight-correction formula.

196. B. T. Kimura, Penetration of Radiation Through Crew Shields, NARF-56-29T (FZK-9-104) (November 30, 1956). (Convair, Fort Worth.) Not for sale to public.

The single-scattering theory of predicting scattered radiation penetrating a crew shield is reviewed. The results of this theory are compared with experimental data taken with the Ground Test Reactor (GTR) operating at one kilowatt.

The experimental arrangement consisted of the GTR, a crew shield containing the detector, and a shadow shield mounted in front of the reactor. The axes of these experimental components were aligned along a line, h = 12.9 feet above the ground. The crew shield and source were separated by a distance a (measured between the centers of the source and crew shield) which took values of 33.5 feet, 48.6 feet, and 66.5 feet. The shadow shield angle α' took values of 10°, 20°, 35°. The detector position, z, relative to the inner wall of the crew shield, was varied between a position near the rear wall and a position near the center of the crew shield.

Comparisons between calculated and experimental dose rates are made for the small cylinder and the thin crew shield.

The magnitude and characteristics of the deviation between calculated and experimental results for a given crew shield depends on α', z, the shield shaping on the crew compartment, and the type of radiation.

197. G. Placzek, "The Angular Distribution of Neutrons Emerging from a Plane Surface," Phys. Rev. 72, 7, 556-558 (October 1, 1947).

The Wiener-Hopf expression for the angular distribution in Milne's standard case is transformed into a form suitable for numerical evaluation. The results of the evaluation carried out by the Mathematical Tables Project are given.

198. F. Johnson, Jr., Source Integrated Material Distribution Attenuation Functions (Distribution Functions), XDC 60-5-120 (May 5, 1960). (General Electric, Cincinnati.) Not for sale to public.

A source integrated material distribution attenuation function is an analytic expression for the effective attenuation of radiation at a receiver point produced by a shielded radiation source of finite size at a large distance. For the sake of brevity, this function is most often referred to as a distribution function. The important feature of a distribution function is that it is derived through the functional integration over the source of functions which express the relative source strength distribution and shield attenuation at the receiver point as functions of the variable(s) of integration. In the strictest sense, the use of distribution functions is limited to radiation which is attenuated on a line-of-sight basis. A distribution function is to a source of finite size as the analytic expression of ψ (attenuation due to material thicknesses) is to a point source. Distribution functions are useful in analyzing the shielding characteristics of source-shield configurations having very irregular geometry. As such, they are used extensively in the analysis of the shielding characteristics of porous shields. Distribution functions are also useful in analyzing the effects of the use of different shield materials in a configuration having a fixed irregular geometry.

The general theory of distribution functions and the derivation of the more elementary of these functions are described herein. A tabulation is also given of the more useful of these elementary distribution functions.

199. J. Fagan, Shield Design Methods for Argonaut-Type Reactors, ANL-6281 (March 1960). Availability: Dep.; $1.00 OTS.

The methods used in calculating the shielding for Argonaut-type reactors are described. The shield designs for the Argonaut-I Reactor, now operated by the International School of Nuclear Science and Engineering at Argonne National Laboratory, and the Argonaut Reactor, which was displayed at the Second United National International Conference on the Peaceful Uses of Atomic Energy, Geneva, 1958, were investigated. The methods used appeared to be satisfactory within the anticipated limits of accuracy.

200. R. L. Kloster, Analysis of the Moments Method Experiment, NARF-59-33T (FZK-9-138) (September 4, 1959). (Convair, Fort Worth.) Not for sale to public.

Monte Carlo calculations show the effects of a plane water-air boundary on both fast-neutron and gamma dose rates. Multigroup diffusion theory calculation for a reactor source shows the effects of a plane water-air boundary on thermal-neutron dose rate. The results of Monte Carlo and multigroup calculations are compared with experimental values.

The predicted boundary effect for fast neutrons of 7.3% agrees within 16% with the measured effect of 6.3%. The gamma detector did not measure a boundary effect because it lacked sensitivity at low energies. However, the effect predicted for gamma rays of 5 to 10% is as large as that for neutrons.

An estimate of the boundary effect for thermal neutrons from a Po-Be source is obtained from the results of multigroup diffusion theory calculations for a reactor source. The calculated boundary effect agrees within 13% with the measured values.

201. D. Yarmush, H. J. Zell, and R. Aronson, The Transmission Matrix Method for Penetration Problems, WADC Technical Report 59-772 (August 1961). Availability: OTS.

A practical machine method for computing transmission and reflection of radiation by laminated slab shields is described, making use of transmission and reflection matrices. These are combined in a way which produces a formal solution of the Boltzmann equation in slab geometry. This result is applied by using representations in terms of energy groups and angular expansions. Two codes for the IBM-704 which together carry out the computation of the transmission and reflection matrices are described and operating instructions are given. Numerical results for gamma rays are presented. The advantages of the transmission matrix method as compared to other methods are discussed. Appendices are included showing how in the P_1 approximation the method gives diffusion theory in a particularly simple way and how it can be applied to other geometries.

202. C. E. Humphries and N. D. King, Neutron and Gamma Air Scattering and Air Secondary Transfer Functions for Shielding Applications, NARF-62-1T (MR-N-283) (March 9, 1962). (Convair, Fort Worth). Not for sale to public.

To aid in the design of shielded compartments, air transfer functions have been generated for four types of interactions: (1) air scattering of gamma rays, (2) production of gamma rays by capture of neutrons, (3) production of gamma rays by inelastic scattering of neutrons, and (4) air scattering of neutrons. Attenuation through various combinations of neutron and gamma shielding thicknesses is considered in these functions, but gamma production within the shielding is not.

203. G. Goertzel, Quota Sampling and Importance Functions in Stochastic Solution of Particle Problems, AECD-2793 (June 21, 1949). Availability: Dep.

In Section 1 a general Stochastic particle problem is defined. Section 3 defines a quota sampling method available for simplifying the solution of such a problem. In Section 4 an importance function, $Q(\alpha)$, is defined such that if it were known and could be used in the quota sampling, the solution of the Stochastic problem is greatly reduced in difficulty. Thus any function $q(\alpha)$, used in quota sampling, should be selected to represent $Q(\alpha)$ as closely as one's knowledge permits.

In Section 5 it is shown how to set up a pair of problems, such that the solution of each is the importance function for solving the other. Thus $Q(\alpha)$ may be estimated Stochastically.

204. B. Carlson, Numerical Solution of Transient and Steady-State Neutron Transport Problems, LA-2260 (October 29, 1959). Availability: Dep.; $3.30 (ph), $2.40 (mf) OTS.

A general numerical procedure, called the discrete S_n method, for solving the neutron transport equation is described. The main topics relate to the derivation of suitable difference equations, and to the problem of solving these, while maintaining generality, accuracy, and reasonable computing speed. A few comparisons with other methods are made.

205. T. S. Dahlstrom and W. E. Thompson, The Angular Distribution of Dose Rate from Gamma Rays Scattered Through Various Thicknesses of Iron and Aluminum, USNRDL-TR-558 (April 19, 1962). Availability: $6.60 OTS as AD-276918.

The angular distribution of dose rate resulting from gamma rays of Co^{60} and Cs^{137} transmitted through various thicknesses of iron (0.737 in. and 1.6 in.) and aluminum (2 in., 4 in., and 9 in.) was measured. These experimental results are compared with Monte Carlo calculations. Comparisons over a limited range of these angular variables are also made with dose-rate distributions calculated from gamma-ray spectrometric data.

206. C. D. Zerby, A Monte Carlo Method of Calculating the Response of a Point Detector at an Arbitrary Position Inside a Cylindrical Shield, ORNL-2105 (June 12, 1956). Availability: Dep.; $3.30 (ph), $2.40 (mf) OTS.

A Monte Carlo method of calculating the response of a point detector at an arbitrary position inside a cylindrical shield is described. The general procedure is developed to obtain the maximum amount of information from each history and to minimize the statistical error.

207. E. L. Kaplan, Backward Monte Carlo and Other Methods for Reversing Stochastic Processes, UCRL-6649 (October 2, 1961). Availability: $0.50 OTS.

When a stepwise Monte Carlo calculation is such that it has to terminate in a very restricted region in order to contribute to the result, it is known that a gain in efficiency may be achieved if the problem is reformulated so that the calculations can begin in the restricted region. The resulting backward Monte Carlo is discussed in terms of matrices and neutron transport; it corresponds to taking the transpose of the matrix of transition probabilities, and is contrasted with the inverse of such a matrix and the inverse probabilities calculated according to Bayes' theorem.

It is shown that in anisotropic scattering, the azimuthal angle is uniformly distributed in the backward process as well as in the forward process. A pair of zero-variance sampling schemes that are related to Bayes' theorem are discussed.

208. J. Spanier, Monte Carlo Methods and Their Application to Neutron Transport Problems, WAPD-195 (July 1959). Availability: $2.00 OTS.

Some applications of Monte Carlo methods to neutron transport problems are introduced after a summary of the relevant probability theory which was placed in a setting of abstract measure theory. This is definitely not for the casual reader.

209. A. R. Ptitsyn, "Use of the Moments Method for Calculating the Spatial and Energy Distribution of the Neutron Fluxes from Plane and Point Sources in an Infinite Medium," Soviet J. At. Energy, English Transl. 10, 2, 109-121 (February 1961).

A method is given for calculating the space-energy distribution of neutron densities from flat and point sources in an infinite medium.

The neutron density $\psi(x,E)$ is sought in the form

$$(\psi,E) = \sum_{i=1}^{N} a_i(E)\,K[b_i(E)\ r] \quad .$$

To a large degree the form of the function $K(x)$ is arbitrary; its selection is based on physical principles. From the 2N space moments of the function $\psi(x,E)$, 2N parameters a_i, b_i are found. The neutron density distribution is found in hydrogen and water. The calculations for water are compared with experimental data. A comparison with the accurate solution of Wick in the case of retardation of neutrons by hydrogen shows that from four moments the suggested method can, with sufficient accuracy, find the spatial distribution of neutrons at distances up to 20 free path lengths.

210. R. E. Beissner, The Application of Invariant Imbedding to Shielding Problems, NARF-61-41T (MR-N-287) (March 9, 1962). Availability: $6.60 OTS as AD-273641.

The fundamental concepts of the theory of invariant imbedding as applied to particle transport processes are reviewed and their applications to shielding problems are discussed. Two new approximate methods for predicting the transmission of particles through slabs of material are derived from the imbedding equations. One-velocity computations of the angular distributions of particles reflected and transmitted by slabs of varying thickness and composition are presented and analyzed.

211. K. M. Case, F. DeHoffmann, and G. Placzek, Introduction to the Theory of Neutron Diffusion - Vol. I, Los Alamos Scientific Laboratory Monograph (June 1953). Availability: $1.25 G.P.O.

This monograph attempts to give a detailed discussion of the general equations of one-velocity neutron diffusion theory and of their solution for the special case of a homogeneous infinite medium with isotropic scattering. The subject matter covered forms, regarding both methods and results, only an almost infinitesimal part of the general theory of one-velocity neutron diffusion but may be regarded as basic for the understanding of many of the papers dealing with more complicated problems in this field.

The volume is based on the first few lectures of a course on neutron diffusion given by me during the summer of 1949 at the RAND Corporation, Santa Monica, California, and at the University of California at Los Angeles. Frederic de Hoffmann recorded the lectures and prepared a first draft of this volume. The task of writing a large portion of the final version of the book was carried out by K. M. Case, who added much original material to it. In particular, the proof for the reciprocity theorem in Section 11 and for the source-free infinite-medium solution in Section 13 are entirely due to him. For Sections 16.1, 17, and 18, I have to assume sole responsibility.

Emphasis has been placed on fairly complete tables and graphs. The numerical work was carried out by the Los Alamos computing group under the direction of Bengt Carlson and Max Goldstein. In those cases where numerical results of other authors were available they have been recomputed except for Table 28, which has been taken from the table of $E_n(x)$ functions prepared by the Mathematical Tables Project. G. Placzek.

212. L. A. Beach, R. B. Theus, P. Shapiro et al., Comparison of Solutions to the One-Velocity Neutron Diffusion Problem, NRL Report 5052 (December 23, 1957). Availability: $3.30 (mf), $7.80 (ph) OTS as PB-129100.

By Fourier transform techniques the exact solutions of the one-velocity neutron diffusion problem have been derived for plane isotropic, plane parallel mono-directional, and plane parallel bidirectional source terms. These exact solutions in terms of Fourier inversion integrals were numerically evaluated upon the Narec to give the angular distribution of the scattered intensity, total scattered intensity, and total intensity. By solving the integrals by contour integration in the complex plane, asymptotic solutions were obtained which are good approximate solutions for deep penetrations and problems with little absorption. Moment method solutions were derived and evaluated to be in good agreement with the exact solutions for symmetric source and intensity functions. Monte Carlo estimates can be obtained that correctly account for source asymmetries and boundary discontinuities and which can be extended to at least 20 mean free paths utilizing suitable bias techniques such as the exponential transformation.

213. R. L. Hickmott, Removal Dose as an Environmental Measurement of X-Rays and Gamma Rays, WADC Technical Report 59-119 (April 1959). Availability: $1.00 OTS as PB-151338.

The Removal Dose of any material is the net energy per unit mass that a limitingly small mass of the material transfers from the incident photon flux to the kinetic energy of the associated flux of charged particles. Removal Dose is compared to both Absorbed Dose and Exposure Dose (roentgen) in terms of cross sections and spectra. Its relation to radiation effects is compared to the similar relation of Absorbed Dose.

214. A. Lorenz, Gamma Energy Deposition in Infinite Source Media, UCRL-6891 (May 15, 1962). Availability: Dep.; $2.00 OTS.

Based on the principle of energy conservation, a time-independent multigroup technique to calculate photon energy deposition in materials composing an infinite gamma-ray source medium has been derived and coded for the IBM 7090 computer. Applicable to homogeneous media, the method is extended to treat nonhomogeneous media by applying self-shielding corrections to the cross sections. The results of sample calculations show good agreement with Monte Carlo results.

The code ultilizes microscopic cross-section input in calculating self-shielding factors, probabilities of gamma-ray interactions and energy down-scatter; these quantities determine the allocation of the photon energies of a given source spectrum to the constituent elements of the medium. With a capacity to handle 30 energy groups and 9 materials, a single problem runs a fraction of a minute on the IBM 7090 computer.

215. J. E. Faulkner, Single Isotropic Air Scattering of Neutrons in the Presence of the Ground (Unshielded Detector), ORNL CF 54-8-96 (August 31, 1954). Availability: Dep.; (mc) $1.80 (ph), $1.80 (mf) OTS.

Calculations are made of the fraction of air scattering occluded by the presence of the ground as a function of altitude for the case in which the source and detector are at equal heights above the ground.

216. W. E. Kinney, A Monte Carlo Calculation of Scattered Neutron Fluxes at an Air-Ground Interface Due to Point Isotropic Sources on the Interface, ORNL-3287 (July 13, 1962). Availability: Dep.; $1.00 OTS.

Scattered neutron fluxes on the interface between SiO_2 ground and air due to 1- to 19-MeV point isotropic monoenergetic sources located on the interface have been calculated by the Monte Carlo method at distances out to 4×10^5 cm from the source. The validity of the method was established by comparing Monte Carlo results for an infinite air medium with moments method results. The two methods were found to yield fluxes that were in good agreement out to 8 source mean free paths for a 2-MeV source. At higher source energies the treatment of inelastic scattering as isotropic elastic scattering overestimated the high-energy flux and underestimated the low-energy flux, while allowing only P_1 anisotropic elastic scattering overestimated the flux near the source and underestimated it far from the source. Scattered neutron fluxes at the interface of black ground and air, which is a useful limiting case, were also calculated.

217. R. Aronson, Neutron Penetration in Hydrogen, NDA 15C-39 (June 11, 1954). Availability: Dep.; $4.00 (ph), $2.25 (mf) OTS.

Results for doses and buildup factors are given for the penetration of neutrons through hydrogen as computed by the moment method. Comparisons are made with a Monte Carlo calculation, and a calculation by a method due to Welton. A discussion of an anomalous flux at zero energy is included.

218. E. S. Troubetzkoy, Continuum Theory of Gamma-Ray Spectra Following Inelastic Scattering, NDA 2111-3, Vol. B (November 1, 1959). Availability: $3.30 (fs), $2.40 (mf) OTS as PB-146778.

A theory predicting gamma-ray spectra following inelastic scattering of fast neutrons is derived on the basis of the continuum model. The problem is reduced to integrals over the solution of a fourth-order differential equation.

219. H. Goldstein and A. D. Krumbein, Moments Method Calculation of the Penetration of Neutrons from a Point Fission Source Through Be and BeO, NDA Memo 2124-1 (May 27, 1960). (Nuclear Development Corporation of America.) Not for sale to public.

The results of moments method calculations (using the RENUPAK code) of fission neutron penetration in Be and BeO are presented. The energy range is from 18 MeV to 1.23 eV. The results supercede the results in NDA-2092-9. The resulting age to Indium of fission neutrons (71.5 cm^2) is in good agreement with Monte Carlo calculations, but is lower than the experimental value.

220. W. R. Burrus, Radiation Transmission Through Boral and Similar Heterogenous Materials Consisting of Randomly Distributed Absorbing Chunks, ORNL-2528 (January 5, 1960). Availability: Dep.; $2.60 (fs), $1.10 (mf) OTS.

Shields that consist of randomly distributed absorbing chunks in a relatively transparent matrix must contain a greater mass of absorber than homogeneous shields which provide the same attenuation. This is the result of radiation "channeling" between the absorbing chunks. Channeling is particularly important for heterogeneous materials when the mean free path for absorption is comparable to the chunk size. A newly developed method for calculating the transmission of radiation through such heterogeneous shields is described. The numerical results of a calculation of the transmission of thermal neutrons by boral (a B_4C-Al mixture) are given, including the effects of energy and angular distributions on the predicted attenuation. The calculated results are in reasonable agreement with available experimental results.

221. J. Spanier, "A Unified Approach to Monte Carlo Methods and an Application to a Multigroup Calculation of Absorption Rates," Siam Review 4, 2, 115-134 (April 1962).

The processes involved in the application of Monte Carlo methods to the transport of particles are described in highly mathematical terms and a specific application, namely that of calculating the spatial distribution of thermal absorption rates in a two-dimensional cell, is shown.

222. W. B. Lane and M. J. Nuckolls, An Improved Zinc Bromide Shielding Window, USNRDL-TR-594 (October 2, 1962). Availability: $1.10 OTS.

An improved zinc bromide-filled shielding window for hot cell use was designed, constructed, and performance-tested.

Bare steel surfaces under an argon atmosphere satisfactorily contained zinc bromide solution. No corrosion or deterioration of optical properties appear in a window which was installed in May 1961. This system is an improvement over the previously used painted surfaces in an air atmosphere.

223. J. Certaine, A Solution of the Neutron Transport Equation. Part II. NDA-Univac Moments Calculations, NYO-6268 (May 31, 1955). Availability: Dep.; $10.80 (ph), $3.90 (mf) OTS.

The techniques used in the UNIVAC phase of the NDA Fundamental Neutron Penetration Program are described. As in Part I (SEAC calculations) the procedure entails the calculation of the (spatial and angular) moments of the angular number flux. The assumption made in this treatment of the degradation process is that (for each element) the product of the scattering cross-section times a given moment is approximable by a sectionally linear function of lethargy. With the existing UNIVAC code one can calculate the moments of the total angular number flux at equally spaced points in lethargy for a point isotropic continuous energy source.

A detailed exposition of the various mathematical steps is included, together with some numerical results illustrative of the accuracy obtained in the calculations. Other applications of the method are also given.

224. J. Certaine, A Solution of the Neutron Transport Equation. Part III. Reconstruction of a Function from Its Moments, NYO-6270 (NDA 15C-6) (July 31, 1956). Availability: Dep.; $9.30 (ph), $3.60 (mf) OTS.

Parts I and II of this report describe methods of calculating the moments of the angular number flux. This part contains methods which have been found useful in the reconstruction of the components of the expansion of the angular flux in a Legendre polynomial expansion.

225. G. R. Keepin, T. F. Wimett, and R. K. Zeigler, "Delayed Neutrons from Fissionable Isotopes of Uranium, Plutonium, and Thorium," Phys. Rev. 107, 4, 1044-1049 (August 15, 1957).

The periods, relative abundances, and absolute yields of delayed neutrons from "fast" fission of six nuclides (U^{235}, U^{233}, U^{238}, Pu^{239}, Pu^{240}, and Th^{232}) and thermal fission of three nuclides (U^{235}, U^{233}, and Pu^{239}) have been measured. "Godiva", the bare U^{235} metal assembly at Los Alamos, was the neutron source. Six exponential periods were found necessary and sufficient for optimum least-squares fit to the data. Despite evident perturbations, general agreement among delayed-neutron periods was obtained for all nuclides. The absolute total delayed-neutron yield for each nuclide has been measured for fast and thermal fission; a spectral effect was not observed. Representative of general delayed-neutron periods (half lives) and abundances are the U^{235} fast fission data:

Half-life (sec)	Relative Abundance
54.51 ± 0.94	0.038 ± 0.003
21.84 ± 0.54	0.213 ± 0.005
6.00 ± 0.17	0.188 ± 0.016
2.23 ± 0.06	0.407 ± 0.007
0.496 ± 0.029	0.128 ± 0.008
0.179 ± 0.017	0.026 ± 0.003

These data have been corroborated in detail by independent period-vs-reactivity measurements on the bare U^{235} assembly. A comparison of the present results with a phenomenological theory of delayed-neutron emission has been made. This treatment has led to the prediction of several new short-period delayed-neutron precursors.

226. J. A. DeJuren and R. K. Paschall, Thermal Neutron Transmission Through Cadmium-Covered Foils, NAA-SR-7770 (March 15, 1963). Availability: Dep.; $0.50 OTS.

The transmission of thermal neutrons through cadmium covers as a function of cadmium thickness was measured in a graphite stack at various heights above the thermal column of the AE-6 reactor. An aluminum disk containing slots for the foils was rotated in the center graphite stringer position. Foils of 40% indium and 60% tin, 0.005 in. thick, were used as detecting foils and were beta-counted. As the thermal flux incident on a cadmium cover is depressed relative to the flux incident on a bare foil, a foil with a thick cadmium cover on one side only was used for the zero thickness measurement and its activity was doubled. The resultant thermal activities as a function of cadmium thickness were consistent with the exponential integral $2E_3(\Sigma_a d)$, where

$$E_3(\tau) = \int_1^{\infty} \frac{e^{-\tau x}}{x^3}\, dx$$

and a value of 2514 barns was used for the absorption cross section of cadmium. For $\tau = 0$, $2E_3(\tau) = 1$.

227. C. R. Hatcher and C. P. Jupiter, Measurement of High-Intensity Gamma Radiation, UCRL-7223 (January 31, 1963). Availability: Dep.; $0.50 OTS.

The use of Victoreen ionization chambers to monitor high-intensity pulsed gamma radiation is discussed. It is shown that the type 154 thimble chamber gives valid readings in pulsed bremsstrahlung fields having peak intensities as high as 5×10^4 r/sec. The Victoreen chambers may then be used to calibrate a plastic fluor-photodiode detector whose saturation level is of the order of 10^{10} r/sec or greater. It is demonstrated that the sensitivity of the plastic fluor detector is constant for a total received dose of $\leq 2 \times 10^5$ r.

228. A. T. Nelms, Graphs of the Compton Energy-Angle Relationship and the Klein-Nishina Formula from 10 keV to 500 MeV, NBS Circular 542 (August 28, 1953). Availability: $0.55 GPO.

The Compton energy versus angle relationship and the differential integral Klein-Nishina cross sections are presented graphically as functions of the energy and direction of the scattered photon and of the recoil electron. These graphs are intended to serve the purpose of tables. Unpolarized primary gamma rays in an energy range from 10 KeV to 500 MeV are considered. The accuracy of all curves is estimated at 1 percent. The advantage of this form of presentation is the convenience and accuracy of two-way interpolation. In general, interpolated values may be obtained with an accuracy of 2 percent.

229. L. R. Day and M. L. Mullender, Shielding Problems Associated with a Cockroft-Walton Accelerator, AWRE NR-1/63 (January 1963). (Great Britain Atomic Weapons Research Establishment.)

Measurements have been made to determine the efficiency of a thick concrete shield subjected to 14-MeV neutrons produced by the H-3(d,n)He-4 reaction, and the results have been used to check the validity of multi-group shielding calculations for 14-MeV neutron attenuation in concrete.

The fast-neutron relaxation length in the shield was found to be 15.0 $\pm$ 0.5 cm.

230. Sigma Center, Angular Distributions in Neutrons - Induced Reactions, BNL-400, Sec. Ed. (Two Volumes) (October 1962). Availability: Dep.; $16.80 (ph), $5.70 (mf) OTS.

This is a compilation of the most significant experimental observations available in October 1962 on the angular distributions of scattered neutrons and of other products of the reactions of fast neutrons with nuclei. The primary angular distribution data are plotted on linear grids for ease in reading off the differential cross-section values. The auxiliary and bibliographic information associated with each graph appears on the page of text opposite the graph. Volume I covers elements 1 through 22. Volume II covers elements 23 through 94.

231. D. C. Anderson, "On the Temperature Dependence of the Thermal-Neutron Flux Kernel," Nucl. Sci. Eng., 7, 5, 468-471 (May 1960).

The thermal-neutron flux kernel for a point fission source in a hydrogenous medium is obtained analytically by representing the epithermal slowing down source in a convenient functional form. Normalization is achieved by invoking an appropriate conservation condition. The temperature dependence is then assessed from experimentally determined variation in the diffusion length and appropriate variation in the fitting parameters for the slowing down source. It is concluded that the kernel for water is rather insensitive to change in the diffusion length, and in fact, the r^2-flux varies to a good approximation as $f(\rho r)$, ρ being the temperature-dependent specific gravity.

232. D. C. Anderson and K. Shure, "Thermal-Neutron Distributions in Metal-Hydrogenous Shields," Nucl. Sci. Eng., 8, 3, 260-269 (September, 1960).

A calculational technique is presented for predicting thermal-neutron fluxes in the primary shields of reactor systems which eliminates reliance on mockup experimental data. A multigroup P_1 approach is employed with the spatial dependence of the neutron attenuation adjusted through use of a point source attenuation kernel for a homogeneous hydrogenous medium. Comparison of calculation with experiment is presented.

233. F. J. Allen, A. T. Futterer, and W. Wright, Neutron Reflection and Flux Versus Depth for Nevada Test Site Soil, BRL-1190 (January 1963). Availability: $16.50 OTS.

Detailed calculated results on neutron reflection and flux versus depth for Nevada Test Site soil are given in the form of machine printouts. The angular and energy distributions of the reflected neutrons along with the energy-dependent and total flux at various depths are contained in tabular form on the printouts. Neutron number current, number flux, and dose transmission as functions of thickness are also given in tabular form on the printouts.

Tables of summary information on reflection are presented. These contain number current, number flux, dose and energy reflection factors as functions of incident energy and angle.

A few figures are presented to illustrate graphically the meaning of the various tabular results.

234. F. J. Allen, A. T. Futterer, and W. Wright, Neutron Reflection and Flux Versus Depth for Iron, BRL-1199 (March 1963). (Ballistic Research Laboratory) Not for sale to public.

Detailed calculated results on neutron reflection and flux versus depth for iron are given in the form of machine printouts. The angular and energy distributions of the reflected neutrons along with the energy-dependent and total flux at various depths are contained in tabular form on the printouts. Neutron number current, number flux and dose transmission as functions of thickness are also given in tabular form on the printouts.

A table of summary information on reflection is presented. This contains number current, number flux, dose and energy reflection factors as functions of incident energy and angle.

A few figures are presented to illustrate graphically the meaning of the various tabular results.

235. J. M. Chapman, Gamma-Ray Dose Rates and Energy Spectra in a 3-Foot-Square Duct, USNCEL R325 (October 1964) Availability: U. S. Naval Engineering Laboratory, Port Hyeneme, California.

Dosimeter measurements and gamma-ray spectra were taken in a 3-foot-square concrete duct, using cobalt-60 and cesium-137 sources. The dosimeter measurements are compared with computer-calculated dose rates, with very good agreement. The spectra show prominent peaks for single scatter and a large amount of lower-energy gamma rays. The spectra are used to determine the dose-rate contributions of individual scattering areas. These dose-rate contributions are compared to calculated values with fair results. The percentage of the dose-rate contribution from one scattering area due to gamma rays that have been previously scattered in the first leg was determined experimentally to be about 40 per cent.

236. W. L. Bunch, Attenuation Properties of High Density Portland Cement Concrete as a Function of Temperature, HW-54656 (January 22, 1958). Availability: Dep.; $5.60 (fs), $1.85 (mf) OTS.

The elevated temperatures used in the test appreciably impaired the neutron and gamma attenuation properties of the two high density concretes. Nonetheless, the concretes would serve as satisfactory shield material if the reduced attenuation efficiency at the elevated temperatures were taken into account in the original reactor design. The two concretes used in the test were: (1) iron limonite, which consists of an aggregate of iron punchings and a grout of limonite, an iron ore containing water of hydration, and (2) magnetite limonite in which the non-hydrated iron ore magnetite is used as an aggregate and the limonite again is the grout material. Significant changes in attenuation effectiveness were found as the concretes were heated to 100 C and 175 C, but very little change was effected in the final heating to 320 C. The density of the concretes was similarly affected, indicating that the bulk of the water had been evolved at 175 C and that the water which remained was more tightly bound. Because of this apparent saturation effect it should be possible to extrapolate the attenuation properties to slightly higher temperatures than those used if future laboratory tests were to show that no significant change in density and water content took place at the proposed temperature.

237. K. Shure and P. A. Roys, "Spatial Distribution of Thermal Neutrons from an N-17 Source in Water," Nucl. Sci. Eng. 2, 2, 170-180 (April 1957).

The thermal-neutron flux in a water medium from a source disk containing water irradiated in the Materials Testing Reactor has been measured using a BF_3 detector. Neutrons from N^{17} and photoneutrons from N^{16} gamma rays have been observed. The radial distribution of thermal neutrons from an isotropic point source of N^{17} neutrons in a water medium out to 35 cm has been deduced, and has been compared with a theoretical calculation of the distribution based on Holte's values of the collision density at 1 eV from a point isotropic source of 1-MeV neutrons. From the results of this experiment and Holte's distribution, the square of the diffusion length plus the age from 1 eV to thermal energies, has been calculated to be 8.2 cm^2. This value is in reasonable agreement with previously reported measured values.

238. P. A. Roys, K. Shure, and J. J. Taylor, "Penetration of 6-MeV Gamma Rays in Water," Phys. Rev. 95, 4, 911-912 (August 15, 1954).

The penetration of 6-MeV gamma rays has been studied out to 190 cm in water. The dose rate has been measured with an anthracene scintillation detector as a function of the distance from the N^{16} source. The results agree closely out to 160 cm with the distribution calculated according to the theory of gamma-ray penetration as developed by Spencer and Fano.

239. C. M. Huddleston, Z. G. Burson, R. M. Kinkaid et al., Ground Roughness Effects on the Energy and Angular Distribution of Gamma Radiation from Fallout, CEX-62.81 (May 22, 1963). (Division of Biology and Medicine, Civil Effects Test Operations, AEC.) Not for sale to public.

The effect of ground roughness on the radiation field above ground that has been contaminated with fallout from the explosion of a nuclear device was investigated on three types of Nevada terrain: (1) a flat dry-lake bed, (2) semirough terrain or land that has a known and controlled degree of roughness, and (3) typical wild desert. A modified Navy 5-ton six-wheel-drive truck was used as a mobile laboratory. A collimator and a scintillation crystal with a 512-channel analyzer were used to measure gamma-ray spectra at various polar angles. Dose-vs-height measurements were also made up to a height of 40 ft.

The project was undertaken to determine the importance of ground roughness as a factor in radiation shielding.

240. M. H. Kalos, "On the Estimation of Flux at a Point by Monte Carlo," Nucl. Sci. Eng. 16, 1, 111-117 (May 1963).

In estimating flux at a point in a Monte Carlo calculation one estimator uses the uncollided flux at a detector from each sampled collision point. This method is shown to have infinite variance. The average value converges to the expected value but the error decreases asymptotically as the inverse cube root of the number of histories. By using the once-collided flux and by proper choice of the intermediate collision point the variance may be made finite. Results of numerical experiments show the finite variance methods to be preferable.

241. G. C. Pomraning and M. Clark, Jr., "The Variational Method Applied to the Monoenergetic Boltzmann Equations, Parts I and II," Nucl. Sci. Eng. 16, 2, 147-164 (June 1963).

The variational method as applied to the monoenergetic integro-differential Boltzmann equation is investigated. It is shown that rendering the Lagrangian stationary with respect to small changes in the directional flux and adjoint directional flux is equivalent to solving the Boltzmann and adjoint Boltzmann equations. Topics discussed include the use of variational weight functions, the inclusion of boundary terms in the functional, the interpretation of a variational optimum for a nonself-adjoint operator, and the second variation. It is shown that, for the general trial function ensemble and within a special restricted trial function ensemble, the variational method is a saddle point principle. The formalism developed is applied to the angular expansion in polynomials of the directional flux.

242. M. H. Kalos, "Importance Sampling in Monte Carlo Shielding Calculations. I. Neutron Penetrations Through Thick Hydrogen Slabs," Nucl. Sci. Eng. 16, 2, 227-234 (June 1963).

It is proposed that importance sampling may be carried out using an approximate analytic form for the importance function. An altered Boltzmann kernel may then be computed. Choosing from this altered kernel is facilitated, in part, by particle splitting. The method is carried out in detail for neutron penetration in hydrogen slabs. Results are given for dose buildup and energy spectrum for slab depths up to 179 mean free paths for 8.1-MeV neutrons.

243. F. H. Helm, "Numerical Determination of Flux Perturbation by Foils," Nucl. Sci. Eng., 16, 2, 235-238 (June 1963).

The method for evaluating thermal-neutron flux measurements by foils, which was developed by Ritchie and Eldridge, has proven to be a valuable tool for the experimenter. However, there is some difficulty in obtaining the proper definitions and numerical quantities essential for its application. Especially, there developed a need for an easy way to find the function $g(\gamma,\tau)$. Therefore, a compilation was made of all important definitions, exact and approximate relations, and numerical values pertaining to this problem.

244. S. Depken, On the Solution of the Neutron Transport Equation, AE-89 (December 1962). Availability: Dep. (mc).

The neutron transport equation has occupied the attention of many authors since Placzek, Wick, and others made their first attempts to solve it. Even in the simple case of energy independent cross sections, and disregarding the motion of the scattering nucleons, it is difficult to find a solution in an analytical form which is easily surveyable and fitted for numerical calculations. In Part I of this paper some new viewpoints will be introduced which enable the solution to be presented in its simplest possible form. Part II is devoted to an investigation of some functions introduced in Part I. In Part III the results are applied to the case of large energy lethargy, and the validity of derived formulas is discussed.

245. L. O. Ricks and D. R. Thomas, Development of Shielding for CVTR, CVNA-160 (January 1962). Availability: Dep. (mc); $6.60 (fs), $2.09 (mf) OTS.

This report describes the shielding developed by the Carolinas-Virginia Tube Reactor (CVTR) with special attention to the following two areas of shield design introduced by the use of heavy water as the moderator and coolant in this plant.

1. Development of the reactor top neutron shield as an integral part of the moderator system, complicated by the poor slowing-down power of the heavy water moderator.

2. Development of the paraffin enclosed lead shield cask for the refueling machine to shield against the neutrons generated by the photoneutron reaction in heavy water.

246. D. F. Luse, Shielding Analysis of the Core and Plenum Viewing Plugs - Experimental Gas Cooled Reactor, TID-18104 (October 7, 1960). Availability: Dep. (mc); $3.60 (fs), $1.37 (mf) OTS.

The details of the shielding analysis of this practical problem are shown including many tables of values needed in the calculation.

247. M. B. Wells, Gamma Dose Rates Resulting from Neutron Capture in Air and Concrete, NARF-59-31T (MR-N-238) (September 1959). (Convair, Fort Worth.) Not for sale to public.

This report describes a Monte Carlo method of obtaining the angular distribution of the gamma dose rate resulting from capture of slow and thermal neutrons in air and concrete. The results of some air-capture calculations are compared with analytic solutions of the Boltzmann transport equation and similar Monte Carlo calculations made at the Oak Ridge National Laboratory. The results of a parameter study of the capture dose rate for thermal-neutron line-beam sources are given for source-detector separation distances of 38.81, 64.00, and 98.42 feet, and a source-detector height of 12.5 feet above a concrete surface. The calculations show the significance of the component of the capture gamma dose rate resulting from neutrons that undergo their first n collisions in one medium and then diffuse into the other medium.

248. R. E. Beissner, The Angular Distribution of Gamma Rays Resulting from Neutron Capture in Air, NARF-61-4T (FZK-9-151) (April 14, 1961). Availability: $9.60 OTS as AD-255185.

An analytic procedure is presented for calculating the angular distribution of gamma rays resulting from neutron capture in air. Numerical results are tabulated for four source-detector separation distances, eight angles of emission at the source, and fifteen initial neutron-energy groups. Agreement with Monte Carlo calculations is within the estimated accuracy of the Monte Carlo data.

249. J. R. Stokes and B. S. Burton, Jr., Air Scattering of Gamma Radiation From Co-60, NARF-57-31T (MR-N-156) (July 16, 1957). (Convair, Fort Worth.) Not for sale to public.

Air scattering of gamma radiation from a Co^{60} source has been investigated experimentally. Measurements were made at 40 feet with a collimated gamma spectrometer which allowed various regions of space to be examined for scattered radiation. The energy spectra were obtained to permit separation of the singly and multiply scattered components of radiation. Additional measurements are required to obtain a complete separation of these components.

250. B. S. Burton, Jr., Spectral and Angular Distribution of Air-Scattered Co-60 Gamma Rays, NARF-57-42T (MR-N-168) (September 18, 1957). Availability: $6.30 (ph), $3.00 (mi) LC as PB-139022.

Angular and spectral distributions of air-scattered gamma radiation from an isotropic Co^{60} source have been measured with a total-absorption gamma-ray spectrometer. The observed spectra exhibit structure which may be associated with singly and multiply scattered gamma rays. The upper energy cutoffs of the spectra agree well with those calculated from the theory of Compton scattering.

The measured intensities at 40- and 89-foot source-detector distances are consistent with an inverse-distance law, i.e., the intensity is proportional to $r^{-0.87}$ where r denotes the distance from the source.

251. J. W. Keller and G. Gonzalez, Analysis of Air Scattering of Co-60 Gamma Radiation, NARF-57-61T (MR-N-183) (December 28, 1957). Availability: $4.80 (ph), $2.70 (mi) LC as PB-139021.

The energy distribution of the singly scattered component of air-scattered radiation from a point isotropic source of monoenergetic gamma rays was calculated for comparison with measured spectral data for a Co^{60} source. The calculations were performed for a collimated detector.

Comparison of measured and calculated spectra for collimator orientations of 10°, 90°, and 135° from the source-detector axis indicates that in general the observed spectra cannot be predicted satisfactorily in terms of the singly scattered component only.

252. L. D. Gates, Jr. and C. Eisenhauer, Spectral Distribution of Gamma Rays Propagated in Air, AFSWP 502-A (January 1954). (Armed Forces Special Weapons Project, Washington, D.C.) Not for sale to public.

Theoretical calculations are presented for the spectral distribution of gamma rays in an infinite homogeneous medium emanating from geometrically idealized sources. The sources considered here are isotropic sources uniformly filling all space (approximation to conditions at the center of a large radioactive cloud), isotropic sources uniformly distributed on an infinite plane (approximating conditions due to ground contamination), and an isotropic point source (approximating initial radiation from an atomic detonation). In the latter two cases, distributions are given at various distances from the source or source plane. Graphical material is presented in the form of differential energy spectra for monoenergetic sources of various energies. These spectra give the energy density of scattered radiation per unit energy range of the individual photons. Derived from these spectra are integral energy and integral dose curves, which give the fraction of energy (or dose) carried by gamma rays whose energy is at most a given value. Also presented are curves which can be used to calculate integral dose spectra for sources composed of several energies. The calculations were based on the so-called "method of moments," and the mathematical explanation underlying this method is presented in some detail.

253. M. B. Wells, Air and Concrete Scattering of Gamma Rays, NARF-59-11T (MR-N-229) (March 20, 1959). Availability: $1.25 OTS.

This report describes a Monte Carlo method of computing the angular distribution and energy spectrum of gamma rays scattered in air and ground. The development of the integrals representing the contribution of each order of scattering to the total scattered flux, and the application of random sampling methods to the evaluation of these integrals, are outlined in detail. Calculation of the flux arising from pair-production interactions is also described. Results obtained in a parameter study for a source-detector height of 12.5 feet above a concrete slab are presented in both graphical and tabular form for various source energies and separation distances.

254. T. E. Fessler and M. L. Wohl, Monte Carlo Studies of Gamma-Ray and Neutron Transport in Infinite Homogeneous Media, NASA TN D-850 (November, 1961). Availability: $1.25 OTS.

Two Monte Carlo codes were written for the IBM 704 computer. One simulates gamma-ray transport from a mono-energetic point source in an infinite medium; the second is a similar code for neutron transport.

Statistical variance reduction techniques designed for use in the codes allowed the calculation of more accurate values of gamma-ray and neutron deep-penetration number and energy spectra than would have been possible in a straightforward Monte Carlo calculation. A form of exponential importance sampling was used to emphasize long particle transit distances and small polar scattering angles, thus favoring deeply penetrating particles.

Studies were made of gamma-ray energy spectra and buildup factors for 1-MeV source gamma rays in water and 6-MeV source gamma rays in tin at various penetration depths. These calculations were done both with and without inclusion of the importance sampling mentioned above. Comparisons were made with moments method solutions for the same problems; the agreement was good at shallow penetrations in both cases. In addition, deep penetration results - at 7 and 10 source energy relaxation lengths - obtained by importance sampling also agreed well with moments method solutions. A low-energy spectral peak was observed in the Monte Carlo calculation for the case of 6-MeV gamma rays in tin; this is due to gamma rays from pair-production positron annihilation and is not present in the moments method results, which do not include the effects of pair production.

Calculations of neutron number spectra for 6-MeV source neutrons at various penetration depths in water show good agreement with moments method results. Importance sampling becomes necessary to produce reasonable agreement at penetration distances greater than 30 centimeters. Monte Carlo results were reliable to as deep a penetration as 90 centimeters, or 12 source energy mean free paths.

255. D. F. Luse, Shielding Analysis - Control Rod Plug - Experimental Gas Cooled Reactor, TID-18105 (October 3, 1960). Availability: Dep. (mc); $5.60 (fs), $1.79 (mf) OTS.

The details of the shielding analysis of this practical problem are shown including many tables of values needed in the calculation.

256. J. J. Begley, Plastic Materials in Lightweight Neutron Shields - General Theory and Practice, PA-TR-3009 (August 1962). Availability: $3.60 OTS.

The advent of thermonuclear weapons requires that armor provide radiological as well as ballistic protection. It has been determined that materials with a high hydrogen content are necessary for neutron shielding if the neutrons are above the thermal energy range. Polyolefins and other organic polymers are considered superior for radiological protection not only because of their high hydrogen content, but also because they are light in weight and relatively inexpensive. These characteristics are especially important for mobile units where restrictive limits are placed on weight and dimensional geometry.

This paper provides a quick, simplified review of the neutron shielding process and describes general criteria for determining the relative effectiveness of various shielding materials. Among these criteria are hydrogen density, the presence of neutron-absorbing materials, and secondary gamma emitters. A graph is presented whereby the hydrogen density of a material can be rapidly determined.

Various shielding materials and fabrication techniques are reviewed, with particular attention to the highly promising cast-in-place method. A list of the major resins and other plastic materials and their relative shielding values is presented. Inspection techniques are discussed as a quality control measure to ensure void-free sections.

257. R. L. French, "Some Convenient Representations for Fission-Neutron Distribution in Air," Health Phys. 8, 3, 299-304 (June 1962).

An analysis of Monte Carlo results for fission-neutron scattering in air has revealed that, for slant ranges of from 500 to 2000 yd, the angular distribution (dose per unit solid angle as a function of angle β measured from the source-receiver axis) of the scattered neutron dose from a point isotropic source may be approximated by

$$\frac{dD_s}{d\Omega} = a + be^{-e\beta} .$$

This expression, coupled with an exponential expression for the total scattered dose as a function of range, permits a simple and convenient approach to many dosimetry and shielding problems which require detailed angular and spatial distribution data. The validity of the Monte Carlo data considered in the analysis has been demonstrated by comparison with measured data from weapons tests.

258. L. V. Spencer and J. C. Lamkin, Slant Penetration of Gamma Rays in H_2O, NBS Report 5944 (July 17, 1958). (National Bureau of Standards.) Not for sale to public.

Tabulations are presented of the gamma-ray dose at various distances from monoenergetic plane slant sources in an infinite water medium. The source energies range from 10.22 MeV to 0.34 MeV in an approximately logarithmic sequence. Source obliquities from emission perpendicular to the source plane to emission parallel to the source plane are treated. The calculations were accomplished with the moment method, using an IBM 704 digital computer.

259. L. V. Spencer and J. C. Lamkin, Slant Penetration of Gamma Rays - Mixed Radiation Sources, NBS Report 6322 (February 27, 1959). (National Bureau of Standards.) Not for sale to public.

Tabulations are presented of the gamma-ray dose at various distances from plane slant sources in an infinite water medium. Various mixed gamma-ray sources are considered, including fission products at various times after fission, Cobalt-60, Cesium-137, and the gamma rays emitted by nitrogen after neutron capture. Source obliquities range from $\cos\theta_0 = 1.0$ (perpendicular incidence) to $\cos\theta_0 = 0$ (grazing incidence). These calculations were accomplished by an extension of previous work for monoenergetic sources, reported in NBS Report 5944.

260. D. K. Trubey, Tabulated Values of Scattered Gamma-Ray Fluxes in Water Interpolated from Moments Method Calculations, ORNL-3466 (July 15, 1963). Availability: Dep.; $2.00 OTS.

Tables of scattered gamma-ray fluxes in water that are suitable for integration over source spectra have been generated on the IBM-7090 computer by Lagrangian interpolation of moments method results for point isotropic gamma-ray sources. Values of the fluxes were obtained for a consistent set of scattered gamma-ray energies taken in steps of 0.1 MeV from 0.1 to 1.0 MeV and in steps of 0.25 MeV from 0.25 to 10 MeV. One table was obtained for each scattered energy for various values of the source energy and of the distance from the source in mean free paths. Tables were then generated in which the distance from the source was converted from mean free paths to centimeters. The latter set may be integrated over any source spectrum to obtain the flux of scattered gamma rays in water, and, with small error, in any material in which the cross section is dominated by the Compton scattering cross section. This includes the low-Z materials such as aluminum and air.

261. J. M. Pictet, Bulk Shielding Studies at the Reactor Saphir, EIR-Bericht NR. 27 (July 1962). Available without charge from CERN European Organization for Nuclear Research, Geneva 23, Switzerland.

The results of measurements of the attenuation of neutrons and gamma rays in water, iron, ordinary concrete, and barytes concrete are presented. The experimental setup at the shielding facility at the reactor SAPHIR is described for these measurements.

262. T. R. Strayhorn, Heat Deposition of Reactor Nuclear Radiation in Aerospace-Vehicle Materials, NARF-62-12T (MR-N-289) (September 10, 1962). Availability: $3.60 (fs) OTS.

In order to determine the importance of radiation heating in nuclear-propelled aerospace-vehicle materials, a series of calculations was performed. The temperatures of the materials as a function of reactor power, distance from the reactor, and time were calculated. The results clearly indicate that the radiation heat input to materials in the vehicle must be determined and, where sufficiently great, be considered in the design calculations.

263. R. J. Holl and D. W. Stephen, Pathfinder Atomic Power Plant Shielding Analysis, ACNP-62016 (July 15, 1962). Availability: Dep. (mc), $10.50 (fs), $4.13 (mf) OTS.

Shielding for the Pathfinder Atomic Power Plant, which incorporates a direct-cycle controlled-recirculation boiling-water reactor with an integral nuclear superheater, has been designed. This report includes the shield design, dose levels for all areas of the plant, and methods of calculation.

Major sources are the reactor complex in the reactor building, the main steam line, turbine, condenser, and feedwater system in the turbine building, and the waste disposal system and spent fuel in the fuel handling building. Biological shield for the reactor consists of concrete with earth fill on one side and 18 ft of shielding water over the reactor. The turbine foundation, condenser water box, the operating floor, and concrete walls serve as shielding for the unrestricted areas of the turbine building. A pool is provided for shielding spent fuel in the fuel handling building, with additional shielding provided for components of the waste disposal and reactor-water purfication system.

264. P. S. Brown, T. J. Thompson, I. Kaplan, and A. E. Profio, Measurements of the Spatial and Energy Distribution of Thermal Neutrons in Uranium, Heavy Water Lattices, MITNE-17 (August 20, 1962). Availability: Dep. (mc), $15.00 (fs), $6.92 OTS.

Intracell activity distributions were measured in three natural uranium, heavy water lattices of 1.010 inch diameter, aluminum clad rods on triangular spacings of 4.5 inches, 5.0 inches, and 5.75 inches, respectively, and in a uranium, heavy water lattice of 0.25 inch diameter, 1.03% U^{235}, aluminum-clad rods on a triangular spacing of 1.25 inches. The distributions were measured with bare and cadmium-covered foils of gold, lutetium, and europium. The gold was used as a 1/v absorber to measure the thermal-neutron density distribution. Because the activation cross sections of lutetium and europium depart considerably from 1/v behavior, their activation depends strongly on the thermal-neutron energy spectrum. Hence, they were used to make integral measurements of the change in the neutron energy spectrum with position in the lattice cell. A method was developed for treating the partial absorption, by cadmium covers, of neutrons at the 0.46 eV europium resonance, and it was found possible to correct the europium activations to energy cutoffs just above and just below the resonance.

The measured activity distributions were compared with those computed with the THERMOS code. In the natural uranium lattices, THERMOS gave excellent agreement with the measured gold activity distributions and very good agreement with the lutetium and europium distributions, indicating that THERMOS gives a very good estimate of the spatial and energy distribution of thermal neutrons in these lattices. In the enriched lattice, THERMOS gave a large overestimate of the activity dip in the fuel for all three detectors. The discrepancy was attributed to a breakdown in the Wigner-Seitz cylindrical cell approximation at small cell radii. However, the measured ratios of lutetium and europium activity to gold activity were in good agreement with the THERMOS values, indicating that THERMOS still gave a good estimate of the degree of spectral hardening.

Neutron temperature calculations were made from the data by using Westcott effective cross sections. The temperature changes so calculated agreed well with those predicted by THERMOS.

Disadvantage factors calculated by the Amouyal-Benoist-Horowitz (ABH) method were in excellent agreement with the measured values in the natural uranium lattices. The agreement was not as good in the enriched lattice.

265. R. Nilsson and J. Braun, Absorption and Flux Density Measurements in an Iron Plug in R-1, AE-13 (November 1958). Availability: Dep. (mc); Sw. Cr. 4, AB ATOMENEGI, Stockholm, Sweden.

Thermal, epithermal and fast-neutron fluxes have been measured in a 60-cm long, "sliced" iron plug, which has been placed in the lower iron lid of the Swedish reactor R 1. Au foils, Cu foils, Mn foils, P packets, Cu wires and small Fe cylinders have been used. The gamma flux has been determined with film dosimeters.

The measurements have shown that only in the first centimeters of the iron is the activation determined by the thermal flux, which decreases with a relaxation length $\lambda = (1.51 \pm 0.02)$ cm. The epithermal flux is entirely predominant already after 10 cm ($\lambda = 16$ cm). The epithermal neutron flux decreases even more slowly than the fast flux ($\lambda = 6.2$ cm).

266. H. Aceto, Jr., M. A. Pick, and L. D. Stephens, Neutron Tissue Dose at Large Distances from an Elevated Unshielded Reactor, UCRL-10559 (March 11, 1963). Availability: Dep.; $0.50 OTS.

The neutron tissue dose at large distances from a fission source was studied by using a water-filled phantom and four different detectors: a BF_3 counter, a polyethylene-lined ethylene-filled proportional counter, indium foils, and nuclear emulsions. The source of fission neutrons was the ORNL Health Physics Research Reactor which was attached to a hoist which was in turn installed on a 1530-foot tower. The reactor could be operated at any elevation from 27 to 1500 feet. The phantom studies were made at horizontal distances from 250 to 1500 yards from the tower. Dose contributions from recoil protons, $H^1(n,\gamma)D^2$ and $N^{14}(n,p)C^{14}$ reactions are considered.

267. S. Block and F. J. Shon, "Neutron Dose Measurements by an Attenuation Technique," Health Phys. 8, 5, 533-541 (October 1962).

A technique has been developed for finding the RBE dose rate and average energy of stray neutron radiation. Neutron energies can be measured by applying the fact that the spatial variation of a thermal flux formed in a moderating material is a function of the fast flux energy distribution. An attenuation curve is plotted using polyethylene disks as neutron absorbers and a BF_3 counter as the detector. Several curves are presented, including relaxation length vs. energy, relaxation length vs. dose rate per unit neutron flux, and the calibration curve of relaxation length vs. dose rate per count per min. The system is compared to the double moderator dosimeter of DePangher.

268. R. Fraki, M. Leimdorfer, and S. Maimskog, The Energy Variation of the Sensitivity of a Polyethylene Moderated BF_3 Proportional Counter, AE-91 (November 1962). Availability: Dep. (mc).

The variation with neutron energy of the sensitivity of a polyethylene moderated boron counter has been investigated experimentally at 2.3 MeV, 550 keV, 210 KeV, 110 keV, 340 eV, and 5 eV, and theoretically by the multigroup diffusion method in the same range. Different moderator thicknesses up to 10 cm were considered.

Results show good agreement between experimental and theoretical sensitivities for the keV and MeV energies, while a discrepancy of the order of a factor 2 at the most is obtained in the eV region.

269. L. V. Spencer and J. C. Lamkin, Slant Penetration of Gamma Rays in Concrete, NBS Report 6591 (November 10, 1959). (National Bureau of Standards.) Not for sale to public.

Tabulations are presented of the gamma-ray dose at various distances from monoenergetic plane slant sources in an infinite concrete medium. The source energies range from 10.22 MeV to 0.043 MeV in an approximately logarithmic sequence. Source obliquities from emission perpendicular to the source plane to emission parallel to the source plane are treated. The calculations were accomplished with the moment method using an IBM 704 digital computer, as described in the similar NBS Report 5944 for water.

270. R. L. Tomlinson, SNAP Shield Test Experiment - Fission Plate Power Calibration, NAA-SR-7432 (August 15, 1962). Availability: Dep.; $0.75 OTS.

The SNAP Shield Test Experiment (STE) was constructed for the radiation evaluation of shield materials and geometries for the SNAP reactor development program. A detailed description of the facility is given in other reports; also the physics tests that were performed on the reactor system are presented.

The facility consists of: a 50-kw water-cooled zirconium-hydride uranium reactor; a neutron thermal column; a fission-source plate; and associated equipment. The uranium fission-source plate is normally located on a wall of the underground shield test room where all the shield measurements are performed. It is driven by the reactor through a graphite thermal column. A shield test carriage is used in conjunction with the fission-source plate to support the shields while undergoing radiation evaluation. Both the fission-source plate and the carriage can be remotely positioned with the aid of a closed-circuit TV system. The fission plate can be driven into a lead shielding cask and the test shields can be retracted from the thermal column thereby allowing operating personnel to assemble the shield experiments in a low-radiation background. Since the fission-source plate serves as the radiation source for many shielding and radiation measurements, an accurate determination of its fission rate is required in order for accurate radiation measurements to be made. This report describes the techniques that were employed in obtaining these calibration measurements.

271. R. L. Tomlinson, R. P. Johnson, and S. G. Wogulis, SNAP Shield Test Experiment - Reactor Physics Tests, NAA-SR-7368 (July 12, 1962). Availability: $0.75 OTS.

The initial physics tests on the Shield Test Experiment reactor and the precriticality rod-drop test data are presented herein.

272. REIC, The Effect of Nuclear Radiation on Transducers, REIC Report No. 25 (April 19, 1963). (Battelle Memorial Institute, Radiation Effects Information Center.) Not for sale to public.

This report presents information relative to the state of the art on the effects of nuclear radiation on various types of transducers that are becoming extremely important in present-day applications. The report consists of two sections. The first section contains information dealing with concepts and characteristics of conventional transducers with estimation of values of radiation thresholds. The approach used in the estimation involves contact with users and manufacturers for their opinions and knowledge on the subject, use of known experimental data for materials used in the fabrication of transducers, and some empirical data from experiments with similar units. Information is also included on research and development programs currently in progress or completed. The section portion of the report deals with information from reports available at the Radiation Effects Information Center (REIC) on irradiation of various transducers and sensing elements. The information presented can serve a useful purpose to designers and electronic-component users as a guide for developing a better understanding of the problems that must be solved prior to final selection of transducers to be used in nuclear environments.

273. M. M. Wasson, The Fast-Neutron Dose Response of Two Organic Phosphors and Their Possible Use for Dosimetry, AERE-R 4244 (April 1963). Availability: Dep.; $0.77 (BIS), 5s 6D. (HMSO).

Using the results of Monte Carlo calculations on the response of two organic phosphors to fast neutrons the report examines the possibility of using them for fast-neutron dosimetry. It also examines experimentally the effect, on their response of decay time discrimination against gamma rays.

274. P. K. Lee, F. M. Cox, and W. H. Schweitzer, Test Cell A - Shielding Study - KIWI B-4A, LAMS-2904 (June 26, 1963). Availability: Dep.; $0.75 OTS.

Gamma doses produced by the Kiwi B-4A reactor test were measured at 49 stations in the Test Cell A area. Most of the stations were located in permanent structures or at locations at which structures served as shadow shields. The study was conducted to determine the reduction in gamma doses caused by the shielding structures.

It was found that the gamma doses in the test cell were reduced by factors ranging from 3×10^{-4} to 3×10^{-3}, and the dose reductions for the J-8 Bunker and the test cell access tunnel were of the order of 2×10^{-3} and 2×10^{-4}, respectively. Doses were reduced at positions in which the test cell served as shadow shield by factors ranging from 10^{-1} to 5×10^{-2}.

275. V. A. Dulin, Yu. A. Kazanskii, V. P. Mashkovich et al., "Investigation of Attenuation Functions in Water for Neutrons from Isotropic and One-Directional Fission Sources," Soviet J. At. Energy (English Transl.) 9, 4, 858-861 (September 1961).

The spatial distributions of neutrons from a BR-5 reactor were measured and the results are presented. The source of neutrons was collimated through a concrete channel.

276. L. N. Zaitsev, M. M. Komochkov, and B. S. Sychev, "High Energy Neutron Attenuation in Concrete," Soviet J. At. Energy (English Transl.) 12, 6, 558-561 (January 1963).

The attenuation of neutrons of energies greater than 20 MeV was measured in concrete with densities of 2.35, 3.2, and 4.1 g/cm^3. The neutrons were produced by protons from a synchrocyclotron impinging on beryllium. The concrete density was increased by using iron ore and steel. The attenuation curves are shown.

277. E. E. Kovalev and D. P. Osanov, "Radiation from a Volume-Distributed Gas-Filled Source Placed Behind Plane Shielding," Soviet J. At. Energy (English Transl.) 13, 1, 657-658 (March 1963).

A formula is given for the gamma-ray dose rate arising from a cylindrical source consisting of a radioactive gas shielded by a plane barrier at the base.

278. D. P. Osanov and E. E. Kovalev, "Dose Adsorption Factor for a Cylindrical Source in the Presence of Shielding," Soviet J. At. Energy (English Transl.) 12, 6, 561-562 (January 1963).

A simple formula is given for predicting the dose rate from a gamma-ray source on the surface of an absorbing cylinder and penetrating a planar shield.

279. S. G. Tsypin, "The Use of Monodirectional Neutron Sources for the Investigation of Shielding (B-2 Apparatus in a BR-5 Reactor)," Soviet J. At. Energy (English Transl.) 12, 4, 318-323 (September 1962).

The report describes the B-2 apparatus, installed in a BR-5 fast reactor, for investigating the passage of neutrons through various shielding materials. It is shown that the monodirectional neutron disc source used in this apparatus makes it possible to obtain detailed information on the spatial-energy and angular distributions of the neutrons in the shielding. The effect of the angular distribution of the radiation leaving the source on the attenuation factor of this radiation in shielding was also investigated.

280. V. A. Dulin, V. P. Mashkovich, E. A. Panov et al., "Energy Distribution in Water of Fast Neutrons," Soviet J. At. Energy (English Transl.) 9, 4, 861-863 (September 1961).

The energy spectra of neutrons in water was measured and the results are presented. The radiation source was a BR-5 reactor. The distances in water were 30 and 60 cm. The results are compared with moments method calculations of Aronson (1954).

281. Yu. I. Bregadze, B. M. Isaev, and V. A. Kvasov, "Determination of Energy Absorption in a Mixed Flux of Fast Neutrons and Gamma Rays by an Ionization Method," Soviet J. At. Energy (English Transl.) 9, 2, 630-636 (August 1961).

The possibility of separate determinations of the energy absorption of fast neutrons and gamma rays in the mixed radiation flux from a reactor has been studied with ionization chambers. Three chambers with different hydrogenous fillers were used: polyethylene with an ethylene filler; graphite with a CO_2 filler and a chamber made from aerion, a conducting plastic, which was filled with a mixture of ethylene and CO_2. Calculations have been carried out to ascertain the sensitivity of these chambers to neutrons with energies ranging from 0.2-8 MeV. Variation of the neutron spectrum over wide limits has no effect on the accuracy in the determination of the absorbed dose in the hydrogenous substrates. A calculation shows that the error in the determination of the absorbed energy for fast neutrons is approximately 15% and is a weak function of the relative doses of neutrons and gamma rays.

282. G. V. Gorshkov and V. M. Kodyukov, "Attenuation of Gamma Radiation from Volume Sources in Iron and Lead," Soviet J. At. Energy (English Transl.) 9, 2, 649-650 (August 1961).

The results of measurements of the attenuation of gamma rays in lead and iron are presented. The volume source had a shape of a truncated cone and the sources used were colloidial Au^{198} and Na^{24} in a salt solution.

283. V. M. Kodyukov, "Attenuation of Gamma Radiation from Point Sources in Various Media," Soviet J. At. Energy (English Transl.) 9, 2, 651-652 (August 1961).

The results of a measurement of the attenuation of gamma rays through slabs of water, iron, and lead are presented. The sources were small radionuclide sources -- Au^{198}, Zn^{65}, and Na^{24}. The results were compared with Moments Method calculations of Fano (1953).

284. A. T. Bakov, S. P. Velov, Yu. A. Kazanskii et al., "Gamma-Ray Yield from Radiative Capture in Iron," Soviet J. At. Energy (English Transl.) 13, 1, 619-624 (March 1963).

A technique is described for measuring the yield of capture gamma radiation from iron by the use of a single crystal scintillation gamma spectrometer. Results are reported for the yield of capture gamma radiation from iron, normalized to the number of neutrons either entering shielding or escaping from it. Measurements were made for the neutron spectrum from a Po-Be source. The dependence of capture gamma radiation yield on shielding thickness is given. The effect of boron-containing impurities on the yield of capture gamma radiation was investigated.

285. A. V. Larichev, D. P. Osanov, and V. I. Popov, "Spectrum of Gamma Radiation from a Homogeneous Cylindrical Source," Soviet J. At. Energy (English Transl.) 13, 2, 744-749 (March 1963).

Results are given for an experimental investigation of the gamma radiation spectrum emitted from large cylindrical sources filled with cobalt sulfate solution containing Co^{60}. The changes in the spectra after passage through various shielding layers were also studied. The results of this work were compared with the data of others. Measurements were made with a scintillation spectrometer having a NaI(Tℓ) crystal.

286. V. I. Kukhtevich, Yu. A. Kazanskii et al., "The Passage of Gamma Rays Through Water," J. Nucl. Energy (English Transl.) 9, 128-134 (1959).

The dosage attenuation in water of gamma rays from ^{198}Au, ^{60}Co, and ^{24}Na has been measured as a function of the source-detector distance over a range from some 3-4 to about 8-12 scattering mean free paths. The primary radiation was collimated in such a direction that none fell on the detector, down to an angle which was varied between 30° and 80°. Theoretical intensities have been calculated by an approximate procedure that replaces the direct solution of the kinetic equation by the reduction of a triple integral and are in satisfactory agreement with observation.

287. Yu. V. Sivintsev, "Ratio Between the Radiation Dose and the Absorbed Dose," Soviet J. At. Energy (English Transl.) 9, 7, 534-543 (July 1961).

The article indicates the difference between the concepts "radiation dose" and "absorbed dose;" specific formulas are given for calculating an absorbed dose from the results of absolute measurements of a radiation dose, and the conditions of electron equilibrium during dosimetric measurements of x-rays and gamma radiations of various energies within the 200 keV - 32 MeV range are stated.

288. D. E. Hankins, New Methods of Neutron Dose Rate Evaluation, LADC-5427 (1962). Availability: Dep. (mc); $2.60 (fs), $0.89 (mf) OTS.

A multisphere unit, utilizing cadmium-covered polyethylene spheres of 2, 3, and 8 in. diameters, and a single 10-in. sphere of polyethylene placed over a LiI thermal-neutron detector have been developed for evaluating neutron dose rates, and simplified spectra including neutrons in the intermediate energy region. A comparison of the total dose rate determined by both techniques and the Hurst type absolute fast-neutron detector is given for the various types of sources and for reactor leakage neutrons.

289. M. J. Schumchyk and H. J. Tiller, Ground Penetration Radiation (Lip Contribution) in a Foxhole from a Fallout Field Simulated By Cobalt-60, NDL-TR-3 (December 1960). (Army Chemical Corps Nuclear Defense Lab, Army Chemical Center, Maryland.) Not for sale to public.

The lip contribution to the total radiation in a cylindrical foxhole (4 ft in diameter and 4 ft deep) from a fallout field simulated by Co^{60} has been measured and calculated. The lip contribution averaged 68% of the total dose rate received at different depth levels in the hole. Of the ground penetrating radiation 99% stems from a 2-ft-wide annulus around the foxhole, the inner circle of which is the rim of the hole.

290. M. A. Schmoke and R. E. Rexroad, Attenuation of Simulated Fallout Radiation by the Roof of a Concrete Blockhouse, NDL-TR-6 (August 1961). Availability: $7.60 OTS as AD-272871.

An experiment was carried out to verify theoretical calculations of the effect of roof thickness on the radiation level inside a concrete blockhouse from a plane of gamma radiation on the roof.

291. F. C. V. Worman, P. K. Lee, and R. W. Henderson, Integral Neutron and Gamma Dose Measurements on the KIWI B-1 Reactors. Part I. KIWI B-1A. Part II. KIWI B-1B, LA-2858 (April 19, 1963). Availability: Dep.; $2.00 OTS.

This report is an evaluation of the direct radiations from the Kiwi B-1 reactors. Included in this report are integral measurements of gamma dose, neutron flux and neutron dose, made on Experimental Plans 4, 5, 5A, and 6 + 6A of Kiwi B-1A, and Experimental Plans 2, 3, and 4 of Kiwi B-1B. Results were obtained from almost every instrumented station.

Gamma doses, in air, on Experimental Plan 6 + 6A ranged from 1.80×10^4 rads at 62 feet to < 10 rads at 1,500 feet. Neutron doses, in air, varied from 5.51×10^5 rads on the privy roof to 2.30 rads at 1,500 feet. Thermal-neutron fluxes dropped from 165 percent of the total fast-neutron flux, at close proximity to the reactor, to 80 percent at 300 feet and 60 percent at 600 feet.

Documentation of the radiation field, in close proximity to the Kiwi B-1A reactor, was accomplished for several studies including an experiment in cooperation with Brookhaven National Laboratory, a meridian ring experiment, and a determination of the shielding qualities of the privy roof.

For Experimental Plan 4, Kiwi B-1B, gamma doses, in air, ranged from 4.85×10^4 rads at 62 feet to 7.50×10^1 rads at 900 feet. Fast-neutron doses, in air, ranged from 1.37×10^4 at 62 feet to 1.32×10^2 at 600 feet. Thermal-neutron fluxes dropped from 192 percent of the total fast-neutron flux at 62 feet to 49 percent at 600 feet.

292. J. J. Schmidt, Neutron Cross Sections for Fast Reactor Materials, KFK 120 (EANDC-E-35 U) (December 1962). Availability: Dep. (mc).

Neutron cross sections and neutron nuclear data tabulated as functions of the neutron incident energy and of the corresponding lethargy in the energy range of 0.01 to 10 MeV are given for H, He, C, O, Na, Cr, Fe, Ni, Mo, U^{235}, U^{238}, and Pu^{239}. Data on fission are presented along with cross sections for neutron scattering, reactions, capture, etc.

293. J. A. Grundl, Study of Fission Neutron Spectra with High-Energy Activation Detector, LAMS-2883 (May 20, 1963). Availability: Dep.; $2.75 OTS.

The energy spectra of neutrons from the fission of U^{235}, U^{233}, and Pu^{239} have been studied with seven high-energy activation detectors covering the range 0.6 to 16 MeV. Interpretation is in terms of the Maxwellian spectral function $E^{1/2} \exp(-E/T)$. By means of a unique cavity arrangement within a moderator to obtain pure thermal-neutron-induced fission neutrons, the average energies of the three spectra are found to be in the ratios

$$U^{235}: U^{233}: PU^{239} = (1): (1.016 \pm 0.003): (1.039 \pm 0.003).$$

Differences between the normalized spectra are most pronounced at high energies as exemplified by the relative $Pu^{239}:U^{235}$ flux ratios 1.17 ± 0.013 for $6 < E < 11$ MeV, and 1.35 ± 0.03 for $E > 11$ MeV.

Comparable detector responses within unreflected critical assemblies of U^{235}, U^{233}, and Pu^{239} metal are consistent with an increase of at least 2% in the average energy of the fission spectrum component of each assembly relative to the corresponding thermal-fission spectrum. This may be associated with the approximately 1.5 MeV average energy of the assembly neutrons inducing fission.

294. REIC, Radiation Effects State of the Art 1962-1963, REIC Report No. 28 (June 30, 1963). Available from the Radiation Effects Information Center.

No abstract needed.

295. K. Shure, The $O^{16}(n,p)N^{16}$ Reaction Cross Section, WAPD-BT-25, 27-35 (May 1962). Availability: Dep.; $8.10 (fs), $2.87 (mf) OTS.

A review has been made of the available information concerning the cross section of the $O^{16}(n,p)N^{16}$ reaction with regard to application to water-cooled reactors. Recent measurements of the excitation function for this reaction, with good precision, have covered a wider energy range than heretofore. This excitation function, when averaged over the fission neutron spectrum, agrees to within experimental uncertainties with the result of an integral activation measurement of 21 ± 4 μb for N^{16} production with a yield of 0.74 high-energy gamma rays per disintegration.

296. K. Shure, Experimental Verification of Neutron Attenuation Kernels, WAPD-BT-17, 19-22 (February 1960). Availability: Dep.; $8.10 (fs) $2.87 (mf) OTS.

Eleven papers are presented on developments in reactor physics and mathematics at Bettis Laboratory. Topics covered include resonance escape probability, fast-neutron conversion factors, neutron attenuation kernels, three-dimensional flux distributions, xenon transients, homogeneous blankets, surface to volume activation by resonance neutrons, multigroup cross sections, Boltzmann equation, slowing down of neutrons, and transient response of spatial flux distribution to changes in materials properties. Separate abstracts were prepared for the eleven papers.

297. D. C. Anderson, Calculation of Buildup Factors for Infinite Plane and Line Sources, WAPD-BT-8, 19-21 (June 1958). Availability: Dep.; $24.30 (ph), $7.50 (mf) OTS.

Although in a typical shield design the radiation sources are finite in extent, available tabulated buildup data are provided only for ideal source geometries such as point isotropic and plane monodirectional. This article provides tabulations of the buildup factors in infinite homogeneous media for infinite isotropic plane and line sources. Ready information is made available to the designer as to the effect of source geometry on the magnitude of the buildup factor.

298. D. C. Anderson, Comparison of Peebles Buildup Data to Experiment and Moments Method Results, WAPD-BT-8, 22-25 (June 1958). Availability: Dep.; $24.30 (ph), $7.50 (mf) OTS.

Estimates of slant penetration effects are of importance in a detailed shield design. This article examines the application of Peebles' data in obtaining slant penetration estimates and compares the results with experimental data. Results indicate that the use of Peebles' calculations is preferable for problems in which slant penetration for angles greater than 45° becomes important.

299. K. Shure, Fission Product Decay Energy, WAPD-BT-24, 1-17 (December 1961). Availability: Dep.; $2.25 OTS.

Knowledge of the energy released from fission products in the form of beta and gamma radiation is necessary in such aspects of reactor design as the calculation of heat generation and refueling shielding requirements. Experimental data has been accumulated which is applicable over different ranges of times after fission, and additional studies based on these data have been made. The information currently available is reviewed, and what is considered to be most reliable is presented in a manner suited for calculation.

300. D. C. Anderson, L. O. Herwig, and W. F. Vogelsang, Comparison of Measured and Calculated Thermal Neutron Distributions in High-Temperature Water Shielding, WAPD-BT-24, 19-25 (December 1961). Availability: Dep.; $2.25 OTS.

Measurements made at 460 F of the thermal-neutron flux distributions in the water shielding surrounding a nuclear reactor core are compared with calculations based on a previously derived analytical kernel. This comparison verifies the applicability of the temperature-dependent kernel in shielding calculations. Additional data at 68 F are also presented.

301. J. M. Ferguson, Ground Roughness Effects for Fallout-Contaminated Terrain Comparison of Measurements and Calculations, USNRNL-TR-645 (May 7, 1963). Availability: $2.60 OTS as AD-410413.

The effect of ground roughness on the radiation field above fallout-contaminated ground is studied. At past weapons tests, the dose rate over fallout-contaminated ground has been measured as a function of height and angle. These measurements are compared with calculations of the same quantities for 1.12-hr fission products uniformly distributed on a smooth plane. None of the experiments is detailed enough to lead to firm conclusions about the ground roughness effect. However, the data indicate that the ground roughness effect can be simulated by assuming that the fallout is buried under a thin layer of material. For desert terrain this thickness of material is equivalent to about 25 $\pm$ 10 ft of air. At 3 ft above the ground this corresponds to a reduction in dose rate by a factor of 0.6 to 0.7, compared to what would be received over a smooth plane.

302. J. Certain, A Solution of the Neutron Transport Equation, NYO-3081 (July 25, 1954). Availability: Dep. (mc).

The neutron transport equation for plane sources is given and transformed by the moments method. In Part I, two methods are described for the treatment of the scattering term by expanding it in powers of M^{-1}, where M is the atomic mass of the scatterer. Applications are made to plane and point isotropic source and to a uniform plane source emitting neutrons in a fixed direction with the normal (the plane conical source). A description is given of the calculations for the moments of the scattered flux (in the case of monoenergetic δ-function sources) and the moments of the total flux (in the case of continuous energy sources) as carried out on SEAC (Standards Eastern Automatic Computer).

303. W. D. Lanning, A Fortran Program to Solve the P-3 Gamma-Ray Equations in Slab Geometry, WAPD-TM-356 (April 1963). Availability: Dep.; $1.00 OTS.

A program has been developed to solve the P-3 gamma-ray equations in slab geometry. The formulation which is utilized to compute the spatial distribution of the gamma energy flux is described. The program was written specifically as a shield design tool for performing gamma energy deposition calculations and for other potential design applications.

304. M. Grotenhuis, Lecture Notes on Reactor Shielding, ANL-6000 (March 1959). Availability: Dep.; $3.00 OTS.

These lectures on Reactor Shielding include the basic information necessary to enable one to compute neutron and gamma-ray flux distributions in an unperturbed shield. The object of the series is to impart the information necessary to design shielding for stationary reactors by hand methods. The lectures have been gradually developed over a period of four years at the International School of Nuclear Science and Engineering at Argonne National Laboratory for the Argonne employees, Two-Year Appointees, as well as the International participants. Prior to that time these notes existed in shorter form as the basis of a two to three-hour lecture series for Laboratory employees and for specific Laboratory training commitments. The lectures now constitute a one-semester hour course, but could be extended to a two-semester hour course by additional emphasis on the problems involving coolant activity, heat generation, voids, irregularities, etc., as they arise in later design and construction stages. Prerequisites for this course are Physics through Applied Nuclear Physics and Mathematics through Mathematics for Engineers and Physicists.

305. K. Shure and G. L. Strobel, Effective Neutron Capture Gamma-Ray Spectra, WAPD-BT-22, 19-30 (March 1961). Availability: Dep.; $2.00 OTS.

The fairly complex energy distribution of the gamma rays arising from the capture of neutrons in several materials has been reduced to simple spectra consisting of two (and in one case, three) energy groups. These results will significantly reduce the computational effort entailed over that required for the more complex spectral descriptions while retaining the adequacy of the information generated. These spectra are sufficiently accurate to be applied in practical problems for the prediction of biological dose rates and of heat generation rates in structural materials.

306. B. I. Sinitsyn and S. G. Tsypin, "The Use of Empirical Constants in Shielding Design," Soviet J. At. Energy (English Transl.) 12, 4, 324-332 (September 1962).

The article contains a generalization of the results of experimental and theoretical investigations relating to the determination and use of removal cross sections. It also considers the questions of the possibility of extending the region of applicability of empirical constants for the design of various kinds of neutron shielding. A table gives the values of empirical constants for calculating the penetration of fast neutrons.

307. J. M. Holmes, H. Pfohl, and E. D. Arnold, Shielding Calculations - POT Calculation Demonstration at Hanford, ORNL-TM-595 (June 21, 1963). Availability: Dep. (mc); $1.60 (fs), $0.80 (mf) OTS.

The results of shielding calculations are presented for the Pot Calcination Demonstration at Hanford. Stored pots containing Consolidated Edison waste cooled 180 days produced the highest dose rate of 18 mrem/hr through a 64-in. concrete shield. Purex waste (180-day cooled) in 8-in.-diameter pots gave 4.2 mrem/hr through the same shield. Purex waste source activities and fission product heating data are also presented.

308. D. H. Day, W. N. Fox, and H. R. Hyder, Measurement and Calculation of Fast-Neutron Flux in a Zero-Energy Reactor, AEEW-R 85 (May 1963). Availability: Dep.; $0.63 (BIS); 4s. 6d. (HMSO).

An activation technique for measuring relative fast-neutron fluxes is described which has some advantages over the normal method using U^{238} fission. The technique is based on the formation of Rh^{103} after inelastic scattering of neutrons above 100 keV in energy. This isomer decays with a 57.4 minute half-life giving an easily measurable gamma activity. The energy dependence of the inelastic scattering cross section of Rh^{103} is similar to that of the fission cross section of U^{238} thus making the results of direct relevance to reactor calculations.

Using the Rh^{103} activation technique, measurements have been made of the fast-neutron flux distribution in a typical pressure tube heavy water lattice and are compared in this report with theoretical calculations using the Monte Carlo method.

309. D. A. Holme and K. Stewart, The Gamma Dose Rate Above an Infinite Plane Source, AWRE E 6/63 (May 1963). Availability: Dep. (mc).

The calculations summarised in this report give the relationships between the exposure radiation dose rate and the energy of the emitted gamma radiation at selected heights above a plane area of contamination. The dose rates above finite plane circular and plume-type sources are also examined.

310. J. N. Anno, Jr., "Fission Plate Power Measurement by a Transient-Temperature Method," Nucl. Sci. Eng. 16, 4, 357-362 (August 1963).

Transient-temperature behavior following a step change in internal heat generation has been analyzed to determine the power generation in the Batelle Shielding Facility fission plate. The fission plate is employed for shielding studies as a radiation source with a fission energy distribution. The plate is a 28-in. diam, 0.0199-in. thick uranium disk containing 3741 g of uranium enriched to 93.14% in U^{235} isotope. It is plated with 0.0007 in. of nickel and clad with 0.025 in. of aluminum on each side and is in intimate contact with a 0.25-in. thick aluminum plate on one side. Ceramic spacers provide airgap insulation of the fission-aluminum plate combination from the surrounding media.

Resistance thermometers were employed to observe the transient-temperature behavior following a step change in the internal heat generation in the plate for fission heating and for cooling tests. The cooling curve data were strictly exponential and rendered a decay constant of 0.0517 min^{-1} which was utilized, along with the physical constants of the assembly, to render a solution to the transient-heating equation and an estimated power of 25.0 $\pm$ 0.6 watts.

311. W. R. Morgan, H. M. Epstein, J. N. Anno, Jr., et al., Shielding-Research Area at Battelle, BMI-1291 (September 18, 1958). Availability: Dep.; $2.60 (fs), $1.10 (mf) OTS.

The design and construction of the shielding facility at Battelle is described in this report. This facility consists of an open pool with a fission plate, an instrument bridge and tower, a control room, and radiation-detection instruments. The shielding pool is located at the end of the thermal column of the Battelle Research Reactor (BRR).

The fission plate is 28 in. in diameter and contains approximately 3.5 kg of U^{235}. The plate was fabricated from three pieces of highly enriched uranium and clad with about 25 mils of 2S aluminum. It generates about 24 w during steady-state reactor operation.

The fission spectra of neutrons and gamma rays produced by the fission plate are free from appreciable background radiations. The ratio of thermal to epithermal neutrons impinging upon the fission plate is approximately 67, indicating a low fast-neutron background. Assuming an average energy of 2 MeV for background gamma rays results in a ratio of thermal-neutron flux to gamma flux of 16.

312. I. O. Andersson and J. Braun, A Neutron Rem Counter with Uniform Sensitivity from .025 eV to 10 MeV, RFA-100 (SSI-14) (November 26, 1962). Not available generally to public.

A neutron monitor has been developed for measurement of the neutron dose rate in rem per hour independently of the energy of the neutrons from 0.025 eV to 10 MeV. The detector consists of a cylindrical BF_3 proportional counter surrounded by two polyethylene layers separated by a 5-mm thick boron plastic layer. This screen provides the appropriate moderation and attenuation to the incident neutrons to give a correct dose response. A simplified theory utilizing a multigroup diffusion program is used to obtain the response in the region between thermal and 35 keV energies where experimental evidence is insufficient. A counter suitable for use in a portable radiation protection survey instrument has been manufactured. The sensitivity is 5 cps per 2.5 mrem/hr and there is no sensitivity to gamma radiation dose rates up to 50 r/hr.

313. J. J. Thompson, Theory and Application of the Double PN Method in Slab Geometry for Isotropic Neutron Sources and Scattering, AAEC/E107 (April 1963). Availability: Dep. (mc).

A simple matrix formalism is developed to facilitate the application of the double P_N method of spherical harmonics to multiple slab configurations, representative of some of the basic neutron transport problems in the theory of nuclear reactors. Special attention is given to the double P_2 and P_3 approximations, and the results of numerical computations are compared with known accurate results.

314. W. M. Veroeven and J. W. McGarvey, Castable Materials for Neutron Shields (Part 2), Rock Island Arsenal Laboratory Technical Report No. 62-654 (February 20, 1962). Not for sale to public.

This is the second report concerned with the development of elastomeric materials for neutron shields. Numerous liquid natural, butyl and polybutadiene rubbers, together with a specially synthesized boron modified, acrylic acid terpolymer were evaluated to determine their curing characteristics and properties pertaining to neutron shields. It was determined that a polybutene extended butyl rubber filled with polyethylene and vulcanized using a quinoid type cure offers the most promise as a room temperature pumpable and curable, high hydrogen density neutron shield matrix.

315. T. E. Northrup, High Density Slag Concrete, ORNL-3473 (July 26, 1963). Availability: Dep.; $2.00 OTS.

Test results are presented that show that a strong concrete weighing approximately 185 lb/ft^3 can be made using water, waste lead slag, and Ciment Fondu. Feasibility, materials, mortar tests, concrete tests, and Ciment Fondu concretes are discussed. A 24-in.-thick concrete shield wall would have to be increased in thickness by 5 in. if slag concrete is used in place of barytes or magnetite concrete. On a pound-for-pound basis, the waste lead slag concrete materials were 30% cheaper than barytes and magnetite concrete materials.

316. W. B. Doe, Zinc Bromide Solution for Use in Shielding Windows, ANL-4879 (September 1952). Availability: Dep.; $4.80 (ph), $2.70 (mf) OTS.

Chemical, radiation, and low-temperature stability, optical transmission as influenced by metallic ions and contamination by container materials, and corrosive effects of Zn bromide solutions were studied. It was observed that the solution is useful as a transparent shielding material because its transmission for light is superior to nonbrowning lime glass. It has freedom from optical inhomogeneities, shielding power for gamma radiation equal to nonbrowning glass or concrete, capability of withstanding nearly the same total dose of radiation as nonbrowning glass, and no observable deterioration.

317. C. F. Deganahl, Reactor and Shield Weights for the CFRMR, PWAC-89 (December 31, 1953). Availability: Dep. (mc); $2.60 (fs), $0.83 (mf) OTS.

Pratt and Whitney has been requested to make a study of propulsion systems using the Circulating Fuel Reflector Moderated Reactor, CFRMR. This report presents shield weights for the CFRMR, together with a brief summary of the assumptions and methods of estimation used.

318. R. B. Theus and L. A. Beach, Pseudorandom Numbers and Collision Parameters for Monte Carlo Shielding Calculations, NRL Report 4695 (February 1956). Availability: $0.50 OTS.

Since the Monte Carlo method involves random sampling under probability distribution functions which describe the phenomenon being studied, a modified "midsquare" method of generating pseudo-random numbers for Monte Carlo calculations has been developed. The numbers have been generated and analyzed for uniformity of population by the NAREC. Distribution functions used in radiation shielding problems were obtained from these numbers and they compare favorably with the theoretical expectations.

319. L. A. Beach, R. B. Theus, J. D. Plawchan et al., Stochastic Estimates of X-Ray Spectral Intensities for Shallow and Deep Penetration, NRL Report 4412 (August 3, 1954). Availability: $2.75 (ph), $2.00 (mf) OTS.

The applicability of several random sampling techniques to gamma-ray diffusion problems has been investigated. Use of the exact physical analogue to estimate the penetration through slab and semi-infinite shields was found to yield estimates of the spectral distributions in agreement with moment calculation for penetration distances up to 4 mean free paths. For deep penetration, several methods of biasing were studied and compared. The exponential transformation exhibited certain advantages over other bias methods and allowed estimation of spectral intensities for penetration distances as great as 18 mean free paths of the primary radiation.

320. N. Edmonson, J. J. Henrick, and T. A. Moss, Application of the Carlson SN-Method to Calculation of Neutron Angular and Total Flux Distributions in a Spherically Symmetric Shield, NARF-60-11T (MR-N-253) (May 16, 1960). (Convair, Fort Worth.) Not for sale to public.

The S_n-method for calculating numerical approximations to solutions of the Boltzmann transport equation is developed for a spherically symmetric shield system. Anisotropic elastic scattering is assumed. Energy-group transfer coefficients are developed by using the Dirac δ-function and by a mapping procedure. Formulae for transforming cross-section data from the center-of-mass coordinate system to the laboratory coordinate system are given. A general matrix formulation for the anisotropic case is constructed. These procedures are applied to the computation of angular neutron distributions and total neutron distributions in water shields 80 cm and 120 cm in radius for Watt fission sources uniformly distributed in a small sphere concentric with the water shields.

321. N. Edmonson and L. M. Bostick, Angular Segmentation Method, NARF-61-43T (FZK-9-172) (February 28, 1962). Availability: $9.10 OTS as AD-272918.

The angular segmentation method for calculating numerical approximations to solutions of the Boltzmann transport equation is developed for a spherically symmetric shield system. Anisotropic elastic scattering is assumed. Energy-group transfer coefficients are developed by using the Dirac δ-function. A general matrix formulation for the anisotropic case is constructed. These procedures are applied to the computation of angular neutron distributions and total neutron distributions.

322. R. L. Kloster, Flux Calculations by the Spherical Harmonics Method, NARF-59-4T (MR-N-219) (February 20, 1959). (Convair, Fort Worth.) Not for sale to public.

The spherical harmonics method of obtaining approximate solutions of the Boltzmann transport equation for neutrons is used to obtain the total flux and angular distribution of neutrons from an isotropic monoenergetic infinite plane source. Plane symmetry is assured by assuming all interfaces to be parallel to the source plane.

Problems involving from two to four regions are considered and the general solutions and boundary conditions derived are applied to these problems.

Results include the flux and angular distribution of neutrons in an infinite hydrogenous medium, and the effect of a slab of heavy material, placed in this medium at various distances from the source, on the flux and angular distributions. Calculation of the flux $\psi_0(x)$, the angular distribution $\psi(x,\mu)$ of neutrons from the first four successive odd approximations, P_1, P_3, P_5, and P_7, indicates that for the geometrical arrangements considered a P_3 approximation is sufficient for prediction of the neutron flux. However, the neutron angular distribution shows a highly fluctuant behavior as the degree of the approximation changes, indicating the need for approximations of higher order.

Application of the method in an investigation of effective removal cross-section calculations is discussed.

323. C. D. Zerby, Transmission of Obliquely Incident Gamma Radiation Through Stratified Slab Barriers, ORNL-2224 (November 29, 1956).

Vol. I. Availability: Dep.; $8.60 (fs), $3.02 (mf) OTS.

Results are presented from an investigation of the penetration of gamma radiation through stratified slabs using a Monte Carlo method of computation. Since this method provides a bulk of information not generally available as a result of the use of other methods of computation, a considerable effort was made to make available all the data obtained without restriction because of its usefulness in many applications. The data are presented in the form of tables and graphs. A detailed discussion of all methods and techniques used throughout the investigation, along with much of the data in the form of tables and graphs for ease of comparison of results of cases with slightly different boundary conditions are included.

Vol. II. Availability: Dep.; $8.60 (fs), $3.11 (mf) OTS.

Results of a Monte Carlo computation of the penetration of gamma radiation through stratified slabs are given. The detailed angular and degraded energy spectrum of penetrating gamma radiation and the energy spectrum of the radiation reflected from many of the slabs are presented.

324. J. J. Taylor, Application of Gamma Ray Buildup Data to Shield Design, WAPD-RM-217 (January 25, 1954). Availability: Dep.; $0.20 OTS.

A method is demonstrated for calculating gamma-ray penetration in infinite homogeneous media by the NBS-NDA procedure, using microscopic cross sections and scattering functions. Required integrations are solved by a reduction to the problem where the attenuation kernel is a simple exponential. Dose and energy buildup factors are calculated in H_2O, Fe, and Pb for point isotropic energy sources in the range 0.5 to 10 MeV.

325. D. D. Babb, J. W. Keller, and E. McCray, Curve Fits of Gamma-Ray Differential-Energy Spectra, NARF-59-36T (MR-N-251) (November 1, 1959). Availability: $3.60 OTS As AD-229401 (PB-159933).

The results of the Nuclear Development Association (NDA) moments method calculations of the penetration of gamma rays through various media from point-isotropic sources have been interpolated and extrapolated to give differential energy spectra for a larger set of initial and final energies. For each of the new pairs of energies the data have been fitted analytically and tabulated as a function of penetration distance and atomic number. The function used for the fit is:

$$\ell n \frac{f(t)}{\mu_o r} = a_1(\mu_o r)^2 + a_2(\mu_o r) + a_3(\mu_o r)Z + a_4 Z + a_5 Z^2 + a_6$$

where f(t) is the function tabulated by NDA. The range of Z was divided into three sections and the fit made independently for each section. Thus, 18 coefficients were required for each of 104 energy pairs.

326. E. Amaldi, "The Production and Slowing Down of Neutrons," Handbuch Der Physik (Encyclopedia of Physics), Vol. 38/2, 1-659 (1959).

A comprehensive survey is presented of the methods of compting the slowing down of neutrons which includes some historical interest. Necessarily the mechanisms of neutron slowing down are fully described.

327. U. Fano, L. V. Spencer, and M. J. Berger, "Penetration and Diffusion of X-Rays," Handbuch Der Physik (Encyclopedia of Physics), Vol. 38/2, 660-817 (1959).

A comprehensive survey of the method of computing the attenuation of gamma rays is presented. The interactions of gamma rays with matter are fully described. Many results of calculations are displayed.

328. D. C. Piercy and D. E. Bendall, The Transmission of Fast Neutrons Along Air-Filled Ducts in Water, AEEW-R 69 (June 1962). Availability: Dep.; $0.77 (BIS), 5S 6d (HMSO).

Measurements of fast flux, by means of the $S^{32}(n,p)P^{32}$ reaction, have been made along air-filled ducts in the water of the LIDO Panel C facility. Using results obtained by the Method of Moments, a technique has been devised for predicting the absolute sulphur reaction rate to better than a factor of 2 for ducts of up to 200 duct radii in length.

329. D. C. Piercy, The Transmission of Thermal Neutrons Along Air-Filled Ducts in Water, AEEW-R 70 (June 1962). Availability: Dep.; $0.77 (BIS), 5S 6d (HMSO).

Predictions and measurements of thermal-neutron transmission along air-filled ducts in water have been made. To aid the analysis, the ducts were shielded in various ways using cadmium sheet.

The predictions were, in general, in agreement to better than a factor 4 for ducts up to 200 duct radii in length.

330. J. M. Miller and C. L. Storrs, Neutron Streaming Through Air Ducts, ORNL CF 54-2-93 (February 15, 1954). Availability: Dep. (mc); $6.30 (ph), $3.00 (mf) OTS.

Studies of neutron streaming through air ducts were continued. Fast-neutron dose-rate measurements and thermal-neutron flux measurements were also made. The fast-neutron data were used to check the Simon-Clifford formula for neutron attenuation through ducts. Data are presented graphically and in tabular form.

331. R. E. Benenson and A. N. Fasano, The Transmission of Fission Neutrons Having Energy Above S-32(N,P)P-32 Threshold by Straight Cylindrical Ducts in Water, WADC Technical Report 57-89 (February 1957). Availability: $6.30 (ph), $3,00 (mf) OTS as PB-133190.

A ray analysis, employing an experimentally derived point attenuation function in water, has been carried out for straight cylindrical air ducts with diameters ranging from 2 inches to 18 inches. An extended fission source and $S^{32}(n,p)P^{32}$ threshold detectors have been used throughout the corresponding experimental determination. The results of the experimental determination of the transmission and the ray analysis agree to within 15% and indicate a significant contribution to the transmission from portions of the source not "seen" directly by the duct.

332. A. Prince, The Scattering of Neutrons by Air Ducts in Shields, XDC 59-7-119 (July 17, 1959). Not for sale to public.

The solution of an integral equation is employed in expressing the scattered neutron flux distribution in cylindrical or annular ducts. Using the "albedo" concept, reasonably simple expressions are derived for cylindrical and annular ducts having any dimensions and composed of any kind of material.

A comparison of this method with experimental data is also given.

333. R. L. Mather, "Gamma-Ray Collimator Penetration and Scattering Effects," J. Appl. Phys. 28, 10, 1200-1207 (October 1957).

Expressions are developed which give the amount of unidirectional radiation from a distant radiation source passing through a cylindrical hole in a slab of material, including the amount of radiation which penetrates the edges of the hole, as a function of angle between the radiation direction and the axis of the hole. The modifications necessary for sources at finite distances are indicated. These functions are integrated over appropriate angles for distant line sources, and for extended sources. It is shown that, to a first approximation, the results are the same as the geometrical aperture for a like diameter hole in a similar slab with one mean free path of material removed from each side. The effects due to scattering from the walls of the collimator are discussed and an approximate method of calculation is presented with results for a specific collimator.

334. F. J. Allen, A. T. Futterer, and W. Wright, Neutron Reflection and Flux Versus Depth for Water, BRL-1204 (June 1963). Not for sale to Public.

Detailed calculated results on neutron reflection and flux versus depth for water are given in the form of machine printouts. The angular and energy distributions of the reflected neutrons along with the energy-dependent and total flux at various depths are contained in tabular form on the printouts. Neutron number current, number flux and dose transmission as functions of thickness are also given in tabular form on the printouts.

A table of summary information on reflection is presented. This contains number current, number flux, dose and energy reflection factors as functions of incident energy and angle.

A few figures are presented to illustrate graphically the meaning of the various tabular results.

Some comparisons are made with results obtained by the National Bureau of Standards.

335. E. S. Troubetzkoy, Minimum Weight Shield Synthesis, UNC-5017 (Part A) (October 15, 1962). Availability: Dep. (mc); $5.60 (fs), $1.79 (mf) OTS.

A simple model for radiation production and attenuation in a spherical multilayer reactor-shield assembly is introduced. Variational methods are applied to this model to minimize the weight subject to given specifications on the dose at the outside of the shield and on the properties of shielding materials. The method provides guidance in the choice of the sequence of materials in the multilayer optimum shield, and gives equations which have to be satisfied by the radii at the boundaries of these layers.

336. F. J. Allen, A. T. Futterer, W. Wright, and A. Budka, The Transmission of Monoenergetic Neutrons Through Polyethylene - Machine Calculations, BRL-1130 (April 1961). Availability: $14.00 OTS as AD-261065.

Detailed results of calculations of the transport of monoenergetic neutrons through plane, laterally infinite, slabs of polyethylene containing eight percent boron carbide by weight are given. The results consist of machine printouts and are in tabular form. Less extensive results are given for pure polyethylene. The calculations were made by the Monte Carlo method.

Sufficient descriptive material is presented so that the printouts can be understood. However, a description of the machine program along with graphs and discussion of the principal results extracted from the machine printouts have been published in a separate report, BRL 1129. This arrangement was selected because of the great bulk of the machine results.

337. C. F. Deganahl, H. Woodsum, and H. V. Garabedian, Supercritical Water Reactor Shield Design Procedure, PWAC-100 (June 1, 1954). Availability: Dep. (mc); $5.60 (fs), $1.97 (mf) OTS.

This report presents the assumptions, data, and methods used to estimate the SCWR shield weights of PWAC-88 and has been prepared as a final report on shielding for that reactor system. The report covers divided shields, representative of what might be used for bomber aircraft, and quasi-unit shields designed for a specific configuration of logistics carrier. Sections II and IV of the report are numerical "recipes" for the two types of shield from which the calculations can be precisely duplicated, while Sections III and V are explanatory in nature and define the reasoning behind the computing "recipes." In order to save time, since the SCWR program has been discontinued, the report has been written for the reader who is thoroughly familiar with the previous SCWR shield work, and only those points which involve recent changes are given detailed explanation.

338. R. B. Theus and L. A. Beach, Gamma-Ray Albedo from Iron, NRL 4701 (February 8, 1956). Availability: $1.10 OTS as PB-163838.

A major problem in designing the radiation shield for a power reactor results from the radioactivity induced in the primary coolant. Since the operating personnel must be protected from the radiation emanating from this circulating coolant as well as from the reactor proper, the Monte Carlo methods have been used to calculate the gamma-ray albedo for the 6.13-MeV photon incident on iron. The spectral and directional distributions of the emerging radiation are given as a function of the angle of incidence.

339. D. R. Otis, The Lid Tank Shielding Facility at Oak Ridge National Laboratory. Part II. Determination of the Fission Rate of the Source Plate, ORNL-2350 (July 7, 1959). Availability: Dep.; $4.60 (fs), $1.70 (mf) OTS.

Following the installation of the second Lid Tank Shielding Facility source plate, a thin uranium disk identified as SP-2, a series of experiments were performed to determine its fission rate. Three independent methods were used, one based on the rate of absorption of the thermal neutrons incident on the plate, another based on the rate of fast-neutron production within the plate, and a third one based on the power dissipation within the plate. Discussions of the methods, all of which yielded consistent results, are presented, along with a description of the source plate. A weighted average of the fission rates determined by the three methods gave a result of 1.66×10^{11} fissions/sec $\pm$ 5% for the standard incident thermal-neutron flux. With an assumed energy dissipation of 200 MeV/fission, a typical reactor having this fission rate would release 5.22 watts of heat power. The calibration of the SP-2 allowed a re-evaluation of the power of the first LTSF source plate, the SP-1. The revised "effective" neutron power for SP-1 is 1.7 watts $\pm$11%.

340. D. W. Cady, The Lid Tank Shielding Facility at Oak Ridge National Laboratory. Part III. Instrumentation, ORNL-2587 (January 4, 1960). Availability: Dep.; $6.60 (fs), $2.09 (mf) OTS.

The instrumentation at the Lid Tank Shielding Facility (LTSF) is described. Drawings or photographs of the gamma-ray, thermal-neutron, and fast-neutron detectors are included, and these detectors are discussed. The methods of detector application at the LTSF are also described, as well as the calibration techniques. The electronics associated with each detector is presented with appropriate references. The range of sensitivity of each detector is also given, along with estimates of the reliability of data from the detection system.

341. J. A. Belcher and L. K. Zoller, The Outside Test Tank and Associated Hardware, XDC 59-11-154 (November 6, 1959). Not for sale to public.

This document describes the General Electric Outside Test Tank (OTT) and the hardware associated with shield testing operation. It is primarily intended to replace the introductory descriptions which accompany each data report and analysis of OTT shielding experiments. A secondary purpose of this report is to serve as an orientation guide for persons planning to utilize the OTT.

342. A. W. Casper, Data Report and Analysis of Battelle Pool Mapping, XDC 59-8-228 (August 25, 1959). Not for sale to public.

This report presents the fast-neutron dose rates, the gamma dose rates, and the thermal-neutron flux measurements made in the GE-ANPD Source Plate Facility at Battelle Memorial Institute. Measurements were made at points on the source plate centerline and at points on lines parallel to the source plate centerline in the pool. Point kernel programs were used to compute the fast-neutron and gamma dose rates and these values are compared with the measurements. The thermal flux measurements are compared with values obtained by using a G-2 program.

343. F. J. Muckenthaler, T. V. Blosser, J. M. Miller et al., Measurements at Beam Hole HB-2 of the Oak Ridge Research Reactor for South Facility Movable Shield Design Study, ORNL-3069 (August 31, 1961). Availability: Dep.; $1.00 OTS.

A series of measurements have been completed at Beam Hole HB-2 of the ORR, for use in designing a movable personnel and instrument shield for the ORR South Facility. They include thermal and fast-neutron spectra, both unattenuated and behind selected shield configurations, and fast-neutron and gamma dose rates behind about 38 configurations. Materials used for the shield mockups included lead, iron, borated polyethylene, polyethylene and concrete. An attempt was also made to measure the intensity of the scattered radiation from these configurations. Despite experimental complications, some general conclusions have been drawn from the latter experiments.

344. A. D. MacKellar, L. Jung, D. R. Mathews, et al., Lid Tank Shielding Facility Measurements Behind the ML-1 Mockup, ORNL-3178 (September 6, 1961). Availability: Dep.; $1.25 OTS.

An experimental evaluation of the shield design for the ML-1 mobile reactor was made at the Lid Tank Shielding Facility. Thermal-neutron fluxes, fast-neutron dose rates, and gamma-ray dose rates were measured behind slab mockups of the basic shield design and a number of possible variations. The designs embodied various combinations of lead, Hevimet, stainless steel, boral, water, and aqueous solutions of ammonium pentaborate at two concentrations. The after-shutdown decay characteristics of the basic design were determined, and data were obtained from a fairly accurate mockup of the stainless steel plenum and gas duct typical of the top and bottom regions of the ML-1 shield. Analysis of results and application to the final shield design are not reported.

345. R. W. Peelle, F. C. Maienschein, and T. A. Love, Energy and Angular Distribution of Gamma Radiation from a Co-60 Source After Diffusion Through Many Mean Free Paths of Water, ORNL-2196 (August 12, 1957). Availability: Dep.; $3.60 (fs), $0.98 (mf) OTS.

Measurements have been made of the energy and angle spectrum of the photon energy flux resulting from 1.17- and 1.33-MeV gamma rays from a point source after diffusion in an infinite water medium. Energy flux spectra were measured for penetration distances corresponding to approximately six and nine mean free paths for the uncollided source photons and for angles of gamma-ray emission varying from 0 to 90 deg. Spectral measurements were made with a Compton spectrometer having a peak-to-total ratio of 0.7 and a resolution of 14% (full width at half maximum) for 1.12-MeV photons. The resulting energy and angle spectra of the energy flux are compared, after integration over angle, with calculated results which have been obtained by the "moments method." The results are also compared with the available calculated differential results.

346. RPCC, Shielding Constants, ANL-5800, Second Edition, 629-666 (July 1963). Availability: Dep.; $7.00 OTS.

Effort has been made to supply a concise selection of data on standard shielding materials for the use of the shield designer. The collection of data on the typical concrete mixes is expected to provide a basis for their comparison as well as some constants for use in determining nominal values for shielding calculations.

Geometry, multilayer effects, and considerations of the general shielding problem are not included since detailed discussions of these subjects are readily available in the open literature.

347. F. L. Keller and O. S. Merrill, Analysis of the Recent TSF Secondary Gamma-Ray Experiment, ORNL-2586 (August 25, 1958). Availability: Dep.; $4.60 (fs), $1.58 (mf) OTS.

Since the existing experimental data on gamma rays produced by the inelastic scattering of neutrons with oxygen and nitrogen is rather sparce, it is necessary to rely upon theoretical cross sections to predict the spectra of inelastic scattering gamma rays produced by an ordinary neutron flux distribution. Because of the model used in the cross-section calculation a considerable amount of uncertainty existed in regard to the accuracy of these results. In order to provide a check on the results of such calculations an experiment was performed at the ORNL Tower Shielding Facility to directly measure the spectra of capture and inelastic scattering gamma rays produced in air by a reactor neutron spectrum. Unfortunately, it was necessary to make these gamma-ray measurements behind a thick shield through which a long water collimator was inserted. Hence, before a direct comparison could be made between the calculations and the experiment it was necessary to first calculate the spectra of capture and inelastic scattering gamma rays produced in air, then to determine the effect of attenuation and buildup in the water collimator, and, finally, to include the detector response functions. When the calculated and measured pulse-height distributions were compared the shapes were found to be in excellent agreement. This indicates that considerably more confidence may be placed in calculations which use spectra of inelastic scattering gamma rays determined from the existing theoretical cross section.

348. A. D. MacKellar, Prediction of Thermal-Neutron Fluxes in the Bulk Shielding Facility from Lid Tank Shielding Facility Data, ORNL CF-59-1-24 (January 12, 1959). Availability: Dep. (mc), $6.30 (ph), $3.00 (mf) OTS.

Predictions of the thermal-neutron fluxes to be expected near the Bulk Shielding Reactor were made on the basis of Lid Tank Shielding Facility experimental data transformed to the BSR geometry. The predicted fluxes are higher than the measured fluxes by 12 to 22% for distances closer than 60 cm from the reactor, but they are essentially in agreement for larger distances.

349. D. R. Otis, Neutron and Gamma-Ray Attenuation for a Fission Source in Water - Comparison of Theory with LTSF Measurements, ORNL CF 57-3-48 (March 12, 1957). Availability: Dep.; (mc), $4.80 (ph), $2.70 (mf) OTS.

Calculations are presented of the fast-neutron dose rate, the gamma-ray dose rate, and the thermal-neutron flux along the source plate axis in the Lid Tank with the tank filled entirely with water. These calculations are compared with experimental measurements. The calculations are absolute and are compared with absolute measurements.

350. BSF Staff, Attenuation in Water of Radiation from the Bulk Shielding Reactor - Measurements of the Gamma-Ray Dose Rate, Fast-Neutron Dose Rate and Thermal-Neutron Flux, ORNL-2518 (Chart) (1958). Availability: Dep. $0.50 OTS.

This is a large chart on which is shown measured gamma-ray dose rate, fast-neutron dose rate, and thermal-neutron flux in the water of the ORNL Bulk Shielding Facility.

351. R. A. Mann and L. S. Burns, Jr., Comparison of a Calculated and a Measured Direct-Beam Gamma-Ray Energy Spectrum Through the Side of a Reactor Shield, XDC 60-5-99 (May 4, 1960). Not for sale to public.

The direct beam gamma-ray energy spectrum was calculated for a position at the side of one of the two-pi reactor shields used in experiments at the ORNL Tower Shielding Facility. A comparison of the calculated and measured spectra shows fair agreement. Uncertainties in the calculation of the distribution of secondary gamma-ray sources in the reactor shield possibly caused greater error in the calculated spectrum than did uncertainties in the spectrum calculations.

352. L. M. Bostick, ASTR Source Terms, NARF-60-2T (FZK-9-144) (April 30, 1960). Not for sale to public.

Source terms [neutrons (photons)/steradian-sec-MeV-watt] for Aircraft Shield Test Reactor (ASTR) Configurations 3, 4, 5, and 15 were calculated by a penetration method which employs moments method solutions of the Boltzmann transport equation. Both primary and secondary gamma rays were considered.

Comparisons are made with direct-beam dose rates measured in the horizontal midplane of the ASTR by the direct-beam-shield technique. The gamma and neutron spectra, calculated at discrete energy levels, are compared with available experimental data. On the average, the calculated total leakage dose rates are within 25% of those measured. The largest discrepancy is for ASTR Configuration 4 where the calculated neutron leakage is 52% high.

In this analysis it is assumed that the ASTR leakage is symmetrical about the cylindrical axis of the ASTR and may be represented by measurements made in the horizontal midplane. It is shown that the effect of the measured irregularities in the leakage outside the horizontal midplane does not alter the conclusions of the analysis seriously.

353. E. P. Blizard, "Nuclear Radiation Shielding," Nuclear Engineering Handbook (Editor - H. Etherington), McGraw-Hill Book Co., Chapter 7, 60-116 (1958).

This article is of a general nature but is slanted toward the shield designer. Many useful tables of values and simple formulae are given.

354. S. Glasstone and A. Sesonske, "Shielding of Nuclear Reactor Systems," Nuclear Reactor Engineering, D. Van Nostrand, 553-627 (1963).

This article is of a general nature but is slanted toward informing the novice. Actual examples of shielding calculations are shown.

355. L. A. Bowman, W. W. Dunn, R. H. Lessig, and D. K. Trubey, A Calculation of the Energy Spectrum and Angular Distribution of Gamma Rays at the Surface of the BSF Reactor, ORNL CF 58-4-75 (April 15, 1958). Availability: Dep. (mc); $3.30 (ph), $2.40 (mf) OTS.

The energy and angular distribution of the gamma rays leaking from the BSF reactor were computed by integrating the sources over the entire reactor and applying an attenuation kernel. The primary gamma-ray spectrum was grouped into six initial energy groups, E_0, of 0.5, 1, 2, 4, 6, and 8 MeV. The uncollided flux in each energy group at a point on the center of the north face was computed as a function of direction given by θ and ϕ. The collided flux was determined for each initial energy group by including the term $[B(\mu R) - 1]$, where $B(\mu R)$ is the energy buildup factor, in the initial integrand and was divided into contributions to that particular energy group and each lower energy group by assuming that the collided flux had a differential energy spectrum of the type computed by NDA for a point source after passage through 1 mfp of water. The total gamma-ray spectrum compared favorably with previously calculated spectrum, but was in the usual disagreement with experimental results. The angular distribution was determined by assuming the collided and uncollided flux had the same angular distribution.

356. R. L. Ashley and D. S. Duncan, The Theoretical Calculation of the Attenuation of Gamma Radiation from a Swimming-Pool Type Reactor, NAA-SR-1921 (1957). Availability: $1.25 OTS.

In order to determine the adequacy of the calculational methods employed by Atomics International to solve shielding problems on such reactor systems as the SRE and OMRE, it is required that the results obtained by using these methods be compared with accurate experimental information. Since the only reactor on which such experimental data is available is the Oak Ridge Bulk Shielding Reactor, the theoretical attenuation of gamma radiation in the BSR shield was evaluated. The results were compared with the measurements of the gamma-ray attenuation from this reactor out to a distance of 700 cm from the core. The analysis performed indicates that it is possible to reproduce the measured data quite accurately over the entire range of water thickness for which measurements have been made, the average value of the calculated to measured dose rates being 1.25 with the maximum deviation from this value being less than 19%. The formulation and assumptions necessary in the analysis are discussed. It is concluded that the spectra and method used will yield results which are more than adequate when dealing with shielding analyses, per se, and are probably adequate for heat generation problems as well.

357. G. deSaussure, A Calculation of the Gamma-Ray Spectrum of the Bulk Shielding Reactor, ORNL CF 57-7-105 (July 31, 1957). Availability: Dep. (mc); $3.30 (ph), $2.40 (mf) OTS.

A calculation was performed to determine the high-energy portion of the gamma-ray flux at the surface of the uranium-aluminum BSF reactor. The flux was considered to consist entirely of the uncollided flux plus the Compton-scattered flux. The four sources of primary gamma rays included in the calculation were prompt fission gamma rays, capture gamma rays from aluminum, capture gamma rays from uranium, and fission-product decay gamma rays. These sources were divided into four energy groups. Although the assumptions used to calculate the Compton-scattered flux tended to overestimate the actual spectrum, the calculated results were considerably lower than a previously measured spectrum.

358. F. N. Watson, Gamma-Ray and Fast-Neutron Dose Rates in Air as a Function of Distance from the TSF Reactor, ORNL CF 58-4-6 (April 15, 1958). Availability: Dep. (mc); $3.30 (ph), $2.40 (mf) OTS.

Measurements of the gamma-ray and fast-neutron dose rates in air were made at the Tower Shielding Facility for distances out to 4000 ft from the reactor. The reactor was shielded with 6 in. of water and was suspended at an altitude of 190 ft. Plots of the results are presented along with analytical expressions fitted to the curves.

359. C. E. Clifford (USA), F. N. Watson et al., Preliminary Study of Fast Neutron Ground and Air Scattering at the Tower Shielding Facility, ORNL CF 54-8-95 (August 23, 1954). Availability: Dep. (mc); $9.30 (ph), $3.60 (mf) OTS.

Measurements of scattered fast neutrons made at the Tower Shielding Facility as a function of the reactor-detector altitude show a peak in intensity at an altitude of 15 to 20 ft for a reactor-detector separation distance of 64 ft. At this altitude the intensity is a factor of 2 larger than that observed at 200 ft. Within the experimental error of ±2%, the intensity observed at 150 ft was the same as that observed at 200 ft, indicating that for a separation distance of 64 ft the ground-scattered contribution was small at these altitudes. Preliminary analysis of the data estimated this contribution to be between 2 and 5% of the total scattered neutrons.

With the reactor radiating at an angle of 30 deg with respect to the reactor-detector axis, the attenuation in water of the neutrons scattered into the sides of the detector tank was approximately exponential with a relaxation length of 5.6 ± 0.2 cm in the direction normal to the shield surface.

360. H. Soodak, "Theory of Neutron Transport," Reactor Handbook, Sec. Ed., Vol. 3, Part A (Physics), Interscience Publishers, Chapt. 3 (1962).

A collection of a brief description of nearly all extenct methods of solving the Boltzmann transport equation are displayed in this chapter.

361. F. J. Muckenthaler, L. B. Holland, and R. E. Maerker, In-Air Measurements in the Vicinity of the Tower Shielding Reactor II, ORNL-3288 (March 25, 1963). Availability: $2.00 OTS.

During the course of an extensive test program at the Tower Shielding Facility for the Army Tank Automotive Center (ATAC), a large number of measurements were made of the thermal-neutron fluxes and fast-neutron and gamma-ray dose rates in air due to radiation emitted by the TSR-II. These measurements, called "free-field" measurements because of a minimum of structural interference in the field, were performed for reactor-detector separation distances up to 1100 ft, reactor heights up to 175 ft, and detector heights up to 145 ft. In addition, the reactor was encased in two different shields, identified as COOL-I and COOL-II, respectively, which were designed to give a fast-neutron to gamma-ray physical dose-rate ratio of 1.5 at large distances from the reactor and a thermal-neutron flux leakage that would cause ~ 30% of the gamma-ray dose rates in the field to be due to neutron captures by nitrogen. The significant difference between the two shields was in the hardness of the gamma-ray and fast-neutron leakage spectra, especially the latter. This report describes the conditions under which the measurements were made and reports the data.

362. R. H. Clark, J. G. Carver, R. F. Brenton et al., Test Data from the Two-Pi Solid-Angle Shield-Cover Experiment, APEX-439 (December 19, 1958). Availability: Dep.; $16.00 (fs), $7.58 (mf) OTS.

The experimental data presented in this report are the results of tests conducted at the Oak Ridge Tower Shielding Facility in which 2π solid-angle covers were used on both the reactor shield and crew shield. These tests were conducted to provide basic air- and ground-scattering data, as well as to determine the minimum requirements for facilities needed to develop a nuclear aircraft shield. The data are presented as recorded. Some of the more important data have been selected and grouped in the text to show the effects of the 2π covers.

363. H. C. Claiborne, C. M. Copenhaver, H. W. Bertini et al., "Calculating Gamma Heating in Reactor Structures," Nucleonics 15, 10, 114-121 (October 1957).

The gamma-ray heating in aluminum, iron, and lead due to gamma rays at the surface of the Bulk Shielding Reactor was calculated and compared with experiment. The calculation was done using energy-absorption build-up factors as calculated by the Moments Method.

364. G. S. Hurst and R. H. Ritchie, "A Count-Rate Method of Measuring Fast-Neutron Tissue Dose," Rev. Sci. Instr. 22: 981 - 986 (1951).

The problem considered is that of measuring the fast-neutron tissue dose. A special proportional counter has been designed whose response to collimated neutrons is such that the number of counts produced is approximately proportional to dose, when the neutron energy is between 0.2 MeV and 10 MeV. The response has been tested experimentally with monoenergetic neutrons of various energies between 365 kev and 5.2 MeV and at 14 MeV and is in good agreement with the dose curve.

365. J. F. Batter and E. T. Clarke, Shielding Effectiveness of the USNRDL Underground Fallout Shelter, TO-B 62-37 (June 20, 1962). Not for sale to public.

Data on the transmission of cobalt-60 radiation into the interior of the NRDL underground shelter are analyzed to determine the shelter protection factor against infinitely-extended fields of 1-hr fallout. Far-field dose rates are calculated, and the conversion from cobalt-60 to 1-hr fallout radiation is made. Results show that the shelter floor offers a maximum protection factor of 100,000 over most of its area and factors of at least 10,000 everywhere except within 6 ft of a 2-ft diameter vent.

366. G. J. Hine and G. L. Brownell, Radiation Dosimetry, Academic Press, Inc. (1956).

The problems of radiation dosimetry are presented as three main parts. First presented are the fundamental principles of dosimetry and some of the basic biological and medical effects of radiation. The second part describes the various instruments available per dosage measurements and their uses for different types of radiation. The third part presents the problems of dosimetry of various radiation fields.

367. J. M. Ruddy, "How to Select Gamma Shielding Doors," Nucleonics 20, 6, 94-95 (June 1962).

This report serves as an introduction to the problem of shielding doors for hot cells, gamma-radiation cells, X-ray machines and accelerators. Considerations for the doors include the mounts and methods to be employed for movement, size, materials - concrete, steel, or lead, space, drives, auxiliary shielding, and cost.

368. V. R. Cain, C. R. Fink, F. N. Greene et al., Penetration of a Fast-Neutron Air-Scattered Dose Rate from the TSR Through the Sides of a Cylindrical Crew Compartment, ORNL CF 57-12-26 (April 29, 1958). Not for sale to public.

The neutron shield design procedure has been refined by the availability of more reliable air-scattering probabilities which have been obtained in recent TSF experiments with the so-called "Boeing" crew compartment. Since the side shield on the crew comparement could be varied from 0 to 8 in. in 2-in. increments, the probabilities could be determined as a function of the thickness, whereas heretofore the attenuation had to be calculated. Also the new probabilities are for scattering to the center of the crew compartment rather than to the inside surface, which eliminates the necessity of using a rather uncertain focusing factor to determine the dose at the center.

369. A. E. Spaller, Study Report - Shielded Viewing Windows and Alpha Seal Window Assemblies for TUFCDF, ORNL CF 63-8-37 (August 19, 1963). Not for sale to public.

Several window designs were studied in order to provide recommendations for the type and size of shielded viewing windows to be used in TUFCDF hot cells. Maximum viewing angles, light transmittance, and estimated cost were determined for each window design. Maintenance, radiation damage, and hazards involved for the various types of construction and shielding media were studied. A conceptual design for an alpha seal window assembly was developed. Conclusions and recommendations are included in the body of this report.

Preliminary copies of this report were issued as Plant and Equipment Division Engineering Document No. 10017-N-100R.

370. G. N. Whyte, Principles of Radiation Dosimetry, John Wiley and Sons, Inc. (1959).

The concepts of radiation dosimetry are presented and their relationships to some basic parameters of the radiation field. The emphasis is on the physical principles of radiation measurements with no attempt made to give a systematic or detailed description of radiation detectors. Topics discussed include: the radiation field; the interaction of radiation with matter; the determination of X-ray spectral distributions; measurements of X-ray intensity; measurements of exposure dose; measurements of absorbed dose per X-ray; dosimetry of charged particles; neutron dosimetry; and the design of secondary radiation detection instruments.

371. J. B. Trice, "Miniature Fission Chamber," Nucleonics 16, 7, 84-87 (July 1958).

To exploit nuclear threshold reactions for neutron-spectrum measurements, miniature fission chambers have been designed and built for use in high-intensity radiation fields. Some measurements made with these chambers and with activation analysis are described.

372. F. C. Hume, "Spray Technique Applies Shielding Concrete," Nucleonics 18, 10, 102-104 (October 1960).

Spraying is an economical way of placing the dense magnetite concrete needed in shielding against intense radiation. This new method sprays rather than pours the concrete. The process was used for the first time with magnetite in the High Level Radiochemistry Facility at Hanford. The spray method has numerous advantages of cost and quality over conventional techniques of concrete placement for high-density applications.

373. R. L. Ashley, "Effective-Energy Method for Spent-Fuel Shielding," Nucleonics 16, 10, 78-81 (October 1958).

Suitable graphs permit substitution of a single photon energy for a fission-product spectrum. Shielding problems reduce to finding the penetration of photons of this energy.

374. D. G. Chappell, "Gamma-Ray Attenuation with Buildup in Water," Nucleonics 16, 7, 80 (July 1958).

This nomogram gives the attenuation factor for a water shielding slab used with a point source of gamma radiation.

375. J. B. Trice, "Measuring Reactor Spectra with Thresholds and Resonances," Nucleonics 16, 7, 81-83 (July 1958).

Neutron spectra in reactors can be measured by making use of resonance and threshold reactors. A miniature fission probe has been employed along with activation detectors to measure spectra in four reactors: The Bulk Shielding Reactor (BSR), the Low Intensity Test Reactor (LITR), and the Graphite Reactor (X-10) at the Oak Ridge National Laboratory and the Materials Testing Reactor (MTR) at the National Reactor Testing Station in Idaho. The method is sensitive to cross-section data, and as these data improve, it can undoubtedly become an incisive approach to knowledge of reactor neutron spectra.

376. S. S. Pawlicki, "Shielding CVTR - A Heavy-Water Pressure-Tube Reactor," Nucleonics 20, 12, 74-82 (December 1962).

Design considerations of the radiation shields for the CVTR, a heavy-water pressure-tube reactor, are discussed. The problems encountered include streaming of radiation around the numerous shield penetrations for the pressure tubes. Precautions to prevent degrading of D_2O with H_2O, the choice and positioning of the thermal shield, and the radiation shields. The methods used to solve these problems are shown.

377. M. G. Chasanov and M. Shatzkes, "Attenuation of Gamma Rays from an Infinite Plane," Nucleonics 16, 6, 63 (June 1958).

A nomogram is given which solves the equation $\frac{1}{2}\,[e^{-b} + E_1(b)]$. This equation is an approximation to the attenuation function of gamma rays from an infinite plane source in light Z materials.

378. J. W. Lindner, "Shielding-Glass Buildup Factors," Nucleonics 16, 10, 77 (October 1958).

Gamma-ray buildup factors for three shielding glasses are presented. This data was computed on the basis of comparing attenuation coefficients which yields an equivalent material whose buildup has been computed.

379. R. W. Brisbane and L. B. Silverman, Photographic Dosimetry - An Annotated Bibliography, UCLA-446 (September 15, 1959). Availability: Dep. (mc), $3.50 OTS.

Annotated references to unclassified report, book, and journal literature, both domestic and foreign, concerning the quantitative evaluation of radiation dosage by means of photographic films and nuclear emulsions is presented. Reports in the field of nuclear physics which contain information of use in the field of dosimetry are included, as are those concerning quantitative autoradiography.

380. R. W. Brisbane and L. B. Silverman, Photographic Dosimetry - An Annotated Bibliography (Supplement), UCLA-516 (August 1963). Availability: $3.50 OTS.

This report, which is intended as a supplement to UCLA-446, Photographic Dosimetry: An Annotated Bibliography, is a list of 500 annotated references to unclassified report, book and journal literature, both domestic and foreign, concerning the quantitative evaluation of radiation dosage by means of photographic materials. It contains references from 1959 to date, plus older references not included in the earlier report. Reports in the field of nuclear physics which contain information in the field of dosimetry are included, as are those concerning quantitative autoradiography. Cross references to the reports listed in UCLA-446 are included.

381. I. E. Lamb and R. R. Tsukimura, Standardization of Fabrication Techniques (Threshold Foils), WADC-TR-59-607 (May 1960). Availability: OTS.

The techniques used to develop and standardize a series of neutron detection foils for measurement of neutron spectra are described. The series includes gold and cobalt foils, and plutonium-239, uranium-238, and sulfur threshold foils. A method of calibrating the foils with a low-flux research reactor is discussed. The instrumentation required to obtain data from the foils is presented, together with the neutron intensities that may be explored with these detectors.

382. E. B. Wagner and G. S. Hurst, "Advances in the Standard Proportional Counter Method of Fast-Neutron Dosimetry," Rev. Sci. Instr. 29, 2, 153-158 (February 1958).

The proportional counter method for measuring fast-neutron dose in the presence of gamma radiation has been considerably improved through (1) redesign of the proportional counter, (2) a study of the behavior of the counter for various filling gases, and (3) development of a more convenient form of the electronics necessary for pulse integration and dose indication. The instrument is suitable for radiobiological research, neutron physics, and shielding research, as well as radiation protection. It will be shown that the response of the instrument is within $\pm 5\%$ of tissue absorbed dose over the energy range of 0.5 MeV to 14 MeV.

383. F. J. Allen and A. T. Futterer, "Neutron Transmission Data," Nucleonics 21, 8, 120-121 (August 1963).

The results of Monte Carlo calculations of neutron transmission through laterally infinite slabs of concrete, water, polyethylene, and Nevada Test Site soil are presented for incident neutron energies from 0.5 to 14 MeV. Splitting was employed in the Monte Carlo calculation.

384. R. L. Lynn and V. L. Dagragnano, Energy Response of Landsverk Ionization Chambers to Monoenergetic Gamma Rays, USNRDL-TR-580 (May 7, 1962). Availability: $2.60 (fs) OTS.

An energy response calibration of Landsverk L-65, L-81, and L-83 pocket chambers was carried out using monoenergetic gamma-emitting radioisotopes. Since the intensity of the sources was low, the pocket chambers were used as ratemeters and the ionization current was read on a vibrating-reed electrometer.

The dose rate of each isotope (Co^{60}, Cs^{137}, Hg^{203}, and Ce^{141}) was calibrated with a scintillation spectrometer.

The angular response of the chambers for the energies of 1.25 MeV, 662 MeV, 279 keV, and 145 keV, respectively, was measured in one quadrant. The effect of wall thickness on the energy response was also investigated.

385. G. S. Hurst, Fast-Neutron Count-Rate Dosimetry, ORNL-589 (February 1950). Availability: Dep.

Calculations have indicated that fast-neutron doses can be measured by simply taking counting rates. The principle used is that of designing a proportional counter such that its sensitivity relates to energy as does the reciprocal of the flux required to produce a tolerance dose. Although the calculations apply specifically to the case where the neutrons have energies between 0.2 and 10 MeV, it is possible that the same principle may be used for higher energy spectra.

386. J. H. Hubbell, "Dose Fields from Plane Sources Using Point-Source Data," Nucleonics 21(8), 144-148 (August 1963).

Analytic point-source buildup-factor equations that lead to analytic solutions for a variety of distributed source problems with rotational symmetry about a source-detector axis are summarized. Extension of these formulae to include other distributed source problems by expanding the point source buildup factors as a power series is illustrated by calculating dose rates from a disk and a rectangular source.

387. D. J. Raso, Monte Carlo Calculations on the Reflection and Transmission of Scattered Gamma Radiation, TO-B-61-39 (Rev.), (AD-277115), (NP-11819), (May 1962). Availability: Technical Operations, Burlington, Massachusetts.

Monte Carlo calculations were performed to determine the backscattering and the transmission of gamma rays having energies between 0.02 MeV and 10.0 MeV from concrete. The radiation was assumed to be incident on a semi-infinite medium and on various slab thicknesses of 0.5, 1.0, 2.0, and 4.0 mfp at angles of $\cos\theta_o$ = 1.0, 0.75, 0.50, 0.25, and 0.10. The case histories of 5000 photons were followed on the IBM 704 digital computer. The information obtained included: (1) The characteristics of emergent photons, which were stored on magnetic tape; (2) A routine that processes these characteristics to give polar and azimuthal angular dose distribution; (3) Detailed results from application of the processing routine to the parameters investigated. The computer program, the processing routine, and the results are appended.

388. J. B. Moore and G. E. DeVore, Thermal-Neutron Capture in the Anthracene Scintillation Dosimeter, NARF-58-33T (MR N-217) (July 29, 1958). Not for sale to public.

The gamma dose rate resulting from thermal-neutron capture in various components of the anthracene scintillation dosimeter (ASD) has been calculated. The calculated gamma dose rate, in terms of the incident thermal-neutron flux, is $4.00(10^{-4})$ mrem/hr per thermal neutrons/cm^2 sec.

To augment the analytical results, one component, that from the mumetal shield was determined experimentally. The measured value was 20% lower than the calculated value for the mumetal shield.

It is concluded that gamma dose rates measured with an ASD should be corrected for thermal-neutron capture gammas when measurements are made in a radiation field which has a thermal-neutron flux-to-gamma dose rate ratio greater than 250.

389. E. T. Clarke and J. F. Batter, "Gamma-Ray Scattering by Concrete Surfaces," Nucl. Sci. Eng. 17, 1, 125-130 (September 1963).

Gamma radiation from point sources of cobalt-60 (1.25 MeV) and iridium-192 (0.4 MeV) was scattered from a concrete surface. Dose rates were measured at heights equal to source heights, as a function of the ratio of height to separation distance. The percent of scattered to direct radiation rises to a maximum of 12% at $h/d = 0.10$ for cobalt-60, and a maximum of 20% at $h/d = 0.15$ for iridium-192. These results are shown to agree with other work and with Monte Carlo calculations.

390. H. Aceto, Jr. and B. W. Churchill, Neutron Depth Dose from (α,n) and (γ,n) Sources in a Tissue-Equivalent Phantom, UCRL-10267 (March 4, 1963). Availability: Dep.; $0.75 OTS.

Neutron depth-dose behavior in tissue-equivalent phantom exposed to SbBe, PoLi, Po mock fission, and PuBe point sources has been examined with respect to existing theoretical and experimental calculations. Dose from exposure to a point source of complex energy spectra can be estimated by applying a $1/r^2$ geometric correction to the incident neutron flux and comparing the result with the known monoenergetic behavior of neutrons equivalent to the average energy of the point source neutrons.

391. K. Shure, P. A. Roys, and J. J. Taylor, Penetration of 6-MeV Gamma Rays in Lead and Iron, WAPD-T-170 (March 1954). Availability: Dep. (mc).

The broad beam attenuation characteristics of plane slabs of lead and iron for high energy (~ 6 MeV) gamma rays have been determined experimentally. Measurements of the relative dose rate have been taken in air and in water through lead thicknesses varying in one inch steps from zero to eight inches and through iron thicknesses from zero to twelve inches. The results permit comparison with recent theoretical calculations of gamma-ray penetration. This comparison is of particular interest for a source energy above or near the minimum in the gamma attenuation coefficient.

392. K. R. Spearman, Jr., "Neutron Sensitivity of Anthracene Dosimeters," NARF-55-67T, Nucleonics 14: 11, 84-6 (1956).

The response to fast neutrons of a gamma-ray dosimeter, which has a 1-in. by $1\frac{1}{2}$-in. anthracene crystal as the detector, was determined by bombarding the dosimeter with monoenergetic neutrons with energies in the interval of 0.5 to 3.5 MeV. A theoretical curve, derived from published values of the scintillation response of anthracene to protons, fitted the experimental data well over the range of energies used. The dosimeter shows considerable response to fast neutrons. The percentage of neutron dose which appears as gamma dose increases with energy from zero at zero neutron energy to a broad maximum of 30% at 11 MeV. When the neutron spectrum and flux are known, the pseudo-gamma dose rate can be computed from the included graphs.

393. A. C. Rand, Jr., Mechanical Features of the Brookhaven Facility, BNL-139 (T-26) (December 1, 1951). Availability: Dep.; $4.80 (ph), $2.70 (mf) OTS.

The mechanical design of the shielding facility at the Brookhaven National Laboratory reactor is reported. The facility is broken up into several major components and the design, special problems, and testing of each is described.

394. REIC, Survey of Irradiation Facilities, REIC Report No. 31 (Part I) (September 15, 1963). Radiation Effects Information Center, Columbus, Ohio.

This report presents a survey of irradiation facilities generally available to U.S. Government Agencies and their contractors. Reactors operated at steady-state and pulsed conditions and gamma facilities have been included in Part I of this report. Wherever possible, the information has been tabulated. Part I supersedes REIC Reports 7, 7 (First Addendum), 11, and 16. Part II of this report surveys particle accelerator machines. Included in Part II are betatrons, fixed-frequency cyclotrons, linear accelerators, potential-drop machines, and synchrotrons.

395. REIC, Radiation at Cryogenic Temperatures on Semiconductor Material, REIC Report No. 30 (September 30, 1963). Radiation Effects Information Center, Batelle Memorial Institute, Columbus, Ohio.

The radiation-effects literature received by the REIC prior to 1963 concerning the combined effects of nuclear radiation and cryogenic temperatures on semiconductor materials has been reviewed. The materials covered include germanium, silicon, tellurium, indium antimonide, indium arsenide, and bismuth telluride. Changes in electrical properties and results of annealing studies are summarized. The bombarding particles considered include electrons, neutrons, alpha particles, deuterons, and gamma rays. The temperatures covered range from 4.2 K to 273 K. The need for more low-temperature irradiation studies is demonstrated.

396. E. P. Blizard and T. A. Welton, The Shielding of Mobile Reactors - II, ORNL-1133, Part 2 (June 30, 1952). Availability: Dep. (mc), $9.30 (ph), $3.60 (mf) OTS.

The methods of applying results of bulk shielding measurements to the design of reactor shields is outlined. Geometrical transformations for the more common shapes are derived, as are approximate means of calculating leakage of radiation.

As an illustration of the methods, the ORNL Lid Tank and Bulk Shielding Facility data are transformed to a standard geometry and compared. Two direct calculations of water attenuation, using cross sections described in the previous article, are finally compared with the experimental data.

397. R. G. Cochran and H. E. Hungerford, Fast-Neutron Dosimeter Measurements for Experiment 2, ORNL CF 51-5-73 (May 11, 1951). Availability: Dep. (mc), $1.80 (ph), $1.80 (mf) OTS.

Measurements of fast-neutron dosage were made behind a 17-in. Fe shield at the Bulk Shielding Facility. The results are tabulated and compared graphically with those obtained previously with a 100% H_2O shield.

398. G. T. Chapman and J. D. Flynn, A Comparison of the Shielding Properties of Iron and Stainless Steel in the ORNL Lid Tank (Exper. 38), ORNL CF 53-6-187 (June 30, 1953). Availability: Dep. (mc), $4.80 (ph), $2.70 (mf) OTS.

Thermal-neutron flux, fast-neutron dose, and gamma dose measurements were made behind several Be-stainless steel and Be-Fe configurations. Also measurements were taken behind various thicknesses of stainless steel alone.

399. F. J. Allen, A. T. Futterer, and W. Wright, Neutron Reflection and Flux Versus Depth for Concrete, BRL-1189 (January 1963). Availability: $9.10 OTS as AD-403802.

Detailed calculated results on neutron reflection and flux versus depth for concrete are given in the form of machine printouts. The angular and energy distributions of the reflected neutrons along with the energy-dependent and total flux at various depths are contained in tabular form on the printouts. Neutron number current, number flux, and dose transmission as functions of thickness are also given in tabular form on the printouts.

A table of summary information on reflection is presented. This contains number current, number flux, dose and energy reflection factors as functions of incident energy and angle.

A few figures are presented to illustrate graphically the meaning of the various tabular results.

400. A. W. Casper, Data Report and Postanalysis of Battelle Source Plate Plug Tests, XDC 59-5-220 (May 8, 1959). Not for sale to public.

This report presents the fast-neutron dose rates measured behind plugs of gadolinium, iron, lead, aluminum, and titanium in the GE-ANPD Source Plate Facility at Battelle Memorial Institute. Measurements made in water along the fission plate centerline are also reported. The attenuating properties of gadolinium are compared with those of the other materials. The distribution of power in the fission plate was measured and used in the point kernel calculations, which are then compared with the measured fast-neutron dose rates. A macroscopic removal cross section of 0.108 cm^{-1} for gadolinium (of density 7.87 gms/cm^3), provides the most consistent fit to the data.

401. J. L. Meem, L. B. Holland, and G. M. McCammon, Determination of the Power of the Bulk Shielding Reactor. Part III. Measurement of the Energy Released Per Fission, ORNL-1537 (February 15, 1954). Availability: Dep.; $13.80 (ph), $4.80 (mf) OTS.

The energy released per fission was measured in the Bulk Shielding Reactor using a specially constructed fuel element. The special element had a removable center fuel plate from which discs were punched to facilitate measurement of their activation by exposure in the reactor. A similar uranium-bearing disc was activated by exposure in a known neutron flux. By comparison, the actual fission rate in the special fuel element was determined.

The special element was equipped with water tubes at top and bottom through which water was pumped at a known rate. Thermocouples were used to measure the temperature rise across the fuel element, special precautions being taken to prevent heat leakage. The power, or rate of energy production, was determined from the water flow and temperature rise, and a comparison with the fission rate gave the heat released per fission in the special element. The net amount of radiation leaking out of the fuel element (less than 0.2%) was calculated as a correction to this figure.

The energy per fission in the reactor was 193 $\pm$ 5 MeV. The parasitic capture in the Bulk Shielding Reactor is calculated to be 2.7 MeV. Accordingly, the energy released per fission (exclusive of neutrino energy) is 190 $\pm$ 5 MeV.

402. M. P. Haydon, E. B. Johnson, and J. L. Meem, Measurement of the Fast-Neutron Spectrum of the Bulk Shielding Reactor Using Nuclear Plates, ORNL CF 53-8-146 (August 31, 1953). Availability: Dep. (mc); $3.30 (ph), $2.40 (mf) OTS.

A nuclear plate camera has been developed at the Bulk Shielding Facility to measure the fast-neutron spectrum of the reactor. A proton-recoil type fast-neutron spectrometer has been used to measure the reactor spectra as attenuated by the beryllium oxide reflector and various amounts of water, and, since data can be obtained much more quickly with the spectrometer, no extensive use of nuclear plates is contemplated. However, the nuclear plate method is of established reliability, and it was considered desirable to check the spectrometer at at least one point. It was found that the total integrated flux as measured by the two methods differs by approximately 11.3%, which is within the statistical accuracy of the two instruments.

403. A. W. Casper and J. G. Carver, Dose Rates Computed for Shielding Arrays at the Battelle Source Facility, XDC 58-11-123 (November 5, 1958). Not for sale to public.

A shield fixture is described which will be used at the Battelle Source Plate Facility; it consists of a slab of iron with a cylindrical hole into which plugs of various materials will be inserted. Computations made to determine the effect of source plate size on centerline dose rates are described. The incremental dose contributions from various annular sections of the source plate are also presented. Plugs of iron, aluminum, and lead will be used in the experiments, so computations were made for these materials.

404. D. D. Babb and R. E. Dewames, An Experiment to Test the Extension of the Moments Method of Calculation to Finite Media, NARF-59-9T (MR-N-228) (March 13, 1959). Not for sale to public.

An experiment was performed at Convair-Fort Worth to determine the applicability of radiation penetration data calculated by the moments method (which assumes a single, infinite medium) to problems involving more than one medium.

In the experiment, fast-neutron and gamma dose rates and thermal-neutron fluxes were measured in the vicinity of an air-water interface, as functions of the distance from the detector to the water surface, with the source-detector separation and the height of the source-detector axis relative to the water surface as parameters. The use of artificial sources, detectors of small sensitive volume, and minimum framework resulted in the geometry approaching that of a point source and detector.

Under the assumption of an infinite water medium, dose rates were calculated from moments method data for comparison with the experimental values. The calculations involved dividing the line segment between the source and detector into a part in water and a part in air (considered a void) and determining the attenuation by the water.

For a Co^{60} gamma source, the use of moments method data was found to be completely valid for degraded energies greater than 0.52 MeV. Validity below 0.52 MeV was not explored because of the limited response of the detectors. For the most realistic geometry, namely source in water and detectors in air, the presence of a boundary caused a 7% decrease in the measured fast-neutron dose rate.

405. F. W. Smith, GTR Dose Rate Distribution in Air, NARF-55-13T (MR-N-56) (March 31, 1955). Not for sale to public.

This experiment consisted of a determination of the dose-rate distribution in air about the GTR in a cylindrical shield tank with an elliptical cross section. Fast-neutron and gamma dose rates were obtained along the reactor centerline to a maximum distance of 90 feet. Measurements were also made over one octant at a radius of 32 feet. Checks were made in the horizontal plane to ascertain the symmetry of the reactor shield.

406. H. J. Kouts and W. W. Pratt, Calibration of the Brookhaven Shielding Studies Fission Source, BNL-2796 (November 29, 1951). Availability: Dep. (mc), $3.30 (ph), $2.40 (mf) OTS.

We have devised and tested a method of calibration of the large (40-in. square)fission source used in conjunction with the Brookhaven shielding studies water tank. The result of such a calibration is an accurate knowledge of the wattage generated by fissions in the plate when it is operating at a given level. There are now two other shielding studies facilities (the Oak Ridge Lid Tank and the Oak Ridge Swimming Pool), both of which have known power generations for their fission sources. This calibration therefore makes possible direct intercorrelation of results of shielding measurements made at Brookhaven with those made at Oak Ridge.

407. P. M. Uthe, Jr., Attainment of Neutron Flux-Spectra from Foil Activations, (Thesis) USAFIT-TR-57-3 (March 1957). Not for sale to public.

Several analytical methods are described for obtaining neutron flux-spectra knowledge from foil data. The response functions used are the nuclear reaction characteristics of threshold, fast fission, and resonance type foils.

Two previously used methods and two new methods are discussed. The previously used methods are the Trice Method and the Cadmium Difference Method. The Trice Method, applicable in the fast-neutron energy region, uses a "step function" cross-section representation. The step function energy threshold is determined by assuming a differential flux function. The Cadmium Difference Method, applicable in the 1/E energy region, uses the resonance activation integrals of resonance type foils as cross-section parameters. Although consideration is given to these methods, major interest is focused on two new methods. These methods are called the Polygonal Method and the Polynomial Method.

The Polygonal Method uses the cross sections in their true form and represents the flux-energy spectrum by a combination of linear functions of energy. In performing the necessary calculations, difficulty in selecting certain energy parameters is experienced.

The Polynomial Method uses the cross sections in their true form and represents the flux-energy spectrum by a weighting function.

408. J. B. Trice, F. J. Muckenthaler, F. W. Smith et al., Two Neutron Energy Measurements in the Bulk Shielding Facility Using Radioactivants, ORNL CF 53-5-139 (October 29, 1953). Availability: Dep. (mc); $4.80 (ph), $2.70 (mf) OTS.

A series of measurements was made in the BSF in order to compare results from proton-recoil measurements, at a reactor operating power of 100 kw. Threshold measurements were obtained in the 1- to 8-MeV range by the use of (n,p) and (n,α) reactions with Al^{27}, Mg^{24}, Si^{28}, S^{32}, P^{31}, and Cl^{35}. The spectral form in this case is similar to a fission spectrum as given by Watt's experimental data. Similar measurements using (n,γ) reactions with Mn^{55}, Cu^{63}, Na^{23}, Cl^{37}, and V^{51} in the energy range 0.0001 to 0.05 MeV are in good agreement with a 1/E spectral function. Whereas spectral structure was discernible only by proton-recoil measurements, the absolute magnitudes from the two methods are comparable. In a second series of tests to measure (n,p) cross sections in O^{16} and O^{17}, neutron flux traverses in the BSF are reported, using Al, Mg, S, Co, and Au radioactivants, placed along a special fuel element through which a controlled quantity of H_2O flowed.

409. D. G. Collins and L. W. McCleary, A Systemization and Penetration Study for Straight Cylindrical Ducts, NARF-63-3T (FZK-9-182) (May 31, 1963). Availability: $4.60 OTS.

A study was conducted to determine the effect that varying the length and diameter of a straight cylindrical duct has on the neutron radiation streaming through the duct. An experiment was set up in which a polonium-beryllium source was located on centerline in the mouth of an aluminum duct penetrating a water slab. The resulting data were analyzed by use of two Monte Carlo procedures developed to analyze radiation streaming through cylindrical ducts. Results of additional calculations are shown to indicate the possibility of extending the experimental findings to duct problems involving different source energy and angular distributions and different shielding materials.

410. K. M. Campe, Geometrical Attenuation Factors for Gamma Radiation from Cylindrical and Spherical Surface Sources, PWAC-397 (September 15, 1963). Availability: Dep.; $0.75 OTS.

This report formulates a method for deriving geometrical attenuation factors of distributed surface sources, and presents them in the form of curves for cylindrical and concave spherical geometries. The factors are plotted versus the ratio, d/R, of the detector-source separation to the radius of the source. A wide range of cylinder sizes is covered by a family of curves with ratios of length-to-radius ranging from 0.5 to infinity. Curves are presented for concave spherical surfaces of various polar angles, between 10 and 90 degrees. The detector-source separation distance ranges from 0.5 to 100 cylinder radii for the cylindrical geometry, and from 0.1 to 10.0 radii for the concave spherical geometry. Beyond these detector distances, the analytical expressions representing the geometrical attenuation are approximated by simpler asymptotic forms in which the cylindrical and spherical sources can be effectively treated as line and point sources, respectively.

411. C. E. Clifford (Can.), Differential Dose Albedo Measurements for 0.66-MeV Gammas Incident on Concrete, Iron, and Lead, DRCL Report No. 412 (August 1963). Not for sale to public.

The angular dependence of the scattered dose produced by 0.66-MeV gammas incident on concrete, iron, and lead was measured for angles of incidence of 0°, 30°, and 60°. A Monte Carlo program was used to study the distribution of distances between the point of incidence and the point at which a backscattered gamma emerged as well as the depth of the point at which a backscattered gamma had its last scattering interaction. The effect of the thickness of the scatterer on the angular distribution of the scattered dose was measured for iron exposed at normal incidence.

The angular distributions of the scattered radiation from the three materials studied were found to be dependent both on the angle of incidence and on the angle of scatter to a detector and could not be related to each other by simple multiplicative factors. As the angle of incidence increased the scattering in the forward directions became increasingly anisotropic.

412. J. G. Carver, W. E. Edwards, and J. E. MacDonald, GE-ANPD Methods of Shield Design, APEX-230 (December 30, 1955). Availability: Dep. (mc) $11.00 (fs), $4.67 (mf) OTS.

A detailed description is presented of the methods of shield analysis and design used by the GE-ANPD shielding group as of November, 1955. Complete sets of working equations are included, together with derivations of the more important relationships. Basic assumptions and limitations of current methods are discussed, as are contemplated changes and improvements now under development. Tables and graphs of useful nuclear data are presented in the appendices.

413. A. J. Breslin, P. Laysen, and M. S. Weinstein, Protection Against Fallout Radiation in a Simple Structure, WT-1462 (August 1963). Availability: Dep.; $1.50 OTS.

A reinforced Butler building was exposed to fallout from Shots Diablo and Shasta, and the resulting dose rates and fallout deposition inside and outside the structure were measured with various instruments and techniques.

Protection factors and roof and ground contributions to the total dose rates at points within the structure were determined from the measurements. Comparisons were made with the results of theoretical and other experimental studies.

Information obtained from this experiment should be of value as basic experimental data for fallout protection work, although it is recommended that additional substantiative data be obtained under more controlled conditions.

414. R. E. Beissner, An Analysis of Fast-Neutron Energy-Angle Distribution, NARF-63-5T (FZK-9-186) (August 15, 1963). Availability: $6.60 OTS.

The energy-angle distributions of fast neutrons transmitted through slab shields are analyzed. Comparisons of calculated and measured angle distributions show that a six-energy group, double P_2 approximation is adequate for polyethylene-slab calculations. Phenomenological arguments are presented which show that the transmitted energy-angle distribution can be described in terms of a simple asymptotic model. Suggestions for improving the asymptotic theory are included.

415. H. M. Stainer, X-Ray Mass Absorption Coefficients - A Literature Survey, Bureau of Mines Information Circular IC-8166 (1963). Availability: free from Mines Bureau, Pittsburgh, Pa.

Published X-ray mass absorption coefficients were compiled and tabulated. These coefficients are required for various X-ray analytical techniques. Values obtained from 28 authors are listed for 87 elements with both experimental and calculated points included; the wave-lengths range from ~ 0.01 to 68 A. Representative graphs are presented for six elements (lithium, nitrogen, iron, silver, holmium, and uranium). A 37-item supplementary bibliography is also included.

416. J. Bourgeois, P. LaFore, J. P. Millot et al., "Methods and Experimental Coefficients Used in the Computation of Reactor Shielding," Second U.N. International Conference on the Peaceful Use of Atomic Energy, A/Conf. 15/P/1190 (May 16, 1958).

Some methods are described which relate to the computation of reactor shields. The methods are generally age theory and diffusion. The definition is given and use of removal cross sections is described. Removal cross sections for a large number of materials included many concretes are presented.

417. R. L. Bernick, Application of the Method of Steepest Descent to Laminated Shield Weight Optimization, NAA-SR-MEMO-8181 (April 17, 1963). Not for sale to public.

An iterative procedure is described whereby the method of steepest descent may be applied to the problem of laminated shield weight optimization. The treatment is general with no reference made to any particular materials or geometries.

418. Sigma Center, Optical Model Analysis of Inelastic Scattering of Neutrons by Heavy Nuclei, BNL-818 (T-317) (August 1963). Availability: Dep.; $1.25 OTS.

Differential inelastic scattering cross sections for Ta^{181}, W^{184}, Au^{197}, Th^{232}, and U^{238} were calculated by using Hauser-Feshbach theory in the neutron energy range from 0.1 to 2 MeV. The transmission coefficients were determined from optical model analysis of the elastic differential cross sections. The elastic differential cross sections are also presented.

419. J. W. Harris, W. P. Kunkel, and W. J. Trautvetter, Penetration of Scattered Co-60 Gammas Through the Engineered Crew Shield, NARF-55-22T (April 27, 1955). Not for sale to public.

This experiment was conducted to determine the dose rate inside the thin crew shield due to scattered Co^{60} gamma radiation. Curves are included to present data obtained as a function of the detector position relative to the crew shield for different shadow shield angles.

420. L. V. Woodruff and E. W. Hessee, Air and Ground Scattering of Gamma Rays and Neutrons from the GTR, NARF-55-19T (April 28, 1955). Not for sale to public.

The purpose of this experiment was to obtain measurements of fast-neutron and gamma dose rates due to air and ground scattered reactor radiations. For this test, the Ground Test Reactor was centered in a small water-filled aluminum cylinder of elliptical cross section to provide a nearly isotropic radiation source.

Both total and scattered (air and ground) dose-rate measurements were made for reactor-detector separation distances from 30 to 100 feet for elevations ranging from 12.5 to 71.3 feet. The scattered measurements were made by shielding dosimeters from the reactor direct beam; variation of scattered dose rate with direct beam shield angle was determined.

Results of the total dose-rate measurements verify previous preliminary studies; the dose rates for both neutrons and gammas vary roughly as the inverse square of the distance.

The functional dependence of the scattered dose rate on separation distance a changes with height above the ground. At the larger separation distances, for scattered gamma radiation, the dose rate is proportional to $1/a^{1.4}$ and $1/a^{1.2}$ at 12.5 and 71.3 feet above the ground; the scattered fast-neutron dose rate varies as approximately $1/a^{1.5}$ and $1/a$ at the 12.5 and 71.3-foot heights.

421. R. Aronson, C. N. Klahr, H. A. Steinberg, and K. Held, Effect of Shield Splitting and Placement in Nuclear Propelled Space Vehicles, TRG-136-FR (March 1, 1961). Not for sale to public.

The effect of splitting and placement of various idealized shadow shield configurations appropriate to manned nuclear rockets has been investigated. The effect on neutron dose in both hydrogenous and nonhydrogenous shield situations was considered, as well as the effect on gamma dose. The results indicate that in some circumstances proper placement or multiple splitting of the shield can affect the dose by a factor of three or more, over and above any r^2 effect due to differences in the relative sizes of reactor and payload.

422. J. R. Streetman and G. A. Graves, A Monte Carlo Calculation of Neutron Heating in a Nuclear Rocket Propellant Tank, LA-2909 (September 16, 1963). Availability: Dep.; $2.00 OTS.

The calculation described here was undertaken to establish the principal features of neutron energy deposition, neutron migration, and neutron absorption within liquid hydrogen contained in the propellant tank of a nuclear rocket. Though the geometry was simplified in comparison to the probable complexity of any operational system, the results are readily generalizable and should assist considerably in understanding such systems.

The bulk of this report is concerned with the presentation and discussion of the neutron study, but the heating values found are compared, both in magnitude and in variation with propellant depth, to those expected from the scattering of gamma rays produced from neutron captures in liquid hydrogen and to those expected from the scattering of gamma rays coming from the reactor itself. An unshielded 1000-Mw reactor system is presumed.

423. C. E. Clifford (Can.), Use of Models for Gamma Shielding Structures, DRCL Report No. 364 (February 1962). Availability: Dep. (mc).

The use of models for shielding studies of buildings exposed to gamma radiation has been evaluated by comparing the dose distributions produced in a blockhouse with movable inside walls exposed to 0.66-MeV gamma radiation with corresponding distributions in an iron 1 to 10 scale model. The effects of air and ground scaling on the readings in the model were also investigated. Iron appeared to be a suitable model material for simple closed buildings but for more complex structures it appeared that the use of iron models would progressively overestimate the gamma shielding protection as the complexity increased.

424. L. A. Beach, R. B. Theus, and W. R. Faust, Penetration of Na-24 Radiation Through Water and Mercury, NRL Report 4581 (July 26, 1955). Not for sale to public.

The dose rate as a function of penetration distance was measured for a point isotropic Na^{24} gamma-ray source in water and in mercury. The experimental results agreed with dose rates predicted by the moment method calculations of NDA Report No. 15C-41.

425. H. O. Wyckoff, A. Allisy, J. W. Boag, H. Franz et al., Radiation Quantities and Units - International Commission on Radiological Units and Measurements (ICRU) Report 10A, NBS Handbook 84 (November 14, 1962). Availability: $0.20 GPO.

The recommendations of the International Commission on Radiological Units and Measurements (ICRU) are presented. The scope and policy of the ICRU is explained. The recommendations are accompanied by the reasons for presenting them.

426. T. V. Blosser and R. M. Freestone, Jr., A Study of the Shield of the University of Virginia Nuclear Research Reactor, ORNL TM-662 (November 25, 1963). Availability: $1.60 OTS, $0.80 (mf).

Detailed measurements were made of the fast-neutron and gamma-ray dose rates and the thermal-neutron fluxes existing at the face of the biological shield of the pool-type University of Virginia Nuclear Research Reactor. Similar data were also obtained at the surface of the pool. The study was made with the ORNL Mobile Radiation Measurement Laboratory. Dose rates and fluxes were generally low, although some streaming around thermal column and access facility closures was observed. Other data obtained included measurements of dose rates at various points in the vicinity of the reactor coolant water heat exchanger, a controlled area during reactor operation.

427. G. T. Chapman and J. D. Flynn, The Effect of Iron Behind Beryllium in the ORNL Lid Tank (Expt. 38), ORNL CF 53-6-185 (June 30, 1953). Availability: $0.15 OTS.

The configurations for the experiment consisted of a 28.5-cm Be slab followed by various thicknesses of Fe. The Be slab (average density 1.23 g/cc) was made of pellets in a steel container (0.32-cm walls), and the Fe was in the form of 2.22-cm thick slabs. Because the Fe slabs were warped some water was distributed in each configuration. In order to determine the effect of this water, measurements made on 15 Fe slabs in water were compared with measurements where the slabs were in a steel dry tank (0.3-cm walls). The thermal-neutron flux fast-neutron dose and gamma dose were measured behind each configuration, and the data are presented. The detectors used for the measurements were: 3-in. fission chamber and 12 1/2-in. BF_3 counter for thermal neutrons; fast-neutron dosimeter for fast neutrons; and anthracene scintillation counter for gamma rays.

428. F. D. Flynn and G. T. Chapman, Attenuation by Lead of Fast Neutrons from a Fission Source, ORNL CF-53-3-166 (March 10, 1953). Availability: $0.10 OTS.

An experiment was carried out in the Lid Tank to determine the effect of a thick hemisphere of Pb on the actual fast-neutron dose. A 15-in. wide dry tank was inserted in the Lid Tank next to the source, giving effectively an infinite slab of air with the 28-in. diameter fission source on one side and water on the other. The dose was then measured in the water next to this tank. A hemispherical Pb shield 7-in. thick was inserted in the dry tank centered on the source axis, and the dose was remeasured in the same position. The ratio of the two readings indicated the effect of the Pb on fast neutrons arriving from the outside. The gamma dose was also measured in the same position with and without the Pb shield.

429. C. E. Clifford (USA), Westinghouse Control Plug Mockup Measurements - Neutron and Gamma Distribution in Water Containing Lead, Steel, and Air, ORNL CF 52-5-183 (May 26, 1952). Availability: Dep. (mc), $1.80 (ph), $1.80 (mf) OTS.

Measurements were completed in the Lid Tank on the short experiment to aid in the design of control-instrument plugs. The plug was designed to increase the thermal-neutron flux locally in the water beyond the pressure shell. This was accomplished by penetrating the water reflector and the steel and water thermal shield with an 8-in.-diameter steel pipe filled with lead, steel, or air. The measurements are compared with those of the unperforated shield. Neutron measurements were taken in water along the axis of the plug and also along the line perpendicular to and intersecting the plug axis. Gamma measurements were taken beyond the air-filled plug only. Neutron-induced secondary rays are predominant in determining the attenuation rate of the pressure shell.

430. H. H. Van Tuyl, F. P. Roberts, and E. J. Wheelwright, Shielding Requirements for Promethium Sources, HW-77375 (April 1963). Availability: Dep.; $1.00 OTS.

A theoretical analysis of the shielding required for large amounts of promethium was made, including an analysis of the radiation contributions from Pm^{146} and Pm^{148} at different times after reactor discharge. The results are compared with experimental determinations on samples of promethium at different ages, and promethium is compared with other radionuclides which may be used in power sources. As a first approximation, shielding requirements for promethium aged less than 2 years from reactor discharge are comparable to those for Sr^{90} or Cs^{137}. For promethium aged over 2 years, shielding requirements are comparable to those for Pu^{238}.

431. L. Burris, Jr. and I. G. Dillon, Estimation of Fission Product Spectra in Discharged Fuel from Fast Reactors, ANL-5742 (July 1957). Availability: Dep.; $5.60 (fs), $1.91 (mf) OTS.

The fission-product spectra in discharged fuel from fast fission power reactors have been calculated over a wide range of operating conditions (reactor operating times from 10 to 10,000 days and cooling times from zero to infinity). These calculations are based on recent yield data for fission of uranium and plutonium. From these calculated data are presented derived values of total curies and energy dissipation rates as functions of operating time and cooling time.

432. W. R. Smith and M. A. Turner, Nuclear Merchant Ship Reactor Shield Design Summary Report, BAW-1144-1 (August 1, 1959). Availability: Dep. $2.50 OTS.

Most of the Nuclear Merchant Ship Reactor (NMSR) shield design work performed by the Babcock and Wilcox Company (B-W) is summarized. Primary and secondary shield results are presented, consistent with contactural requirements and designated design criteria. Methods of calculation and basic parameters are shown.

433. A. Foderaro and F. Obenshain, Fluxes from Regular Geometric Sources, WAPD-TN-508 (June 1955). Availability: Dep. (mc); $16.80 (ph), $5.70 (mf) OTS.

Graphs and equations are given for computing the flux from various geometric source arrays. Most of this data has been reproduced in Reactor Shielding Design Manual (edited by Rockwell).

434. A. Tattegrain and H. Satge, Shielding Measurements on the Ulysse Reactor During Startup, AEC-tr-6010 (CEA Note No. 414) (1963). Availability: Dep. (mc); $1.60 (fs), $0.80 (mf) JCL or OTS.

These measurements were made in order to verify the calculations of shielding for the ULYSSEE reactor. We measured the thermal and epithermal fluxes and the fast flux when the latter was high enough to be measured. The gamma flux was only measured in the HO_1 thermal column. We made measurements in the horizontal channel of the North side and in the channel of the thermal column on the West side.

435. E. D. Callahan, L. Rosenblum, and J. R. Coombe, Shelter from Fallout, TO-B-60-30 (November 18, 1960). Availability: $9.10 OTS, as AD-276392.

This report summarizes the results of a survey it presented of the existing fallout shelter potential in basements and mines in the United States, and in boats on bodies of water of sufficient size and depth. Also presented is an analysis of the design, construction, and habitability of a minimum-type, improvised home basement family fallout shelter, and the shelter potential in an actual suburban community in the Northeast.

The survey shows that about 60% of the population in the U.S. would have access to basement shelter, with the figures ranging from better than 80% in OCDM Regions 1, 2, and 4 to less than 20% in Regions 3, 5, and 7. Mine shelter could be an important shelter resource for two to four million people in some 16 states, including West Virginia, Illinois, Michigan, Missouri, Kansas, Oklahoma, and New Mexico. Shelter in covered boats on lakes, rivers, and ocean is likely to provide the best available means of protection for several million people, particularly in the states of New York, Delaware, Maryland, Virginia, Florida, Louisiana, California, Oregon, and Washington.

A family-size, sand-bag fallout shelter can be readily constructed in the basement corner by one person for a materials' cost of about $60. The shelter, which offers a protection factor of 100 against outside radiation levels, can be assembled in an hour if the materials are suitably stored along the basement walls, and realistic excursion schedules appear possible after two days even in the heaviest fallout areas. A survey of public and private buildings in a typical northeastern suburban city of 25,000 population indicated that the basements of schools, churches, and other large buildings do not offer significantly better protection than that of the average home basement (i.e., about a factor of 20).

In terms of the number of people per state who do not have even remote access to any fallout shelter (including home basements), the four most needy states are California, Texas, Florida, and Georgia.

436. V. R. Cain, F. J. Muckenthaler, and C. E. Clifford (USA), Study of Gamma Ray Produced by Neutron Interaction in Air, ORNL CF 58-4-77 (May 29, 1958). Availability: Dep. (mc); $4.80 (ph), $2.70 (mf) OTS.

An experiment was performed at the Tower Shielding Facility to study the production of secondary gamma rays in air by thermal-neutron capture in nitrogen and by the inelastic scattering of fast neutrons. A beam of neutrons from the TSF reactor irradiated a cone of air, and the gamma rays resulting from neutron interactions within the cone were observed by a gamma-ray spectrometer several feet away. The ratio of the nitrogen capture gamma rays observed to the other gamma rays observed was varied by covering the reactor neutron beam with boron-impregnated plexiglas for some of the measurements. The difference between the measurements was then used to separate the pulse-height spectrum associated with nitrogen capture gamma rays from the spectrum associated with other sources. In this investigation, the spectrum which remained after the subtraction of the capture gamma-ray spectrum was presumed to be due to inelastically scattered fast neutrons; however, contributions from other sources must be considered. The results of the experiment are to be compared with theoretical calculations of the pulse-height spectra to be expected from various sources.

437. C. E. Clifford (USA), Gamma Dose Behind Iron-Water Thermal Shield of Various Thicknesses as a Function of Water Reflector Thickness (Investigation for NDA), ORNL CF 52-5-40 (May 7, 1952). Availability: Dep. (mc); $3.30 (ph), $2.40 (mf) OTS.

Gamma-ray dose-rate measurements were taken in water behind various iron-water configurations in the ORNL Lid Tank Shielding Facility. Up to six 2.2-cm-thick iron slabs, separated by 1.3 cm of water, were placed at various distances from the source.

438. C. E. Clifford (USA), Gamma Dose Behind Iron-Borated Water Thermal Shield with 18-, 20-, and 30-cm Water, ORNL CF 52-5-4 (May 7, 1952). Not for sale to public.

Gamma-ray dose-rate measurements were taken in borated water behind various iron-borated water (1% boron solution) configurations in the ORNL Lid Tank Shielding Facility. Up to eight 2.2-cm-thick iron slabs, separated by 1.3 cm of water, were placed at various distances from the source. With a 20-cm-thick water reflector and four iron slabs, the dose rate at 80 cm is reduced by boration to 70% of the pure water value.

439. D. R. Smith, R. E. Beissner, J. E. Ward, and C. F. Malone, Analysis of ASTR Centerline Dose Rate Measurements, NARF-57-8T (FZK-9-107) (April 4, 1957). Not for sale to public.

This report presents an analysis of the results of a ground-level experiment in which the gamma and fast-neutron dose rates along the Aircraft Shield Test Reactor (ASTR) centerline were measured in air for several direct-beam shield angles. Predictions were made of the dose rates to be expected due to single scattering, multiple scattering, and neutron-induced gamma radiation.

A comparison of the predicted gamma dose rates with experimental data shows reasonably good agreement in magnitude except for ASTR Configurations 1 and 5. The predicted values for these two configurations are about one-half the experimental values. Inasmuch as these configurations have the highest neutron-to-gamma ratios, a need for further investigation of neutron-induced gamma radiation is indicated.

In general, the prediction of fast-neutron dose rates agreed well with the experimental data. The slopes of the predicted and experimental curves agreed quite well. Agreement in magnitude was best for 0° direct-beam shield angles, which was expected since the method of calculating the multiply scattered dose rates was most accurate in this case.

440. E. A. Burrill, Neutron Production and Protection, High Voltage Engineering Corporation, Burlington, Massachusetts (1963).

A brief survey of neutron sources is given with emphasis on those associated with accelerators. The yield for many reactions is given in graphical form.

441. D. J. Raso, Transmission of Scattered Gamma Rays Through Concrete and Iron Slabs, TO-B-59-13 (October 9, 1959). Not for sale to public.

The penetration of gamma rays undergoing one or more scatterings in concrete and iron of various slab thicknesses (3.75 to 60.0 g/cm^2) has been investigated by the Monte Carlo technique. Radiation with energy of 350, 660, and 1250 keV and also the gamma-ray spectra produced by the products (omitting volatile materials) of U-235 fission at 1.12 and 23.8 hours after fission were assumed to be incident on the medium at angles of $\cos\theta_0 = 0$, 0.25, 0.50, 0.75, and 1.0. The case histories of 5000 photons were followed on the IBM 704 digital computer at the National Bureau of Standards, Washington, D.C. The information obtained included:

1. number and energy transmitted,
2. the energy distribution of the transmitted radiation integrated over all directions,
3. the angular distribution integrated over all energies,
4. the angular distribution of the transmitted energy, and
5. the angular distribution of the transmitted "dose".

These computed quantities refer to transmitted current and an approximation is given for determining flux density. Comparisons are made with other theories and an experiment performed at USNRDL.

442. R. L. Kloster and M. B. Wells, Neutron Cross Sections for Nitrogen and Oxygen, NARF-62-17T (MR-N-298) (May 31, 1963). Availability: $8.10 OTS.

Neutron cross sections for nitrogen and oxygen that are of primary interest in shielding and transport calculations are compiled in this report.

Cross sections for the following processes are compiled for both oxygen and nitrogen: total, elastic scattering, nonelastic collisions, inelastic scattering, differential elastic scattering, (n,α) reactions, (n,p) reaction, and (n,d) reaction.

The following cross sections are compiled for nitrogen only: gamma-ray production, (n,t) reaction, (n,2n) reaction, and (n,2α) reaction. Cross sections of nitrogen for excitation of the various levels of the residual nucleus in the (n,αj), and (n,n'j) reactions are also compiled.

443. J. P. Nichols, E. D. Arnold, and D. K. Trubey, "Evaluation of Shielding and Hazards on the Transuranium Processing Plant," Nucl. Sci. Eng. 17, 3, 486-491 (November 1963).

The shielding and containment criteria for the Transuranium Processing Plant obtained by calculations and experiments are given. The shield evaluation studies (for cell walls, cell windows, and fission source carriers) utilized experiments at the ORNL Lid Tank Shielding Facility and IBM-7090 computer calculations for determination of neutron transport, neutron activation, and gamma penetration. These studies also included an evaluation of the effects of credible accidents occurring in the facility.

444. A. B. Chilton and C. M. Huddleston, "A Semiempirical Formula for Differential Dose Albedo for Gamma Rays on Concrete," Nucl. Sci. Eng. 17, 3, 419-424 (November 1963).

A semiempirical formula is developed which yields values for the differential dose albedo of gamma rays on concrete. Gamma rays of incident energies 0.2, 0.5, 1.0, 2.0, 4.0, 6.0, and 10.0 MeV are considered. Results of the semiempirical formula are compared with values derived from Monte Carlo calculations for the backscattering of gamma rays from a semi-infinite slab of concrete.

Results show that the two-parameter formula gives satisfactory agreement with the Monte Carlo calculations.

The principal assumption involved in the theoretical analysis is that the actual reflection process can be approximated by two terms, one involving a single Compton scattering event and the other involving isotropic processes. The two parameters used involve the contributions of the two terms.

445. D. J. Raso, "Monte Carlo Calculations on the Reflection and Transmission of Scattered Gamma Rays," Nucl. Sci. Eng. 17, 3, 411-418 (November 1963).

Monte Carlo calculations were performed to determine the backscattering from and transmission through concrete of 0.2, 0.5, 1.0, 2.0, 4.0, 6.0, and 10.0 MeV gamma rays. The radiation was assumed to be incident on a semi-infinite medium and on slab thicknesses of 0.5, 1.0, 2.0, and 4.0 mfp at angles of $\cos\theta_0$ = 1.0, 0.75, 0.50, 0.25, and 0.10. The case histories of 5000 photons were followed on the IBM 704 digital computer for each of the 175 geometries (874,000 histories). The information obtained included: (1) The characteristics of emergent photons, which were stored on magnetic tape; (2) a routine that processes these characteristics to give polar and azimuthal angular dose distribution; and (3) detailed results from the application of the processing routine to the parameters investigated. The computer program, the processing routine, and the detailed results are given in Ref. 1.

446. J. J. McInerney, "Some Aspects of Neutron Slowing-Down Distributions in Hydrogeneous Media," Nucl. Sci. Eng. 17, 3, 392-397 (November 1963).

Some aspects of the relation between the energy dependence of the scattering cross section and the spatial distribution of neutrons slowing down in a hydrogeneous moderator are discussed. The neutron population in an infinite medium is analyzed using the mean square slowing down distance, $\overline{r^2(E)}$, as a measure of its spatial extent. It is shown that, contrary to what is perhaps expected, the neutron population in many cases is distributed closer to its source with decreasing energy. In the analysis of this behavior it is found that diffusion theory may in some cases be expected to give good results close to the spatial position of the source. A correction term is also found for the "first fight" transport kernel often used in practical calculations.

447. M. J. Berger and J. Doggett, "Reflection and Transmission of Gamma Radiation by Barriers - Semianalytic Monte Carlo Calculation," J. Res. Nat. Bur. Std., 56, 2, 89-98 (February 1956).

The transport equation for photons which have been Compton-scattered repeatedly with a specified sequence of energy losses and deflections was solved analytically, by an "orders of scattering" approach, to obtain conditional reflection and transmission probabilities. These probabilities were then averaged over all possible intermediate angular and energy configurations in a Monte Carlo calculation carried out on the NBS automatic computer (SEAC). The efficiency of the Monte Carlo calculation was increased by taking advantage of the close correlation between the transmission of radiation through thick barriers and the corresponding penetration problem in an infinite homogeneous medium for which an exact solution was available. The reflected and transmitted energy flux (integrated over all spectral energies and directions) has been calculated for 0.66-, 1-, and 4-MeV radiation incident on water barriers, and 1-, 4-, and 10-MeV radiation incident on iron, tin, and lead barriers. Angular and energy spectra were obtained for water at 0.66 MeV. The ratio of the scattered energy flux, transmitted through a barrier to the scattered flux at equal depth in an infinite medium, was found to be constant for distances from the source greater than four mean free paths of the source radiation.

448. S. Tomoeda, A. L. Frank. and M. B. Hastings, The Effect of Source Distance on Buildup Factor for Gamma Rays Penetrating into a Compartmented Structure, USNRDL-TR-681 (October 8, 1963). Availability: $3.60 OTS as AD-424880.

In order to correlate some basic measurements of radiation attenuation made on an aircraft carrier to other situations and geometries, experiments were carried out to study the effects of source distance on the dose distribution inside a model-sized compartmented structure intended to simulate the carrier.

Data are presented in the form of buildup factors (B_r) as a function of source-to-detector distance for two configurations of the compartmented structure. Buildup factors are also presented as a function of the slant path through the interposing steel plates for the case where the source is considered far from the structure.

Comparisons of the buildup factors within our complex structure with the more basic single-slab dose-buildup factors obtained by other investigators are also presented.

In every case, the buildup factor for the compartmented structure was found to be consistently and significantly lower than the single-slab data. The greatest differences in buildup (B_r-1) amounted to 30 percent.

Starting with the source irradiation point against the center of the structure front face and moving away along the perpendicular to the front face, the buildup factor at the detector position of the center compartment nearest the source increased smoothly while the buildup factors at the detector positions of the two further center compartments remained approximately constant. The buildup factors of all the detector positions in the noncenter compartments decreases smoothly. At a distance of about 20 in., or roughly twice the largest dimension of a compartment, the buildup factors for all detector positions had become nearly constant so that doubling or tripling the source-to-detector separation distance produced practically no change in B_r.

Among the several effects that contribute to the buildup, the slant path of the primary radiation and the effects due to the geometry of the detector and slab appear most important.

449. H. J. Kopp, "Synthetic Method Solution of the Transport Equation," Nucl. Sci. Eng. 17, 1, 65-74 (September 1963).

A general residue iteration method (the synthetic method) is described and applied to the solution of the neutron transport equation. For example, a possible solution is the synthesis of a sequence of solutions to diffusion equations, each having a residue source. Only time independent systems consisting of isotropic scattering material of arbitrary geometry are considered herein. Exact solutions, approximate solutions, and the refinement of existing approximate methods can be accomplished within the framework of the synthetic method.

450. G. C. Pomraning and M. Clark, Jr., "Orthogonal Polynomial Angular Expansion of the Boltzmann Equation," Nucl. Sci. Eng. 17, 1, 8-17 (September 1963).

The angular dependence of the solution of the monoenergetic Boltzmann equation in slab geometry with isotropic scattering is expanded classically in the set of Jacobi polynomials which are orthogonal in the interval -1 to +1 with respect to the weight function $w(\mu) = (1 - \mu)^{\alpha}.(1 + \mu)^{\beta}$. The low-order solution obtained by retaining only the first two terms in the expansion is investigated in detail. In this low order it is shown that a proper choice of α and β leads to the exact asymptotic transport eigenvalue. With this choice of α and β a significant improvement in the linear extrapolation distance and the critical size of a bare slab over the usual (P - 1) diffusion theory is obtained. However, it is shown that, in general, the truncated set of classical Jacobi equations does not conserve neutrons. A modification in the truncation procedure is made in order to obtain neutron conservation while retaining the advantages of the Jacobi expansion. The choices $\alpha = \beta = -1/2$ and $\alpha = \beta = -1$ are discussed in some detail and shown to have advantages over the corresponding Legendre ($\alpha = \beta = 0$) expansion.

451. T. V. Blosser and R. M. Freestone, Jr., A Radiation Study of the Westinghouse Testing Reactor, ORNL TM-332 (December 11, 1963). Availability: $2.60 OTS, $0.98 (mf).

Gamma-ray, fast-neutron, and thermal-neutron dose-rate measurements utilizing instruments of high sensitivity have been made over a major portion of the biological shield of the Westinghouse Testing Reactor (W.T.R.) at Waltz Mill, Pennsylvania. Observed rates were well below the rate of 1 mr/hr predicted for the shield for a reactor power of 60 Mw. Selected regions of the area within the W.T.R. security fence were examined to determine radiation levels due to gamma-ray "shine" from the elevated primary coolant tank. Dose rates ranging from 0.08 mr/hr to 10 mr/hr were observed. The existence of a considerable fast-neutron flux in the reactor coolant exit stream at distances as great as ~ 245 ft from the core was established, but the source for these neutrons was not definitely determined.

452. T. V. Blosser and R. M. Freestone, Jr., A Study of the Shield of the Virginia Polytechnic Institute UTR-10 Reactor, ORNL TM-290 (December 12, 1963). Not for sale to public.

A study of the radiation leakage from the shield of the Virginia Polytechnic Institute UTR-10 nuclear reactor was completed in February of 1962. Fast-neutron, thermal-neutron, and gamma-ray dosimeters of high sensitivity were employed to obtain a detailed network of data over all surfaces of the shield. Supplementary experiments evaluated gamma-ray dose rates in a corridor and in a classroom adjoining the reactor room and also determined a rough value for argon dispersal through the reactor room ventilating system.

The results of the detailed survey showed generally satisfactory performance of the shield for fast- and thermal-neutron radiations, but a number of regions were identified in which the gamma-ray dose rate was higher than that predicted by the shield design.

453. A. B. Chilton, Gamma-Ray Reflection Factors from Concrete, CONF-39-9 (June 1963). Availability: AED-CONF-63-060-42, Gmelin Institute, 40-42 Varrentrappstrasse, Frankfurt (Main), West Germany.

The field from a radioactive point source near a concrete-air interface was mapped using the differential dose albedo formula devised by Chilton and Huddleston. The conditions for the validity of the results are that the distances involved are larger than a mean free path in concrete, but substantially smaller than a mean free path in air.

454. K. Sehested, A. Brynjolfsson, and N. W. Holm, Determination of the Absorption Curve in Water of Co-60 Gamma Rays and Determination of the Strength of a Six Curie Co-60 Source by Means of Ferrous-Sulphate Dosimetry, RISO Report No. 62 (May 1963). Availability: Dep. (mc), Riso, Roskilde, Denmark.

The gamma-ray dose rate in water from a Co-60 source was measured with a Fricke chemical dosimeter. The experimental results were compared with a calculation using the moments method results reported by Goldstein and Wilkins.

455. D. L. West, Data Report and Analysis of Fast-Neutron Attenuation in LiH, XDC 60-1-17 (December 30, 1959). Not for sale to public.

Fast-neutron dose-rate measurements were taken within a 36-inch slab array of LiH at the GE-ANPD Source Plate Facility operated by the Battelle Memorial Institute. Analysis of these data indicates that the fast-neutron removal cross section of the Li in LiH is of the order of .07 cm^{-1} for a density of 5.64×10^{22} lithium atoms/cm^3, which represents a microscopic removal cross section of 1.24 barns.

456. R. H. Chesworth, Army Gas Cooled Reactor Systems Program - ML-1 Shielding Design Report, IDO-28609 (UC-80) (October 1963). Availability: $2.75 OTS.

The ML-1 is a closed cycle, gas-cooled nuclear power plant developed under the Army Gas-Cooled Reactor Systems Program (AGCRSP) to demonstrate the feasibility of such a plant for military field use. The shielding was designed to comply with the ML-1 performance specifications which stipulated that the biological shield should attenuate the radiation to certain specified levels within 24 hours after the reactor was shut down from full-power operation. Additional shielding was to be provided as possible within the size and weight limitations to reduce levels associated with reactor operation.

Preliminary designs, final design, analysis, and experiments which resulted in the final design for the ML-1 shield, as well as preliminary evaluations of shield performance, are described.

The work reported herein was performed by Aerojet-General Nucleonics under Contract AT(10-1)-880 between Aerojet-General Corporation and the U.S. Atomic Energy Commission.

457. L. J. Cannon, L. A. Schmittroth, Computer Generation and Testing of Random Numbers, IDO-16921 (August 1963). Availability: $1.00 OTS.

This report discusses the need for random numbers particularly in regard to Monte Carlo applications and the two methods most commonly used to obtain random numbers. These methods are the multiplicative congruential method and the mixed congruential method, and both methods involve somewhat arbitrary parameters. Proofs for the lengths of period for both methods and a derivation of an expression for the serial correlation coefficient in terms of the parameters of the mixed congruential method are presented. The results of a series of statistical tests made to determine satisfactory parameters are listed. The length of run test was found to be the most powerful test in that it rejected the largest number of parameters as unsatisfactory. The multiplicative method (for the parameters tested) was found superior to the mixed congruential method. The multiplicative method multiplier $\lambda = 2^{18} + 3$ satisfactorily passed all tests and has a short multply time.

458. D. W. Green, Attenuation of Gamma Radiation in a Two-Legged 11-Inch Rectangular Duct, USNCEL Technical Report R-195 (NP-11657) (May 2, 1962). Not for sale to public.

Results are presented of experiments carried out with an 11 x 11-in. duct in concrete with one right-angle bend, using Co^{60} as a gamma-ray source. Experiments cover the effects of moving the source and detector off the center-line position, of replacing the inside concrete corner lip with one of lead, and of removing various scattering areas in the corner. Measured values of attenuation factors for the corner and second leg are compared with calculated values, using the LeDoux-Chilton analysis. The comparisons suggest that the analysis provides an adequate treatment of the single-scatter effect in the duct. Multiple-scatter effects, however, which are not taken into account by the analysis, apparently are responsible for approximately half of the radiation reaching the detector. An empirical expression is obtained which adequately represents most of the experimental results of duct measurements available at the present time.

459. V. R. Cain, Calculations of Thermal-Neutron Flux Distributions in Concrete-Walled Ducts Using an Albedo Model with Monte Carlo Techniques, ORNL-3507 (January 21, 1964). Availablity: Dep.; $2.75 OTS.

When radiation shields are penetrated by ducts, the gamma rays resulting from the radiative capture of low-energy neutrons in the duct walls can be principal contributors to the total dose along the duct. In order to calculate the effect of these capture gamma rays, the distribution of low-energy neutrons in the ducts must be known. This report presents calculations of low-energy neutron distributions in concrete-walled ducts by a method in which an albedo model and a Monte Carlo technique were used. One series of calculations was performed for straight ducts so that comparisons could be made with the Simon-Clifford analytic approximation. Another series was for three-legged rectangular ducts similar to those used in a Tower Shielding Facility experiment. When an albedo similar to that to be expected for a pure thermal-neutron source was used, the results agreed very closely with the TSF data. A digital computer code was written to perform the calculations.

460. J. Certaine, H. Goldstein, M. H. Kalos, and P. S. Mittelman, Penetration of Neutrons from a Point Fission Source Through Carbon and Hydrocarbons, NDA 12-18 (June 30, 1956). Availability: Dep. (mc); $3.30 (ph), $2.40 (mf) OTS.

Moment calculations have been performed to study the penetration of fission spectrum neutrons through carbon and hydrocarbons. For pure carbon, the neutron spectrum as a function of penetration depth and the "age" to 1.44 eV have been determined. For hydrocarbons having the formulae CH, $CH_{1.9}$, $CH_{2.0}$, and $CH_{2.1}$ the fast-neutron dose has been determined as a function of penetration depth and an effective removal cross section of carbon has been evaluated.

461. R. G. Cochran, J. D. Flynn, K. M. Henry et al., Reactor Radiations Through Slabs of Graphite, ORNL CF 54-7-105 (July 30, 1954). Availability: Dep.; $4.80 (ph), $2.70 (mf) OTS.

Measurements have been completed to determine the attenuation of large thicknesses of graphite next to a reactor. These measurements are of interest for evaluating a graphite reflector as a shield component, and they also provide a direct comparison with LTSF determinations of the carbon removal cross section. Graphite thicknesses of 1, 2, and 3 ft were used, and the usual gamma-ray, thermal-neutron, and fast-neutron dose measurements were made behind each slab thickness. In addition, the fast-neutron spectrum (above 1.3 MeV) through 1 ft of graphite was measured.

462. J. I. Marcum, Monte Carlo Calculations of the Transport of 14-MeV Neutrons in the Atmosphere, RM-3531-PR (April 1963). Availability: $10.50 OTS as AD-405401.

This report presents new results for the transport of 14-MeV neutrons in the atmosphere. Two corrections have been made to the RAND neutron Monte Carlo transport code, and curves from several previous reports are presented along with the new results given by the corrected code so that comparisons can be made and the previous incorrect results can be changed.

For the first time, results are given for the transport of 14-MeV neutrons near an air-ground interface and also in an infinite homogeneous air medium.

An auxiliary smoothing and graphing code has been developed for use in conjunction with the Monte Carlo transport code. This code is described and sample results are shown.

463. J. Certaine and R. Aronson, Distribution of Fission Neutrons in Water at the Indium Resonance Energy, NDA-15C-40 (June 15, 1954). Availability: Dep.; $2.60 (ph), $2.00 (mf) OTS.

Calculations have been made by the moment method of the neutron flux distribution at 2.03 eV from a point isotropic fission source in water. Comparison has been made with flux measurements by Hill, Roberts, and Fitch at the indium resonance energy (~ 1.44 eV). The shapes of the two flux curves agreed beyond 10 cm. However, the maximum of the theoretical flux curve is ~ 5 cm from the origin while the experimental maximum is ~ 7.5 cm away. The age, defined herein as $r^{2/16}$ (r being the distance from the origin), is calculated to be 26 cm^2, compared to a measured value of 31 cm^2.

464. J. F. Scoles, Calculated Gamma-Ray Spectra from U^{235} Fission Products, NARF-58-37T (FZK-9-132) (August 29, 1958). Availability: $24.30 (fs), $7.50 (mf) OTS as PB-142179.

The gamma rays from fission products remaining in a reactor after operation are usually the most important contributors to the dose from a shutdown reactor or spent fuel assembly. To improve the accuracy of shielding calculations, the spectra of these gamma rays were calculated for various combinations of reactor operation time and time after shutdown. A literature survey was performed to obtain U^{235} thermal fission yields and decay information. These data were used in a digital computer program to obtain the gamma-ray energy in each of 28 energy groups. The results obtained are substantially more accurate than previous calculations of the same type. The improvement was obtained primarily by using more recently available nuclear data, by extending the length of the decay chains considered, and by a finer division of energy groups.

465. D. H. Jansen, E. E. Hoffman, and D. M. Shepherd, "Lead-Lithium Shielding Alloy - Metallurgical Studies," J. Nucl. Mater. 1, 3, 249-258 (1959).

Alloying and casting studies conducted on a lead-0.69 wt% lithium alloy, a proposed shielding material, are presented. The information obtained during this investigation was used in connection with the making of an experimental, laminated slab, 90 cm square and 10 cm thick, of the lead-lithium alloy. This slab was used to determine the neutron and gamma-ray shielding characteristics of the material. Shielding tests have demonstrated the effectiveness of the alloy in suppressing secondary gamma-ray production.

Mechanical property tests were conducted on the lead-lithium alloy and the results compared to those for a lead- 0.06 wt% copper alloy, also being considered for shielding applications. It was found that the lead-lithium alloy has superior strength at room temperature, but the lead-copper alloy has superior creep strength at 110°C.

Tests at 110°C in air indicate that the lead-0.69 wt% lithium alloy oxidizes more rapidly than pure lead. The lead-lithium alloy is attacked by water with an accompanying volume increase and cracking. Corrosion was much more severe in boiling water than in room-temperature water.

The lead-lithium alloy has good rolling characteristics. No cracks or surface defects were observed on a piece which was rolled to 92.5% reduction.

466. C. M. Park, C. B. Agnihotri, and J. Silverman, Interim Report on Scattering of Gammas Through Ducts, TID-17014 (UMNE-2) (June 1, 1962). Availability: Dep. (mc); $5.60 (fs), $1.97 (mf) OTS.

An area-wise dose-rate calculation method using differential angular dose albedos is presented for application in radiation shielding design. A FORTRAN program employing differential dose albedos was set up to calculate the dose rate resulting from singly reflected photons in a straight rectangular duct and an L-shaped two-legged rectangular duct. A sample calculation is included.

467. V. R. Cain, A Study of the Radiation Shielding Characteristics of Basic Concrete Structures at the Tower Shielding Facility, ORNL-3464 (January 21, 1964). Availability: Dep.; $1.75 OTS as AD-428230.

In the first of a series of experiments performed for the Department of Defense to investigate the protection afforded by various typical structures against prompt weapons radiation, radiation-intensity measurements were made at the Tower Shielding Facility in two concrete-shielded bunkers and in an interconnecting tunnel. Prompt weapons radiation was simulated by the Tower Shielding Reactor II (TSR-II), which was operated 100 ft above the ground. The distance between the reactor and the bunkers was approximately 700 ft. The bunkers were each 12-ft cubes and were constructed so that the shield thickness on the front face of one and on the top face of the other could be varied in 4-in. steps from 0 to 20 in. The thickness of concrete and dirt surrounding all other faces was sufficient to make them black to incident radiation.

The immediate goals of the experiment were to study (1) the attenuation of radiations by various thicknesses of ordinary concrete slabs, (2) the buildup of radiation intensities within the cavities by scattering of radiation in the walls, and (3) the transmission of radiation down a tunnel with two right-angle bends. The gamma-ray and fast-neutron dose rates and thermal-neutron fluxes measured at various positions within the bunkers and in the tunnel and the pulse-height spectra from a 3-in. sodium iodide crystal determined at one position in the top bunker and one position in the tunnel are reported.

468. W. L. Bunch, Radiation Shielding Program at Hanford, HW-78274 (April 27, 1963). Availability: Dep.; $0.75 OTS.

The experimental shielding program carried on at Hanford has been built around two reactor facilities which can be used as sources of reactor radiation. A cross-sectional drawing is shown of one of two wells located in the top shield of the DR reactor. A graphite reflector and cast iron thermal shield are located permanently between the reactor and these wells. The wells are dry. Slabs of the material under investigation are removed and replaced only while the reactor is not operating. Suitable detectors are placed between the slabs to measure the neutron or gamma distribution.

The 36-inch shield plug facility is located on the side shield of the C reactor. This is a dry facility, similar to the two DR wells, and differs primarily in the fact that only a graphite reflector remains permanently between the facility and the reactor. The slabs of material under investigation are loaded vertically into a crate which is moved into and out of the shield opening by a hydraulic piston.

Calculations performed at Hanford are briefly described.

469. Z. Haider, M. K. Husain, and F. Shaikh, "Marbolite - A New Synthetic Material. Part II. Gamma-Ray Shielding Measurements on Marbolite and Other Materials," Pakistan J. Sci. Ind. Res. 6, 2, 90-92 (April 1964).

Gamma-ray shielding (absorption and scattering) properties of a new synthetic material, "Marbolite," have been studied and compared with other shielding materials, especially lead and natural marble. For temporary shielding, Marbolite turns out to be much cheaper than lead and marble for the same percentage absorption. Ease of handling and fabrication in any desired size and shape coupled with acid and alkali resistant property, makes it an excellent shielding material for radiochemical laboratories.

470. A. G. Pieper and L. A. Beach, Stochastic Estimates of the Penetration of Gamma Rays Through Slab Shields, NRL Report 5981 (September 26, 1963). Availability: $3.60 OTS.

The energy penetration and dose-rate buildup factors have been calculated by Monte Carlo techniques on a digital computer for a plane, monodirectional source of gamma rays of 1.0, 2.5, and 6.0 MeV incident at various angles upon finite slab shields of lead, iron, ordinary concrete, and water. The exponential transformation and equivalent integration over the spaticl variable were utilized to obtain results for slabs up to 18 mean-free-path lengths in thickness. A simple expression was derived which relates a given buildup factor at any incident angle to the corresponding buildup factor at normal incidence.

471. M. L. Randolph, "First Collision Doses Produced by Fast-Neutron Fluxes of Wide Energy Distributions," Radiation Res. 19, 3, 492-511 (July 1963).

Analytic methods for evaluation of the ratios of first-collision dose to fast-neutron flux for five classes of neutron spectra have been developed. Results in terms of rads in tissue and bone have been calculated for examples of each distribution and in terms of rems for all but one distribution. Doses for several other hydrogenous materials may be obtained by multiplying tissue doses by stated factors. The neutron distributions describe fission spectra, evaporation spectra, multigroup spectra, and spectra resulting from cosmic rays. Vagaries in this method of dosimetry are discussed briefly. The ratio of dose in bone to that in tissue is found to be 0.44 in every case examined. A fast-neutron dose rate of 10 to 20 millirems/day in the upper atmosphere is estimated.

472. E. P. Blizard, T. B. Cook, Jr., P. S. Harris, et al., Nuclear Radiation Criteria for Hardened ICBM Systems, STL/TR-59-0000-00735 (December 1959). (SECRET)

473. K. Parker, Neutron Cross Sections of Carbon in the Energy Range .025 eV to 15 MeV, AWRE Report No. O-71/60 (August 1961). Availability: Dep. (mc).

Recommended values are given for the neutron total and partial cross sections of carbon in the energy range 0.025 eV - 15 MeV based on available experimental and theoretical evidence. Details of energy and angular distributions are given so that the data are complete for the purpose of neutronics calculations.

The values given in this note are available on punched cards as part of the AWRE Punched Card Library of Neutron Cross Sections.

474. M. A. Schmoke, Attenuation of Fallout Radiation as a Function of Concrete Blockhouse Wall Thickness, NDL-TR-43 (October 1963). Availability: $9.10 OTS as AD-431063.

This experiment was conducted to verify theoretical calculations of wall thickness effect on the shielding characteristics of a concrete blockhouse in a uniformly contaminated fallout field.

Two gamma emitters, cobalt 60 and cesium 137, were used to simulate uniform planes of contamination. The dose rates at various locations within blockhouses with wall thickness of 48 psf, 93.7 psf, and 139 psf were measured with ionization-chamber dosimeters. Reduction factors were calculated from the data taken at the center detector positions and compared with reduction factors computed from the theoretical calculations of Dr. L. V. Spencer, National Bureau of Standards.

1. Experimental and theoretical reduction factors 3 feet and 6 feet above the center of the concrete blockhouse agreed within ±15 percent for a uniformly contaminated plane of cobalt 60, and within ±20 percent for cesium 137.

2. Cobalt 60 and cesium 137 radiation show approximately exponential attenuation of dose rate as a function of wall thickness ranging from 48 to 139 psf for detector heights of 0 (ground level), 3, and 6 feet.

475. L. Degelman, A. Foderaro, and G. Kowal, Evaluation of the Shielding Characteristics of Structures for Simulated Residual Radiation, Pennsylvania State University NUC E 15 (November 1963). Availability: $4.60 OTS as AD-432876.

A semiempirical expression has been developed for the gamma-ray dose rate at any point within a concrete structure resulting from a point isotropic source lying anywhere on the ground outside the structure. This point-to-point kernel has been integrated over an infinite area to check existing theories of fallout protection and to determine the variation of dose rate as a function of position within the structure.

476. G. D. Joanou and H. Fenech, "Fast Neutron Cross Sections and Legendre Expansion Coefficients for Oxygen-16," J. Nucl. Energy Parts A, B 17, 10 (December 1963).

The fast-neutron cross section for oxygen-16, as well as the Legendre expansion coefficients for the angular distribution of elastically scattered neutrons, has been reported by Joanou et al. (1961). Since the publication of that report, new experimental data on the (n,α) and (n,p) cross sections have been reported (Harvey, 1959; Bettis Atomic Power Laboratory Report, 1962) and have motivated the present paper. The recent results obtained by Phillips (1962) on differential elastic scattering were used in the determination of the Legendre expansion coefficients. The total cross section has been revised to include the three resonances at the laboratory neutron energies of 1.651, 1.830, and 1.903 MeV. No data are as yet available on the differential elastic scattering cross section for neutrons in the energy range 1.75 to 2.00 MeV, but since this energy region is of importance in reactor physics, an analysis using the Blatt-Biedenharn (1952) formalism was made to determine the angular distribution of elastically scattered neutrons and the corresponding Legendre expansion coefficients in this range.

The procedure adopted to obtain the Legendre expansion coefficients from the available experimental data was the same as previously reported by Joanou et al. (1961).

477. C. S. Pemberton, Scavenging Action of Rain on Non-Wettable Particulate Matter Suspended in the Atmosphere, FWE-233 (AWRE O-53/58) (February 1959). Availability: United Kingdom Atomic Energy Authority. (CONFIDENTIAL)

478. E. W. Beale, Computation of Fallout Pattern. Part I. General Theory, FWE-219 (AWRE E1/58) (August 1958). Availability: United Kingdom Atomic Energy Authority (CONFIDENTIAL)

479. C. L. Storrs and J. M. Miller, Shielding Properties of Uranium, ORNL CF 53-9-96 (September 30, 1953). Availability: Dep. (mc); $4.80 (ph), $2.70 (mf) OTS.

A basic study of the shield properties of uranium was made. A slab 3 in. thick by 3 ft square was placed in the water of the ORNL Lid Tank at various distances from the source plate. Gamma-ray and thermal-neutron measurements at each position with (a) no thermal neutron on the slab, (b) a 3/16 in. slab of plexibor on the source side, and (c) the plexibor on both sides. Thus, the contribution to the measurements from fissions in the slab were determined.

480. T. V. Blosser, R. M. Freestone, Jr., and J. M. Miller, Study of the Shield of the University of Illinois TRIGA Mark II Research Reactor, ORNL TM-178 (April 23, 1962). Not for sale to public.

Detailed measurements have been made of the fast-neutron and gamma-ray dose rates and the thermal-neutron fluxes existing at the surfaces of the biological shield of the University of Illinois TRIGA Mark II Research Reactor. Dose rates and fluxes were found to be extremely low. A comparison was made by means of threshold foil techniques between the fast-neutron flux in a beam hole of the reactor during steady-state operation at 1 kw and during a nominal 250 Mw, 30 msec pulse produced by rapid insertion of ~ two dollars of excess reactivity. The flux over the duration of the pulse was approximately four times that for steady-state operation for 1 hr at 1 kw. Fast-neutron dose rates and thermal-neutron fluxes were measured at each of the four beam ports of the reactor at operating powers of zero, one and five watts. Some gamma-ray dose-rate data was also obtained at zero and 1 watt. An appendix briefly describes the threshold foil technique used.

481. M. K. Hullings, The Effect of Some Liquid Metal Ducts on Reactor Shields, ORNL CF 53-11-53 (February 12, 1954). Availability: Dep. (mc); $10.80 (ph), $3.90 (mf) OTS.

A series of neutron attenuation experiments designed to test engineering mockups of liquid-metal ducts have been completed. The primary purpose of the test was to facilitate calculations of additional shielding required by loss in efficiency of a reactor shield when perforated with liquid-metal ducts.

The test was performed at the Thermal Column Facility located on the ORNL Graphite Reactor (X-10 Pile) with ducts having one or two 90-deg bends.

482. N. F. Shoemaker and C. M. Huddleston, "Economy of Experiment in Dose Albedo," Nucl. Sci. Eng. 18, 1, 113-115 (January 1964).

Treatments of the differential dose albedo of gamma rays on concrete have supposed that the albedo value is a function of the energy of the incident gamma radiation, the polar angle of incidence, the polar angle of reflection, and the azimuthal angle of reflection. It is demonstrated here that, if certain assumptions (which appear reasonable) are made regarding the mechanism of reflection, it is not necessary to investigate variations in albedo with azimuthal angle of reflection. Once differential dose albedo has been determined for a complete set of incident and reflected polar angles with zero azimuth, albedo at any azimuth can be derived by a suitable transformation.

483. W. K. Talley, Analytic Expressions for Calculating Neutron Distribution Functions by Monte Carlo, NP-13117 (1962). Availability: (Thesis) University Microfilm, Inc., Ann Arbor, Mich., $2.75 (mf), $5.00 xerox, Order No. 64-9098.

In Monte Carlo calculations, the successive directions and energies of a particular history of a neutron slowing down in a homogeneous medium can be determined without reference to the locations of the collisions. For each set of such discrete directions, the analytic form of the spatial distribution of any particular collision is developed by integrating over that of the previous collisions. Not only is the statistical error of such a method less than that of ordinary Monte Carlo, but an analytic expression for the energy distribution of the last collision enables one to calculate a sample of the vector flux density at chosen positions and energies.

Previously, the use of analytic spatial distributions has provided scattering probability densities for plane source neutrons in an infinite homogeneous medium and for a plane source on the boundary of a finite slab. The solution to the more general problem of a plane source anywhere within the finite slab is presented here; however, the main portion of this thesis treats the more complicated three-dimensional problems involving point sources.

The scattering probability distribution functions for a point source in an infinite and semi-infinite medium are developed, and the spatial moments of the former are given. The results for the plane source are shown to follow those from the point source. The extension of this method to finite bodies is discussed. The problem of round-off error (due to machine truncation) is examined, and a method to circumvent it is set forth.

484. C. L. Bennett, Shielding Properties of As-Cured Barite Concrete, HW-71113 (September 20, 1961). Availability: Dep. (mc); $2.60 (fs), $1.04 (mf) OTS.

This preliminary report contains experimental data on neutron and gamma attenuation by barite concrete in the as-cured condition. The program was designed to test barite concrete in the as-cured condition and heated to 300°C as part of Hanford's continuing shielding studies of high-density concretes.

The mean density of the barite concrete tested was 3.51 g/cm^3.

The fast-neutron attenuation was found to be approximately exponential. The measured relaxation lengths for Tests 1, 2, and 3 of barite concrete in the as-cured condition are respectively 10.55 cm, 10.65 cm, and 11.69 cm. This difference in relaxation lengths was attributed to the water loss (0.07 g/cm^3) that occurred while the barite concrete was in the test wells, the water being driven off by heat absorbed from the thermal shield.

In the first test a measured fast-neutron removal cross section of 0.0892 cm^{-1} was obtained for a concrete density of 3.55 g/cm^3. This value agrees within about 7 percent with the calculated removal cross section of 0.0831 cm^{-1}. In the last test the variation between experimental and calculated values was about 6 percent. These variations can be attributed largely to uncertainties in the composition of the concrete.

485. N. M. Newmark, Design of Openings for Buried Shelters. Chapter 5. Effects Input Data. Chapter 6. Radiation Shielding Design Parameters, NP-13087 (July 1963). Not for sale to public.

The shield design section is a very simple introduction to shelter shield design problems for the engineer new to the field.

486. J. F. Perkins, Decay of U^{235} Fission Products, RR-TR-63-11 (AD-415052) (July 25, 1963). Availability: $1.60 OTS.

Decay properties of mixed fission products from thermal fission of U^{235} have been calculated by a method employed earlier but with revised input data describing decay of individual nuclides. Input data and calculated decay rates and rates of beta and gamma energy release are presented.

487. W. C. Ingold, Some Applications of a Semiempirical Formula for Differential Dose Albedo for Gamma Rays on Concrete, USNCEL TN 469 (November 30, 1962). Availability: $2.60 OTS as AD-291133.

This theoretical study illustrates the usefulness and versatility of a semiempirical formula through its application to three specific problems:

1. In generating a table of albedo values for various combinations of angles of scatter.

2. In a study of first reflection contribution to dose rate in a round duct compared to that in a square duct.

3. In the development of a relatively simple and unsophisticated program for multiple scatter contributions.

488. T. R. Fowler and C. H. Dorn, Gamma-Ray Attenuation in a 12-Inch Diameter Round Concrete Duct, USNCEL TN-465 (November 5, 1962). Availability: $3.60 as AD-289349.

Dosimeter and Geiger tube measurements were made along the centerline of a 12-inch diameter circular concrete duct with two right-angle bends. Measurements showed that the buildup factor in the first leg was the same as that found in earlier work for the square duct. As with the square duct, the dosage received down one leg of the duct follows closely the functional form Kx^{-m}, where x is the distance along the axis of the duct, and K and m are empirical constants. The value of m for a round duct has been found to be about 10 percent smaller than that found for square ducts of similar size. The attenuation of the corner, however, is greater for a round duct than for a square one. A theoretical treatment of the first leg using single-scattering theory is included in the appendix.

489. H. Mochizuki, Y. Tanaka, Y. Higashihara et al., Experimental Research on the Penetration of Co-60 Gamma Ray Through Finite Homogeneous and Heterogeneous Media, NP-tr-1139 (August 1962). Availability: Dep.; $4.60, $1.61 (mf) JCL.

Dose-rate buildup factors were obtained from the results of measurement of Co-60 gamma-ray transmission through multilayers of water, iron, and lead. An empirical formula for the buildup factor for multilayered shields was obtained on the basis of the experimental results obtained and by a generalization of the formula derived by Kalos for two layers. The experimental and calculated results agree well for thickness less than six mean free paths.

490. Z. G. Burson, Experimental Radiation Measurements in Conventional Structures. Part II. Comparison of Measurements in Above-Ground and Below-Ground Structures from ... Fallout Radiation, CEX-59.7B (February 1963). Availability: Dep.; $1.50 OTS.

An experimental study designed to provide a basis for estimating protection against fallout radiation was made on two types of structures at the Nevada Test Site. This study was sponsored by the Civil Effects Test Operations, Division of Biology and Mecidine, U.S. Atomic Energy Commission. The two buildings studied were a lightly constructed building with a basement, and an underground group shelter.

An idealized fallout radiation field was simulated by the use of the Mobile Radiological Measuring Unit (MRMU). The unit employed a sealed radioactive Co^{60} source that was pumped at a uniform speed through a long length of flexible tubing evenly distributed over the area of interest. Radiation levels at selected points inside the structures were measured with sensitive ionization-chamber detectors.

These measurements were compared with measurements taken under actual fallout conditions at an earlier time and were also compared with the theoretical calculations.

Protection factors from fallout data and MRMU data at the basement structure compared roughly within a factor of 2. This was good, considering the limitations of the two sets of data and other factors affecting the differences. Comparisons between protection factors from fallout data and MRMU data at the underground group shelter were excellent. MRMU data and theoretical calculations also compared satisfactorily.

491. F. J. Shore, R. D. Schamberger, and H. P. Sleeper, Jr., The Streaming of Neutrons in Carbon Steel, Including the Geometrical Dependence, BNL-265 (December 1, 1953). Availability: Dep. (mc); $4.60 (ph), $2.70 (mf) OTS.

A study has been made of the streaming of neutrons through rectangular carbon steel slabs which were placed near the bottom of the water tank. The principal aim of the investigation was to find an empirical relationship correlating the streaming in the steel with the dimensions of the steel. Such a relation was found and is discussed. The validity of the empirical law, however, might conceivably be limited to a particular kind of energy spectrum. Experimental evidence for its independence of the incident energy spectrum is presented. For the purpose of building patches to shield against the streaming neutrons, the shapes of the flux distributions around the end of the slabs are important. Information useful in this connection is given. The data obtained for right circular cylinders are discussed. The connection with the empirical law for rectangular slabs is indicated. A brief study is made of the decreased streaming noted when thin water gaps are included between slabs. Finally, experiments which were done to determine the energy and origin of the streaming component are described. In the discussion, an effort is made to correlate the results of the previous sections.

492. R. L. Kloster, Analysis of Lead Shadow Shield Equipment, NARF-57-9T (FZK-9-113) (February 27, 1957). Not for sale to public.

Gamma dose rates behind lead shadow shields in water were calculated and compared with experimental measurements. The Ground Test Reactor served as the source for these measurements. The results are shown as functions of three variables: source-shield separation distance, shield thickness, and source-detector separation distance.

The dose rates are analyzed in relation to the geometrical arrangement of the lead shield, detector, and source. Interrelationships of the dose rate components (primary source gammas, water-capture gammas, lead capture gammas, and gammas scattered around the shadow shield) are discussed.

The analysis showed that primary and water-capture gamma rays are the important contributors to the total dose rates for most realistic shadow shield designs. However, under certain circumstances, lead capture and scattering around the shield become important.

493. J. M. Chapman, Computer Calculation of Dose Rates in Two-Legged Ducts Using the Albedo Concept, USNCEL R-264 (October 24, 1963). Not for sale to public.

This report gives the results of calculations of gamma-ray dose rates in two-legged rectangular concrete ducts. The calculations were performed on an IBM-1620. They are based on the differential dose albedo and include multiple scattering effects. The results of the calculations are compared with experimental data for ducts whose widths vary from 11 in. to 6 ft, using Co-60, Cs-137, Au-198, and Na-24 gamma-ray sources. The calculated dose rates agree to within ± 30 percent for all ducts and all sources, except for small ducts with Cs-137 sources, ducts with very short first legs ($L_1/W \leq 1.33$) and for Au-198 sources.

494. N. R. York, R. L. MacNeil, and R. J. Brodeur, The Effect of Limited Strips of Contamination on the Dose Rate in a Multistory Windowless Building. Vol. IV. 20 PSF Wall and 80 PSF Floor Thickness, TO-B 62-49 (July 31, 1962). Not for sale to public.

The effects of limited field of contamination on the dose rate within a multistory structure are investigated using the modeling technique in this fourth volume of reports on Contract OCS-OS-62-14. The model structure represents a six-story building similar to an apartment or office building. It has a wall mass thickness of 20 psf and a floor mass thickness of 80 psf. Otherwise, it is the same structure described in Vol. II of this series.

The results are presented as fractions of infinite field dose for different areas of simulated contamination. The results of this study verify the conclusion of Vol. II: presentation of finite-field correction factors as a function solely by the ratio W_c/h is incorrect for structures with thick floors. Allowance must be made for the shadow effect of the thick floor on direct radiation. The finite-field correction factor presently used in the National Shelter Survey Computer Program overestimates the actual dose received on upper stories from finite fields of contamination located on the ground.

495. D. Spielberg, Shielding Characteristics of Air, Soil, Water, Wood, and Other Common Materials. Vol. I. Air Scattering of Gamma Rays and Neutrons. Vol. II. Attenuation of Gamma Rays and Neutrons, WKNL-89 (February 28, 1957).. Availability: Vol. I, $3.00 OTS as PB-151164; Vol. II, $3.00 OTS as PB-151165.

A large amount of data is given for air scattering and attenuation in soil, sand, granite, limestone, sandstone, wood, water, and ice. Neutron and gamma-ray air scattering, based on single scattering with no attenuation, was calculated for three types of source-shielded geometries for many energies. The results are given in Vol. I. The attenuation data is presented in Vol. II for isotropic plane sources. The gamma-ray data was based on published moments method results. The neutron data was generated using Age Theory and Bethe-Tonks-Hurwitz theory for non-hydrogeneous materials and a multigroup procedure for hydrogeneous materials. Secondary gamma-ray results are also given.

496. R. A. Cameron, Jr. and P. H. Huff, Radiation Shielding and Response Studies of AEC Test Structures, WT-1723 (December 10, 1963). Availability: Dep.; $1.25 OTS.

The two primary objectives of Project 34.2 were to determine the variation of the radiation dose in different positions inside four small shallow-buried reinforced-concrete test structures and to compare the accelerations inside two of these structures with the accelerations in the adjacent free field. Film badges, chemical dosimeters, and energy-dependent film badges were placed throughout all four of the structures. In addition, in two of the structures, a self-recording Ballistic Research Laboratories accelerometer was located in a corner on the floor. Two more accelerometers of the same type were located in the ground about 20 ft from the two stations.

Even for the small structures considered in this test, considerable variation was observed in the radiation dose for the various locations in the structures. The average radiation readings in the structures were higher by a factor of about 3 than those predicted. The energy-dependent film badges, which also were strongly direction dependent, indicated that the radiation near the bottom of the structures was approximately isotropically distributed. However, calculated doses at individual badge locations based on the isotropic distribution did not agree well with observed doses.

In all cases the peak accelerations recorded inside the shelters were higher than the corresponding free-field peaks. Consequently it is recommended that until further information is available, the predicted free-field peak motions (acceleration, velocity, and displacement) be multiplied by 1.5 to predict the corresponding peaks inside structures similar to those observed in this test.

497. C. R. Mehl, A Monte Carlo Calculation of the Neutron Flux from a Monoenergetic Point Source in Air, SC-4174 (TR) (April 1958). Availability: Dep. (mc); $19.80 (ph), $6.30 (mf) OTS.

A Monte Carlo calculation of the penetration of 6, 5, 3.5, 2.5, 1.5, 0.7, 0.4, 0.1 MeV point isotropic sources in air has been carried out. Anisotropic elastic scattering and inelastic scattering were accounted for in the calculation. Splitting was utilized to increase the collision density at the deeper penetrations. Flux information was determined from crossings of detector spheres. Angular and energy distribution data is given. The results agree fairly well with moments method data of Holland and Richards.

498. W. E. Thompson, J. M. Ferguson, and R. L. Mather, Neutron Distributions Near an Air-Soil Boundary, USNRDL-TR-478 (November 18, 1960). Availability: $8.10 OTS as AD-247598 or PB-158319.

Given an air burst of a nuclear weapon at height x, then for horizontal distances from ground zero greater than x the vertical spatial distribution of neutrons is assumed to remain essentially invariant. A method applicable in such regions, using discrete space-energy "bins" and a number of assumptions, is described by which energy and spatial distributions of neutrons above and below an air-soil boundary can be predicted. A number of results derivable from the theoretical treatment are compared with experimental results obtained in the field.

The method found involves the iterative "multiplication" of matrices representing spatial and energic scattering probabilities. The method is restricted to distances from ground zero greater than the burst height. A machine computation was performed using the composition of Nevada Test Site soil as parameters. This solution compares favorably with experimental results obtained at NTS.

499. F. M. Tomnovec and P. A. Read, The Effect of Collimation on the Resolution and Peak-to-Total Ratio of a NaI(TL) Crystal, USNRDL-TR-543 (January 31, 1962). Not for sale to public.

Restricting the incident radiation to a small area of a NaI(Tl) scintillation crystal improves the resolution and peak-to-total ratio of the spectrometer. This effect was studied with a series of tapered lead collimators, all with the same apex but with differing apertures, in front of a 4 in. by 4 in. diameter NaI(Tl) scintillation crystal. Sources of gamma rays with energies from 0.28 to 2.76 MeV were located at the apex of the taper. Optimum resolution of about 25 percent greater than broad-beam resolution was obtained with a 0.25-in. diameter aperture. Anomalous effects were observed with a smaller opening.

500. R. Aronson, J. Certaine, H. Goldstein, and S. Preiser, Penetration of Neutrons from a Point Isotropic Fission Source in Water, NYO-6267 (NDA 15C-42) (September 22, 1954). Availability: Dep.; $4.80 (ph), $2.70 (mf) OTS.

The method of moments has been used to compute fast-neutron doses and differential number spectra in the energy range 0.33-18 MeV out to 120 cm of water for point isotropic fission neutron sources. Four problems, which differ in the mesh width used in the numerical integration and in the treatment of the term describing scattering by oxygen, have been done. It is concluded that for hydrides of elements heavier than oxygen the heavy component can be taken as having infinite mass. The attenuation length in the 60-120 cm range is 10 cm. A short discussion of the general shape of the spectra and of the effect of the details of the oxygen cross section is included.

501. R. Aronson, J. Certaine, and H. Goldstein, Penetration of Neutrons from Point Isotropic Monoenergetic Sources in Water, NYO-6269 (NDA 15C-60 (December 15, 1954). Availability: Dep.; $9.30 (ph), $3.60 (mf) OTS.

Doses and differential number spectra at distances of up to 120 cm are presented for neutrons in water from several point isotropic sources. For most of the problems the sources emitted nearly monoenergetic neutrons, in narrow energy bands centered respectively at 2, 4, 6, 8, 10 and 14 MeV. Energy degradation on scattering by oxygen was ignored in all but two 14-MeV source problems. One of these used the nearly monoenergetic type of source, and the results showed the effect of degradation to be small. The other had a monoenergetic delta-function source. However, the numerical procedure proved inadequate to handle the problem because of the strongly forward scattering by oxygen at 14 MeV.

For the low-energy portion of the spectra (from 300 keV down to 50 keV for some problems) it was found that spatial equilibrium is established quite close to the source. The same was observed for a fission source at energies as low as 2 eV. It is shown that in the 2 eV - 300 keV region the energy dependence can be given by a simple formula in terms of the hydrogen cross section. At distances of 30 cm and more from the source the spatial dependence of the spectrum closely parallels that of the fast-neutron dose curve.

502. F. G. Moote, "Dose Rates from Rectangular Plaques," Nucleonics 19, 7, 102-103 (July 1961).

The analytical procedure for calculating the dose rate from rectangular sources is given. The linear form of the buildup factor for a Co^{60} source in a water medium was found to give the best empirical fit. Curves of solid angle fraction and attenuation factors are given.

503. J. H. Smith and M. L. Storm, "Generalized Off-Axis Distributions from Disk Sources of Radiation," J. Applied Physics 25, 4, 519-527 (April 1954).

In the field of shielding it is frequently necessary to calculate biological radiation doses at any point in space due to disk sources of radiation. Such a source is the leakage radiation from an end of a cylindrical reactor. If $G(R)$ is the dose at distance R from a unit point source (e.g., $G \sim e^{-\sigma R}/R^2$) then the dose due to a disk source of radius "a" with source strength $S(\rho)$ is given by

$$D(z,a,\epsilon) = \int_0^a \int_0^{2\pi} G[(z^2 + \rho^2 + \epsilon^2 - 2\epsilon\rho \cos\theta)^{\frac{1}{2}}] \; S(\rho) \; \rho d \rho d\theta,$$

where (z,ρ,θ) are the usual cylindrical coordinates, $z = 0$ is the source plane, and $(z,\epsilon,0)$ locates the dose point.

This integral is quite general, arising in many physical problems. In counter measurements, use of $G = z/(4\pi R^2)$ $S = 1$ yields the solid angle subtended by the counter. In neutron physics the slowing down density of neutrons from a circular source of finite radius can be calculated in the age approximation with $G \sim \exp(-R^2/\tau)$.

Since for most G's the integral cannot be evaluated in closed form, there are derived three alternative series for $D(z,a,\epsilon)$ each of which has a direct physical interpretation and a different region of convergence. The results are obtained for a class of analytic functions $G(R)$, the possible singularities of which occur only at $R = 0$ for finite R; and for $S(\rho)$ of the form $\Sigma_p, S_p \rho^{2p}$. The approach in each case is to expand either the double integral or one of the two repeated integrals in a power series in $\cos\theta$, ϵ, or ρ prior to performing the detailed integrations. Numerical results are given for the case of a typical shielding function.

504. D. P. Osanov and E. E. Kovalev, "A Shield for Sources of Rectangular-Form Gamma Radiation," Soviet J. At. Energy (Engl. Transl.) 6, 503-506 (1960).

The integral

$$\int_{\phi} \int_{\rho} \frac{d\phi \, d\rho}{\rho^2 + b^2} \exp\left[- \mu d \sqrt{\rho^2 + b^2} / b \right]$$

which is the attenuation function for a detector located a distance d from a plane source has been evaluated for various rectangular sources and shield thicknesses. The shield thickness is μd mean free paths. Graphs of the results are given for four detector distance ratios and five source dimension ratios for μd between 0 and 7. Comparisons with line and point source attenuation functions are also given.

505. D. R. Doty, Dose Measurements for Neutron Streaming in Ducts, USNCEL R 282 (March 26, 1964). Availability: $3.60 OTS as AD-436183 (DASA 11026).

The purpose of this report is to describe an operating facility and present some preliminary findings concerning the streaming of neutrons in ducts of protective shelters. The experimental results are compared to a simplified theoretical treatment using the albedo approach. Dose measurements were obtained in a 3 x 3 ft concrete duct for thermal, 2.5-MeV, and 14.7-MeV neutrons using a 12-in. spherical energy-independent dosimeter. In the second leg, for all energies, the dose was found to have a higher drop-off than the theory predicted. Reasons for the discrepancy are examined and a comparison of effective albedos is made for various energies.

506. W. Zobel, Experimental Determination of Corrections to the Neutron Activation of Gold Foils Exposed in Water, ORNL-3407 (March 21, 1963). Availability: Dep.; $6.60 (fs), $2.12 (mf) OTS.

The corrections required to deduce unperturbed neutron fluxes from measurements with foil absorbers have been determined experimentally for the case of gold foils exposed in a light-water medium. The results are in good agreement with the theoretical predictions of Dalton and Osborn and of Ritchie and Eldridge.

507. R. J. Engelmann, Rain Scavenging of Particulates, HW-79382 (December 1, 1963). Availability: $3.00 OTS.

The process of collection of particles by falling raindrops is discussed in detail, and related to the flow field of air about a raindrop, as inferred from the literature. An extensive literature review treats the theoretical predictions of collection efficiencies, and the contributions to collection by both the wake of the drops and by electrical charges. The present state of knowledge of scavenging by impaction is then summarized. Several experiments are conducted using artificial rain in the outdoors with zinc sulfide tracer particles. Comparisons are made to the theoretical predictions of Langmuir, and to the measurements with freely falling drops made by Kinzer and Cobb, Walton and Woolcock, and McCulley *et al*.

508. V. R. Cain, C. E. Clifford (USA), and L. B. Holland, Measurements of Radiation Intensities in Vertical Concrete-Lined Holes and an Adjoining Tunnel at the Tower Shielding Facility, ORNL-3513 (March 1964). Availability: Not for sale to public.

Fast-neutron and gamma-ray dose rates within 4-ft-diam, 20-ft-deep, concrete-lined holes have been measured at the ORNL Tower Shielding Facility. The radiation source was the Tower Shielding Reactor II (TSR-II) enclosed in a shield which modified the neutron to gamma-ray ratio of the reactor leakage spectrum to more closely resemble that of a weapon spectrum. The holes were located at horizontal distances of 100, 228, and 450 ft from the reactor. From the hole at 100 ft extended a reinforced concrete-lined tunnel, 6 ft high, 2 1/2 ft wide, and 20 ft long, with its ceiling 10 ft below ground level. The experimental measurements consisted of vertical traverses in the three holes and horizontal traverses in the tunnel. The parameters varied included distance from the reactor, the angle of elevation of the reactor with respect to the horizontal at the hole, and the material and thickness of the shield over the hole. Reactor elevation angles ranged from 15 to 90°. The shields over the holes were concrete, iron, and laminated iron and concrete slabs.

509. C. M. Huddleston and W. L. Wilcoxson, Gamma-Ray Streaming Through Ducts, USNCEL R 289 (February 6, 1964). Availability: $4.60 OTS as AD-430603 (DASA-11026).

A survey is presented of the current status of experimental and theoretical investigations of the problem of gamma-ray streaming through air ducts in concrete. Data are tabulated and plotted for a variety of experiments. Comparisons are made between theory and experiment, inconsistences are pointed out, and areas needing further investigation are indicated.

510. J. R. Askew and R. J. Brissenden, Some Improvements in the Discrete Ordinate Method of B. G. Carlson for Solving the Neutron Transport Equation, AEEW-R 161 (August 1963). Available from Her Majesty's Stationery Office, London, ENGLAND.

This report gives an account of the DSN method for simulating neutron transport, together with methods of solution developed to deal with problems in the physics of thermal reactors, for which previously available computer programmes were unsatisfactory. The methods described are those incorporated in the programmes WINFRITH DSN written in FORTRAN language for the IBM 7090 and STRETCH computers.

511. K. T. Spinney, Neutron Attenuation in Concrete, AERE T/R 2507 (1958). Availability: From Her Majesty's Stationery Office, London, ENGLAND.

The observed behavior of the penetrating neutrons in a concrete shield is accounted for by a simple theory making use of a calculated 'transport' cross section for fast neutrons. This method is then linked with conventional age theory in an empirical manner so that a good fit is obtained to the observed neutron attenuation at all distances in the shield.

512. A. Langsdorf, Jr., R. O. Lane, and J. E. Monahan, Neutron Scattering Angular Distribution, ANL-5567 (Rev.) (October 1961). Availability: dep. (mc); $13.00 (fs), $5.87 (mf) OTS.

The accompanying 41 graphs present the results of a survey of the angular distribution of neutrons scattered from 36 elements, two compounds, and one alloy. The data are presented in the form of an expansion in a series of Lengendre polynomials,

$$\sigma(\theta) = \frac{\sigma_t}{4\pi} \sum_{\ell=0}^{4} \omega_\ell \, P_\ell(\cos\theta) \quad ,$$

in the laboratory coordinate system. The graphs give σ_t, ω_1, ω_2, ω_3, and ω_4 as a function of neutron energy.

Since the detector system used BF_3 proportional counters immersed in an oil moderator so arranged as to have constant (within 10%) counting efficiency for neutrons of all energies from 100 keV to 1500 keV, $\sigma(\theta)$ is the total differential scattering cross section.

The actual scattering detectors were positioned at five scattering angles: 22°, 54°, 92° (on the transmission side of the plate scatterer), 112° and 144° (on the reflection side). Therefore $\sigma(\theta)$, as given by the above expansion, will be correct at these angles even if the value thus extrapolated to other angles is slightly erroneous.

The angular distribution is measured usually at about 80-keV intervals for each element; the resolution was ~ 100 keV at the lower energies, decreasing to ~ 60 keV at the higher incident energies. For the transmission angles the angular resolution is ~ 5°, while for the reflection angles it is ~ 3°.

A statistical analysis of part of the data, as well as an observation of the general consistency, indicated standard deviations of ~ ± 0.05 for ω_1 and ± 0.10 for ω_4, with the accuracy of ω_2 and ω_3 being intermediate between these values. These deviations are roughly independent of the values of the ω_ℓ. In most cases, the total cross section, σ_t, is probably accurate to better than 10% statistically.

513. J. J. Griffin, Beta Decays and Delayed Gammas from Fission Fragments, LADC-6002 (1961). Not for sale to public.

A theoretical study is made of the gross behavior of beta decays following nuclear fission in times from 10^{-2} to 10 seconds. First a simple model is formulated to describe the situation in terms of a few parameters. Then the most uncertain of these parameters are chosen to fit the observed rate of delayed gamma emission (assumed proportional to the beta-decay rate) for the U^{235} (n,f) process. The description is extended to other isotopes by assuming that they differ only by small shifts of their initial fragment distributions away from that appropriate to neutron-induced fission of U^{235}. The result is a theoretical summary of the presently available data which can be used in predictive extrapolation of that data to situations not yet studied experimentally.

514. J. T. Lence, R. R. Liguori, and A. Lowery, Hand Calculation of Neutron Flux, APEX-652 (August 1961). Availability: Dep. (mc); $3.60 (fs), $1.19 (mf) OTS.

Empirical fits of neutron flux distributions were obtained for a large number of shielding configurations (LiH, BeO, and stainless steel). In shield tests, foils were exposed so that fluxes might be separated into a subcadmium flux, an epi-cadmium flux, and one or two fast fluxes. The objective was to develop simplified hand calculations to approximate these experimental results.

515. D. Spielberg and A. G. Duneer, Jr., Dose Attenuation by Soils and Concrete for Broad, Parallel-Beam Neutron Sources, AN-108 (May 1, 1958). Availability: $12.00 (fs), $5.18 (mf) OTS.

This report presents the theory and results of certain neutron shielding studies performed to supply information primarily for underground bomb shelter design. Specifically, for neutron source energies of 0.025 eV, 0.5 MeV, 2.5 MeV, 7.5 MeV, 10 MeV, and 14 MeV, and various angles of incidence upon the ground, results are presented that show the total neutron plus secondary gamma dose rates which would be received by individuals located in shelters at various distances below ground. To cover the entire range of possibilities, four basically different soil types were analyzed with water contents ranging from zero to saturation. This range also includes concrete. Also, dose rates were calculated for depths ranging from 30 to 1,000 grams/cm^2.

The information presented is considered to be "engineering-type" data of sufficient accuracy for the intended use. The multigroup theory which was utilized has obvious deficiencies in that certain nuclear scattering characteristics are simplified or ignored. Therefore, the results should not be construed as "scientific-type" data where comparison with experiment can result in gaining some insight regarding the primary nuclear scattering processes.

516. W. A. Biggers, L. J. Brown, and K. C. Kohr, Space, Energy, and Time Distribution of Neutrons at the Ground-Air Interface, LA-2390 (January 29, 1960). Availability: Dep.; $7.60 (fs), $5.51 (mf) OTS.

The work described in this report is the result of attempts to correlate neutron data obtained during several years by Los Alamos Scientific Laboratory.

A calculation, named NHM for Neutron, Hydrodynamic, Monte Carlo code, has been done on the IBM 704 with the following objectives, the results of which are included in this report.

1. To determine if such a method is sufficient, with present cross-section knowledge, to give neutron fluxes and spectra from nuclear devices as a function of distance and time.

2. To ascertain the validity of assumptions of the location of the source of gamma rays from neutron capture in nitrogen.

3. To record the number and energy of neutrons entering and leaving the ground as a function of distance and time.

All of the above were calculated for the case of a 10 to 15 kt fission device detonated 300 feet above Nevada soil.

The degree of agreement with experimental data is shown.

517. C. F. Ksanda, A. Moskin, and E. S. Shapiro, Gamma Radiations from a Rough Infinite Plane, USNRDL-TR-108 (January 18, 1956). Availability: $1.60 OTS as PB-159458 (AD-113588).

Various calculations have been made relating the gamma radiation intensity above a land area to the activity concentration on the ground. These calculations have generally assumed the ground to be a smooth infinite plane. It has been recognized that the natural roughness of any real surface will reduce the radiation intensity from the ideal smooth-plane value, and various correction factors have been proposed to account for this effect. In this report it is suggested that the assumption of a uniform mixture of radioactivity through a thin top layer of the ground surface, whose effective thickness depends on the degree of roughness, offers a good approximation to the actual case.

For a given uniform activity concentration on the ground, this assumption results in computed radiation intensities which may differ markedly, at certain heights, from those computed for the smooth-plane case. If a radiation detector is moved upward from the surface, its distance from all the radioactive sources is increased. Where these sources are on a smooth plane, the path lengths in the attenuating medium (air) are also increased. Consequently, the radiation intensity above a smooth plane always decreases with increasing height. For the mixture case, the path lengths in air also increase with increasing height, but the path lengths through the ground decrease. As a result, the intensity also decreases with increasing height if the measurement is made over a region from which no radioactivity has been removed. However, if measurements are made over a small area from which the radioactivity has been removed, the intensity first increases rapidly with increasing height, reaches a maximum value, and then decreases.

A practical application where this behavior is of importance is in the determination of the radiation intensity in buildings resulting from radioactive contamination of the surrounding terrain. In this case the area covered by the building itself corresponds to the cleared area.

518. M. B. Wells, "A Comparison of Some Recent Calculations of Neutron Transport," Health Physics, 8, 543-549 (1962).

Monte Carlo computations performed by the Rand Corporation, the Los Alamos National Laboratory, the Sandia Corporation, and General Dynamics/Fort Worth to determine the transport of 3 and 14 MeV neutrons in air and air-ground geometries are compared, and the differences between these computations are discussed.

Air-transport calculations for a 3-MeV neutron source were performed at GD/FW, with the neutron cross sections used in the Sandia and Los Alamos calculations serving as cross-section input data. The GD/FW results were found to be in good agreement with the Los Alamos air-ground calculations, but to disagree with the Sandia calculations for air only.

Monte Carlo calculations of the transport of neutrons from a point isotropic fission source in air, as made at GD/FW, are compared with results obtained from a moments-method calculation performed at the Nuclear Development Corporation of America. The differences between these calculations are discussed.

519. M. B. Wells, A Monte Carlo Calculation of Gamma-Ray and Fast-Neutron Scattering, FZM-2027 (October 27, 1960). Not for sale to public.

The results of a Monte Carlo calculation of the scattering of gamma rays and fast neutrons from a point isotropic source in an infinite homogeneous medium of air are presented in terms of energy spectra, angular distributions, and tissue dose rates at source-detector separations of 180 to 1500 yards. The calculations were made for gamma-ray source energies of 0.5, 1, 2, 4, and 6 MeV, and neutron source energies of 0.75, 2, 4, 6, 8, 10, and 14 MeV as well as for a fission-neutron source.

520. M. B. Wells, Monte Carlo Calculations of Fast-Neutron Energy Spectra in Air, FZM-1867 (October 14, 1959). Not for sale to public.

Results are presented of Monte Carlo air-scattering calculations for source-neutron energies above the threshold energy of inelastic scattering. Energy and angular distributions of the scattered neutron flux and angular distributions of the scattered dose rate are given for point isotropic monoenergetic sources at source-detector separation distances of up to 200 feet. The results of this study are compared with other calculations made by Holland and Richards and by Zerby in order to evaluate the effect of inelastic scattering on the scattered energy spectrum.

521. M. B. Wells, Monte Carlo Analysis of Air and Ground Scattering of Gamma Rays, FZM-1046 (November 19, 1957). Not for sale to public.

Results are presented of a Monte Carlo analysis of air and ground scattered gamma flux from a point isotropic source. The results are based on a program, coded for the IBM 704, that allows for multiple scattering, arbitrary separation distances of source and detector, and arbitrary heights of the source-detector system. For purposes of this program, the ground is assumed to be concrete. Results are in good agreement with measured dose rates made with a nondirectional anthracene scintillation counter from a 70-curie Co^{60} source.

522. G. H. Peebles, Gamma-Ray Transmission Through Finite Slabs, R-240 (December 1, 1952). Availability: Rand Corp., Santa Monica, Calif.

The quantity which is considered in this report to be of first importance is the buildup factor for a monoenergetic, monoangular beam of photons incident on a slab of finite thickness but infinite extent. This quantity is obtained for a set of discrete values of slab thickness, of incident energy, and of incident angle. Slab thickness, in mean free paths, spans the interval (0,20); incident energy, in units of mc^2, the interval (1,20); and the incident angle, in degrees, the interval (0,90). Two materials are used for the slab -- lead and iron.

The processes used to obtain the set of "buildup" factors and to establish their accuracy within reasonable limits are such that a considerable amount of supplementary information is produced. This information is also presented. The supplementary results consist in the probabilities and expected energies corresponding to the transmission of photons with exactly one, two, or three scatterings; distributions of the transmitted photons over energy and angle; some results for slabs of air and of the pure Compton scatterer; some buildup factors in an energy range just below that stated above; photon and energy densities for three source configurations; and other quantities which on occasion might prove interesting or useful. Also, a discussion of the effect of variation of the total absorption coefficient on transmission probabilities is given which leads to estimates of buildup factors for a material of arbitrary atomic number Z.

523. F. H. Gregory, Calculation of Radiation Dose Levels for the Army Pulse Experimental Reactor Assembly, BRL Memorandum Report No. 1438 (November 1962). Availability: OTS $2.60 as AD 402147.

Calculations are made to estimate, as a function of distance, the radiation dose levels which would occur during full capacity operation of the Army Pulse Reactor Assembly (APRA) scheduled for construction at Aberdeen Proving Ground, Maryland. Calculations of the neutron dose are made by two methods. The first calculation (Method A) is made by utilizing an experimentally determined effective dose mean free path, which probably includes the effect of ground scattering. The second calculation (Method B) is made by utilizing the result of a Monte Carlo calculation of neutron transmission in air. The results of this calculation indicate considerably higher dose levels than indicated by Method A. Experimental data are available which substantiate the result of Method A. The gamma dose at distances of interest is found to be unimportant when considered in light of the uncertainty in the neutron dose.

524. M. J. Bina, The Transmission of Fast Neutrons Through Iron, Aluminum, and Lead (Thesis), AD-236498 (March 1960) Availability: Air Force Institute of Technology, Wright-Patterson AFC, Ohio.

The dose transmission of polonium-beryllium neutrons through iron, aluminum, and lead was investigated. A Hurst type, tissue equivalent, neutron detector was used in conjunction with an electronic integrator circuit which gave directly a number proportional to neutron dose rate. It was experimentally determined that the transmission number was a function of source-detector distance and a method was developed whereby the transmission number could be obtained for an effective infinite source-detector separation. The absorbing material consisted of concentric hemispherical shells 15/16 inches thick and three configurations were possible; the spherical absorber, the hemispherical absorber, and the hemispherical reflector. Five shells were used for each configuration with a total resulting thickness of 4 11/16 inches. The dose transmission through iron, aluminum, and lead was found to be an exponential function of thickness for the hemispherical absorber configuration and, from this data, relaxation lengths were calculated to be 14.5 cm for iron, 23.6 cm for aluminum, and 20.6 cm for lead. Data was taken for aluminum in the hemispherical reflector and spherical absorber configurations. The results show that approximately four inches of aluminum constitute an infinite reflector for fast neutrons and as a consequence, the relaxation length for the aluminum spherical absorber changes to that of the hemispherical absorber for thicknesses greater than four inches.

525. J. Butler, The Status of Theoretical Methods for Reactor Shield Design, AEEW-R-361 (1964) Availability: Dep. $1.10 (BIS); 5s.6d. (HMSO)

The current techniques employed for the design of shields of power reactors in the U.K. are reviewed. The general approach is based on the new RASH E removal age-diffusion code in conjunction with the line of sight corrections for streaming. The validity of this technique has been established by comparison with theoretical spectra computed by transport codes and with integral experimental data from a wide range of bulk shield and duct configurations in the LIDO Panel facilities at Harwell. As an example of its application, the theoretical predictions are compared with measurements made during the commissioning and operation of the gas cooled Magnox reactors at Calder Hall and Chapelcross.

526. J. L. Meredith, Method of Evaluation of Experimental Radiation Measurements Over a Rectangular Source, NDL-TR-11 (April 1961) Availability: Nuclear Defense Lab., Edgewood Arsenal, Maryland.

Data analysis of field radiological decontamination tests involves evaluation of detector response over finite test areas. A method of calculation is derived for the theoretical detector response over a rectangular source by numerical integration of the function dA/r^2, where A is the source area and r is the distance between an element of the source area and the detector. Numerical results for a rectangle of dimensions of 20 by 100 feet and a detector height of 3 feet are presented. Effects of uneven distribution of the contaminant over the source area are considered, and a method of statistical analysis of data is derived.

It was concluded that:

1. The theoretical relative detector response, I/K, for any position over a rectangular source can be calculated by numerical integration of the integral $I/K = \int_A dA/r^2$. The results agree with computations done by other methods of integration.

2. Data obtained from this project's field decontamination tests of rectangular areas can be analyzed for the effectiveness of decontamination by methods developed in this report.

527. C. Eisenhauer, An Engineering Method for Calculating Protection Afforded by Structures Against Fallout Radiation, NBS 7810 Availability: National Bureau of Standards, Washington, D. C.

This report is a discussion of the technical assumptions underlying the methods currently recommended by the Office of Civil Defense (OCD) for calculating protection afforded by structures against fallout radiation. It shows in detail how each technical chart in the OCD Professional Manual is derived from basic data on radiation penetration developed by Dr. L. V. Spencer. Charts from the Engineering Manual and relevant curves from Spencer's work are included in this report in order to make it self-contained.

528: U. S. Naval Ordnance Laboratory Report, The Monte Carlo Method: Neutron Reflection by Water, AD-220676 (NAVORD Report 6227) (January 29, 1959). Availability: U. S. Naval Ordnance Laboratory, Silver Springs, Maryland.

The Monte Carlo Method with applications of interest to the physicist is discussed. Papers are listed which:

1. Utilize the technique in the solution of physics problems, and

2. Desribe methods of increasing the efficiency of the technique.

Comments are made on the general usefulness of the technique, its limitations and pitfalls. The treatment is introductory rather than comprehensive.

The Monte Carlo Method is applied to the problem of estimating the probability of normally impinging monochromatic neutrons scattering back from deep circulating pure water.

529. G. T. Western, Energy and Angular Distribution Experiment. Vol. II. Angular Distribution of Fast-Neutron and Gamma-Ray Number-Energy Flux from Slabs, NARF-62-16T (September 30, 1963). Availability: $6.60 (OTS) as AD-295718.

An experiment in which a narrow-beam slab geometry was used to project radiation from the Aerospace Shield Test Reactor onto test materials of 3% borated linear polyethylene and of lead was performed at the Nuclear Aerospace Research Facility. Except as noted, measurements for polyethylene and lead were made of (1) the radiation source terms; (2) the flux distribution of thermal and fast neutrons within the test materials; (3) the angular distribution of thermal- and epithermal-neutron flux, fast-neutron dose rate, and secondary gamma rays from the test materials; (4) the reflection of neutrons from 3% borated polyethylene and from steel; (5) the angular distribution of fast-neutron number-energy flux; and (6) the angular distribution of gamma-ray dose rate and number-energy flux resulting from primary gamma rays scattered from the test materials.

Data for items 1, 2, 3, and 4 were published in Volume I of this report; data for items 5 and 6 are presented in this volume, Volume II.

530. F. J. Allen, A. T. Futterer, and W. P. Wright, Neutron Transmission Versus Thickness for Some Common Materials, BRL-1174 (September 1962) Availability: $6.60 (OTS).

Curves of neutron dose transmission versus thickness are presented for laterally infinite slabs of several common materials for neutrons incident at several fixed energies and angles. The materials are: water, polyethylene (borated), iron, concrete, Nevada Test Site soil (area 7, dry and 100 percent saturated), laminated slabs containing one inch of iron on the outside, a variable thickness of iron on the inside, and 3, 6, 9, or 12 inches of polyethylene sandwiched between the layers of iron. The neutron source energies are 0.5, 1, 2, 3, 5, and 14 MeV; the incident angles are 0°, 30°, 45°, and 70°.

Sufficient additional material is presented for interpretation, evaluation, and use of the results given.

531. C. E. Clifford, Effects of Ground Roughness on the Gamma Dose from Cs^{137} Contamination, AD-408304 (DCRL-401) (March 1963). Availability: Dep. (mc).

The effects of ground roughness on the dose observed above a contaminated field were measured with Cs^{137} sources and a concrete ground with a 45° sawtooth profile of 6 in. depth. The effect of rough ground was to reduce the dose observed at a height of 1 meter by a factor that varied markedly with the choice of ground roughness and with distribution of contamination. The results indicate that failure to allow for ground roughness could be the main source of error in the determination of radiation hazards near ground level from the aerial survey of a contaminated region.

532. R. W. Henderson, F. C. V. Worman, F. M. Cox, and P. K. Lee, Integral Gamma and Neutron Measurements on KIWI B-4A, LA-2885 (May 29, 1963). Availability: Dep. $1.50 OTS.

This report presents data on integral neutron fluxes and gamma and neutron rad doses measured on the experimental plans involving criticality of the KIWI B-4A reactor. Studies included a 1,500 foot free-line, two meridian rings, and an equatorial bar.

Neutron fluxes and gamma and neutron rad doses show a good approximation to the inverse square relationship with distance, with air attenuation, along the freeline. The angular distribution of the leakage radiation is highly influenced by the core configuration. The equatorial bar shows the experimentally determined flux in the horizontal plane to be isotropic over the arc of the bar.

Gamma to fast-neutron dose ratios varied from about 1 at 62 feet to 1.8 at 900 feet. The ratio of thermal flux to total fast flux varied from 0.6 at 62 feet to 0.4 at 1,500 feet.

533. B. G. Carlson, Numerical Formulation and Solution of Neutron Transport Problems, LA-2996 (1964). Availability: Dep. $1.25 OTS.

The numerical techniques for solving the neutron transport equation in arbitrary geometry in the case of discrete treatment of neutron velocity and direction are summarized. The transport problems considered may be stationary or time-dependent and may involve isotropic or anisotropic sources with or without up-scattering. The special case of the diffusion approximation is also briefly discussed.

534. W. H. Steigelman, Radioisotope Shielding Design Manual, NYO-10721 (July 1963). Availability: Dep., $2.50 (OTS).

A simple and straightforward method for determining the required thickness of a material to reduce the dose rate to a specified level from a radioactive source is presented. Many graphs have been given for use in this simple calculation. Information is given for concrete, ordinary concrete, iron, lead, and uranium for many common radioisotopes and several mixtures of fission products.

535. B. P. Fairand and E. N. Wyler, A Survey of Current Research and Developments in the Field of Dosimetry, REIC-33 (September 30, 1963) Availability: Radiation Effects Information Center, Battelle Memorial Institute, Columbus, Ohio

A survey of development in dosimetry techniques and devices during the last three to four years is presented. Three general types of radiation environments are considered: steady state, pulsed, and space.

Dosimetry procedures for selectively measuring the neutron or gamma fields in mixed neutron-gamma environments and procedures for measuring pure gamma environments are discussed in the steady-state section.

Techniques for measuring the radiation environments of pulsed reactors, and pulsed gamma and neutron sources are reviewed in the pulsed-radiation-environment section.

The section on space dosimetry is divided into a discussion of techniques and devices for measuring galactic cosmic rays, solar-flare radiation, Van Allen radiation and radiation from secondary sources.

This report complements but does not supersede REIC Reports Nos. 6 and 6A, "A Survey of Current Research and Developments in the Field of Dosimetry," Respectively dated May 31, 1958, and March 31, 1959.

536. M. B. Wells, A Monte Carlo Study of the Transport of Neutrons Resulting from Proton-Induced Nuclear Evaporation, RRA-M43 (June 12, 1964) Availability: Radiation Research Associates, Fort Worth, Texas.

Monte Carlo methods are being used in performing a parameter study of the transport of neutrons through slabs of aluminum, polyethylene, carbon, and iron for use in further machine calculations of the neutron dose in space vehicles which results from proton-induced nuclear evaporation. The calculations being performed are for plane isotropic monoenergetic neutron sources located at various depths within a slab shield. The source plane locations and source energies were selected so that data for slabs of up to 30-cm thick and source energies between 0.5 MeV and 14 MeV could be obtained by interpolation from the Monte Carlo calculations.

The results obtained to date are mainly from aluminum, and show that the amount by which the boundary effects influence the dose rate transmitted through slabs is highly dependent on both the location of the plane isotropic source within the slab and the slab thickness. Differences of as much as a factor of 3 in the transmitted dose rate are noted between the slab results and similar calculations for a plane isotropic source in an infinite medium.

537. R. S. Hubner, Analysis of Neutron Scattering from Small Volumes of Carbon, Beryllium, Aluminum, Iron, and Polyethylene, NAA-SR-9418 (March 1, 1964) Availability: Dep. (mc); $4.60 (fs), $1.52 (mf) OTS.

The results of the analyses of the ORNL Tower Shield experiments on neutron scattering from small volumes are presented and compared to the experimental data. Satisfactory agreement in the scattered dose is noted with both experiments and ORNL analytic methods. The calculated scattered spectra and nth collision doses are presented. The Monte Carlo techniques used in the analysis are described in the appendix.

538. F. J. Allen, A. Futterer, and W. Wright, Dependence of Neutron Albedos Upon Hydrogen Content of a Shield, BRL-1224 (October 1963) Availability: OTS $4.60.

Curves are presented which enable one to determine the neutron number current, number flux, energy, and dose albedos as functions of the hydrogen content of a shield material. The curves are given for eight incident energies: 0.1, 0.25, 0.50, 1.0, 2.0, 3.0, 5.0, and 14.0 MeV and four incident angles: 0, 30, 45, and 70 degrees for each energy.

A method is presented by means of which one can calculate changes in neutron albedos caused by changes in cutoff (or effective cutoff) energy.

539. J. C. LeDoux, Nuclear Radiation Shielding Provided by Buried Shelters, AD-230086 (Tech. Rept. 025) (October 1959) Availability: L.C. (mi) $4.50 (ph) $12.30.

The U. S. Naval Civil Engineering Laboratory is charged by the Bureau of Yards and Docks with the responsibility of developing design principles for the construction of atomic warfare shelters for use by the Naval Shore Establishment. Part of this problem is the evaluation of the shelter system as a shield for nuclear radiation from atomic weapons.

The objective of the present study is to advance existing knowledge of shielding for nuclear radiation and to verify theoretical concepts in this field in order to furnish the shelter designer with a formula for effective shielding against AW attack.

This report presents methods of calculating the attenuation of initial gamma, initial neutron, and residual gamma radiation incident upon buried shelters from fission-type weapons. Attenuation curves are presented which take into account variations in the moisture content and density of the soil cover. The effect of the various energy spectra for each type of radiation has been considered. Three types of shelter configurations have been investigated: the rectangular or slab, the hemisphere, and the arch.

Although a simplified model has been assumed, comparison with experimental values of attenuation factors for flexible arch-type structures indicates that the methods developed in this report will predict doses or attenuations within a factor of two.

540. R. D. Shelton, J. F. Perkins, E. Feinauer, Ground Scattered Dose Calculations Using a Monte Carlo Albedo, CVAC-266T (MR-N-54 (September 10, 1954). Availability: $2.75 (fs), $2.00 (mf), OTS.

An approximate analytic expression for the multiple scattering predicted by the Monte Carlo albedo calculations for Co^{60} has been obtained in a form readily applicable to ground scattering calculations for various source-detector geometries. It is seen that this expression, which contains the assumption that the ground scattering can be described as a surface effect, does not agree with the albedo assumptions usually made for the purpose of calculating ground scattering dose.

The dose per unit solid angle per photon, $D(\theta_o,\theta_1,\phi)$, was found to be given by a curve of the form

$$D(\theta_0,\theta_1,\phi) = (A + B\theta_0^2) \cos\theta_1 \, [1 + C\theta_0^2 \, (\pi/2 - \phi)]$$

where θ_0 is the angle of incidence of the photon, θ_1 is the angle of scatter, and ϕ is the angle of rotation of the scattered photon path from the vertical incidence plane.

The expression for the multiply scattered dose, when added to the known expression for the single scattered dose, should permit a calculation of the total ground scattered dose which is more accurate than calculations assuming single scattering only or single scattering plus the usual albedo assumptions.

541. B. L. Jones, J. W. Harris, and W. P. Kinkel, Air and Ground Scattering of Co^{60} Gamma Radiation, CVAC-170T (FZK-9-051) (March 30, 1955) Availability: Nuclear Aerospace Radiation Facility, General Dynamics, Fort Worth, Texas

Air and ground scattering of Cobalt 60 gamma radiation has been experimentally investigated for source-receiver separation distances from 7.5 feet to 70 feet for heights above ground from 9 feet to 57 feet.

The experimental results show that the dose rate at the detector due to scattered radiation was negligibly affected by changes in height after a height of about 45 feet had been reached. It was concluded, therefore, that the data for heights greater than 45 feet was about equal to that which would result from air scattering alone in the absence of the ground. The raw data for heights greater than 45 feet and for separation distances greater than 35 feet closely fit a straight line plot on log-log paper. The variation of the air-scattered dose rate thus fits a straight line corresponding to a variation $1/\underline{a}^n$. The best value of the exponent of $\underline{a}$ was found to be $n = 1.02 \pm 0.02$.

542. R. Asley, A. Foderaro, R. Aronson, H. C. Claiborne, K. Shure, H. Gerardo, and H. Goldstein, Neutron Attenuation in Optically Thick Shields, ANS-SD-1 (April 23, 1964). Availability: American Nuclear Society, Hinsdale, Illinois, $3.50.

The Shielding Division of the American Nuclear Society sponsored the calculation of a group of neutron attenuation problems. These problems ranged from point sources in homogeneous media to typical, but simplified, reactor shield designs. The results were presented at the November 1963 meeting of the ANS.

Approximately 90 individual problem solutions were submitted. The results of each problem are summarized by individual authors.

543. G. Goertzel and M. H. Kalos, "Monte Carlo Methods in Transport Problems," Progress in Nuclear Energy, Series I, Physics and Mathematics, Pergamon Press, New York, Vol. 2, pp. 315-369.

The article is devoted to the discussion of the applications of the Monte Carlo method in the field of nuclear energy. An account of the theory is given, including preliminary material on random and pseudorandom numbers and on choosing from probability distributions. The target game and the transport game are described in detail, with emphasis put on generality. The final section deals with specific applications to some shielding and reactor core calculations.

544. E. Ehret, The Determination of Epithermal Neutron Spectra by Means of Resonance Probes (Sandwich Method), AERE-Trans-906 (June 1962) Availability: JCL $2.60 (fs), $.83 (mf).

A method of measuring the epithermal neutron flux is described. A detailed description is given of the fabrication "sandwich" probe used and the analysis of the measurements made. Some comparison of the measurements made are made with calculations. The results are generally good. It is pointed out that I^{155}, Au^{197}, W^{186}, L^{139}, and Mn^{55} are good materials for such detectors and resonance energies such that measurements may be made in each logarithmic energy interval of the slowing-down spectrum.

545. R. W. Klingensmith, R. G. Jung, W. A. Lindgren, H. M. Epstein, and J. W. Chastain, Jr., Fast-Neutron and Gamma Spectrum and Dose in Beryllium Oxide, BMI-1493 (January 11, 1961) Availability: Dep. (mc); $2.60 (fs), $1.10 (mf) OTS.

Neutron and gamma penetrations through and behind BeO were measured in the Battelle-GE-ANPD Lid Tank Shielding Facility. Neutron-spectra measurements by threshold foil techniques indicated practically no change in the fast-neutron spectrum above 2.5 MeV in 13 in. of BeO. Fast-neutron dose measurements with Hurst dosimeters through 21 in. of BeO were compared with those calculated by the NDA moments method. The experimental results were within 30 percent of those calculated by GE-ANPD based on the moments calculations. The removal cross section for BeO was determined from fast-neutron dose rates measured in the water behind the slabs to be 2.02 barns. The gamma dose rate decreased with a relaxation length of about 11 cm near the source. The relaxation length increases with distance through BeO indicating spectrum hardening. Gamma spectra were measured at 4-in. intervals behind BeO slabs from 0 to 21 in. thick with a collimated 8 by 8-in. sodium iodide crystal. The 6.8- and 3.41-MeV beryllium capture gammas become dominant as the distance through the BeO increases. Thermal flux was measured through and behind the BeO with both a 3/4-in. diameter fission chamber and 1/4 by 1/4-in. by 1-mil-thick gold foils.

546. J. W. Haffner, Duct Mockup Experiments at the Convair Reactor, APEX-364 (July 1956) Availability: (mc); $3.60 (fs), $1.13 (mf) OTS.

An analysis of the Convair Ground Test Reactor experiments on the simplified annular duct mockup has been carried out. The fast-neutron duct leakage was found to be estimable to within a factor of 2 for annular ducts in water. Gamma-ray duct leakage is approximately one order of magnitude less than the fast-neutron duct leakage.

547. P. A. Read, Calculations and Nomograms for Use in a Neutron-Induced Activity Prediction System, USNRDL-TR-686 (Oct. 29, 1963) Availability: U. S. Naval Radiological Defense Laboratory, San Francisco, California.

A method is described for calculating the expected induced activity gamma dose rate, given a thermal neutron flux distribution varying only with depth in the ground. The results of a number of such calculations are discussed. A set of nomograms incorporating the results of the calculations is presented, and their construction is discussed. The nomograms themselves constitute a rapid and convenient calculating device for use in an induced activity dose rate prediction system.

548. Radiation-Effects State of the Art, 1963-1964, REIC Report No. 34 (June 30, 1964) Availability: Radiation Effects Information Center, Battelle Memorical Institute, Columbus 1, Ohio.

No abstract needed.

549. J. Lamberieux, Dosimetry and Spectrometry of Fast Neutrons by Radioactivation, HW-TR-59 (Translated from GE Report CEA-2251, 1963) Availability: JCL $6.60 (fs), $2.24 (mf).

A complete description, including worked-out examples, of the use of threshold detectors at CEA, France, is given. The means of obtaining the fast neutron dose-rate and energy spectra are derived. A Pu-Be neutron source is used as an example of spectrum determination. Tables of detector properties, including cross-sections, are included.

550: Harvey J. Amster and Wilson K. Talley, "Spatial Distribution Functions for Calculating Neutron Densities by Monte Carlo," Nucl. Sci. and Eng., 20(1), 53-59 (1964).

When describing neutrons interacting with homogeneous media, Monte Carlo can generate a sequence of energies and flight directions without locating the positions of the collisions that produced them. If the spatial distribution of one of neutron density can conveniently be obtained at specified positions and energies and at discretely sampled angles. Previous applications for plane sources in infinite media with real cross sections are here generalized for plane sources in slabs and point sources within infinite half spaces. A modified treatment for heterogeneous media is also formulated. These extensions to other geometries can provide not only additional calculational standards, but also theoretical results that could disagree with experimental facts only because of the assumed nuclear data.

551. R. E. Rexroad, M. A. Schmoke, H. J. Tiller, A. Foderaro, L. Degelman, and G. Kowal, "Point-to-Point Kernel Analysis of an Experiment to Determine Wall Penetration of a Squarebased Concrete Structure by Gamma Radiation, Nucl. Sci. and Eng. 20(1), 66-79 (1964).

An analysis was made of shielding data obtained from an experiment designed to determine the wall attenuation properties against residual gamma radiation of a simple 12 ft x 12 ft x 8 ft concrete blockhouse for three different wall thicknesses (48, 93.7 and 139 lb/ft^2). Roof thickness was chosen so that radiation scattered through the roof could be considered negligble. A plane source was simulated by successively positioning point isotropic Co^{60} and Cs^{137} sources on the ground surrounding the structure. The experiment was designed to obtain experimental evidence for the validity of certain aspects of L. V. Spencer's Theory of Structure Shielding against Fallout Radiation as presented in NBS Monograph 42. From analysis of a total of 9700 measurements, a point-to-point kernel was developed which permits calculation of the dose rate at any point within the structure from a point isotropic source on the ground outside. Doses calculated from the kernel differ less than 10 to 50 per cent from the experimental doses, except at extreme angles of incidence where the error can exceed a factor of 2. A computer program has been written which integrates the point-to-point kernel over infinite or finite rectangular areas. The kernel was integrated over an infinite area and results were compared with those obtained from Spencer's theory. It was found that the difference in reduction factor D/D_o from 'experimental' point-to-point kernel data and Spencer's theory did not exceed 40 per cent. Free-field (unshielding detector) dose rates obtained from integrating the point-to-point kernel over Co^{60} and Cs^{137} infinite-plane isotropic sources have been compared with those obtained experimentally by Rexroad and Schmoke. The difference was found to be 9 and 7 per cent respectively for the Co^{60} and the Cs^{137} source fields.

552. G. D. Joanou, A. J. Goodjohn, and N. F. Wikner, "Moments Calculations of the Fermi Age in Moderators and Moderator--Metal Mixtures," Nucl. Sci. and Eng. 13(2), 171-189 (1962).

Essentially exact calculations of the Fermi age in the pure moderators and in moderator-metal mixtures have been performed by computing the first few moments of the slowing-down distribution. The treatment of energy degradation takes into account the anisotropy of the elastic scattering to sixth order in a Legendre expansion, using data not previously available. Energy degradation through inelastic and (n,2n) processes is explicitly included.

553. K. F. Hansen, Techniques for Nuclear Shielding Calculations, AD-407300 (May 1963) Availability: $3.60 (OTS)

Reviews and critiques of selected methods for nuclear shielding calculations are presented. Those techniques considered are numerical integration of the Boltzmann equation, moments method, Monte Carlo, method of successive scatterings, and removal cross-section method. An outline of the advantages and disadvantages of each particular method is included, along with a new simplified calculation procedure.

554. L. A. Beach, R. B. Theus, and W. R. Faust, Analysis of Scintillation Spectrometer Observations of the Penetration of Cs^{137} Gamma Radiation Through Water, NRL Report 4277 (Dec. 1, 1953) Availability: $1.50 (ph), $1.50 (mf), OTS as PB 111291.

Spectral modifications of Cs^{137} gamma radiations produced by penetration and diffusion through water have been observed. A single crystal of NaI(Tl) was used to observe the pulse height distributions, produced by radiation from both plane parallel and point height distribution was carried out by use of an analogue computer to obtain the spectral distribution of radiation.

555. Arthur Reetz, Neutron and Gamma-Ray Heating in Iron Slabs, GAMD-3690 (October 24, 1962) Availability: $6.60 (OTS).

Radiation heating in iron slabs 2 to 8 inches thick is calculated for monoenergetic neutrons and gamma rays incident normal to the slab. Transmission, reflection, and absorption of the energy and the number of neutrons are presented for neutrons of energies from 1.9 ev to 14 Mev. The heat deposition and energy transmission are presented for gammas of 0.5 to 8 Mev. Monte Carlo techniques are used for the high-energy (> 0.5 Mev) neutron calculations and the one-dimensional neutron transport code DSN is used for the low-energy neutron calculations. Monte Carlo techniques and gamma-ray differential energy spectra are used in the gamma calculations.

556. D. K. Trubey, Calculation of Fission-Source Thermal-Neutron Distribution in Water by the Transfusion Method, ORNL-3487 (August 1964) Availability: $1.25 OTS.

A calculational method which combines transport theory and diffusion theory, called the "transfusion" method, has been used to obtain the thermal-neutron flux from a point isotropic fission source in water. The transport theory code RENUPAK, which utilizes the moments method, was used to calculate the energy-spatial neutron distribution above 1 keV, and the multigroup diffusion code MODRIC was used to continue the calculation down to thermal energies. The resulting thermal-neutron fluxes agree very well with experimental results for distances out to 70 cm from the source, but the calculated fluxes are as much as 40% higher for greater distances.

557. W. E. Edwards and J. D. Simpson, Reactor and Shield Physics, APEX-918 (May 31, 1962) (Confidential)

This is one of twenty-one volumes summarizing the Aircraft Nuclear Propulsion Program of the General Electric Company. This volume describes the experimental and theoretical work accomplished in the areas of reactor and shield physics.

The reactor physics technology for all ANP reactor types is presented in its most advanced stage; i.e., no attempt is made to present chronologically the development of the technology.

The use of automated techniques for power-mapping critical experiments in the reactor physics program are discussed, with particular attention to the use of high speed computer programs employing the IBM 704 and IBM 7090 computing systems.

In the nuclear shielding program, efforts were concentrated in two main areas: (1) the optimum placement of shield materials to reduce radiation levels, and (2) the calculation of specific nuclear data, such as nuclear heating and activation, which are important to the design of an efficient, safe power plant.

Methods were developed for determining, at any position in the reactor-shield assembly, the total flux and the angle and energy distribution of neutron and gamma rays, as well as the response of any detector used to measure radiation effects.

Important shielding computer codes described are the point kernel and single scattering codes and the more recently developed Monte Carlo's codes.

558. W. Poenitz, Determination of Absolute Thermal Neutron Flux by Gold Foils, KFK-180 (April 1963) Availability: dep. (mc).

A relation for the activation rate of a gold foil is discussed, together with the necessary corrections. The method used for absolute determination of ^{198}Au activity at Karlsruhe is then discussed. The effective beta self absorption factor was determined by two techniques involving 4π β-γ coincidence counting. The necessary corrections are established and evaluated. The two methods gave results in good agreement. Using the self absorption factors, the ^{198}Au activities can be determined by 4π β-measurements with an accuracy of 0.2 to 0.3%. The absolute thermal neutron flux determination should be accurate to 0.5%. (D.C.W.)

559. D. C. King, Neutron Reaction Data on Aluminum and Oxygen for the UKAEA Nuclear Data Library, AEEW-M-445 (February 1964) Availability: H. M. Stationery Office, Seven Shillings

This memo describes the new data files for aluminum and oxygen which have been prepared in the period July 1963 - February 1964 for the U.K.A.E.A. Nuclear Data Library.

Data from recent compilations of L. Forsberg, G. D. Joanou and H. Fenech, and J. J. Schmidt have been used extensively in the preparation of these new data files which cover the energy range from 15 MeV down to 0.0006 eV for aluminum and 0.0001 eV for oxygen.

560. K. Parker, Neutron Cross-Sections of Beryllium in the Energy Range 0.025 eV - 15 MeV, AWRE O-27/60. Availability: Dep. (mc)

Recommended values are given for the neutron total and partial cross-sections of beryllium in the energy range 0.025 eV - 15 MeV based on available experimental and theoretical evidence. Details of energy and angular distributions are given so that the data is complete for the purposes of neutronics calculations.

Whilst every effort has been made to ensure the general reliability of the data, no responsibility can be accepted for errors, whether from the interpretation of the original experimental data or from the transcription of the intended data to the graphs.

561. K. Parker, Neutron Cross-Sections of U^{235} and U^{238} in the Energy Range 1 keV - 15 MeV, Part I: Best Cross-Sections for U^{238} Based on Microscopic Experimental and Theoretical Data Available at December 1961, AWRE O 79/63 (January 1964) Availability: Dep. (mc).

Recommended values are given for neutron total and partial cross-sections of U^{238} in the energy range 1 keV - 15 MeV. The available experimental evidence is supplemented by estimates based on nuclear theory. Details of energy and angular distributions are given so that the data are complete for the purposes of neutronics calculations. Experimental and theoretical data which have become available since the "best" values were determined are discussed as far as possible.

The values given in this note have been combined with older data (1/U238) below 1 keV and written on punched cards (3/U238 data, nuclide identification number = 156) as part of the Aldermaston Nuclear Data Library.

Part II [1] of this series describes similar work for U^{235}. Part III [2] of this series describes the simultaneous adjustment of the U^{235} and U^{238} data, within the limits of experimental error where possible, in order to bring experimental and calculated parameters for a large number of simple integral systems into agreement.

562. K. Parker, Neutron Cross Sections of U^{235} and U^{238} in the Energy Range 1 keV - 15 MeV, Part II: Best Cross Sections for U^{235} Based on Microscopic Experimental and Theoretical Data Available at March 1962, AWRE O 82/63 (December 1963) Availability: Dep. (mc).

Recommended values are given for neutron total and partial cross-sections of U^{235} in the energy range 1 keV - 15 MeV. The available experimental evidence is supplemented by estimates based on nuclear theory. Details of energy and angular distributions are given so that the data are complete for the purposes of neutronics calculations. Experimental and theoretical data which have become available since the "best" values were determined are discussed as far as possible.

The values given in the note have been combined with older data (1/U235) below 1 keV and stored on punched cards (3/U235 data, nuclide identification number = 155) as part of the Aldermaston Nuclear Data Library.

Part I [1] of this series describes similar work for U^{238}. Part III [2] of this series describes the simultaneous adjustment of the U^{235} and U^{238} data, within the limits of experimental error where possible, so as to bring experimental and calculated parameters for a large number of simple integral systems into agreement.

563. E. D. Pendlebury, Neutron Cross Sections of Li-6 in the Energy Range 0.001 eV - 15 MeV, AWRE O - 60/64 (July 1964) Availability: UKAEA, Atomic Weapons Research Establishment, Aldermaston, Berks.

Recommended values are given for neutron total and partial cross sections of Li-6 in the energy range 0.001 eV - 15 MeV. The available experimental data are supplemented where necessary by estimates based on nuclear theory. Details of energy and angular distributions are given so that the data are complete for the purposes of neutronics calculations.

The recommended data given in this report have been written on to punched cards (3/Li-6 data, nuclide identification number (NIN) = 175) as part of the Aldermaston Nuclear Data Library. A modified data set 3A/Li-6 (NIN = 181) which is perhaps slightly better than the 3/Li-6 data and which can be used as an alternative to it is referred to in the Appendix.

AWRE Report No. O-61/64 describes similar work for Li-7.

564. E. D. Pendlebury, Neutron Cross Sections of Li-7 in the Energy Range 0.001 eV - 15 MeV, AWRE O - 61/64 (July 1964) Availability: Dep. (mn).

Recommended values are given for neutron total and partial cross sections of Li-7 in the energy range 0.001 eV - 15 MeV. The available experimental data are supplemented where necessary by estimates based on nuclear theory. Details of energy and angular distributions are given so that the data are complete for the purposes of neutronics calculations.

The recommended data given in this report have been written on to punched cards (3/Li-7 data, nuclide identification number (NIN) = 176) as part of the Aldermaston Nuclear Data Library.

AWRE Report No. O-60/64 describes similar work for Li-6.

565. H. Hurwitz, Jr., J. B. Nims, Jr., G. M. Roe, and M. L. Storm, "Gamma-Ray and Neutron Heat Generation," Chapter 22, The Physics of Intermediate Spectrum Reactors (UC-81) Availability: Naval Reactors Branch, Division of Reactor Development, U. S. Atomic Energy Commission, Washington, D. C.

In part A the geometries of different regions of a reactor are idealized by considering plane, spherical, and cylindrical bodies with various surface and volume sources of gamma radiation. This idealization permits the analytic evaluation of most of the integrals arising in the absorption calculations. In addition, the "straight-ahead" approximation is assumed at all times; this assumes that, in all scattering events, the gamma-rays keep their direction unchanged. The gamma-ray attenuation is assumed to be exponential. The validity of these assumptions is discussed in B, where the selection of effective energy-absorption coefficients is discussed.

The study in Part B investigates the accuracy with which the absorption may be represented by an exponential approximation and determines the effective cross-section to be used in this representation. To consider this question for SIR-type reactors, a multigroup calculation was performed to determine the spectral energy distribution and energy deposition rate as a function of distance from a point source along a straight-line path starting in the core and continuing into the reflector. The "straight-ahead" approximation (wherein the gamma-ray photons lose the proper amounts of energy on collisions with electrons but are assumed to keep their direction of travel unchanged) was used, and various representative path lengths in core and reflector material were chosen. Although the treatment of gamma-ray attenuation by the straight-ahead approximation and the multigroup method would not be entirely satisfactory in the case of shielding studies, this approach is considered to be useful for gamma-ray heating calculations, where the distances involved are comparatively small.

In Part C of the chapter, we obtain the magnitude and distribution of the heat generated in an intermediate reactor by the absorption of gamma radiation, by neutron degradation, and by (n,α) reactions in beryllium. Account is not taken of the kinetic energy of fission products and fission betas in the core. All values are obtained for steady-state operation at 1 Mw.

566. J. A. Rich, and R. E. Slovacek, "Gamma-Ray Heating Measurements in the PPA," Chapter 23, The Physics of Intermediate Spectrum Reactors (UC-81) Availability: Naval Reactors Branch, Division of Reactor Development, U. S. Atomic Energy Commission, Washington, D. C.

The distribution and magnitude of the heating that arises from gamma-ray and neutron interactions in the reflector and blanket of an intermediate reactor are of importance in the proper design of the cooling system for those regions. Calculations of the neutron and gamma-ray heating in such a reactor are described in Chapter 22, this volume. The measurements described here were made in PPA-20 (an assembly similar to the reactor) to determine experimentally the distribution and magnitude of the gamma-ray heating.

567. F. Allen, W. Wright, and A. Futterer, Angular Distributions and Energy Spectra of Neutrons Transmitted Through Polyethylene, BRL-1148 (September 1961) (CONFIDENTIAL)

This report contains angular and energy spectral distributions calculated by the Monte Carlo method for neutrons which penetrate various thicknesses of polyethylene. The forms of these distributions are correlated with physical ideas. Some generalizations of the computed results are given. It is shown that a quasi-equilibrium state is reached for any given source energy within a few inches of the incident face. Methods utilizing the angular and energy distributions are developed which permit the calculation of number and dose attenuation factors for deep penetrations. Relaxation lengths are also obtained.

It is shown that the reflected neutrons in a thick-walled enclosure are distributed uniformly and isotropically to a good degree of approximation. It follows that the theory of black body radiation is applicable with the independent of the shape of the enclosure. Therefore the increase in dose due to internal reflections can be calculated to good approximation in slab geometry.

Neutron absorption versus depth distribution curves are also given for various source energies and angles of incidence.

568. F. Allen, C. Cialella, A. Futterer, O. Lyman, and W. Wright, Experimental and Theoretical Determination of the Transmission of Monoenergetic Neutrons Through Laminated Slabs of Iron and Borated Polyethylene, BRL-1156 (December 1961) (CONFIDENTIAL)

Experimental and theoretical determinations of neutron number current transmission factors are presented for laminated slabs of iron and borated polyethylene. Good agreement between theory and experiment was obtained. The configurations contained up to 12 inches of polyethylene and 8 inches of iron. Incident neutron energies employed in the experiments were 1, 5 and 16.6 MeV. The theoretical dose transmission factors are presented for the same configurations and for incident neutron energies of 0.5, 1, 2, 3, 5, and 14 MeV and incident angles of 0°, 30°, 70°.

569. S. Buscaglione and R. Manzini, Build-Up Factors: Coefficients of the J. J. Taylor Equation, ORNL-TR-80 (Rev.) (February 1964) Availability: Dep. (mc); $1.60 (fs), $0.80 (mf) JCL. (Translated from Comitato Nazionale Energia Nucleare RT/FI(65)7).

The coefficients for the build-up factors expressed on the basis of the J. J. Taylor equation were determined for some of the most common materials for the construction of gamma-radiation shielding.

The materials investigated were H_2O, Al, Fe, Pb, Sn, W, U as well as normal concrete, baryte concrete ferro-phosphorus concrete and magnetite concrete.

570. M. J. Barrett, W. E. Barch, and R. L. MacNeil, Transmission of Gamma Radiation Through Thin, Finite Slabs, TO-B-64-44 (May 1964). Availability: $2.00 (fs), $0.50 (mf) OTS.

The shielding of gamma radiation -- at photon energies where the Compton scatter process is dominant--results in a transmitted beam that is largely composed of scattered photons. An approximate formula has been developed through experiment to calculate the scattered radiation and therefore determine the exact transmission, i.e., dose build-up factor (B), for thin shields in the range of photon energies from 0.6 to 2 MeV. The results of this experiment also give rise to a simple method of measuring any microscopic scattering cross section by an array of detectors without loss of the definition of the scattering angle.

571. M. J. Barrett and R. L. MacNeil, Dose Rate Distribution in a Compartmented Structure, TO-B-64-1 (June 1964) Availability: $3.00 OTS as AD 604192.

Measurements are reported of the gamma radiation-induced dose in a compartmented steel structure. The experimental dose profile is compared with predictions based on computational techniques in the Office of Civil Defense Engineering Guide, as well as with computations performed by the Monte Carlo technique. Although agreement (within experimental accuracy) is reached between the Monte Carlo and the experimental measurements, the OCD predictions appear to overestimate the dose because of an incomplete consideration of the effect of spacing of the structure slabs. The degree of conservatism does not appear to be serious, however.

572. Albert W. Starbird and John F. Batter, Angular Distribution of Skyshine Radiation at the Surface of a Plane of Fallout Contamination, TO-B-63-40 (March 1964) Availability: OTS $3.60 as AD 436311.

An experimental program was conducted to determine the angular distribution of the air-scattered component of gamma radiation (skyshine) at the surface of a plane of fallout contamination. Measurements were made by placing a radiation detector in a vertical collimator recessed flush with the ground surface and placing cobalt-60 sources on the ground along a line extending from the center of the collimator to a radius of about 40 ft to simulate fallout contamination. Detector depth in the collimator was varied to measure skyshine within the cone formed by the collimator opening and the detector at given solid angle fractions for source positions extending from the collimator lip to the 40-ft radius. The experimentally-determined results of this program and of other investigators were compared with theoretical calculations with the following conclusions. Calculated accumulative angular dose rate distributions of skyshine from point and plane sources are probably valid. Analytically-derived curves for total scattered radiation for cobalt-60 and cesium-137 underestimate dose rates from close-in sources and slightly underestimate them from distant sources when an earth or asphalt interface is substituted for free or compressed air values. The analytical equation for calculation of total skyshine is reasonably accurate.

573. A.Schaarschmidt, "Interpolated Energy Dist. of Cs^{137} Gammas in Water," Nuclear Instr. and Methods, 27, 311 (1964).

Using the moment-method spectra of Goldstein and Wilkins complete energy distributions of the number flux of multiple scattered γ-rays in water, starting from a point isotropic source of Cs^{137}, are evaluated by interpolation for $\mu_0 r = 1 \ldots 20$. Also the corresponding energy build-up factors are given. Some remarks concerning the recent experimental work of Sybesma are added.

574. R. L. Henry, L. G. Mooney, and R. J. Prevost, Study of Radiation Penetration and Reflection from Shield Materials, FZK-183 (August 31, 1964) Availability: Not for sale to the public.

Two computer procedures were developed: the Slab Penetration and Reflection Calculation (SPARC) code and the Combine code. The SPARC code is a Monte Carlo procedure used to generate penetration and reflection probabilities due to radiation incident upon multilayer slabs. The Combine code is an integration procedure that combines the environment parameters, the reflection and transmission probabilities, and the shield geometry to calculate the dose inside the shield.

An experiment was performed to evaluate the SPARC calculations. The experiment utilized a collimated beam of radiation from the ASTR normally incident upon selected slabs of shielding material. The measurements were obtained of the radiation reflected from and transmitted through these materials. The measurements compare very well with the SPARC calculations for neutrons.

The results of the analysis of the neutron data show that the reflected component of the radiation is a significant portion of the measured dose. The magnitude of the reflected component is much greater from the materials of higher density, such as steel and lead, than from the low-density 6% borated polyethylene. Also, it was demonstrated that this reflection component can be effectively suppressed by placing 6% borated polyethylene on the beam-incident side of the steel slabs.

The gamma-ray analysis is rather limited because of the neutron effect on the gamma-ray detector. However, indications are that the reflections of gamma-rays is not a significant portion of the measured dose rate.

575: R. W. Klingensmith, H. M. Epstein, and J. W. Chastain, Shielding Studies on Salt Slabs, Gadolinium, and Water, BMI-1384 (October 7, 1959) Availability: Dep. (mc); $3.60 (fs), $1.40 (mf) OTS.

A lid-tank shielding facility, similar to the one in operation at Oak Ridge, has been constructed and is in operation at Battelle. This facility was described in BMI-1291. The fast-neutron and gamma dose rates and thermal-neutron fluxes were mapped in a symmetry plane of the shielding tank.

Fast-neutron and gamma dose rates were measured behind a series of shielding plugs along a line in the water perpendicular to the fission-plate center. The plug materials included aluminum, iron, lead, titanium, and gadolinium. In order to use small-diameter plugs the effective diameter of the fission plate was reduced for these experiments by placing an iris between the thermal column and the fission plate. The purpose of the test was to determine the removal cross section of gadolinium by comparison with other materials, using a small amount of gadolinium. The removal cross section for gadolinium is about the same as for lead.

Experiments were conducted on ordinary and enriched LiH slabs 4 to 32 in. thick. Fast-neutron and gamma dose rates in addition to thermal-neutron fluxes were measured on the water center line behind the slabs.

576. D. E. Evans and D. M. Wheeler, Porous Plug Experiment III, NARF-58-28R (June 30, 1958) Availability: Not for sale to the public.

With the Ground Test Reactor as the source of radiation, measurements were made in the Outside Tank Facility of the dose rates behind two porous plugs and one slab plug used as a standard. This report presents the results of these measurements and a description of the Outside Tank Facility.

Fast neutron results indicate a shielding weakness in the interior of the two porous plugs (nut not the slab plug) such that the fast neutrons are effectively focused at some distance from the plug surface. These weaknesses cause a significant increase in the fast neutron dose rate at angles off the centerline and at distances of 20 to 100 feet from the reactor core center.

Where possible, relative comparisons are made between measurements taken with these plugs in pool and in the Outside Tank Facility. Definite conclusions cannot be reached about the shielding effectiveness of these plugs until a further analysis of the internal construction of the plugs is completed.

Results of previous porous plug experiments are reported in FZK-108 and MR-N-187.

577. W. A. Rhoades, Photon Streaming Around Offset Cylindrical and Square Access Plugs in Thick Shields, NAA-SR-MEMO-5001 (March 3, 1960) Availability: (mf) $.83, (fs) $2.60, OTS.

Equations for three separate contributions to the photon streaming are developed. Streaming from the annulus above the offset is the dominant streaming source in the case of a 4 inch cylindrical plug in a 42 inch magnetite concrete shield. Radiation scattered across the offset is found to be unimportant in this case. For a 1.8 Mev source, optimum location of the offset would reduce the maximum streaming dose rate to one half of that obtained with the offset located at the midplane.

578. S. G. Wogulis and K. L. Rooney, Experimental Studies of Neutron Attenuation in Natural Lithium Hydride Shields, NAA-SR-9364 (February 12, 1964) Availability: Not for sale to the public.

Results of a program to measure the attenuation of fission neutrons in lithium hydride at the SNAP Shield Test Facility are presented, with the methods used.

Experimental and analytical corrections to the data are given, to allow reduction to the infinite medium case.

579. R. G. Rose, "Gamma-Ray Attenuation Coefficient," Nucleonics, 22(8), 126 (August 1964)

Attenuation coefficients for a number of chemical elements have been calculated by Blizard from the data of Grodstein. It was intended to use these coefficients to calculate the attenuation of gamma radiation in concretes and in lead glasses, but chemical analysis indicated the presence of chemical elements for which attenuation coefficients are not listed by Blizard. We obtained the attenuation coefficients of the additional chemical elements by interpolation and, in a number of cases, extrapolation of the Grodstein data.

580. S. T. Friedman, Shielding Tests Performed in the General Electric Outside Test Tank in the Period March-June 1958, APEX-581 (February 1961) Availability: General Electric Co., Cincinnati, Ohio.

Dose and flux data were obtained outside the General Electric Outside Test Tank (OTT) (formerly Outside Tank Facility) when configurations of realistic shield material slabs were in the tank. Tables and graphs of these data are presented. Information is given about secondary gammas, photoneutron production, radiation falloff rate, and other shielding variables.

581. E. D. Arnold, Handbook of Shielding Requirements and Radiation Characteristics of Isotopic Power Sources for Terrestrial, Marine, and Space Applications, ORNL-3576 (April 1964) Availability: $3.00 OTS

Calculations have been made of the radiation intensities from shielded and unshielded sources fabricated from seventeen isotopes that show promise for use in isotopic power or radiation applications. Source sizes in the range of 100 to 20,000 thermal watts were evaluated. All shielded sources were assumed to be attenuated by iron, lead, and uranium; and in those cases where the source also emitted neutrons, neutron and gamma attenuation through water was determined.

The isotopes studied and their physical form are as follows: Co^{60} (metal), Kr^{85} (liquefied gas), Sr^{90} (oxide and titanate), Zr-Nb^{95} (oxide), Ru^{106} (metal), Cs^{137} (glass), Ce^{144} (oxide), Pm^{147} (oxide), Tm^{170} and Tm^{171} (both as oxides), Tl^{204} (metal), Po^{210} (metal matrix with void space for gas collection), U^{232} (oxide), Ph^{228} (oxide matrix with void space for gas collection), Pu^{238} (oxide), Cm^{242} (oxide matrix with void space for gas collection), and Cm^{244} (oxide).

582. W. G. Miller, B. W. Shumway, S. Tomoeda, G. E. Plummer, and M. B. Hastings, Gamma-Ray Penetration Experiments for a Destroyer Exposed to a Distant Source, Data Report, USNRDL-TR-728 (February 1964) Availability: Not for sale to the public.

Presented are the gamma-ray dose rates received at various locations aboard a destroyer extensively exposed to a distant collimated point source of Co^{60}. Part I contains the dose rates received at twenty-one detectors which were distributed throughout the ship. Part II contains the dose rates received at closely spaced detectors which comprised arrays upon each of four decks in a forward section of the ship. In general, radiation was incident on all detectors from numerous directions.

583. G. E. Langer and B. W. Shumway, Some Gamma-Ray Shielding Properties of Finite Steel Slabs, USNRDL-TR-707 (January 21, 1964) Availability: $4.60 OTS.

Some gamma-ray shielding properties of plane steel slabs placed between a point source and a point detector have been investigated. The dose buildup (B_r-1) resulting from 8 by 9 ft slabs 0.25, 1.00, and 1.50 in. thick (0.36, 1.45, and 2.17 mean free paths) placed between a Cs^{137} point source and a small hemispherical ionization chamber was measured. The distance from the source to the detector was 16 ft. Each slab was placed at five selected points between the source and the detector in order to observe the variation in B_r-1 with slab position. The dose buildup due solely to singly scattered photons was calculated both for the 8 by 9 ft slabs and for slabs of infinite extent. The measured values of dose buildup are compared with the calculations in this report.

A similar set of measurements has been made for 1.25-Mev photons incident upon steel slabs 0.25 and 0.75 in. thick (0.26 and 0.78 mean free paths). The distance from the source to the detector was 25 ft for these measurements.

The effect of interchanging two separated slabs of different thicknesses was measured for two slab pairs; one 0.25 and 0.75 in. thick, the other 0.50 and 1.00 in. thick. Dose buildup due to separated slab pairs was measured. It was also estimated from single slab data, and the results were compared.

584. L. K. Zoller, "Fast-Neutron-Removal Cross Sections," Nucleonics, 22(8), 128 (August 1964).

A complete and consistent set of calculated fast-neutron-removal cross sections is provided in the table for use in point-kernel and moments-method shielding calculations.

The cross sections were calculated from empirical expressions that permit relatively quick and simple determination of the fast-neutron effective removal cross section for any element. These expressions eliminate most of the inconsistencies that are inherent in the usual graphical determinations of removal cross sections for elements that have not been studied experimentally. Moreover, the expressions are ideally suited for incorporation in computer programs wherein self-generation of the removal-cross-section data is desired.

585. R. E. Schreiber and R. J. Allio, "Gamma Heating in Reactor Experiments," Nucleonics, 22(8), p. 120 (August 1964).

The important sources of gamma heating in a material placed in or near the reactor core are shown. We present a hand-calculation method here for making a good estimate with reasonable effort of the gamma-heat generation in in-pile experiments.

586. Akira Tsuruo, The Spectrum Matrix Method in Gamma-Ray Transmission Problems for Multiple Layers, ORNL-TR-241 (1964). Availability: JCL $1.10 (fs), $0.80 (mf).

An approximate calculation method in γ ray transmission problems is shown for multiple layers.

In this method, the differential energy flux of incident γ rays is expressed by a vector, and the shielding property of each slab is represented by a matrix. The matrix is constructed by a relatively simple procedure, using the data of penetration in infinite media calculated by the moments method and the data of the effect of finite slabs calculated by the Monte Carlo method. The compound shielding property of multiple layers is also represented by the matrix which is obtained by the multiplication of the matrix corresponding to each slab. The differential energy flux of transmitted γ rays is given by multiplying the incident vector by the matrix.

The results of the method are compared with those of the Monte Carlo calculation for Pb-H_2O and H_2O-Pb stratified slabs, and also with those of experiments for Pb, Fe and Al stratified slabs. The comparison shows that this method is useful for all combinations of materials for high incident energy, and when low energy γ rays enter the heavy layer followed by a light or heavy layer.

587. R. E. Rexroad, M. A. Schmoke, and J. L. Meredith, Evaluation of Internal Radiac Detector Positions for an M60 Tank, NDL-tr-49 (April 1964) Availability: Army Nucl. Defense Lab., Edgewood Arsenal, Maryland.

The objectives of this work were (1) to determine the suitability of placing a radiation detector within an M60 tank to provide a radiation survey and detection capability in a residual-gamma-radiation field, (2) to determine the optimum internal position for such a radiation detector, and (3) to determine the necessary correction factors pertinent to such an internal detector to yield the free-field dose rate.

A cobalt 60 point-source-circulation system was used to simulate a uniformly-contaminated residual-gamma-radiation area having a radius of 600 feet. The dose rate was measured with ionization-chamber dosimeters at various points within the tank where it would be possible to mount such a radiac detector. The free-field dose rate was measured and comparisons were made to determine the appropriate correction factors. The applicability of these correction factors for radiation areas of smaller size, or for areas in which extensive hot spots might exist, was determined.

588. Haruyuki Fujita, Keishuke Kobayashi and Tomonori Hyodo, "Backscattering of Gamma Rays from Iron Slabs," Nuclear Science and Engineering, 19, 437-440 (1964).

Spectra of backscattered radiation from finite thick slabs of iron were measured by means of a scintillation spectrometer. A point Co^{60} gamma source was placed in close contact with a scattering slab. Variation of number and energy albedo values with increasing the slab thickness are given as experimental formulas and graphical representations.

589. R. L. French and M. B. Wells, "An Angle-Dependent Albedo for Fast-Neutron-Reflection Calculations," Nuclear Science and Engineering, 19(4), 441-448 (1964).

An albedo model for calculating the dose due to fast neutrons reflected from materials of low to moderate hydrogen content has been developed through analysis of extensive Monte Carlo data. The model, which was developed from reflection data for iron, concrete and three types of soil, is for reflection to a unit non-directional receiver and is of the form

$$\alpha(E_o)\cos^{2/3}\theta_o\cos\theta$$

where

$\alpha(E_o)$ is a coefficient tabulated as a function of incident energy,
E_o, for the various materials,
θ_o is the angle of incidence and
θ is the angle of reflection (both measured from the normal).

The differential albedo, in unit of reflected dose/steradian per unit dose incident at angle θ_o, may be converted to a total albedo by multiplying by π. The total dose albedo for normally incident fission neutron was found to be closely approximated by $0.435\ (\Sigma_T-\Sigma_H)/\Sigma_T$ where Σ_T is the macroscopic total cross section of all elements of the material, and Σ_H is the macroscopic cross section of the hydrogen of the material, both weighted by the fission spectrum.

590. P. N. Haubenreich, J. R. Engel, et al., MSRE Design and Operations Report. Part III. Nuclear Analysis, ORNL-TM-730 (February 1964) Availability: Dep. (mc); $14.00(fs), $6.32 (mf) OTS.

The design of the biological shield for the Molten-Salt Reactor Experiment is based upon conventional shield design techniques for shielding a stationery reactor. The additional sources of induced activity in the molten salt were taken into account. The shield as designed reduces the calculated radiation dose rates to permissible levels.

591. S. T. Friedman, An Analysis of Late 1957 and Early 1958 Lid Tank Experiments Pertaining to XMA-1 Shield Design, APEX-524 (June 1958) Availability: Not for sale to the public.

This report contains an analysis of experiments performed at the ORNL Lid Tank Shield Facility (ORNL-LTSF) during the last half of 1957 and the early part of 1958. A lengthy discussion of secondary gammas and their relevance to XMA-1 shield design in light of these experiments is given. Information about the suppression of secondaries with boral and about the relative merits of Hevimet, uranium, and stainless steel as gamma shielding materials is presented.

592. H. S. Davis, "Development and Construction of Concrete Biological Shields for the New Production Reactors at Hanford," TID-2507, Nuclear Science and Technology, Vol. 1A, Issue 2, p. 343-360 (March 1955) (CONFIDENTIAL)

Presented in this article are data and photographs describing the steel and concrete biological shields constructed recently at Hanford. These shielding structures enclose the two new production reactors, 105-KW and 105-KE, and are unprecedented in several respects, such as size, materials used, uniformity of composition, ease of construction, dimensional accuracy, and quality of workmanship.

High-density concrete containing steel aggregate and limonite was placed in prefabricated steel forms by means of the prepacked method. In those areas where high density was not required, the steel aggregate was replaced with magnetite. Type 1 concrete, made with steel aggregate and limonite, weighed 270 lb/cu ft, and type 3 concrete, made with magnetite and limonite, weighed 216 lb/cu ft. Type 1 and type 3 concrete cost $678 and $390 per cubic yard in place, respectively. The two biological shields cost $942 per cubic yard to construct, or $465 for concrete and $477 for steel forms, gas seals, and aluminum reflectors. These cost figures include only direct construction costs for labor, materials, and equipment.

By including indirect costs, it is estimated that a savings of over $11,000,000 was realized in the total project cost of these two reactors by using high-density concrete instead of steel and masonite in the biological shields. In addition, these steel and concrete shields provide greater resistance to deterioration, earthquake, and blast forces than do the steel and masonite shields constructed previously at Hanford.

593. P. W. Schreiber, Measured Gamma Spectra Behind Various Thicknesses of Beryllium, Beryllium Oxide, and Lithium Hydride, APEX-700 (June 1961) (CONFIDENTIAL-RESTRICTED DATA)

Angular gamma spectral data measured behind various thicknesses of beryllium, beryllium oxide, and lithium hydride at the GE-ANPD Source Plate Facility located at Battelle Memorial Institute are reported. Measurements were made with an 8-inch-high by 8-inch-diameter NaI(Tl) scintillation spectrometer. The contract was terminated before the collimated spectrometer was calibrated. However, a tentative method for approximating the response functions from measured data and available Monte Carlo calculations is given.

594. P. W. Schreiber and F. D. Kodras, Measured and Calculated Radiation Levels Within and Behind Beryllium, APEX-701 (July 17, 1961) (CONFIDENTIAL)

The data obtained from extensive shielding measurements, using beryllium slabs in the GE-ANPD Source Plate Facility at Battelle Memorial Institute, are reported. The measurements include fast neutron dose rates, thermal fluxes, gamma dose rates, resonance foil activations and threshold foil activations. Dose rate measurements and thermal flux meaxurements were made within a 32-1/2-inch slab array of beryllium and in water behind various thicknesses of beryllium. Resonance foil measurements were made to a penetration of 16 inches of beryllium, and threshold foil measurements were made to a penetration of 12 inches of beryllium.

In addition, a number of the measurements are compared with calculated values determined by the use of point kernel and diffusion computer programs.

595. C. E. Humphries and P. Tawil, Investigation of the Two-Component Method, APEX-745 (July 1961) (CONFIDENTIAL)

The two-component method, as outlined as John Moteff, is used to predict fast-neutron dose rates outside the (1) OTT General Electric Slab configurations (Series 43, 47, and 62), (2) Configuration I of the General Electric solid shield, and (3) the Pratt and Whitney reactor shield. The results are compared with measured dose rates and those calculated by the Albert-Welton point kernel method and the moments method. Although the investigation is not conclusive, the results indicate that the accuracy of the two-component method depends completely on the ability to predict the magnitude and angular distribution of the diffuse component at the shield surface.

596. M. Morgenthau, H. B. Maieran, et al., Local Fallout from Nuclear Test Detonations. Vol. II. Compilation of Fallout Patters and Related Test Data. Part I. Trinity through Redwing, Part II. Plumbbob through Hardtack, DADA-1251, Vol. II, Parts I and II (August 1963) (SECRET).

597. J. M. Ferguson, Neutron-Induced Radioactivity in Beaches, USNRDL-TR-494 (February 17, 1961) (SECRET).

598. C. Ciallella, O. Lyman, and H. Schuckler, The Transmission of Monoenergetic Neutrons through Armor, BRLM-1327 (January 1961) (CONFIDENTIAL).

599. J. M. Ferguson, Early-Time Gamma Radiation from Nuclear Weapons, USNRDL-TR-600 (June 13, 1961) (SECRET).

600. R. L. Mather, Neutron Energy Effects and Induced Activation, USNRDL-TR-465 (September 28, 1960) (SECRET).

601. L. A. Webb, F. M. Tomnovec, R. L. Mather, R. F. Johnson, and C. S. Cook, Analysis of Gamma Radiation from Fallout from Operation Teapot, USNRDL-TR-106 (August 27, 1956) (CONFIDENTIAL).

602. W. E. Thompson, Spectrometric Analysis of Gamma Radiation from Fallout from Operation Redwing, USNRDL-TR-146 (April 29, 1957) (CONFIDENTIAL).

603. C. S. Cook, R. F. Johnson, R. L. Mather, F. M. Tomnovec, and L. A. Webb, Gamma-Ray Spectral Measurements of Fallout Samples from Operation Castle, USNRDL-TR-32 (January 26, 1955) (CONFIDENTIAL).

604. C. S. Cook and F. M. Tomnovec, Radioactivity in Ship Structural Material Induced by Neutrons from Nuclear Weapons, USNRDL-TR-274 (October 30, 1958) (CONFIDENTIAL)

605. J. M. Ferguson, The Problem of the Calculation of Radioactivity Induced in Soil by Neutrons, USNRDL-TR-210 (March 12, 1958) (SECRET).

606. J. M. Ferguson, W. E. Thompson, and F. M. Tomnovec, Gamma-Ray Fallout Spectra for Operation PLUMBBOB, USNRDL-TR-286 (December 8, 1958) (SECRET).

607. R. L. French, "A First-Last Collision Model of the Air-Ground Interface Effects on Fast Neutron Distribution," Nuclear Science and Engineering, 19(2), 151-157 (1964).

A method has been developed for predicting the effect of an air/ground interface on the fast-neutron flux or dose at large distances from a point isotropic source of neutrons in air. The method yields numerical values for functions $f(H_S)$ and $f(H_D)$ that may be used to express the fast-neutron intensity as a function of source height H_S, receiver height H_D, and source receiver separation distance R, in terms of the corresponding infinite air intensity I(R). Thus

$$I(H_S,H_D,R) = f(H_S)f(H_D)I(R) \quad .$$

The method is called the "First-Last Collision Model" because it is based on the influence of the ground upon the distribution of "first" collisions of neutrons about the source and of "last" collisions about the receiver. Generalized numerical results have been computed, and means have been developed for applying these results to specific cases. Comparisons of these results with those derived from Monte Carlo calculations, and from experiments performed at the ORNL Tower Shielding Facility and the Nevada Test Site indicate that the first-last collision model predicts the fraction of the infinite air intensity within 5 per cent in almost all cases.

608. H. Waldinger, J. Agresta, G. Goertzel, "Numerical Integration of the Spherical Harmonics Equations," Nuclear Science and Engineering, 18(4), 459-467 (April 1964).

A method is formulated for numerical integration of the spherical-harmonics equations in the case of cylindrical geometry. This method avoids many of the difficulties of the usual analytical techniques and allows space-varying sources as well as regions of low neutron cross section and large physical size.

The usual spherical-harmonic equations (truncated) are presented in cylindrical geometry. To obtain a set of equations which (because they are more intuitive in form) lead to readily manageable numerical solution, the equations are converted to the discrete ordinate form in cylindrical geometry. From the discrete-ordinate equations, one may readily discuss inward- and outward-going neutrons. Based on this, reflection matrices are introduced at each radius r, one describing the reflection of inwardly directed neutrons by the medium inward of r and the other describing the reflection of outwardly directed neutrons by the medium outward of r. The complete source-independent properties of the medium are described by these reflection matrixes. Furthermore, the matrices can be obtained by numerical integration in a single pass, one by integrating from the center out and the other by integrating from the outside in.

The source can be treated by considering at each radius r the flux that escapes outward due to sources inward of r and by considering separately the flux that goes inward due to sources outward of r. The first of these escape fluxes is obtained by integration outward from the origin, using the corresponding reflection matrix, the second by integration inwards.

Once the above quantities have been found, the fluxes are obtained by solution of simultaneous algebraic equations (no further integrations).

Numerical results necessary for the use of this method in the P_3 approximation are also given.

609. W. H. Martin and D. M. Clare, "Determination of Fast-Neutron Dose by Nickel Activation," Nuclear Science and Engineering, 18(4), 468-473 (1964).

Fast-neutron dose measurement by the activation of nickel foils involves a correction for thermal-neutron burnup of Co^{58}, the daughter product of the (n,p) reaction. Fast-neutron irradiation of nickel produces Co^{58} in its ground and excited isomeric states, and recently the isomer has been shown to have a high thermal-neutron-absorption cross section. This paper considers how the determination of fast-neutron dose by nickel activation should be corrected for thermal-neutron burnup of both ground and isomeric states of Co^{58}. Results, which have been fully corrected, are compared with results obtained at low reactor power where the thermal-neutron burnup of Co^{58} and Co^{58m} is negligible. All the data considered were obtained from foils irradiated in rigs in hollow fuel elements in reactors of the DIDO type. The data demonstrate that accurate fast-neutron dose measurements, using nickel activation in high-flux facilities can only be made if the thermal-neutron cross sections of Co^{58} and Co^{58m} and the branching ratio of the Ni^{58} (n,p) reaction have previously been determined in the neutron spectrum being utilized.

610. K. Shure, "P-3 Multigroup Calculations of Neutron Attenuation," Nuclear Science and Engineering, 19(3), 310-320 (1964).

Neutron penetration in water and in iron/water shields has been calculated using a P-3 multigroup program. The thermal-neutron flux from a point fission source in water obtained from calculation and experiment agree to within 18% in the region between 15 and 140 cm, covering more than 9 decades of attenuation. The calculated neutron spectrum compares favorably in shape and magnitude with moments-method results out to 120 cm of water. The observed variations of the thermal-neutron flux in an iron/water shield are predicted by the P-3 program. Some of the differences between experiment and the predicted thermal flux within a thick iron region are due to the single-energy-group treatment in the calculations. Uncertainties in the high-energy cross sections for iron are of sufficient magnitude to account for differences between calculation and experiment noted in the water region following iron.

611. F. F. Haywood, J. A. Auxier, and E. T. Loy, An Experimental Investigation of the Spatial Distribution of Dose in an Air-Over-Ground Geometry, CEX-62.14 (October 2, 1964) Availability: $4.00 - Clearinghouse for Federal Scientific and Technical Information, NBS, U.S. Dept. of Commerce, Springfield, Virginia.

An experiment was conducted to determine the effect of the presence of the air-ground interface on the measurement of neutron first collision tissue dose and gamma-ray exposure in air above the earth's surface. Measurements of neutron dose were made in radiation fields from the Health Physics Research Reactor operating at steady-state power and supported at several heights above the earth's surface (up to 1500 feet). Measurements of gamma-ray exposure were made in radiation fields from the HPRR and from a Co^{60} source supported in a similar geometry.

To substantiate the experimental data, comparisons were made to data which were collected during nuclear weapons tests. Values of relaxation length for neutron dose and gamma-ray exposure were compared in both cases with satisfactory agreement.

Several comparisons of experimental data have been made with computed data. In comparing experimentally determined boundary correction factors K(exp), for neutrons, with values of K computed by Ritchie, $\pm$ 10 per cent agreement was obtained from a slant range of 100 meters to a slant range of 1000 meters. Ritchie and French have computed the decrease in the infinite medium neutron dose, at large slant ranges, caused by the presence of the air-ground interface. A comparison was made between these values and experimental data, but it was not found to be acceptable. Experimental values of dose as a function of detector height were compared to French's computed values indicating $\pm$ 2 per cent agreement from 10 feet to 100 feet above the interface and $\pm$ 6 per cent at the interface.

It was experimentally determined that, in general, the presence of the air-ground interface enhances the measured radiation at slant ranges close to the source and decreases the measured radiation at large slant ranges.

612. N. E. Scofield, R. L. Lynn, and W. E. Kreger, Penetration of Plane Normal and Plane Slant Gamma Rays Through Slabs of Aluminum and Steel. I. Angular and Energy Distributions (Experimental Pulse Heights), USNRDL-TR-272 (October 31, 1958) Availability: Not for sale to the public.

Measurements have been made of the scattered gamma radiation emerging from slabs of aluminum and steel exposed to Co^{60} and Cs^{137} gamma radiation with plane normal and plane slant (only for Co^{60}) incidence. The measurements were made using a 4-in. diameter by 4-in. long NaI(Tl) scintillation detector unit with a lead shield and collimator. Pulse-height spectra were recorded in a 100-channel analyzer for detection angles of 5°, 10°, 20°, 30°, 45°, and 60° for the normal incidence case. Slab thicknesses used were from 1 to 5 mean free paths for the primary source gamma radiation (1 to 5 inches of steel and 3 to 15 inches of aluminum for Co^{60}, 3/4 to 3 3/4 inches of steel and 2 to 10 inches of aluminum for Cs^{137}). Slant angles of radiation incidence were -50°, -40°, 20°, 30°, 40°. Corrections are discussed and results are plotted as families of pulse-height distributions for the various materials, source energies, etc. General conclusions are drawn from the data.

613. N. E. Scofield and L. G. Haggmark, Penetration of Plane Normal and Plane Slant Gamma Rays Through Slabs of Aluminum and Steel. II. Angular and Energy Spectra (Photon Number Flux), USNRDL-TR-475 (October 25, 1960) Availability: Not for sale to the public.

Experimentally measured differential energy and angle spectra of the scattered gamma-photon number flux are presented for Cs^{137} and Co^{60} source gamma rays normally incident upon slabs of aluminum and iron varying in thickness from one to five mean free paths. Spectral measurements for oblique angles of source photon incidence were also made for Co^{60} and slabs of iron with slant thicknesses of two and five mean free paths. Detection angles of 5°, 10°, 20°, 30°, 45° and 60° were used. Spectra are compared to show the effects of slab material, slab thickness, source energy, detection and incidence angle as well as the proportion of single and multiple scattering. The results have been normalized to a primary incident flux of one photon cm^{-2} sec^{-1}.

614. J. T. Serduke and C. V. Smith, Penetration of Plane Normal Gamma Rays Through Slabs of Aluminum and Steel. III. Processed Numerical Data, USNRDL-TR-730 (February 17, 1964) Availability: OTS $1.75 as AD 600392.

This report presents in tabular and graphical form data from experimental measurements of scattered gamma radiation from slabs of aluminum and steel as reported in USNRDL-TR-475. Monoenergetic gamma rays from Cs^{137} and Co^{60} sources with plane-normal incidence were used as irradiation sources. Detection angles used were 5, 10, 20, 30, 45, and 60 deg with effective thicknesses of one, two, three, four, and five "mean-free-paths" of slab material. Since it was desirable to have spectra for larger angles (75 and 85 deg), a method of extrapolation was devised. The extrapolated spectra are presented in the text. Also included in this report are calculations of dose and build-up factors. There is also shown (in graphical form) how the over-all dose rate is affected by the added extrapolated spectra.

615. J. Shapiro and W. D. Welte, An Investigation of Neutron Attenuation by Castable Polyethylene, RAS-33 (December 15, 1960) Availability: General Dynamics Corp., Groton, Connecticut.

Radiation attenuation measurements were made on unirradiated samples of castable polyethylene and on samples that had been exposed to 10^7 rads of reactor and Cobalt 60 gamma radiation. An investigation was conducted on the effect of voids on fast neutron dose rates through shields. Gaps in shields were traversed with various detectors and a probing method utilizing a 5 curie Pu-Be neutron source and neutron scintillation survey meter was found suitable for the detection of voids in polyethylene shields.

A soil moisture detection method based on the thermalization of neutrons was applied to the measurement of the hydrogen content of castable polyethylene samples and results were encouraging for use of the method in quality control of manufactured castable polyethylene shields.

616. R. B. Walton and R. E. Sund, Final Report - Research on Delayed Gamma Rays from Fission, RTD-TDR-63-3072 (November 1963) Availability: General Dynamics Corporation, San Diego, California.

Absolute delayed gamma-ray intensities from the photofission of U^{238}, U^{235} and Th^{232} were measured as a function of time after bursts of fissions produced by the bombardment of samples with 20-Mev X rays from an electron linear accelerator. For all three target nuclei, components of short-lived gamma rays contributed markedly to the gamma-ray activity for times up to 800μ sec after the beam pulse. Beyond 800μ sec the gamma-ray activity decreased little for times up to about 10^{-1} sec. After 0.1 sec the gamma-ray decay curves have the same shapes as those observed by other investigators for the fission of the same target nuclei with neutrons, but the absolute intensity from photofission is approximately a factor of two less than that from neutron fission.

Techniques were developed to use neutrons produced by the electron linear accelerator to investigate delayed gamma rays from neutron fission. Exploratory measurements on U^{235} (n,F) for times between 200μ sec and 30 msec after fission confirmed the existence and spectral composition of the isomeric component observed in photofission.

617. S. G. Tsypin, V. I. Kukhtevich and Yu. A. Kazansky, "The Penetration of Gamma Rays through Water, Iron, Lead, and Combinations of Iron and Lead," Soviet J. of Atomic Energy (English Translation) 1(2), 217 (1956).

The dose rate attenuation of γ-rays is measured in iron, water, and lead for "infinite" geometry. The dose rate attenuation of γ-rays is measured in mixtures of iron and lead for "barrier" geometry. The experimental data obtained is compared with the results of calculations as presented in reference(1).

618. V. V. Mateev, A. D. Soklov, R. S. Shlyapnikov, "Energy Distribution of Gamma Quanta from a Point Source of Gamma Radiation in an Infinite Sandy Medium," Soviet J. of Atomic Energy (English Translation) 1(4) 507 (1956).

Gamma spectra from point sources (Cr^{51}, Zn^{65}, Ra and MsTh) in an infinite sandy medium were studied with the aid of the scintillation spectrometer. The results obtained indicate that when gamma radiation passes through a thickness of sand there takes place an accumulation of soft scattered radiation of energy $\sim$ 50 kev independently of the energy of the initial radiation.

619. B. P. Bulatov and E. A. Garusov, "Albedo of Co-60 and Au-198 Gamma Rays from Various Substances," Soviet J. of Atomic Energy (English Translation), 5(6), 1563-1570 (1958).

Absolute values are obtained in the present investigation for the energy albedo of the γ-radiation of Co-60 (1.17, 1.33 MeV) and Au-198 (0.410 MeV) from various substances, at angles of 0°, 45°, and 60° from the original beam.

The dependence of the magnitude of the albedo on the effective atomic number and thickness of the scattering material was investigated; in addition, the angular distribution of the intensity of the scattered radiation and its spectra composition were determined.

The energy flux of the primary radiation was determined by use of an ionization chamber and "Roentgen KH-5" photographic film. The intensity of the scattered radiation was measured with a gas-filled counter with a practically constant sensitivity to γ-quanta of different energies. The spectral composition of the scattered radiation was determined by the method of absorption in lead.

620. O. I. Leipunskii, "Gamma Rays in the Atomic Explosion," Soviet J. of Atomic Energy (English Translation), 6(1), 23 (1959).

The question of γ-radiation in an atomic explosion is discussed. The physical factors which are essential for the appearance of γ-radiation are analyzed. The influence of a shock wave and atmospheric heating on the passage of γ-rays are introduced into the calculation, as is the presence of capture gamma-rays.

A calculation is made of the γ-ray dosage, which, because of the effect of the shock wave, increases at a more rapid rate than proportionally to the explosion energy. It is shown that in the course of time the dosage rate passes through a minimum and a maximum. A hard component is present in the γ-radiation and appears at great distances.

621. J. H. Hubbell, "A Power-Series Buildup Factor Formulation. Application to Rectangular and Off-Axis Disk Source Problems," J. Res. Nat. Bur. Std., 67C(4), 291-306 (October-December, 1963).

The response (e.g., dose-rate of an isotropic detector to primary radiation from a finite plane source may, for points less than a mean-free-path distant, be evaluated as the sum over an infinite series. This series is derived as expanding the exponential dependence exp $(-\mu_o r)$ as a power series in $\mu_o r$ and integrating, numerically or analytically, each term over the source array. If the medium is effectively infinite and homogeneous, the scattering properties of this medium can be characterized by a point isotropic source buildup factor. Buildup factor data are often approximated by formulas having simple analytic dependences on $\mu_o r$ and numerical parameters independent of $\mu_o r$. Any such set of parameters can be used to generate a set of weight functions b_n for an infinite series buildup factor representation

$$\exp(+\mu_o r) \sum_{n=0}^{\infty} b_n \cdot (-\mu_o r)^n/n!$$

which can be used with the above primary radiation series solution to give instead, a series solution including both primary and scattered radiation. Tables of b_n's (air-dose buildup only) derived from coefficients of cubic polynomials fitted to the Goldstein-Wilkins data (NYO-3075) are given for $0 \leq n \leq 13$ at primary photon energies of 0.5, 0.7, 1.0, 1.5, 2.0, 3.5, 5.5, 7.5, and 9.5 Mev in water, Al, Fe, Sn, W, Pb, and U. The method is applied to give: (1) a comparison with dose-rates measured by a detector separated from a Co-60 rectangular plaque food irradiator by a layer of steel and a layer of water and (2) dose-rate profiles at constant heights across a cleared circular area in a Co-60 infinite plane source in air.

622. D. L. Broder, Yu. P. Kayurin, and A. A. Kutuzov, "Transmission of Gamma Radiation Through Heterogeneous Media," Soviet J. of Atomic Energy (English Translation), 12(1), 26-31 (January 1962).

The paper describes an experimental determination of the buildup factor in heterogeneous media consisting of different combinations of materials (polyethylene, Al, Fe, Pb) for an isotropic Co-60 point source (mean radiative energy 1.25 Mev). The experiment showed that the existing recommendations usually give unsatisfactory results in calculating dosages for shielding consisting of two layers of different materials, especially at large distances from the source. On the basis of the experimental data on empirical formula is deduced, corresponding to physical concepts and permitting rapid and facile computation of the buildup factors for heterogeneous mixtures of any number of layers from the known buildup factors for the homogeneous media comprising the ensemble. The experimental buildup factors for heterogeneous combinations of media -- polyethylene, Al, Fe, Pb -- with thicknesses of 0.7 to 15 mean free paths for a Co-60 source differ from those calculated on the basis of the given formula seldom by more than 10%. It may be assumed on the basis of the monotonic variation in the buildup factor with increasing energy to the value E_{cr} (where E_{cr} is the energy for which the absorption of the given material becomes a minimum), that the derived formula can also be used for large energies.

623. B. R. Bergel'son, "Calculating the Streaming of Fast Neutrons Along the Cylindrical Channels in a Biological Shield," Soviet J. At. Energy (English Translation), 10(4), 378-380 (April 1961).

An embellishment of the Simon-Clifford equation for a straight duct is given using some age theory calculations.

624. Ya. Rumyantsev, "Boundary Conditions in the Method of Spherical Harmonics," Soviet J. of Atomic Energy (English Translation), 10(1), 22-32 (January 1961).

The equations of the method of spherical harmonics are derived for use with the single-velocity transport equation. From an analysis of the equations, conditions are obtained for boundary between two different scattering media. It is shown that these conditions completely define a solution in any-order approximation (we are concerned here with the so-called P_N approximations) including even-order approximations. The possibility of using even-order approximations is of major practical interest. Thus, for example, the relatively simple P_2 approximation may serve as a method for improving the elementary diffusion theory without introducing excessive complexity.

Even-order approximation, and in particular the P_2 approximation, have not as yet been used in calculations.

625. Yu. A. Kazanskii, "Angle-Energy Distribution of Gamma Radiation Scattered in Water and Iron," Soviet J. of Atomic Energy (English Translation), 8(5), 364-370 (May 1960).

In this paper we investigate the detailed characteristics of multiple scattering of γ-rays, in particular the angle-energy distribution, a knowledge of which allows us to calculate the attenuation of γ-radiation in complex geometries. Measurements have been made of the angle-energy distribution of γ-rays from a Co^{60} source in water and iron for a semiinfinite geometry and it has been established that these distributions have a maximum close to the energy corresponding to single scattering at the minimum angle. It is shown that the angular distributions are exponential and that the exponential factor is a linear function of the atomic number of the medium. The energy distributions are compared with calculations carried out by Goldstein and Wilkins.

626. D. L. Broder, A. A. Kutuzov, V. V. Orlov, and A. V. Turusova, "Passage of Fast Neutrons Through Lead and Iron," Soviet J. of Atomic Energy (English Translation), 7(4), 797-803 (October 1959).

The present article describes the results obtained in measuring the spatial distribution of fast neutrons, which are emitted from sources of monoenergetic neutrons, with E_0 = 4 Mev and E_0 = 14.9 Mev, as well as neutrons from atomic reactors, in lead and iron.

In order to calculate the space-energy distribution of fast neutrons at large distances from the source, we have developed a method of solving the kinetic equation for media where the neutrons are moderated due to inelastic scattering on nuclei. The anisotropy of elastic scattering is taken into account. The energy losses of neutrons in elastic scattering are neglected.

627. M. Leimdorfer, "A Monte Carlo Method for Analysis of Gamma Radiation Transport from Distributed Sources in Laminated Shields," Nukleonik, 6(2), 58-65 (March 1964).

A description is given of a method for calculating the penetration and energy deposition of gamma radiation, based on Monte Carlo techniques. The essential feature is the application of the exponential transformation to promote the transport of penetrating quanta and to balance the steep spatial variations of the source distributions which appear in secondary gamma emission problems. The estimated statistical errors in a number of sample problems, involving concrete shields with thicknesses up to 500 cm, are shown to be quite favorable, even at relatively short computing times. A practical reactor shielding problem is also shown and the predictions compared with measurements.

628. M. Leimdorfer, "On the Transformation of the Transport Equation for Solving Deep Penetration Problems by the Monte Carlo Method," Transactions of Chalmers University of Technology, Gothenburg, Sweden, 1964.

The present report describes a modification of the "exponential transformation" of the Boltzmann transport equation which removes a singularity in the transformed scattering kernel, discussed by the author in an earlier paper. The effect is obtained by limiting the tracks of the particles to the problem boundary, whereby a zero "virtual" cross-section may be permitted. A sampling technique for the new inter-collision distance is given.

629. D. J. Raso, "Scattering of Gamma Rays in Laminar Media," Radiation Research, 19(2), 384-391 (1963).

Energy deposition, transmission, and reflection data for 0.66 Mev and 1.25 Mev gamma rays normally incident on water slabs are presented. The results were generated by a Monte Carlo calculation. The fraction of energy backscattered by 1 mfp of water is about 85% of the semi-infinite case for 1.25 Mev photons and about 75% for 0.66 Mev photons.

630. H. A. Mehlhorn, E. T. Clarke, R. Gold, and R. McMath, Final Report on Attenuation of Point Source Gamma Radiation in Slabs, TO-B-62-13 (DASA 1318), (September 1962) Availability: OTS $3.60 as AD-436311.

Data are reported for an experimental program providing measurements of attenuation of gamma rays in steel slabs, to compare with theoretical work being done at the National Bureau of Standards. Gamma rays from a cobalt-60 source, suspended in air, strike a steel slab at three angles of incidence. Transmitted radiation is detected by five ionization dose-rate meters at various distances below the slab. Information is given for 2-ft square slabs from 1/4 to 2 in. thick, and detectors from 1 to 6 ft distant. It was found that slab-scattered radiation was approximately proportional to the solid angle subtended by the slab, that less radiation was transmitted at large polar angles of incidence, and that at least 2 in. of steel is needed to produce shielding improvement in a slab-covered hole.

631. Herman Kahn, "Random Sampling (Monte Carlo) Techniques in Neutron Attenuation Problems -- I and II," Nucleonics, 6(6), 60-65 (June 1950).

A basic description of random sampling is given as an introduction to Monte Carlo. A simple Monte Carlo neutron penetration calculation is described and improvements to the analog procedure are suggested. These improvements include most of the known importance sampling techniques. Warnings regarding the reliability of Monte Carlo calculations are also given.

632. George A. Bennett, "Nomogram for Calculating Shielding for Co^{60}," Nucleonics, 8(4), 55-58 (April 1951).

The cascade decay of cobalt-60 complicates shielding calculations for this γ-emitter. A nomogram for use with a point source eliminates mathematical analysis in individual problems.

633. Engineers of the Service, Handbook for the Calculation of Reactor Protections, AEC-tr-6349 (1963) Availability: JCL $12.00 (fs), $5.12 (mf).

This is a general reference but more oriented toward the design of shielding for graphite moderated reactors. There are many tables and formulas in the appendices.

634. S. K. Penny, Derivation of First- and Last-Flight Estimators for Monte Carlo Computations of Scattered Particle Fluxes in Attenuating Media with Spherical or Plane Symmetry, ORNL-TM-974 (October 1964) Availability: OTS $1.00 (fs).

Estimators for the first, last, and in-between flights are derived for use in Monte Carlo computations of scattered particle fluxes in attenuating media with spherical or plane symmetry. The derivation is based upon the integral form of the Boltzmann Equation.

635. D. L. Broder, A. P. Kondrashob, A. A. Kutuzov, V. A. Naumov, Yu. A. Sergeev, and A. V. Turusov, "The Use of Multigroup Methods in Biological Shielding Calculations," Soviet J. of Atomic Energy (English Translation), 12(2), 139-150 (February 1962).

The article describes a seven-group method and a ten-group method, which are applicable to the design of biological shielding for nuclear reactors.

The results of the calculations of the space-energy distribution of neutrons for several variants of the shielding were checked experimentally.

Good agreement with experimental results was obtained both for the seven-group method, based on a semi-empirical determination of the fast-neutron group, and for the ten-group method.

The results of the calculations and experiments are listed.

636. V. N. Mironov, "Passage of Radiation Through a Test Hole," Soviet J. of Atomic Energy (English Translation), 12(3), 220-224 (March 1962).

This paper discusses the passage of monoenergetic radiation through rectangular, cylindrical, and ring-shaped test holes. It is assumed that the sources are located at the far end and on the lateral surface of the test hole. Formulas are derived for calculating the radiation flux in test holes passing through both the shield and the core of the reactor. The results of the theory are compared with experiment.

637. U. I. Kukhtevich and B. P. Shemetenko, "The Spatial Distribution in Water of Multiply Scattered Gamma Quanta from Unidirectional Au-198, Co-60, Na-24 Sources," Soviet J. of Atomic Energy (English Translation), 12(3), 214-217 (March 1962).

We have measured the spatial distribution of the dose of γ-radiation from collimated Au-198, Co-60, and Na-24 sources, where the radiation has been multiply scattered in an infinite aqueous medium. To show the use of a unidirectional source based on the obtained data we have plotted the attenuation in water of γ-rays with energies of 2.76, 2.07, 1.25, and 0.411 Mev for a point isotropic source and with an energy of 1.25 Mev for a plane unidirectional source. The results are in good agreement with the data of (1) except for γ-rays with an energy of 0.411 Mev. We have also given the empirical dependences of the angular distribution of the dose on the initial energies of the γ-rays and the distance between the source and the detector.

638. V. A. Dulin, Yu. A. Kazanskii, and I. V. Shugar, "The Energy Distribution of Scattered Neutrons in Water," Soviet J. of Atomic Energy (English Translation), 14(4), 411-412 (April 1962).

Experimental results are presented for the energy spectrum of neutrons penetrating water from a 15-Mev source. These are compared with moments method calculations.

639. N. G. Gusev, V. P. Mashkovich, and B. V. Verbitskii, "Universal Tables for Calculating Gamma Attenuation in Thin Filters," Soviet J. of Atomic Energy (English Translation), 13(5), 1101-1102 (November 1962).

A variety of filters is commonly used to attenuate γ-radiation (walls of irradiation capsules and source containers, cutoff filters and smoothing filters). It is not possible to calculate beforehand, for all of the different sources, the emission characteristics to be found after the radiation has passed through the various filters encountered in practical work. It is consequently feasible to set up tables similar to the tabulated arrangements found in (1), to compute the attenuation of γ-radiation in filters made of different materials and in different sizes.

640. B. P. Bulatov, "Some Characteristics of Gamma Ray Scattering at the Boundary of Two Media," Soviet J. of Atomic Energy (English Translation), 13(5), 1057-1061 (November 1962)

The characteristics of γ-ray scattering ($0.33 \leq E_\gamma \leq \sim 1.25$ MeV) at the boundary between two media were experimentally investigated. It was established that the scattered radiation was produced mainly by a narrow beam of primary radiation. The effective scattering area was contained within a region with a radius $r \leq 10H$ (for $h \leq z$). The contribution to the total energy from radiation scattered by the medium increased for angles of incidence of the primary "narrow" beam up to $\alpha \simeq 85\text{-}88^\circ$, and then decreased monotonically.

An estimate was made of the effective dimensions of the area from which scattered quanta were emitted into the backward half-space when an isotropic source was located at the surface of the scatterer.

The energy flux of the scattered radiation was measured with a gas counter which had a practically constant sensitivity for γ-rays of different energies.

641. M. A. Van Dilla and G. J. Hine, "Gamma-Ray Diffusion Experiments in Water," Nucleonics, 10(7), 54-58 (July 1952).

The gamma ray dose rate in water from point radioactive sources was measured in an infinite medium and near an air-water interface. The dose rate as a function of water level is given.

642. G. J. Hine and R. C. McCall, "Gamma-Ray Backscattering," Nucleonics, 12(4), 27-30 (April 1954).

The intensity and energy distribution of backscattered gamma rays were measured with a NaI (Tl) crystal. Two types of scattering experiments were performed. First, backscattering materials of different atomic numbers were placed in close contact with Hg^{203}, Cs^{137}, and Co^{60} sources. In the second group of experiments, aluminum was always used as the scattering material, but the geometry of the scatterer and its position relative to the source were varied.

643. Fred Maienschein and Temple Love, "Gamma-Ray Spectrum of the Bulk Shielding Reactor," Nucleonics, 12(5), 6-8 (May 1954).

Scintillation spectrometers were used to measure the gamma ray spectra at various distances in water from the Bulk Shielding Reactor. At the highest energy, the experimentally-measured attenuation length agrees with the value expected from the cross section. At lower energies, the effect of buildup is reflected in the larger measured attenuation lengths.

644. Harold S. Davis, "How To Choose and Place Mixes for High-Density Concrete Reactor Shields," Nucleonics, 13(6), 60-65 (June 1955).

Selecting the best concrete for shielding means choosing from among densities of 150-410 lb/ft, at material costs from $0.50 - 25.40/ft. This article tells how high-density concrete can be designed and placed at lowest cost. Properties of available heavy aggregates and usable mixes are given, plus construction techniques.

645. David G. Chappell, "Gamma-Ray Attenuation," Nucleonics, 14(1), 40-41 (January 1956).

A nomogram is presented to aid in the calculation of gamma-ray attenuation factors.

646. David G. Chappell, "Gamma-Ray Scattering from Thin Scatterers," Nucleonics, 14(7), 36-37 (July 1956).

Nomograms are given which permit rapid estimation of the photon flux reflected from a thin object.

647. David G. Chappell, "Gamma-Ray Streaming Through an Annulus," Nucleonics, 14(11), 87 (November 1956).

A nomogram is presented to aid in calculating gamma ray streaming through an annulus. Scattering effects are not accounted for.

648. David G. Chappell, "Gamma Attenuation with Buildup in Lead and Iron," Nucleonics, 15(1), 52-53 (January 1957).

Nomograms are presented which give gamma-ray point source attenuation factors for lead and iron shielding slabs.

649. David G. Chappell, "Gamma-Ray Streaming Through a Duct," Nucleonics, 15(7), 65 (July 1957).

A nomogram is presented to calculate geometrical attenuation of the gamma-ray flux streaming through the duct from an emitting surface upon which the duct abuts perpendicularly. Scattered effects are not accounted for.

650. F. T. Binford, "Gamma-Ray Heating in BSR," Nucleonics, 15(3), 93-97 (March 1957).

To determine the rate of gamma heating in aluminum, lead, iron, and copper structures near reactor cores, temperatures of these materials were measured during heating in the gamma field of the Bulk Shielding Reactor and during cooling after removal from the field. Heat generation rates were determined for most of the samples at distances of up to 4 inches from the reactor face. In the case of lead, data is obtained at distances up to 9 inches.

651. R. L. French, "Calculating Gamma Spectra from Reactors," Nucleonics, 18(3), 114-117 (March 1960).

A method was developed to calculate gamma ray energy spectra- and dose rate in laminated media from homogeneous medium data (such as that generated by the moments method). The moments method data is successively applied to each layer between the source and detector. The results compare well with experiment.

652. R. L. French and J. B. Eggen, "Fast-Neutron and Gamma Spectra for BSR," Nucleonics, 18(3), 117-120 (March 1960).

Detailed tables of gamma ray and fast neutron spectra in the water of the ORNL Bulk Shielding Facility are presented. The data was generated by integrating moments method data. The results agree well with experiment.

653. I. O. Huebsch, The Development of a Water-Surface-Burst Fallout Model: The Rise and Expansion of the Atomic Cloud, USNRDL-TR-741 (April 23, 1964) Availability: AD-441983 Not for sale to the public.

A water-surface-burst fallout model is being developed in three parts: (1) cloud rise and expansion, (2) particle formation, and (3) particle fall through the atmosphere. Part (1) is discussed in detail. The cloud is described by a set of equations for rate of rise, temperature, water content and other parameters, using an "entraining parcel" method. The questions allow for variations in atmospheric temperature, pressure and humidity as well as in initial cloud height, temperature, and air/water energy partition. Temperature and volume are calculated by extensions of standard meteorological equations.

The equations have been programmed for machine computation. Comparison of numerical results with atomic-cloud observations indicates that a single empirical parameter, the entrainment constant λ, (which is the proportionality constant in assumption 1) can also be used as a measure of the rate of generation of turbulence. Correct predictions are obtained for rate of rise, cloud size, final cloud height and the later horizontal expansion of high yield clouds. This last has not been previously predicted by a cloud model. Numerical results also indicate that for low yields, the effect of initial water content of the cloud is minor compared with that of atmospheric humidity while for high yields the opposite is true.

654. Emanuele Fumagalli, The Use of Lead Glycerolates for Highly Effective Biological Shields, AEC-tr-6355 (July 1963) Availability: JCL $1.60 (fs), $0.80 (mf).

In continuation of the research program already conducted at ISMES on concretes for biological shields of power reactors, highly effective shielding mixes with organic binders have been subjected to study, which are particularly indicated for applications in research reactors and for transport containers.

It appeared that cements on a litharge and glycerine base chemically designated as lead glycerolates, could be effectively useful for this purpose.

In a preceding publication, we already had the occasion to consider the possibility of utilizing heterogeneous mixes in order to obtain a more equilibrated shielding action. Moreover, since highly efficient materials are necessary, it was considered suitable to supplement the shielding action of lead, which always constitutes the base of the mix, by the addition of iron.

It is not the purpose here to furnish information on the mixing ratios to be utilized for the two; these will be determined on the basis of the type of incident radiation and when necessary subsequent to the verification of the calculation for the project.

After formulating the mixes, which are applicable in practical installations, experiments are carried out in order to determine the physico-mechanical characteristics which are of interest for application purposes.

655. William H. Steigelmann, Jr., "Gamma-Ray Shielding Design Curves," Nucleonics, 19(11), 148-150 (November 1961).

Attenuation functions for various radioactive gamma ray sources are presented. These curves permit rapid and accurate determination of the dose rate from a point source with either a lead or concrete shield.

656. F. Allen and A. Futterer, Lightweight, Radiological Shielding Materials for Armored Vehicles, BRL Memo 1424 (July 1962) (CONFIDENTIAL)

The neutron shielding potentialities of several high hydrogen content materials are studied for possible application to an armored vehicle. It is found that for the type of shielding arrangement presently under consideration, materials with high hydrogen weight fractions (even though this implies relatively low physical densities!) are the best.

657. F. Allen, A. Futterer, W. Wright, and A. Budka, The Transmission of Monoenergetic Neutrons Through Borated Polyethylene, BRL-1129 (April 1961) (CONFIDENTIAL)

A description of a versatile Monte Carlo neutron transport code for multilayer, laterally infinite, plane shields is presented. Curves are given which show the principal results obtained for the transmission of monoenergetic neutrons through, and their reflection from, polyethylene slabs containing 8 per cent boron carbide by weight. A discussion of the results is also given. Because of their great bulk, detailed computing machine calculations are being issued in a companion report, BRL 1130.

658. R. H. Karcher, B. J. Garrick, Design Information for Protection Against Initial Radiation, HN-179 (May 29, 1964) (SECRET)

This manual summarizes information on shield design methods for protection against radiation emitted during the first minute after the detonation of a nuclear explosion. A large amount of tabulated and graphical data is presented.

659. W. E. Edwards, C. W. Moon, J. M. Crook, L. S. Burns, Gas-Cooled High-Temperature Nuclear Reactor Design Technology, Section 7. Shield Design, APEX-800 (Part E), (June 30, 1962) (CONFIDENTIAL).

The technology developed by General Electric's Aircraft Nuclear Propulsion Department (GE-ANPD) was sufficiently accurate and versatile to assure the feasibility of designing a shield system light enough in weight for aircraft application. This section:

1. Describes the procedures used in designing shields for gas cooled aircraft nuclear power plants
2. Describes the technology developed and used in these procedures
3. Discusses the treatment of special problems encountered
4. Suggests development procedures for future shielding tasks of comparable complexity.

660. W. E. Francis and L. L. Marsden, "Gamma-Ray Dose and Heating from Spent MTR Fuel Elements," Nucleonics, 15(4), 80-83 (April 1957).

Gamma ray dose rates were measured in water close to MTR fuel elements which had different histories. The decay was observed for 120 days. Heating was also measured by comparison with electric heating.

661. Eric T. Clarke and James O. Buchanan, "Radiation Shielding Against Fallout," Nucleonics, 20(8), 143-146 (August 1962).

Data is presented for evaluating the protection offered by various structures against gamma radiation emitted by fallout from nuclear weapons.

662. F. E. Obenshain, The Theoretical Buildup Factor for Ordinary Concrete, WAPD-P-646 (March 1955) Availability: Dep. (mc); $1.80 (ph); $1.80 (mf), OTS.

The build-up factor for ordinary concrete (2.3 gm/cm^3) presented in this note was theoretically determined for gamma rays having the initial energy between .5 and 10 mev. The values given were derived from the N.D.A. report 15C-41. The result was found to be in excellent agreement with experiment at 6.1 mev.

663. F. J. Allen, A. T. Futterer, W. Wright, Neutron Reflection and Flux Versus Depth for Aluminum with an Appendix -- Neutron Dose Transmission Versus Thickness, BRL-1238 (February 1964) Availability: OTS $9.10 as AD 439 641 or AD 600 755.

Detailed calculated results of neutron reflection and flux versus depth for aluminum are given in the form of machine printouts. The angular and energy distributions of the reflected neutrons along with the energy-dependent and total flux at various depths are contained in tabular form on the printouts. Neutron number current, number flux, and dose transmission, as functions of thickness, are also given in tabular form on the printouts.

A table of summary information on reflection is presented. This contains number current, number flux, dose, and energy reflection factors as functions of incident energy and angle.

A few figures are presented to illustrate graphically the various tabular results.

An appendix contains the dose transmission results as a function of incident angle and slab thickness for the various incident energies.

664. D. E. Lamkin and V. Keshishian, A Study of the Applicability of the DTK Code to the Calculation of the Thermal Neutron Flux in a Reactor Shield, NAA-SR-Memo 9458 (January 1964) Availability: OTS $2.60 (fs), $0.83 (mf).

It was found that the DTK code does not compute the magnitude of the thermal neutron flux accurately at deep shield penetrations. However, the calculated neutron spectrum was observed to be fairly accurate. Using this fact, the thermal neutron flux in two shields was estimated quite accurately by using the DTK spectra in conjunction with fast removal theory to determine the magnitude of the fast flux.

665. C. R. Greenhow, G. O. Mueller, F. E. Sabian, Use of Diffusion Theory Codes to Predict Gamma Heating, KAPL-M-DNA-5 (January 1963) Availability: Dep. (mc); $5.60 (fs), $1.91 (mf) OTS.

A method utilizing diffusion theory and existing nuclear codes has been developed which eliminates the necessity of transforming the true geometry and spatial gamma source distribution into an "equivalent" problem with one of the standard geometrical sources and a simple analytical spatial source distribution. This method also does away with the use of build up factors, which in a case with finite, complicated geometry; more than one medium; and slant penetration; can be only crudely estimated.

A three step process is involved in obtaining gamma heating values by diffusion theory. First, a standard neutron diffusion problem is solved for the core or configuration being investigated.

The second step of the calculation is to use "pseudo fission cross sections" to generate the gamma source from the converged multigroup fluxes of the neutron diffusion problem. Spatial and energy variations of the gamma source may thus be calculated to the same accuracy as the original neutron diffusion problem.

As a last step to this gamma diffusion problem the resulting multigroup gamma fluxes are multiplied by suitable energy absorption coefficients to obtain the gamma heating. These energy absorption coefficients are supplied by GAM I.

Calculations using this method have been made for several cases with simple geometry and source distribution and the results compare favorably with other, more widely accepted calculational methods. While these calculations are for regions near the core where the gradients are not steep, the use of a transport correct diffusion constant and a pre-calculated gamma source will give accurate values for gamma heating even in regions that are several mean free paths from the reactor core.

666. E. G. Peterson, Shielding Calculations for a Power Reactor, HW-72844 (February 28, 1962) Availability: Dep. (mc); $1.10 (fs), $0.80 (mf) OTS.

Five types of thermal shielding enclosed by ordinary concrete were analyzed. Thermal shields include ordinary water, borated carbon steel, cast iron, a mixture of magnetite and water, and borated graphite. For the current study it is concluded that the use of boron and cast iron as a thermal shield cannot be justified economically although they possess excellent shielding properties. The magnetite-water mixture (light water with magnetite are added) is the best choice for a thermal shield because of its relatively low cost, excellent heat transfer properties, and the fact that cooling pipes are not needed in the concrete.

The calculations show the thermal neutron flux, total neutron dose rate, and gamma dose rate distribution through the various shield assemblies.

667. Lead Development Association, London, Lead for Radiation Shielding - No. 2. The Design and Construction of Lead Shielding Containers, NP-13691 (July 1963) Availability: Dep. (mc).

This discussion of the design and construction of lead shielded containers is intended for the guidance of designers so that they can take into account the problems of manufacture in preparing designs, and thus facilitate the use of techniques that best overcome these problems.

Types of containers, construction methods and design requirements of lead and lead-clad-steel containers are discussed.

668. A. W. Casper, Modified Fast Neutron Attenuation Functions, XDC 60-2-76 (1960) Availability: Dep. (mc); $2.60 (fs), $0.95 (mf) OTS.

New coefficients were determined for the Albert and Welton fast neutron attenuation function by constraining it to pass through three experimental data points. These coefficients are suggested for use in making fast neutron dose rate calculations in hydrogenous materials.

$$\alpha_1 = 1.5964 \times 10^9$$
$$\alpha_2 = 0.34921$$
$$\alpha_3 = 0.42231$$
$$\alpha_4 = 0.69842$$

Coefficients of the polynomial exponent in the following hydrogen attenuation function were determined by constraining the function to pass through four experimental data points.

$$\psi(X) = e^{20.523 - .0941X + .000014374X^2 + .0000008443X^3}$$

It would be valuable to make the use of this function optional in the shielding programs. Both functions give dose rates in mullirep per hour per watt. A careful study was made of computations using these functions and better results were obtained in all cases studied.

669. C. E. Johnson and C. A. Goetz, "SNAP-8 Reactor and Shield," Am. Inst. Aeron. Astronaut. J., 1(10), 2355-2361 (October 1963).

This paper summarizes the SNAP 8 reactor design and shielding requirements. It is hoped that the information will be useful for preliminary design studies of spacecraft employing SNAP 8. SNAP 8 is a 35-kw electrical power supply for use on spacecraft. It employs a nuclear reactor heat source to drive a mercury Rankine cycle turboalternator. The system is being developed jointly by the Atomic Energy Commission and NASA. The system has been under development for approximately three years. The major performance o jectives of the SNAP 8 reactor are 600 kw of thermal power, 1300°F NaK outlet temperature, 10,000-hr endurance, orbital startup and automatic control, high reliability, and low weight. To meet these performance requirements, a 500-lb reactor has been designed and is described in this paper. Use of a reactor imposes restrictions on the spacecraft design. It is shown that shield weight can vary from a few hundred pounds to several thousand pounds, depending on the payload dose tolerance and the spacecraft configuration. Approximation relationships are presented to facilitate shield weight estimates for preliminary design studies of unmanned spacecraft using the SNAP-8 reactor.

670. S. E. Van den Broek, "Lead Radiation Shielding," Nucl. Energy, 186-191 (July 1963)

The fabrication and use of lead as a shielding material is discussed.

671. Philip G. Berman, Neutron and Gamma Measurements. Mark I and Mark F Triga Reactors, DOFL-TR-1104 (HDL) (April 30, 1963) Availability: AD 444787 not for sale to public.

Integrated neutron-flux (fluence) and neutron-spectrum data have been obtained on the General Atomic Mark I and Mark F reactors. Limited gamma-ray dose data have also been obtained.

The data are presented graphically to facilitate determination of the radiation environment provided at the various test positions in the above reactors.

Gamma-ray doses were as high as 145,000 r/pulse; and neutron fluence above sulphur threshold (3 mev) were greater than 10^{14} n/cm^2.

672. Lazuhisa Shirayma, "Properties of Radiation Shielding Concrete," J. Am. Conc. Inst. 60(2), 261-279 (February 1963).

Reports on a study of boron-containing aggregates from the United States and England and of seven heavy aggregates from Japan. Data are presented and discussed concerning mix proportions, workability, unit weight, strength, drying shrinkage, and absorption coefficients for γ- and x-rays.

Based on data obtained for mix proportions, formulas for estimating the unit weight of concrete from the specific gravity of the aggregate are proposed.

The hematite ore used was found to be unsuitable for concrete aggregate because of an excessive amount of fines coating its surface. The tests indicated that, of the boron-containing material, the deleterious effects of colemanite on concrete strength and setting time is greater than that of borocalcite. This effect increases with increased fineness.

The barite concrete provided better shielding against x-rays than anticipated from density calculations. Barite or magnetite concretes were more suitable in shielding against gamma rays.

673. R. Schamberger, J. Celnik, R. Ross, Shield Small Source Experiments, UNC-5072 (November 8, 1963) Availability: Dep. (mc); $6.60 (fs), $2.27 (mf) OTS

A shielding experiment has been performed which provided a proof test of the shield calculation methods used in the MCR shield design and provides a partial confirmation of cross-section data. Measurements were made with a gamma ray spectrometer behind a configuration consisting of a layer of gamma shielding material followed by a neutron shield, with the configuration exposed to neutrons from a Po-Be source.

Secondary gamma rays produced in the assembly were measured with the spectrometer and compared directly with calculations. Calculations were performed with United Nuclear's ADONIS three-dimensional rectangular geometry Monte Carlo code, which is derived from the one-dimensional Monte Carlo codes used in the basic MCR shield analysis.

Experiments were performed with samples of iron, lead, and stainless steel, and calculations were completed for the no-sample configuration and for an iron sample. Comparison of experimental and calculated detector response for these two cases is shown. Agreement is within a factor of two at all energies, and the calculated energy deposited in the crystal is within 25% of that observed for the no-sample case and within 15% for the iron case. This agreement demonstrates the validity of the experiment and of the basic calculation techniques used, and gives support to the secondary gamma production cross sections used for iron. Modifications of the inelastic gamma spectrum used for iron is suggested by the results.

674. W. E. Edwards, Gamma Ray Production Cross Sections, GEMP-278 (April 7, 1964) Availability: Dep. (mc); $3.60 (fs), $1.28 (mf) OTS.

Gamma ray production cross sections are presented in this report as a function of 16 neutron lethargy groups and 11 photon energy groups for 13 elements and isotopes. They were computed by Program GAMMA-P and account for the production of gamma rays by neutron radiative capture, neutron inelastic scattering, neutron induced fissioning, and (n,α) reactions. The production cross sections are available on IBM decimal cards for use with transport theory Program S-X.

675. T. Atoda, and Y. Sasa, "Neutron-Shield Materials, (III) Rolling of Boral-Ingot into Sheet," Sci. Papers Inst. Phys. Chem. Res. (Tokyo) 57(2), 84-89 (June 1963).

Boral-ingots prepared according to the method reported previously by us are proved to have favourable qualities, capable of being rolled into sheets. Boral sheets, composed of a layer of aluminum and boron carbide mixture sandwiched between aluminum sheets, are produced successfully by rolling Boral-ingots enclosed in an aluminum frame and sheets. Hot as well as cold rolling is applicable for the sheet production. Small Boral-sheets of the size (400 ~ 800) mm X (65 ~ 70) mm X (3 ~ 6) mm, composed of inner layer of thickness of 2.0 ~ 4.6 mm and top and bottom wrappers of thickness 0.5 ~ 1.0 mm each, and larger sheets of the sizes, (800 ~ 900) mm X (715 ~ 1370) mm X (5.0 ~ 6.0) mm and 230 mm X 2000 x 6.0 mm, composed of inner layer of thickness 2.3 ~ 3.6 mm and top and bottom wrappers of thickness 1.2 ~ 1.6 mm each, are produced. Strength of bonding between inner layer and wrappers is found excellent.

Comparison of boron contents of Boral-ingots and -sheets determined by analysis with those of ingots and sheets calculated from the ratios in amount of constituents of the raw materials provided the successful execution of the series of studies--the preparation of Boral-melt, production of Boral-ingot, rolling of ingot into Boral-sheet and measurement of thermal neutron absorption coefficient.

676. E. R. Berry, A Generalized Shielding Study for Nuclear Electric Space Power Plants, SSD-TDR-63-217 (August 28, 1963) Availability: 5.60 OTS.

This report presents generalized shield weight and thickness curves suitable for providing rough estimates of the shield weights required for nuclear-reactor-powered electrical space powerplants, such as the SNAP series (Systems for Nuclear Auxiliary Power), over a wide range of conditions. The curves presented should also be applicable in many cases to nuclear rocket reactors operating in space. The shields considered are two-layer tungsten-lithium hydride shadow shields optimized to provide minimum shield weights.

677. B. A. Engholm, Shielding Studies for Target Reactor, GAMD-4526 (May 1, 1964) Availability: Dep. (mc); $4.60 (fs), $1.61 (mf) OTS.

Shielding analysis of the TARGET reactor concept has been limited to those problem areas in which the feasibility of the system or key subsystems might be in question. Areas investigated include the fast-neutron flux incident upon the vessel liner, nuclear heating in the side and top regions of the reactor, neutron activation of the circulator, radiation dosage to the gas-turbine lube system, and biological dose rates external to the vessel.

678. W. G. Homeyer, Shadow Shielding of a Nuclear Reactor: Effect of Reactor Shape on Weight of Shield, GA-1528 (April 24, 1964) Availability: Dep. (mc); $1.10 (fs), $0.80 (mf) OTS.

A brief study has been made of the weight of a shadow shield required for a reactor to be used in space. Shield weight was calculated as a function of the diameter of the reactor in order to determine the reactor configuration that would minimize the weight of the shield. An approximate general relationship was derived from which the optimum reactor dimensions can be calculated for reactors and shields for space applications.

679. V. D. Svikis, "Dense Lithium Fluoride for Gamma-Ray-Free Neutron Shielding," Nucl. Instr. and Methods, 25(1), 93-105 (December 1965).

A simple method of processing powdered LiF into dense and strong LiF, suitable for self-supporting neutron shielding, was developed. The processing was carried out in two stages. In the first stage, chemically pure LiF powder was slip-cast and sintered for periods of from 1 to 5 hours at temperatures ranging from 500° to 840° C, in order to establish the best conditions for making dense and volume-stable LiF aggregate. Aggregates were prepared with densities in the range 52 to 92% of theoretical, depending on sintering temperature and time. Maximum density was achieved by sintering at 700°C for 3 hours. In the second stage, dense and stabilized LiF aggregate was graded, pressed into shapes at 4000, 6000 and 8000 psi, and heated at 600° and 800°C to find the best conditions for making dense compacts. LiF compacts, as large as 8 3/4 by 4 1/2 by 3 in., with densities in the range 78 to 85% of theoretical, were prepared. Maximum density and good mechanical strength were attained when the graded aggregate was pressed into shapes at a pressure of 8000 psi and the shapes were heated for five hours at 800°C. The material can be cut and machined to close, dimensional tolerances.

680. F. Cap and J. Mennig, "Analytical Method for Determining n-Group Neutron Fluxes in Cylindrical Shielding Problems Using Lie Series," Nukleonik, 6(3), 141-147 (May 1964).

Calculation of fluxes in shields of cylindrical reactors by means of n-group diffusion theory illustrates the usefulness of Lie series in reactor theory. While it is possible to use difference techniques to solve the problem in cylindrical geometry, computing times are relatively long. Analytical methods are faster, but lead to numerical difficulties for large arguments of the Bessel functions. By using Lie series, the combined flux and current vector in n-group theory may be written in the form $\vec{g}^{cylinder} = \vec{g}^{plane} - \vec{g}^{Correction}$ for arbitrary arguments. Integration of the n-group diffusion equations by means of Lie series results in an explicit expression for $\vec{g}^{correction}$. The vectors are functions of $\xi_\mu = r - r_{\mu-1}$ denotes the inside radius of the μ'th zone. This eliminates numerical difficulties associated with the evaluation of Bessel functions depending on r and $r_{\mu-1}$ separately when the arguments become large. Recursion relations permit calculation of an arbitrary number of terms in the Lie series and lead to solutions of any desired accuracy within the framework of diffusion theory. Separating out $\vec{g}^{plane}$ results in rapid convergence and essentially reduces the cylindrical problem to one in plane geometry which can be solved on a relatively small computer.

681. J. O. Juliano, R. H. Ibarra, O. V. Gabriel, M. G. Natera, E. E. Reyes, and G. J. Rodriquez, A Study of the Shielding Effectiveness of the Parc Cobalt-60 Garden, PAEC(D) PH-634 (June 1963) Availability: Dep. (mc).

The shielding effectiveness of a wall around a 40-ft diameter Co^{60} garden was evaluated. Experimental dose rates were obtained with a 6.6 curie source. The skyshine measurements agreed well with calculations.

682. M. J. Barrett, Gamma Heating Core Structure Metallic Components. Experimental Gas Cooled Reactor, TID-18095 (April 21, 1960) Availability: $4.60 (fs) OTS.

The heat production in the steel structures used to support the EGCR were calculated by kernel integration. Gamma rays and thermal neutrons were considered.

It was found that 90% of the heat production in the steel structure is due to capture gamma rays originating in the structures themselves.

683. M. J. Barrett, Neutron Streaming from the EGCR Core Experimental Gas Cooled Reactor, TID-18096 (May 11, 1960) Availability: $3.60 (OTS).

A calculation was made of the neutron current streaming from the fuel channels in the graphite core of the Experimental Gas Cooled Reactor.

684. M. J. Barrett, Shielding Analysis of the Service Machine. Experimental Gas Cooled Reactor, TID-18097 (June 1, 1960) Availability: $7.60 (OTS).

This study presents (1) an analysis of the normal operation schedules of the EGCR service machine, (2) data on the resulting radiation within the service machine, and (3) a shielding system for the service machine to meet design specifications.

685. R. West, An Isotope Handling Calculator (Mk. III) for Gamma-Ray Sources with Lead Shielding, RCC-R-165 (July 1963) Availability: BIS $0.30; HMSO, 1s 6d.

The report describes a circular slide-rule calculator which at a single setting inter-rolates in direct terms the following factors for the isotopes Na-24, Co-60, Radium, Cs-137, Ir-192, I-131 and Au-198:

Source strength
Source distance
Gamma-ray dose-rate
Lead thickness

The calculators are available from the Radiochemical Centre, Isotope Production Unit, A. E. R. E., Harwell, Berkshire.

686. J. A. Belcher, Comparison of Methods of Calculating Fast Neutron Scattering from Beryllium, NAA-SR-Memo-9214 (November 12, 1963) Availability: OTS $1.60 (fs), $0.80 (mf).

The flux from fast neutrons (greater than 0.1 Mev) scattered by the SNAP 10A/2 control drums has been calculated with ray tracing codes 14-0 and SCAR 1. A Monte Carlo code, ANNEX 2, has been extremely successful in predicting the results from some recent scattering experiments at Oak Ridge. It was decided to check the accuracy of the SNAP 10A and SNAP 2 control drum scattering calculational methods with a devised problem which would be analyzed with 14-0, SCAR 1, and ANNEX 2. This comparison indicated that no major error has been made in SNAP 10A or SNAP 2 control drum scattering calculations performed to date.

687. E. W. Bennett, Energy Deposition in the Atmosphere by Delayed Gammas from Fission, LAMS-3059 (April 24, 1964) Availability: $0.50 OTS.

Calculated source functions are provided for the determination of energy disposition in the atmosphere by delayed gammas from fission of U^{236} and U^{238}. The gamma spectra and the source strength for debris resulting from 1 kt of fission yield in each case are included. Calculations are done for both "good" geometry and "bad" geometry. Energy deposition rates are determined from 10^{-2} to 20 seconds after fission and extrapolated as $t^{-1.2}$ for times > 20 seconds.

688. J. J. Griffin, Other Details of Post-Fission Beta Decay, LA-2811 (Add.) (October 1, 1963) Availability: $0.75 OTS.

In this report are summarized various detailed numerical results obtained in calculations reported in LA-2811, but not specifically relevant to the purpose of that report. These include the time dependence of average beta disintegration energies, and the average number of beta decays for various nuclides, as well as plots of these quantities at certain specific times as a function of the single parameter $\bar{z}$, which characterizes the fissioning nuclide. It is hoped that the latter will be useful for extending the results to other nuclides not calculated explicitly.

689. W. E. Selph, T. C. Hinckson, and W. J. Rainwater, Jr., Experiment and Analysis of Neutron Attenuation in Zirconium Hydride, Report 2-53010/4R-534 (February 1964) Availability: Ling-Temco-Vought, Inc., P. O. Box 5907, Dallas, Texas.

Data are presented on the fast neutron attenuating properties of zirconium hydride, and comparisons are made between experimentally and analytically determined values.

The General Atomic Linear Accelerator (LINAC) was used to produce neutrons with an energy spectrum approximating that for a fission source. Energy spectral measurements were made of the neutrons incident on and penetrating $Z_rH_{1.9}$ slabs of various thicknesses. Spectral measurements were made as a function of exit angle for a slab of 87 g/cm^2.

Analysis of the experimental results was performed at LTV with the ADONIS Monte Carlo program developed by United Nuclear. In addition, the experimental results were analyzed at General Atomic with the GAPLSN transport program.

690. M. H. Kalos, A Monte Carlo Calculation of the Transport of Gamma Rays, NDA-56-7 (July 1956) Availability: OTS $7.80 (fs), $3.30 (mf).

A program for the calculation of the transport of gamma rays by Monte Carlo methods is described. It is concluded that the "doubling" technique and statistical estimation used in this program are simple and effective methods for improving the efficiency of the sampling.

691. J. E. Braley and R. E. Donaldson, Neutron Penetration by the Monte Carlo Method, ZJ-015 (May 1957). Availability: Convair, San Diego, California.

A calculation has been made of the penetration of plane slabs of lead and boron leaded polyethylene by a plane parallel source of fast neutrons. The neutron energies used were 14, 10, 6.5, and 3 MeV. The slab thickness ranged up to 6 mean free paths and in one instance to 9 mean free paths. The spectral distributions of the transmitted and reflected neutrons were recorded as well as the volume distributed source of gamma-rays resulting from captures and inelastic scatterings.

692. E. D. Cashwell and C. J. Everett, A Practical Manual on the Monte Carlo Method for Random Walk Problems (1959), Availability: Pergamon Press.

This book summarizes the Monte Carlo method as it applies to problems involving the interactions of neutrons and photons with bulk matter in geometric systems of varying complexity. Flow charts are given and computer program subroutine are described in detail.

693. D. A. McKown, Dose Rates, Activity, and Shielding Transmission Factors for U-235 Fission Products After Short Irradiation, LAMS-2963 (March 6, 1964). Availability: $0.50 OTS.

The calculated gamma dose rates and shielding dose transmission factors with buildup for U-235 fission products are tabulated for various times after fission. The transmission factors are listed for lead, iron, concrete, aluminum, and water. The calculations are based on the U-235 fission product gamma-ray spectra and activities presented by Ann T. Nelms and J. W. Cooper in Health Physics, V. 1, No. 4 (March 1959), 427-441. Point source geometry is assumed, and the buildup factors are for infinite media.

694. C. McDonnell, J. Velletri, A. W. Starbird, and J. F. Batter, Description, Experimental Calibration, and Analysis of the Radiation Test Facility at the Protective Structures Development Center, PSDC-TR-14 (September 1, 1964) Availability: Protective Structures Development Center, Fort Belvoir, Virginia.

The initial calibration experiments performed at the Radiation Test Facility of the Protective Structures Development Center are described and their results analyzed. The dose rate above an open field and the attenuation afforded by the steel frame of the test structure is calculated and found to agree well with experiment when modified calculational procedures are used. The cumulative angular distribution of direct radiation is found to be as much as fourteen percent above that predicted by theory over the range investigated. Several modifications of presently used calculational techniques are suggested.

695. R. L. Henry, Multilayer Shield Experiment IV, OTF III, NARF-58-40R (October 9, 1958) Availability: Convair, Fort Worth, Texas.

The Aircraft Nuclear Propulsion Department of General Electric has sponsored a series of shield tests at the Nuclear Aircraft Research Facility at Convair-Fort Worth to obtain shield design information for nuclear aircraft reactors. This document contains the experimental results of one of these tests.

In the experiment, slabs of various shielding materials were assembled in different arrangements into shield mock-ups. These assemblies were then placed in the Outside Tank Facility and subjected to radiation from the Ground Test Reactor. External measurements were taken with dosimeters and BF_3 counters; and internal measurements, with foils and chemical dosimeters. Results indicate that zirconium hydride is a good attenuator of gamma rays and fast and thermal neutrons, that the relaxation lengths for zirconium hydride in a BeO matrix are the same for fast neutrons and gammas, that fast neutrons and gamma rays are attenuated exponentially in zirconium hydride, and that there is relatively little production of secondary gamma radiation in zirconium hydride due to thermal neutron capture and inelastic scattering of fast neutrons. Lead, especially in large thicknesses, seems to produce a relatively high secondary gamma dose rate. Although thermal neutron shielding, when placed forward of uranium in a shield mock-up, does lower the gamma dose rate, the reduction is not great.

696. R. S. Hubner, Results of the Analysis of the ORNL Tower Shield Experiments, NAA-SR-Memo-9060 (October 1, 1963) Availability: OTS $1.60 (fs) $0.80 (mf)

The results of the analysis of the ORNL Tower Shield experiments using Monte Carlo techniques are presented and compared to the experimental data. Satisfactory agreement with the experiment is noted. Scattered dose, no collision dose, and scattered spectra are tabulated.

The experiments were performed using the ORNL Tower Shield Reactor as a source of neutrons. A collimated beam of neutrons was incident on a cylinder in a direction normal to the cylinder axis. The scattered dose was recorded at angles measured from the direction of the incident beam in a plane perpendicular to the cylinder axis. Dose measurements were made with a Hurst Dosimeter. The incident beam spectrum and dose intensity distributions were measured. The target cylinders were 6 inches high and 1, 2, 4 and 6 inches in diameter; cylinders of Be, C, Al, Fe, and CH_2 were used. Detector angles were 45, 90, and 135 degrees.

697. K. L. Hall, M. W. Tobriner, M. A. Sweeney, J. A. Dubay, R. F. Klaver, and R. O. Bolt, Dosimetry of Reactor Radiation from the Shield Test Pool Facility, Report No. 22, TID-19678 (June 30, 1963) Availability: OTS $11.50 (fs), $4.88(mf).

The neutron- and gamma-dose rates were determined separately in reactor radiation from the shield Test Pool Facility (STPF). The separation was achieved by using a series of calorimeters of varying neutron to gamma responses. The method of analysis required a knowledge of the neutron spectra. This was determined both theoretically from reactor physics and empirically using the foil activation technique. The results are expressed by an equation interrelating the neutron- and gamma-heating rates as a function of position within each of two canisters, one for exposures to fast neutron-rich and the other to gamma-rich radiation.

698. J. H. Hubbell, "Response of Large Sodium-Iodide Detector to High-Energy X-rays," Rev. Sci. Instr., 29(1), 65-68 (January 1958).

The response of a four-inch long by five-inch diameter sodium-iodide crystal to axially incident 0.01 to 8-Mev x-rays has been expressed as a 28 by 28 matrix. Measured pulse-height distributions were combined with Monte Carlo calculated distributions for the input data. The matrix was inverted on an automatic computer for ease of application to spectra. The mesh is uniform in the square root of energy, each interval including an entire photopeak. Such a coarse mesh makes the response matrix suitable for the unscrambling of continuous but not line spectra.

699. P. W. Reinhardt and F. J. Davis, "Improvements in the Threshold Detector Method of Fast Neutron Dosimetry," Health Physics, 1(2), 169-175 (September 1958).

Improvements have been made in the threshold detector method for measuring fast neutron spectra and tissue dose. The sensitivity of the counters used to measure the induced activity of the detecting elements has been increased by a factor of 12 to 16 times by improving counting geometry. Holding the gain of the scintillation counting system constant by an automatic monitoring arrangement reduces the time required for standardization. The method of P-32 recovery from S-32 by burning out the sulfur has been shown to be reproducible and to have a recovery efficiency of 93.5 per cent. The neutron cross-sections for various detection elements used in the threshold detection system are reviewed.

700. M. S. El-Nesr and K. A. Mahmoud, "Penetration of Gamma Rays in Paraffin," Proc. Math. Phys. Soc. (U.A.R.), No. 25, 59-65 (1961).

The spectra of gamma-rays penetrating paraffin have been studied for Cs(137), Co(60) and Na(24) gamma-ray isotopic sources. Experiments were performed using a calibrated single channel gamma-ray spectrometer and the results were compared satisfactorily with the derived theoretical computations.

These experimental investigationsof paraffin proved to be of good protectional value against scattered radiation in its different orders and particularly at energies greater than 0.5 Mev.

701. G. S. Hurst and R. H. Ritchie, "On Energy Resolution with Proportional Counters," Rev. Sci. Instr., 24(8), 664-668 (August 1953).

The shape of a proportional counter pulse due to an extended track of ionization is determined, before and after passing through a linear pulse amplifier, for various values of track extension and positive ion collection time. It is found that the maximum height of the amplifier output pulse is essentially independent of the length and orientation of the track in the counter, provided the time of collection of the outermost electrons is not longer than the amplifier time constant.

702. W. F. Hornyak, "A Fast Neutron Detector," Rev. Sci. Instr., 23(6), 264-267 (June 1952).

A fast neutron detector with an efficiency of a few percent and the capability of detecting neutrons in the presence of intense gamma-ray backgrounds has been developed. The detector consists of a ZnS and Lucite molded cylinder used in conjunction with a 5819 photomultiplier tube. This detector is well suited to experiments requiring "good geometry" and pulse speeds of the order of 0.1 μsec.

703. Rosemary T. Berger, "The X- or Gamma-Ray Energy Absorption or Transfer Coefficient: Tabulations and Discussion," Rad. Res., 15(1), 1-29 (1961).

The quantity known as the X- or γ-ray energy absorption coefficient is defined in various ways in the literature and has also been given a variety of other names. The present paper attempts to clarify this confusing situation by presenting a careful definition and suggesting a new name, "energy transfer coefficient," which is more descriptive of the definition. The significance of the energy transfer coefficient for the calculation of absorbed dose is discussed, and a tabulation is given of energy transfer coefficients for various materials for photon energies between 3 kev and 10 Mev. This tabulation is a revision and extension of previous work of G. R. White Grodstein and takes into account the latest values of the relevant cross sections. Limitations and uncertainities of the tabulated values are discussed. Auxiliary data are given of the parameters (such as cross sections and conversion factors) which enter into the calculation of the energy transfer coefficient to enable the reader to construct energy transfer coefficients with correction terms appropriate to a particular application.

704. Ulf Tveten, A Shielding Analysis for JEEP II. Part I. Computer Programmes, KIR-E-155 (March 1962) Availability: Institutt for Atom Energi, Kjeller Research Establishment--Norway.

The following report is the first part of a shielding analysis of JEEP II. It contains a description for computations of the shielding properties of the bulk shield performed by electronic computer. All shielding constants utilized are given in the tables at the end of the report.

705. Ulf Tveten, A Shielding Analysis for JEEP II. Part II. Streaming Problems, KIR-E-156 (March 1962) Availability: Institutt for Atom Energi, Kjeller Research Establishment--Norway.

This report is the second one in a series treating various aspects of the shielding of the reactor "JEEP II." This part contains a description of the methods used for calculating the streaming through the reactor shield.

706. Ulf Tveten, A Shielding Analysis for JEEP II. Part III. The Basement, KIR-E-157 (April 1962) Availability: Institutt for Atom Energi, Kjeller Research Establishment--Norway.

This report, which is the third in a series, contains calculations for the thicknesses of the walls in the basement of JEEP II. Directly under the reactor there is a room into which the tubes for the heat exchanger enters from above, from the reactor tank, and exists through a side wall into the heat exchanger room. Through the tubes from the reactor there is a strong neutron and gamma streaming, and the activation of the coolant makes the heat exchanger itself as well as the tubes strong radiation source.

707. Ulf Tveten, A Shielding Analysis for JEEP II. Water Tank, KIR-E-161 (May 1963) Availability: Institutt for Atom Energi, Kjeller Research Establishment--Norway.

This report is number four in a series treating various aspects of the shielding of the reactor "JEEP II." This part contains description of the methods used for calculating the fluxes through the intertank, water tank and the surrounding shields, as well as the fluxes outside the storage block from stored fuel elements and plugs. The shielding constants used are tabulated at the end of the report.

708. M. J. Berger and L. V. Spencer, "Some Radiological Applications of Gamma Ray Transport Theory," Rad. Res., 10(5), 552-570 (May 1959).

Gamma-ray transport theory is applied to the calculation of energy dissipation in bounded water media under conditions approximating those found in radiological problems. A combination of analytical, numerical, and Monte Carlo techniques is used. Two typical situations are considered: (1) a medium in which an isotropic source of radiation is distributed; (2) a medium irradiated by an external beam of radiation. The spatial pattern of energy dissipation is worked out in detail for a source emitting 1.28-Mev photons. The results of problem 2 are applied to the calculation of the depth dose produced by the irradiation of medium with Co-60 beams with cross-sectional areas up to 400 cm^2. The calculated depth-dose distributions are in good agreement with experimental distributions obtained by Johns and his collaborators.

709. M. J. Berger, "Calculation of Energy Dissipating by Gamma Radiation Near the Interface Between Two Media," J. Appl. Phys., 28(12), 1502-1508 (December 1957).

A Monte Carlo calculation was made of the energy dissipation by gamma radiation in a medium with the scattering properties of water which consists of two semi-infinite regions, one of which is much denser than the other. The calculations are for a 1.28-Mev point-isotropic source at or near the boundary plane separating the two regions. The results are presented in terms of boundary correction factors which express the change of energy dissipation compared to that prevailing at the same location in a medium of uniform density. The analysis of the results leads to the conclusion that the boundary effect is rather insensitive to the atomic number of the medium, and is mainly a function of the density change. The Monte Carlo results are in good agreement with an experiment of Soole on the propagation of radiation in the air above the ground, and an experiment by Titus on the dose distribution in iron.

710. C. E. Clifford, J. A. Carruthers and J. R. Cunningham, "γ-Radiation at Air-Ground Interfaces with Distributed Cs-137 Sources, Can. J. Phys., 38(3), 504-507 (March 1960).

Measurements of gamma-ray dose rate were made in air over an air-ground interface and compared with Monte Carlo calculations of Berger. Similar results were also obtained near a concrete-polyfoam interface and a lead-foam interface.

711. Iwao Umeda, "Calculation of the Transmission of Singly Scattered Gamma-Rays Through Finite Thin Slabs," J. Nucl. Sci. Technol., 1(2), 37-40 (May 1964).

The intensity and number of transmitted singly scattered photons were calculated for 1.0, 2.0, 4.0, 6.0, 8.0 and 10.0 MeV γ-rays normally incident on slabs of lead, iron, concrete and water varying from 1 to 4 mean free paths in thickness.

Energy build-up factors were obtained by integrating them using Simpson's formula. The calculation for composite shields included 1.0, 3.0 and 6.0 MeV γ-rays normally incident on a lead-water shield. The results obtained show good agreement with the values calculated by the Monte Carlo method and the moments method.

The approximation can be used for the case of γ-rays above 2.0 MeV incident on slabs less than 3 mean free paths in thickness. For γ-rays under 1.0 MeV, it can be also used for iron and lead slabs less than 1 mean free path in thickness.

712. F. S. Kirn, R. J. Kennedy and H. O. Wyckoff, "The Attenuation of Gamma Rays at Oblique Incidence," Radiology, 63(1), 94-104 (July 1954).

Attenuation data have been taken for radiation incident obliquely upon barriers of lead, concrete, or concrete-equivalent material. The gamma radiation from cobalt-60, cesium-137 and gold-198 was used at angles of incidence 0°, 30°, 50°, 60°, and 70°. The results show that for oblique angles of incidence the Compton scattered radiation may have shorter path lengths through the barrier. For the larger angles of incidence, the obliquity effect must be considered in designing barrier requirements. For a diverging cone of radiation, the effect is not too significant for attenuations less than 100. It is also shown that, in the range of energies below 0.7 mev, the addition of a layer of lead behind a low atomic number barrier can be effective in reducing the thickness requirements of the barrier.

713. A. T. Nelms and J. W. Cooper, "U-235 Fission Product Decay Spectra at Various Times after Fission," Health Physics, 1(4), 427-441 (March 1959).

Tabulations are given of the β- and γ-ray spectra produced by the total products of U-235 fission at times after fission from 31.2 min to 119 years. Spectra are also given which exclude radiation from fission products that form volatile compounds.

714. K. W. Morton, "Scaling Neutron Tracks in Monte Carlo Shielding Calculations," J. Nucl. Energy, 5(3/4), 320-324 (July/August 1957).

A simple technique is given to enable neutron tracks generated in a Monte Carlo study of a given slab of material to be used for slabs of any other thickness. It is applied to the interpretation of some thick sample scattering data.

715. K. W. Morton, "A Generalization of the Antithetic Variate Technique for Evaluating Integrals," J. Math. Phys., 36, 289-293 (1957).

Hammersley and Morton (1956) have introduced a new Monte Carlo technique which they have called the antithetic variate method. In their paper they investigated fully the application of the technique to the evaluation of single integrals; and then considered multiple integrals as repeated integrals to which this theory could be applied approximately. Now, it is for integrals of fairly high dimensionality that Monte Carlo can potentially be of the most use: this treatment of them is rather unsatisfactory from both an aesthetic and a practical point of view, requiring as it may a great deal of judicious guesswork and complicated approximate reasoning, as the practical example in Hammersley and Morton's paper shows. The main result of this note is a generalisation of their technique from single to multiple integrals; this will, in general, require too much initial computation to be used exactly in its present form, but it forms a much firmer foundation on which to base approximation.

716. J. M. Hammersley, "Conditional Monte Carlo," J. Assoc. Compt. Mach., 3, 73-76 (1956).

Trotter and Tukey have discovered a powerful technique which they have called conditional Monte Carlo. Although they were aware that this method had generalizations, they confined themselves to rather special cases. The following more general specification of conditional Monte Carlo examines its underlying structure and exhibits its considerably flexibility.

717. J. G. Wendel, "Groups and Conditional Monte Carlo," Anals of Math. Statistics, 28, 1048-1052 (1957).

The conditional Monte Carlo technique advanced by Tukey, et al. has been explained in analytic terms by Hammersley. This note offers an alternative explanation, wherein the group-theoretic aspect of the problem plays the dominant role. The method is illustrated on an example simpler than that treated in (1,2).

718. J. M. Hammersley and K. W. Morton, "A New Monte Carlo Technique: Antithetic Variates," Proc. Cambridge Phil. Soc., 52, 449-475 (1956).

Workers on both sides of the Atlantic have recognized the possibility of increasing efficiency by using correlated samples; but all work which we have hitherto encountered is restricted to the use of positive correlations. We shall utilize negative correlations, for which we propose the name 'antithetic variate method' to distinguish it from the term 'control variate method' already applied by British workers to the use of positive correlations. The general term 'correlation methods,' hitherto used by American authors for positive correlations, can then be freed for the inclusive description of both antithetic and control variate methods.

719. S. M. Ermakov, V. G. Zolotukhin, and E. E. Petrov, "Calculation of the Penetration of Neutrons Through a Plane Layer of Polyethylene," Soviet J. Atomic Energy (English Translation), 15(3), 960-962 (September 1963).

The article discusses the penetration of neutrons through a plane layer of polyethylene (CH_2). The solution of this problem is of great practical significance in the design of shielding for nuclear power plants. The only method free from considerably simplifications is evidently the Monte Carlo method, which is essentially effective for small thicknesses H of the shielding layer. This was the method used in the present study.

720. V. S. Galishev, "Theory of the Passage of γ-Quanta Through Matter," Soviet J. of Atomic Energy (English Translation), 14(5), 469-473 (May 1963).

An analytic treatment is given of the problem of the passage of γ-quanta through a plain parallel slab. The integro-differential equation including boundary conditions for the problem of the scattered γ flux density in the nth approximation reduces to a system of 2nd integro-differential equations of the first order for the coefficients in the Legendre polynomial expansion of the scattered γ flux density. The values of the 2n arbitrary constants are found from the 2nd boundary conditions. The final results are given only for the case n = 1, which gives the spectral distribution of the scattered γ flux density at a fixed distance from the source.

721. G. Vestey, "Production of Lead Bricks for Temporary and Movable Shields," Lead 62, London, Lead Development Association (1962).

Various techniques for the manufacture of lead Bricks are described. These include casting, extrusion, machining and cold pressing.

722. H. Arnold Jenkinson, "The Manufacture of Special Lead Shields for Nuclear Shielding," Lead 62, London, Lead Development Association (1962).

The manufacture of various special lead shields is described. These shields include bricks, sheets, special castings and extrusions.

723. H. S. Davis, F. L. Browne, and H. C. Witter, "Properties of High-Density Concrete Made with Iron Aggregate," J. Am. Conc. Inst., 27(7), 705-726 (March 1956).

Data are presented on the physical properties of several types of mortar and concrete made with iron-bearing aggregate for use in radiation shields. Heavy aggregates used in this investigation included steel punchings, fine steel shot, ferrophosphorus, magnetite, and limonite.

Concrete made with natural heavy aggregates had densities of from 185 to 225 per cu ft. One type of concrete made with steel punchings and limonite weighed 273 lb per cu ft. Both the prepacked method and conventional methods were used for fabricating concrete specimens. The results of tests performed on several types of heavy mortars made with fine sand are also reported. Data obtained at elevated temperatures are presented, as well as data obtained from standard tests on aggregate, concrete, and mortar.

The data and experience obtained during this investigation demonstrate that concrete suitable for radiation shields can be made with iron-bearing aggregates by either the prepacked or conventional methods of construction.

724. H. S. Davis, "High Density Concrete for Shielding Atomic Energy Plants," J. Am. Conc. Inst., 29(11), 965-977 (May 1958).

Concrete properties required for gamma ray absorption and neutron attenuation are outlined. "Fixed" water content and its role in slowing down fast neutrons are explained. Grouts and mortars, heavy aggregate properties, and use of boron are also covered. Placement methods evaluated include: conventional, pumping, prepacking, and puddling. Formwork problems including dimensional tolerances are outlined, and shields made of high-density concrete block are described. Tabulated data cover costs of materials as well as finished shielding structures.

725. J. O. Henrie, "Properties of Nuclear Shielding Concrete," J. Am. Conc. Inst., 31(1), 37-46 (July 1959).

General shielding requirements are discussed with particular emphasis on fast neutron shields. Concrete materials which are effective in meeting the requirements are outlined and the basic principles involved are explained. The concrete mix, type and source of materials, and strength properties of the shield of four similar nuclear reactors are compared. Effects on concrete setting time and strength of adding boron in the form of mineral colemanite and borocalcite and the counter effects of calcium chloride are described.

726. H. S. Davis and O. E. Borge, "High Density Concrete Made with Hydrous-Iron Aggregates," J. Am. Conc. Inst., 30(10), 1141-1147 (April 1959).

Data are presented on the physical properties of a high-density concrete tested under standard conditions and after heating at temperatures of 85, 200, and 350 C. The concrete has a density of 218 lb per cu ft and a fixed water content of about 4.3 percent at 85 C, which makes it a desirable material for biological shields around atomic power reactors. Computed values of attenuation lengths are included for fast neutrons and gamma radiation.

727. J. M. Raphael, "The Structural Properties of Magnetite Concrete," J. Struc. Div. Am. Soc. Civil Engineers, 84, ST1, Paper 1511 (January 1958).

Magnetite concrete for the Engineering Test Reactor (ETR) biological shield was made from a combination of crushed magnetite aggregate and a low-alkali Type I portland cement. Some combinations of cement and aggregate used in concrete are potentially self-destructive through internal volume changes caused by alkali-aggregate reaction or by thermal growth. Tests proved that this particular combination was non-reactive and without deleterious thermal growth. The elongated shape and high density of the crushed magnetite particles tended to cause harshness and segregation in mixing the concrete, but by paying very careful attention to the proportion of fines and the composition of the cement paste in the mix, workable concrete mixes were designed having densities ranging from 225 to 239 psf. It proved important to prevent the cement paste from becoming too fluid, as this caused excessive bleeding and segregation. It was found that up to 4% of the weight of the mix could be lost by drying of small cylinders of the concrete, but this moisture loss could be completely controlled by encasing the shield in steel plate. Concrete of any required strength could be made, but for concretes having compressive strengths less than 4000 psi at 28 days, extra fines had to be added to the mix in place of cement to lend supporting power and lubricity to the paste. Elasticity and Poisson's ratio for this dense concrete were about 50% higher than for normal weight concretes of comparable compressive strength, varying with the density of the concrete. At the end of a year, the drying shrinkage was about 0.06%, and the coefficient of thermal expansion was 5×10^{-6} in/in/^{o}F. When tested against narrow and wide beam radiation, the mass absorption coefficient for this magnetite concrete was found to be 0.054 cm^2/gm, and the build-up factor was 2.57 at an absorbent thickness of 80 gm/cm^2.

728. H. A. Bethe, L. Tonks, and H. Hurwitz, Jr., "Neutron Penetration and Slowing Down at Intermediate Distances through Medium and Heavy Nuclei," Phy. Rev., 80(1), 11-19 (October 1950).

This paper reports theoretical results and numerical calculations of neutron slowing down and penetration to fairly large distances from the source assuming constant mean free path and no absorption. The Placzek one-velocity problem which results from a Laplace transformation of the Boltzmann equation is solved approximately by keeping only a finite number of harmonic terms in the Laplace transform of the scattering function but not restricting the expansion of the neutron density. The Laplace inversion is carried out by the method of steepest descent. Numerical results are given for nuclei of mass infinity and 9, carried out to three orders of approximation, namely zeroth, first, and second. They show the expected transition behavior between age theory and the asymptotic exponential behavior. The accuracy of the method is high, being greater than that of the numerical work over the useful range of the calculations. Rough agreement with Wick's asymptotic results for M = 12 is found.

729. J. B. Eggen, R. L. French, and A. Reetz, Jr., "Fast-Neutron Spectra and Dose-Rate Calculations," Health Physics, 5 (3/4), 119-125 (June 1961).

This article describes a method and its application for calculating fast-neutron spectra and dose rates about the Lid Tank Facility source plate, the Bulk Shielding Reactor and the Aircraft Shield Test Reactor. The method may be applied to any reactor whose shield is composed largely of hydrogenous materials. Dose rates and spectra calculated by the method are useful in shield design and in the analysis of experiments performed in reactor radiation fields.

The method is based upon fast-neutron differential energy spectra and dose rates calculated by the Nuclear Development Corporation of America for a point isotropic fission source in infinite media. These data represent a moments method solution of the Boltzmann transport equation for water and hydrocarbons. By representing a reactor core as an arry of point isotropic sources and assuming that any material along the line-of-sight between a point source and a point detector may be represented by an equivalent thickness of water or hydrocarbon, the fast-neutron attenuation through any material may be determined by use of the results for these reference materials. The equivalent thickness of non-hydrogenous material is determined from the ratio of the effective removal cross-sections.

Results from dose-rate calculations by this method agree with measured BSR and LTSF pure water centerline data within 10 per cent. Calculated and measured spectra for the BSR and the ASTR compare favorably, although the ASTR calculation involves penetration of considerable amounts of non-hydrogenous materials.

730. R. L. French and M. B. Wells, "Calculations of Weapons Radiation Penetration in Air," Health Physics, 5(3/4), 108-118 (June 1961).

The spatial, angular and energy distributions of the initial radiation from a nuclear weapon were calculated on a theoretical basis. The radiation components considered were fission neutrons and prompt fission, fission product and air-capture γ-rays. The principal simplifying assumptions employed in the theoretical model were neglect of (1) self-attenuation in the nuclear weapon; (2) blast effect on air density; and, (3) ground effects. Comparisons were made with measured data which clearly demonstrate that the spatial, angular and energy distributions of the initial radiations from a nuclear weapon are calculable.

731. L. Costrell, "Scattered Radiation from Large Co-60 Calibrating Sources," Health Physics, 8(3), 261-272 (June 1962).

Radiation beams from multicircle Co-60 sources are widely used for instrument calibration and for radiation treatment. It is often assumed for purposes of analysis that such beams contain only the 1.17 and 1.33 MeV γ-rays. However, it is recognized that this assumption is not strictly valid because of the energy degradation due to Compton scattering in the source material and the source collimator. A scintillation spectrometer with a 5-in-dia. by 4-in-long sodium iodide crystal has been used to determine the amount and energy of the scattered radiation as well as its variation with source dimensions and collimator geometry. Since the sources currently in common use have appreciable mass, source scattering is a far greater contributor of degraded radiation than is the collimator. Thus for multicurie sources, minimization of scatter involves primarily an increase in specific activity. The scattered intensity from the sources investigated ranges from 3.3 to 14.6 per cent of the intensity in the primary beam. Quantitative data are presented.

732. R. L. Mather, R. F. Johnson and F. M. Tomnovec, "γ-Radiation Field Above Fallout-Contaminated Ground," Health Physics, 8(3), 245-260 (June 1962).

Measurements were made of the energy and directional distributions of photons at a point 106 cm above fallout-contaminated level ground 9 days after detonation of a fission device. The radiation was measured with a NaI (tl) scintillation spectrometer and the results converted to an absolute flux of photons/sec-cm^2-keV-sterad. The direct and scattered radiations could be distinguished by their different angular distributions. The direct radiation was analyzed, using simple theory, to give a source spectrum and a thickness of overlying absorber. The results are shown to conform with other knowledge of the situation. The observations of scattered radiation show all of the expected qualitative characteristics.

733. John F. Batter, Jr., "Transport Calculation of the Net Current of Photons in an Isotropically-Scattered Atmosphere," J. Opt. Soc. of Am., 48(6), 371-74 (June 1958).

The net photon current is obtained from the exact transport solution for photon density in an infinite isotropic atmosphere. An analytical-numerical solution for the photon current is developed and presented graphically together with the diffusion theory current versus non-dimensional distance.

734. R. O. Lane and W. F. Miller, "Energy-Dependent Multiple Scattering," Nucl. Instr. Meth., 16, 1-16 (June-August 1962).

The Monte Carlo calculation described will yield relative fluxes for first-, second-, and third-order elastic scattering of neutrons in thin plates. The method specifically includes the effects due to energy loss by the neutron following elastic scattering, together with the corresponding energy dependence of the cross sections for subsequent collisions. Such effects have not been included in previous treatments. Illustrative examples of some typical cases are presented.

735. L. Costrell, "Scattered Radiation from Large Cs-137 Sources," Health Physics, 8(5), 491-498 (October 1962).

Measurements have been made to determine the scattered radiation to be expected from typical multicurie Cs-137 installations. For the cesium pellet sources investigated the scattered intensity ranged from 12 to 21 per cent of the total incident radiation intensity. The source capsule is responsible for scattered intensity of about 8 per cent with the source material contributing the bulk of the balance. The source material scatter, for a source diameter of 3.6 cm, is about 5 per cent per centimeter for source thicknesses of up to about 1.5 cm.

736. C. Sharp Cook, "The Energy Spectrum of γ-Radiation from Fallout," Health Physics, 4(1), 42-51 (October 1960).

This report compiles and analyzes the results of fallout sample gamma-ray spectral measurements using a NaI (Tl) crystal scintillation spectrometer. The summary of the information indicates that, for those detonations at which γ-rays spectral characteristics of deposited fallout have been measured, there appears to be no positive correlation between spectral characteristics and weapon yield, weapon type or conditions of detonation. Comparisons have been made between the experimental results and the calculational predictions both of Nelms and Cooper, and of Dolan for the γ-radiation from the products of slow neutron fission of U-235. Although there is considerable scatter in the experimental data, agreement between experiment and calculations appears relatively good, except for variable effects produced by neutron-induced radioactivity, such as Np-239. The mean energy per photon in the time interval between 10 and 3000 hr after detonation appears to be not too different from 0.7 MeV/photon if some shielding is provided to screen out low-energy photon radiations.

737. P. H. Hemmings, A Compilation of Nuclear Data for Natural Zirconium, AHSB(S) R-62 (1963) Availability: Her Majesty's Stationery Office, 2s 9d.

A survey has been made of experimental cross-sections and other nuclear data for Zirconium. These have been collected and put on to punched cards for use in neutronics calculations. In particular, the cards have been prepared within the schemes of (a) The Aldermaston Punched Card Library, and (b) The Aldermaston/Winfrith Nuclear Data File.

738. D. J. Raso, "Transmission of Scattered Gamma Rays Through Concrete and Iron Slabs," Health Physics 5(3/4), 126-141 (June 1960).

The penetration of γ-rays undergoing one or more scatterings in concrete and iron of various slab thicknesses (3.75-60.0 g/cm^2) has been investigated by the Monte Carlo technique. Radiation with energy of 350, 660 and 1250 keV and also the γ-ray spectra produced by the products (omitting volatile materials) of U-235 fission at 1.12 and 23.8 hr after fission were assumed to be incident on the medium at angles of cos θ_0 = 0, 0.25, 0.50, 0.75 and 1.0. The case histories of 5000 photons were followed on the IBM 704 digital computer. The information obtained included:

(1) Number and energy transmitted.
(2) The energy distribution of the transmitted radiation integrated over all directions.
(3) The angular distribution integrated over all energies.
(4) The angular distribution of the transmitted energy.
(5) The angular distribution of the transmitted "dose."

These computed quantities refer to transmitted current and an approximation is given for determining flux density. Comparisons are made with other theories and an experiment performed at USNRDL.

739. Iwao Umeda, "Monte Carlo Calculations on the Transmission of Gamma-Rays from ^{60}Co and ^{137}Cs Point Isotropic Sources in Spherical and Cylindrical Shields," J. Nucl. Sci. Technol., 1(3), 87-92 (June 1964).

For the case of point isotropic source, the number and energy current build-up factors, number of collisions, energy current spectra of γ-rays transmitted through spherical and cylindrical shields, and angular distribution through spherical shields were obtained by Monte Carlo calculations. Radiation with energy of 1.25 and 0.662 MeV was assumed to be emitted from the center of an iron and a lead shield varying from 1 to 5 mean free paths in radius. The case histories of 2,000 photons for a spherical shield and 4,000 photons for a cylindrical shield were followed on the IBM 7070 digital computer. Further, a ^{60}Co point isotropic source was placed at the center of a cylindrical plastic scintillator, and pulse height distribution caused by energy absorption was measured for comparison with the Monte Carlo values. The results obtained by the experiment showed good agreement with the results of the Monte Carlo calculation.

740. J. P. Jain and M. M. Gupta, "Optimum Shape of Shielding Barrier for Disc Source," Health Phys., 10(5), 337-40 (May 1964).

In this communication an attempt has been made to discuss the optimum shape of a shielding barrier for a disc source. The problem is divided into two parts. In Section A the attenuation of gamma radiation is supposed to be due to the linear absorption only in the shielding material. In Section B the effect of build-up factor on the optimum shape of the shielding barrier is discussed. For calculations, a suitably approximated form of build-up factor for gamma ray is used. One can draw quite a legitimate conclusion that build-up factor plays a paramount role in optimising the shape of a shielding barrier.

741. G. During, R. Jansson and N. Starfelt, "Experimental Fast Neutron Spectra in Al and Fe," Arkiv for Fysik (Stockholm), 26(4), 293-307 (1964).

Fast neutron spectra was measured using an organic scintillator and a L.I. (eu) crystal. The source was either deuterium or tritium bombarded by 230 kev deuterons giving 2.95 and 14.9 Mev neutrons. The spectra after transmission through Fe slabs, Fe spheres, or Al spheres compared well with Monte Carlo results of Leimdorfer.

742. W. H. Ellett, A. B. Callahan, and G. L. Brownell, "Gamma-Ray Dosimetry of Internal Emitters. Monte Carlo Calculations of Absorbed Dose from Point Sources," Brit. J. Radiol., 37(433), 45-52 (January 1964).

Monte Carlo calculations of the absorbed dose delivered by point source γ emitters to tissue equivalent phantoms have been performed and the results tabulated in tables of A.F.; i.e. that fraction of the emitted energy absorbed by the phantom. The use of these tables makes it possible to calculate γ dose-rate in a manner analogous to that used for β-ray dosimetry.

$$R_\gamma = \Sigma_\gamma \; C \; (A.F.)_{M,\; E\gamma}$$

where Σ_γ is a constant for a particular isotope, C the average isotopic concentration and $(A.F.)_{M,E\gamma}$ the A.F. for a particular phantom mass and γ energy.

The effects of changing phantom mass, shape and source configuration have been considered and the results compared to analytical treatments of the same problem performed by others.

Of some clinical interest are the following conclusions:

(1) Tables of absorbed fraction afford an accurate and convenient method of estimating the total body γ dose;
(2) The absorbed fraction is relatively independent of body height, shape, and the position of radioactivity within the patient;
(3) A restricted distribution of activity in the trunk can be treated as a point source;
(4) The use of spheres as patient models and the ICRP method of γ dose calculation should be discontinued.

743. Nuclear Engineering, "Shielding Material Review -- Emphasis on Neutrons," Nuclear Engineering, 9(98), 244 (July 1964).

A large number of unusual shielding materials are described briefly. In many cases, manufacturers are listed.

744. W. E. Thompson and F. C. Engesser, Measurements of the Effects of Compartmentalization on the Gamma-Ray and Neutron Dose Rate in the Radiation Field of a Reactor, USNRDL-TR-722 (February 17, 1964) Availability: $5.60 OTS.

Data illustrating the effects of various components of a simply compartmented structure on the dose-rate field near a reactor were gathered at the Tower Shielding Facility (TSF) of Oak Ridge National Laboratory (ORNL) and are presented in forms allowing convenient comparisons between gamma-ray and neutron dose rates for given configurations.

These effects were found to be nearly equivalent to the additive effects of the sides of the compartment considered separately, thus adding credence to the assumption that each slab contributes to the total dose rate independently. By weighting these contributions according to solid angle, a method is worked out which consistently predicts the dose rate for any configurations not involving double transmission.

745. M. J. Barrett and J. Waldman, Experimental Gamma Ray Backscattering by Various Materials, TO-B-64-68 (July 1964) Availability: Not for sale to the public.

The backscattering of ^{60}Co gamma-rays from steel, concrete, water, and lead was measured. The differential dose albedoes obtained from analysis of these measurements were compared with calculations using the Chilton-Huddleston formula.

746. M. J. Barrett, J. L. Jones and E. M. E. Murray, Experimental Evaluation of Fallout Protection Using Modeling Techniques - Final Report, TO-B-64-62 (July 1964) Availability: Not for sale to the public.

Work performed at Technical Operations Research under Contract OCD-OS-62-14 from 1961 to 1964 is briefly described. This work consisted of experimental measurements of radiation attenuation through complicated shields such as would be encountered in the study of the protection afforded by existing buildings in the event of a nuclear fallout hazard. A list of the principal parameters involved in the assessment of the fallout protection factor is included, and reports on the individual parameters noted. Curves required to relate results obtained with cobalt-60 test sources to results obtained with the fallout spectrum are appended.

747. Dominic J. Raso and Stanley Woolf, The Dose Resulting from 1.25 MeV Plane Source Behind Various Arrangements of Iron Barriers (Final Report), TO-B-64-49 (June 1964) Availability: Not for sale to the public.

Calculations were performed by the Monte Carlo method to determine the dose at various positions behind parallel barriers (circular or rectangular). Also, calculations were made for a blockhouse geometry. The results were obtained for a 1.25 MeV plane monodirectional source (angles of incidence given by $\cos \theta_0 = 0.0, 0.25, 0.50, 0.75, 1.0$, and isotropic) incident on 20.5, 41.0, and 61.5 psf of iron. Comparisons were made with roof reduction factors obtained by Spencer and with experiments performed at Technical Operations Research.

748. J. E. Watson, Jr., An Analysis of Calculated and Measured Fission Product Activities, BRL-1239 (February 1964) Availability: Not for sale to the public.

Calculations and measurements of delayed beta and gamma activities and spectra following nuclear fission are presented. There is good agreement between various calculations and measurements of gamma activity and spectra of U-235 fission products. There is a degree of agreement in calculated and measured beta activity. U-235 fission product beta and gamma activities are integrated over time to determine the total number of betas and the total gamma energy emitted per fission. Analysis of various information reveals that: (1) Gamma activity and spectra following fission of U-235 by thermal neutrons coincide with that of fission of U-235 by fission spectrum neutrons. (2) Gamma activities of U-235 fission products and Pu-239 fission products are identical within limits of experimental error. (3) Initial beta and gamma activities of U-238 fission products are approximately a factor of 2 higher than for U-235 fission products.

749. Kaye D. Lathrop, Anisotropic Scattering Approximations in the Boltzmann Transport Equation, LA-3051 (February 14, 1964) Availability: $2.25 OTS.

The effect of anisotropic scattering upon solutions of two elementary problems of monoenergetic neutron transport is examined. Solutions to the infinite homogeneous medium problem for a medium containing isotropic and anisotropic scatterers and a fission source, in which the diffusion length appears as an eigenvalue, are derived for several scattering approximations. Solutions are also derived for spatially dependent neutron flux in the problem of a plane source in an infinite medium. In each problem, examination of an exact solution for an anisotropic scatterer with purely forward and backward scattering is illuminating. The relative effectiveness of the scattering approximations developed is evaluated numerically, and extensive tables of solutions for elastic hydrogen scattering are tabulated for reference. In overall effectiveness, in order of increasing accuracy, the approximations considered are ranked as follows: the 1) transport, 2) forward-backward, 3) first-order Legendre, 4) extended transport, and 5) high-order Legendre, scattering approximations.

750. S. Buscaglione and R. Manzini, On the Build-Up of Concrete: Coefficients of the Polynomial Representation, ORNL-TR-349 (June 1964) Availability: JCL $1.60 (fs), $0.80 (mf).

Bivariant polynomial coefficients of the gamma ray dose for a point isotropic source were derived for ordinary concrete, magnetite concrete, ferrophosphorus concrete and barytes concretes.

751. J. Harrison, "Shielding Materials," Nucl. Eng., 7(75), 305-309 (August 1962).

This is a general survey of shielding principles and materials which includes a list of manufacturers in Austria, Belgium, Canada, France, Germany, Italy, Netherlands, United Kingdom, and United States.

752. G. T. Chapman, K. M. Henry and J. D. Jarrard, Pulse-Height Spectra of Gamma Rays Emitted by the Stainless-Steel-Clad Bulk Shielding Reactor II (BSR-II), ORNL-3609 (November 1964) Availability: $3.00 Clearinghouse for Federal Scientific and Technical Information, National Bureau of Standards, U. S. Department of Commerce, Springfield, Virginia.

Measurements have been made of the pulse-height spectra of gamma rays emitted from the core of the Bulk Shielding Reactor II (BSR-II). Preliminary analysis of the data indicates that the gamma rays having energies above 5 MeV originate primarily from thermal-neutron capture in the stainless steel reactor structure. Below 5 MeV the spectrum is apparently composed of the expected continua of prompt-fission and fission-product gamma rays plus a large contribution from thermal-neutron captures in the hydrogen of the water that moderates and reflects the core. Except for reduced intensity, little change was noticed in the structure of the distribution as the amount of water between the reactor surface and the spectrometer was increased from 1 to 60 cm. Angular-distribution measurements were also made about two points at 10 and 25 cm from the surface of the reactor. A simple calculation of the expected spectra at the surface of a homogeneous reactor compared favorably with a measured spectrum.

753. M. B. Wells, Reflection of Thermal Neutrons and Neutron-Capture Gamma Rays from Concrete, RRA-M44 (June 12, 1964) Availability: Trans. Am. Nucl. Soc. 7, 41 (June 1964)

Monte Carlo methods were used to compute thermal-neutron and thermal-neutron capture-gamma ray differential albedos for TSF and Portland concretes. The results obtained for the thermal-neutron differential albedo for Portland concrete were fitted by the equation

$$0.21 \cos^{2/3}\theta_0 \cos\theta$$

where θ_0 and θ are the angles of incidence and reflection respectively. The total thermal-neutron flux reflected per unit incident flux was found to be given by $0.414\ \pi\cos\theta_0$ for TSF concrete and $0.42\ \pi\cos^{2/3}\theta_0$ for Portland concrete.

The Monte Carlo calculations of the capture-gamma dose reflected from concrete per unit incident thermal-neutron flux were fitted by the expression.

$$1.046 \times 10^{-5} \cos^{5/4}\theta_0 \cos^{2/3}\theta$$

for TSF concrete and by the expression

$$6.986 \times 10^{-6} \cos\theta_0 \cos\theta$$

for Portland concrete. The units of the above albedos are $\text{erg}\cdot\text{g}^{-1}\cdot\text{hr}^{-1}\cdot\text{steradian}^{-1}$ per unit incident thermal-neutron flux.

The Monte Carlo calculations of the total thermal-neutron reflection from both TSF and Portland concretes were found to be in reasonably good agreement with albedo expressions derived by both Fermi and Halpern, et al. Calculations of the thermal-neutron capture-gamma dose rates inside 4-ft-diameter concrete-lined cylindrical holes dug in the ground at the Oak Ridge National Laboratory's Tower Shielding Facility show, when compared with measured data, that the major component of the gamma dose rate in the holes is that resulting from thermal-neutron capture in the concrete walls.

754. D. J. Raso, Monte Carlo Computation of 100,000 1-MeV Gamma Rays Scattered by Concrete Slabs, TO-B-64-31 (March 1964) Availability: OTS $7.60 as AD 438 908.

Monte Carlo calculations were performed to determine the backscattering of 1.0 MeV gamma rays from concrete and their transmission through slabs of this medium. The radiation was assumed to be incident at various angles θ_0 ($\cos \theta_0$ = 1.0, 0.75, 0.50, 0.25, and 0.10) on a semi-infinite medium and on slabs with thicknesses of 0.55, 1.1, 2.2, and 4.4 mfp. Case histories of 100,000 on an IBM 7094 digital computer for each of the six incident angles. Differential dose and number distributions as a function of azimuthal and polar angles are reported.

755. R. I. Hildebrand and W. C. Outten, Energy and Angular Distribution of Gamma Rays, Bremsstrahlung and Annihilation Radiation in a Lead Cylinder, ER-10485 (November 1958) Availability: Martin Company, Baltimore, Maryland.

A Monte Carlo Program for the IBM 704 has been developed for calculating the energy spectrum of gamma rays reaching a detector at any point inside a finite open-ended cylindrical shield. The shield may comprise up to ten coaxial regions representing different materials. The energy spectrum and angular distribution of the gamma rays incident upon the outside of the cylinder are assumed to be known.

756. Alexander E. Anthony, Jr. and Eugene Omoda, The Transport of Neutrons through the Atmosphere for a Burst Height of 500,000 Feet, SWC-TDR-62-17 (February 1962) Availability: OTS, $3.60 as AD 272 546.

This report is the sixth in a series of AFSWC technical reports presenting the results of neutron transport in a rarefied atmosphere.

A series of curves is presented showing the scattered and direct neutron flux per initial neutron from a burst height of 500,000 feet and sampling altitudes of 100,000, 200,000, 500,000, and 1,000,000 feet. The scattered flux data were calculated from a Monte Carlo Code developed at the RAND Corporation by Dr. Jesse I. Marcum and Dr. Albert L. Latter. The direct flux data were calculated by the inverse square method.

These data may be used to determine the following: (a) the fraction of the flux or dose attributable to the neutrons of various initial energies and selected energy bands; (b) the total flux or dose due to the scattered neutrons; (c) the total flux or dose due to the unscattered neutrons; and (d) the total flux or dose arriving at a designated distance in space from a nuclear detonation at 500,000 feet.

757. Alexander E. Anthony, Jr., The Transport of Neutrons through the Atmosphere for a Burst Height of 70,000 Feet, AFSWC-TR-61-85 (October 1961) Availability: OTS $5.60 as AD 268 958.

A series of curves is presented depicting the neutron flux per initial neutron from a burst height of 70,000 feet and sampling altitudes from 50,000 to 100,000 feet. The scattered flux data for these curves were calculated from a Monte Carlo code developed at the RAND Corporation by Dr. Jesse I. Marcum and Dr. Albert L. Latter. The direct flux data were calculated by the standard inverse square method.

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ABSTRACTS

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758. Alexander E. Anthony, The Transport of Neutrons Through the Atmosphere for a Burst Height of 50,000 Feet, AFSWC-TN-61-2 (January 1961) Availability: OTS $3.60 as AD 255 920.

A series of curves is presented showing the scattered and direct neutron flux per initial neutron from a burst height of 50,000 feet and sampling altitudes from 30,000 to 100,000 feet. The data for these curves were calculated from a Monte Carlo Code developed at the RAND Corporation by Dr. Jesse I. Marcum and Dr. Albert L. Latter.

759. J. I. Marcum, Comparison of Monte Carlo Calculations with Experimental Results for the Propogation of Gamma Rays Near an Air-Ground Interface, RM-3399-PR (December 1962) Availability: OTS $2.60 as AD 293 864.

The RAND gamma ray Monte Carlo code is used to predict dosages from point isotropic monoenergetic sources of 1.28 and 0.661 MeV near an air-ground interface. These results are compared with experimental results for identical geometry obtained by the Nuclear Defense Laboratory. In general the agreement between the Monte Carlo estimates and the experimental results is quite good.

760. T. O. Sauar, A Monte Carlo Calculation of Scattering from Collimated Co-60 Gamma Rays in Air and Comparison with Measurements in Water, FFIF-F-443 (October 1963) Availability: Norwegian Defence Research Establishment Library, P. O. Box 25, Kjeller, Norway.

The Flux-factor describing the secondary photon flux around a 1.25 MeV gamma ray from a monodirectional source in air has been determined by a Monte Carlo method.

Calculations have been performed for penetration depths up to 1000 meter and discrimination energies from 0.8 - 0.25 MeV.

The results have been applied in predicting the secondary photon flux around a narrow Co-60 beam in water.

There is good agreement for discrimination energies $E_d \geqq 0.4$ MeV (within 10%), while the measured values are 20 - 30% above those calculated at $E_d = 0.25$ MeV.

761. John R. Stehn, Murrey D. Goldberg, Benjamin A. Magurno, and Renate Wiener-Chasman, Neutron Cross Sections, Vol. 1, Z = 1, 20, Suppl. No. 2, BNL-325 (May 1964) Availability: CFSTI $2.50.

This supplement to BNL-325 presents new information on neutron cross sections accumulated in the four years since the previous supplement was issued. It supercedes previous issues only when there is new information. This issue covers only the first 20 elements.

The format differs from preceding versions in several ways. The more normal size, arrangement of all data by nuclide, and fluid format is likely to make this supplement more convenient for the user.

762. J. E. Powell and J. V. Walker, "A Determination of the Cadmium Absorption of Resonance Neutrons in Cadmium-Covered Indium Foils," Nucl. Sci. Eng., 20(4) 476-480 (December 1964)

Theoretical and experimental determinations of the cadmium correction factor F_{Cd} have been made for various thicknesses of indium foils in an isotropic epithermal neutron flux.

It was found that for indium foils with 0.040-in. cadmium covers F_{Cd} varied slightly with foil thickness and reached a maximum value of 1.123 for thicknesses greater than 90 mg/cm^2.

763. G. R. Dalton and R. K. Osborn, "The Effect of Flux Anisotropy on Neutron-Detector Foils," Nucl. Sci. Eng. 20(4), 481-492 (December 1964).

An integral form of the one-speed neutron-transport equation is applied to the case of a neutron-detecting foil placed in a homogeneous medium with an initially non-isotropic neutron population. A series of numerical calculations have been carried out to investigate the effect on the self-shielding flux-depression factor of anisotropy in the initial undisturbed flux. The case of a square foil of gold placed in a light-water medium is investigated. It is found that the existence of anisotropy in the initial flux leaves the flux correction factor essentially unchanged. However, the presence of anisotropy implies spatial non-uniformity of the scalar flux. Thus, movement of the center of mass of a foil in a flux which has a gradient, or rotation of a foil in a flux which has a second derivative can alter the undisturbed flux and the disturbed flux to which a foil is exposed, though the flux correction factor remains unchanged.

764. K. Mather and A. L. Felts, Jr., High Strength, High Density Concrete, U. S. Army Eng. Waterways Exp. Stat. TR 6-636 (November 1963). Availability: U. S. Army Eng. Waterways Exp. Stat., Vicksburg, Miss.

Any sound concrete, in sections of sufficient thickness, can be used to construct a satisfactory biological shield. When space permits, conventional concrete is the most economical, satisfactory shield against radiation. However, when space is a consideration, high-density concrete can be used for shielding. Most high-density concrete for shielding has been made with naturally occurring iron ores, titaniferous iron ores, "hydrous iron ores," and barite. Although structural strength is an important factor in high-density concrete, workability and density have been the major factors previously studied. This investigation showed that high-strength concrete can be made using materials that provide high density, and that high-density concrete can be made using proportions that ensure high strength. When high strength and high density are both desired, they can be attained by a direct combination of the standard practices developed for attaining each condition separately. Concrete made as part of this study using magnetite aggregate or ilmenite aggregate having a hardened unit weight of approximately 230 lb per ft had a compressive strength of 9000 psi at 7 days and 11,000 psi at 28 days age.

765. R. L. French and C. W. Garrett, Gamma-Ray Energy and Angular Distributions Near the Air/Ground Interface from Plane Fallout and Point Co^{60} Sources, RRA-M42 (June 1, 1964) Availability: AD-612027, CFSTI $2.00 (fs), $0.50 (mf).

Monte Carlo techniques were applied in a detailed study of the gamma radiation environment three feet above the air/ground interface due to two types of gamma-ray sources: 1) fallout uniformly distributed on the ground surface, and 2) a Co^{60} point source also three feet above the ground at separation distances of 100 to 800 feet. In each case, the scattered and direct-beam fluxes were computed in each of ten receiver energy intervals between 0.04 and 3.5 Mev and eighteen equal receiver angle intervals between 0° (nadir) and 180° (zenith). All results were integrated over azimuthal angle at the receiver to reflect distribution in polar angle only. The source energy was used as a parameter in the fallout calculations and the results thus computed for each of ten different source energies may be weighted and combined to obtain data for fallout of any desired age. Comparisons of the Monte Carlo data with measured data and with other calculated data, both for fallout and for Co^{60} point sources, show favorable agreement.

766. R. L. French, Gamma-Ray Energy and Angular Distributions Above an Infinite Fallout Field, RRA-T43 (May 1, 1964). Availability: AD-612030 CFSTI $3.00 (fs), $0.75 (mf).

Monte Carlo techniques were applied in a detailed study of the gamma radiation environment produced in air three feet above fallout uniformly distributed on a smooth ground surface. The basic computations included the energy and angular distributions of the scattered photon flux above monoenergetic plane isotropic gamma-ray sources for ten different source energies from 0.10 to 3.5 MeV. For each source energy, the flux arriving at the receiver in each of ten energy intervals between 0.04 and 3.5 MeV and eighteen equal receiver angle intervals between 0° (nadir) and 180° (zenith) was calculated. Separate calculations were performed for the uncollided fluxes from the infinite plane isotropic source. In addition to the fluxes, the dose arriving through each ten degree interval of receiver angle was also computed. The results of the monoenergetic calculations were combined to obtain energy and angular distributions of the photon flux from fallout of four different ages: 1.12 hours, 23.8 hours, 4.57 days, and 9.82 days. Comparisons with measured and previously calculated data, both for actual fallout fields and for simulated infinite plane sources, show favorable agreement.

767. R. L. French, "The Last-Collision Approach to Calculating the Angular Distributions of Fast Neutrons Penetrating a Shield," Nucl. Sci. Eng. 23(1), 28-33 (August 1965).

To perform fast-neutron penetration calculations, a new method is described which offers some of the advantages of the Monte Carlo method and other highly sophisticated methods, yet retains some of the features of the line-of-sight kernel methods. The method includes the use of effective flux removal cross sections to predict a distribution of "last-collision" centers in a shield and uses statistical estimation to obtain the flux at the receiver from each last collision. The chief advantage of the method is that it provides an approximate angular distribution of the neutron flux at the receiver and includes the effects of boundaries, but is much less costly to apply than Monte Carlo. The principal limitation to the accuracy of the method stems from the assumption of no change in neutron direction and energy prior to the last collision. Detailed formulations for slab and for cylindrical geometries are given, along with results of an initial evaluation based on comparisons with Monte Carlo and with measured data for lithium hydride shields.

768. M. E. Anderson and W. H. Bond, Jr., The Neutron Spectrum of a Plutonium-Beryllium Source, MLM-1131 (September 4, 1964). Availability: $0.75 OTS.

The neutron spectrum was obtained for a plutonium-beryllium source (containing 80 grams of plutonium) with the use of nuclear emulsions. Intensity maxima exist at less than 1.0 MeV and at 3.2, 4.7, 7.7, and 9.7 MeV. Those at 4.7, 7.7, and 9.7 MeV are well accounted for in a calculated spectrum. Maxima at less than 1.0 and 3.2 MeV are partially explained. Comparison with other published spectra shows more low energy neutrons for a plutonium-beryllium source with 80 grams of plutonium than for a source containing 13 grams of plutonium. The preparation, neutron flux distribution and neutron growth of plutonium-beryllium neutron sources appear in the appendices.

769. A.Bardett, Radiation Measurements through the Biological Shield of a Calder Hall Reactor, AEEW-R-75 (1961) Availability: Seven Shillings Net, H. M. Stationery Office

Measurements of gamma dose rate, thermal neutron flux and fast neutron flux, through the biological shield of a Calder Hall reactor are described, and compared with theoretical predictions made using methods recommended by A. F.Avery et al. in AERE R-3216.

770. J. Braun and K. Randen, Neutron Streaming in D_2O Pipes, AE-98 (February 18, 1963) Availability: Dep. (mc).

An investigation has been carried out concerning the attenuation of neutrons inside D_2O-filled pipes penetrating a concrete shield. As the purpose has been to simulate the conditions around a heavy water power reactor, pipes surrounded by an annular air gap have also been considered.

Thermal, epithermal and fast neutron fluxes have been measured in three separate pipes (15, 22 and 28 cm in diameter and 100 cm long) with annulii ranging from 0 to 9.7 cm in width.

The thermal flux distribution has been predicted theoretically by assuming it to be composed of three components originating from a fast exponential volume source and two surface sources at the origin; a 1/E-distributed source and a thermal source, of which the latter proved to be negligible. The fast flux distribution has been approximated by a single exponential expression for the configuration with no annulus and with the sum of two exponentials when an annulus is present.

The agreement between measured and calculated values for the thermal flux is better than a factor 1.5 after about 10 cm. For deep penetration (> 40 cm) the agreement is within 20% and only the fast volume source contributes appreciably to the thermal flux. This holds for all cases both with and without annulus.

The agreement in the case of the fast flux is within a factor 1.3 (> 40 cm) for no annulus geometry and about 2-3 with annulus.

Curves are presented for obtaining necessary parameter ('removal' cross section and extrapolated radius) for other geometries than those covered in this report.

771. ICRU, Physical Aspects of Irradiation, Recommendations of the International Commission on Radiological Units and Measurements, NBS Handbook 85 (March 31, 1964) Availability: $0.70 GPO.

This report deals broadly with the physical aspects of irradiation with a considerable amount of new material added since the 1959 report. It includes an extensive discussion of the various techniques for the measurement of absorbed dose as well as exposure. Characteristics of radiation instrumentation are covered in some detail including the more sophisticated work on standards. The section on spectra has been up-dated and a new section added on neutron measurements and standards. Available data for stopping power ratios and the average energy (W) required to produce an ion pair in a gas have been reviewed. On the basis of this review it has been necessary to modify the previous ICRU tables for these factors. This modification amounts to about 1 or 2 percent change in stopping power ratios and up to 1 percent in W.

772. K. O'Brien, Wayne M. Lowder and Leonard R. Solon, A Semi-Empirical Method of Calculating the Energy-Absorption Buildup Factor with an Application to a Uniformly Contaminated Space Having Spherical Boundaries, HASL-2 (October 14, 1957) Availability: $0.50 CFSTI.

A form for the point-source gamma-ray energy absorption build-up function valid for a material in the energy range where it is essentially a Compton scatterer is suggested as a modification of the asymptotic forms derived by U. Fano. Its parameters are evaluated by means of an energy equilibrium condition and a fit to experimental data. The results are compared with those obtained by other methods, and an application to the problem of uniformly and continuously distributed point sources is discussed.

773. H. S. Davis, Use of High Density Prepacked Concrete in Reactor Construction, HW-36572 (May 1955) Availability: $4.80 (ph), $2.70 (mf), CFSTI.

The erection of a concrete biological shield at Hanford, Washington presented several unique construction problems. This shield encloses the radioactive core of a reactor and is designed to protect operating personnel from the harmful effects of pile neutrons and associated gamma radiations. Permanent steel forms were erected to close tolerances around the reactor and filled with high density concrete by means of the prepacked method. Prefabricated sections of the steel forms, or crates, were five to eight feet deep and weighed up to 18 tons each. During erection, the process holes through successive crates were kept within 0.012 in. of their nominal distances. The left side, top shield, and right side of this biological shield were constructed integrally to form a rigid frame. The front and rear crates, which contain numerous process holes, were keyed into the side and top shields so as to permit relative expansions of shield components and to provide for their lateral support. These massive steel and concrete walls provide considerable resistance to blast and earthquake forces, support numerous process facilities, contain pile gas, as well as protect personnel from nuclear radiations.

774. C. E. Clifford, Gamma Shielding Provided by Ducts, DRCL-370 (1962) Availability: AD-282393, Defence Research Chemical Laboratories, Canada.

Dose distributions have been measured in 8", 12" and 3 foot wide ducts exposed to an off-axis source of 0.66 Mev γ's (Cs^{137}). In the main ducts scattered radiation from surfaces not exposed to the incident radiation accounted for at least 25% of the total dose. The variation of the dose due to radiation scattered from the exposed areas was compared with the calculated distributions obtained using various assumptions about the angular differential scatter of radiation. The additional dose reduction provided by side ducts was less than the solid angle fraction subtended by the entrance of the side ducts and varied with their location along the main duct.

775. J. A. Auxier, F. F. Haywood, and L. W. Gilley, General Correlative Studies - Operation BREN, CEX-62.03 (September 1963) Availability: $1.00, CFSTI.

This report describes the physical facilities used during Operation BREN with both the ORNL HPRR and a 1.2-kilocurie Co^{60} gamma source. Tabular data are given with appropriate formulas so that corrections can be made in the data to compensate for the significant changes in air density during Operation BREN.

As the reactor was raised from 27 ft to 1500 ft, an apparent increase in neutron flux was measured at 790 and 1000 yards from the tower; the increase was approximately 2.5.

The relaxation lengths for neutrons and gamma rays in air, for a point fission source at 1125 ft above the ground, are 181 and 245 yards, respectively, when data are reduced to an air density of 1.293 g/liter. The relaxation length for gamma rays in air from a Co^{60} point source exposed at 300 ft is 180 yards at 1.293 g/liter.

The energy spectrum for HPRR leakage neutrons was measured at a distance of 44 in. from the core by utilizing threshold detectors.

776. W. C. Farries, First-Collision Moments Method, NARF-61-12T (MR-N-274), September 30, 1961 Availability: General Dynamics Division, Fort Worth, Texas.

Dose rates resulting from fast neutrons that penetrate the walls of a shielded compartment are calculated by a First-Collision Moments (FCM) Method. The particular advantage of this method is that it can be applied to incident neutrons having arbitrary angular and energy distribution.

A combination of single-scattering theory and moments-method data are used in the development of the method. The results are presented in the form of a ratio of the transmitted dose rate to the initial dose rate for initial energies (E_o) of 15, 10, 8, 5, 4, 2, and 1 Mev and d incident angles (β) of 30°, 60°, and 90°.

Calculated fast-neutron dose rates are compared with available measured dose rates.

777. J. W. Neuberger and R. L. Johnston, Spatial Distribution of Neutrons Beyond a Straight Cylindrical Duct Through a Shield, NARF-57-1T (MR-N-137) (PB-159391) (January 15, 1957) Availability: CFSTI $3.60 (fs).

In the design of experiments using neutrons which issue from a reactor beam hole, it is important to know the spatial distribution of these neutrons beyond the biological shield. In this report, a method for obtaining this distribution suitable for high-speed electronic computation is described together with results obtained from an IBM-704.

In this method, the intensity at the field point is assumed to consist of the unscattered neutron beam and a scattered component arising from internal scattering in a straight cylindrical duct through the shield. The problem is discussed using a general source distribution function and then specialized to the case of a plane cosine source distribution for particular collimating systems.

778. C. E. Humphries and B. J. Workman, A Comparison Between BSR Centerline Dose Rates and Dose Rates Calculated by the Moments Method, NARF-55-23T (March 1956) Availability: General Dynamics, Fort Worth, Texas

Preliminary calculations have been made for gamma and fast neutron dose rates from the BSR and the GTR. The calculations were based on the moments method described in NYO-3075 for gamma radiation and NYO-6269 for neutron radiation. Gamma production due to thermal neutron capture in water was included.

The calculations compared favorably with the centerline gamma and fast neutron dose rates for the BSR and GTR as functions of distance from the core face. A normalization factor was applied to the fast neutron spectral measurements taken at the bare core of the BSR to force the dose rate then calculated to agree with the centerline fast neutron dose rate.

Comparison was made between gamma fluxes calculated by the moments method for the GTR and gamma fluxes as evaluated by use of an empirical spectrum and experimental data. When converted to dose rates the values calculated by the moments method were approximately 50% higher than those obtained by use of the experimental data and the empirical spectrum.

779. E. P. Blizard, T. V. Blosser, and R. M. Freestone, Jr., The Radiation Leakage Survey of the Shield of the Nuclear Ship Savannah, ORNL-3336 (August 15, 1962) Availability: $0.75, CFSTI

A detailed survey of the radiation from the N.S. Savannah shield was made over the period from May 20 through May 25, 1962. The study was accomplished with the reactor at its full power of 69 megawatts during a cruise of 2821 miles in the Atlantic Ocean, out of Yorktown, Virginia. The data were obtained with the ORNL Mobile Radiation Shielding Laboratory No. 1 which was designed and built especially for this survey.

The results of the detailed shield survey show that the design dose rate of 0.5 rem/yr for passenger spaces is not exceeded in any region to which passengers have access. In the areas restricted to crew access the design dose rate of 5.0 rem/yr is nowhere exceeded except in a small region on "D" deck, in a room to which entry can readily be controlled. Specifications for cargo-hold dose rates, 0.5 rem/yr at 1/5 full power, are satisfactorily met.

Additional data obtained include measurements of special points for comparison with dose rates predicted in design calculations, and an experiment examining the result of loss of water from the Savannah's subreactor shield tanks. Predicted dose rates were conservatively higher than the dose rates measured, and the result of the water-loss experiment showed that even with a total emptying of the tanks a safe condition exists only a few feet from the shield.

780. J. S. Rosen, M. O. Burrell, D. L. Cribbs, C. W. Hill, W. B. Ritchie, and J. C. Whiton, Nuclear Weapons Radiation Doses in Armored Vehicles, Volume I, NR 82 (NP 9851 Vol. I)(July 1960) Availability: Lockheed Aircraft Corp., Nuclear Products, Georgia Division.

This is an analytic study for predicting the doses within the interior of armored vehicles due to the initial neutron and gamma radiation from any specific nuclear weapon. Basic data are generated by Monte Carlo techniques for the penetration of neutrons and gamma rays through iron slabs, neutron capture, inelastic scattering of neutrons, and neutron albedos.

A computer program for calculating crew doses contains independent sections which can accommodate new vehicle geometry or weapons radiation environments.

781. L. K. Donovan, Dose Attenuation Factors for Concrete Slab Shields Covered with Fallout as a Function of Time after Fission, USNCEL-R-137 (NP-10306)(AD-258246), (June 1961) Availability: CFSTI, $4.60 (fs).

For radiation shielding, underground or buried fallout shelters have an important advantage over other types of shelters, because the attenuation of the radiation in such a shelter is primarily a function of the thickness of the material in the roof only.

This study was made to investigate the dose attenuation of fallout gamma radiation by various thicknesses of concrete roofs of buried fallout shelters as a function of time after a nuclear detonation. A spectrum of energies is used for the fallout source rather than a single average energy as has been done in previous studies. Dose attenuation factors are derived and presented as a function of the above parameters. The Office of Civil and Defense Mobilization recommends a two-week shelter-stay time in the event of a nuclear attack; therefore, also presented is an average dose attenuation factor for any fourteen-day stay time as a function of time of arrival of the fallout or of shelter-entry time, for various roof thicknesses.

782. R. Corcoran, H. C. Crumpacker, and D. M. Leppke, Radiation Shielding Study, ACNP-5912 (June 22, 1959) Availability: Dep. (mc), $7.80 (ph), $3.30 (mf) CFSTI.

This report describes the alternate concepts studies for the various shielding requirements in the Pathfinder Atomic Power Plant.

It concludes that a concrete shield surrounding the reactor pressure vessel is an adequate biological shield. This shield is made up of two parts; an outer two foot section which is both shield and structural support for the various floors in the reactor building and the inner shield which is strictly a radiation shield. By segregating these sections, localized thermal cracking in the inner shield may be tolerated. Air cooling of this shield is feasible. Containment vessel space requirements are minimized by an off-center arrangement for the reactor and utilizing earth shielding on the outside of the containment shell.

A 26'-0" diameter self-supporting carbon steel tank with protective coating appears to be the most economical concept for the shield pool above the reactor. This tank is surrounded by a two foot thick concrete wall for radiation protection.

The availability and cost of aggregate for high density concrete in the Sioux Falls area is discussed. Except under special and unusual circumstances, its use is not justifiable because the transportation cost of shipping the aggregate to the site is too high.

783. Lead Development Association, London, Lead for Radiation Shielding, No. 3. Design and Erection of Space Shielding (March 1964) Lead Development Association, 34 Berkeley Square, London.

The two previous booklets in this series on Lead for Radiation Shielding dealt, respectively, with the material and its properties, and with the design and construction of lead-shielded containers for storage or transport purposes.

There are however circumstances, particularly in hospitals and factories, which require the use of a collimated beam of radiation or sometimes even an uncollimated source inside a room or other relatively large space, and protection will have to be provided in these cases for those outside. Lead sheet or lead laminates are often used for this type of shielding, which is usually referred to as room shielding, structural shielding, or SPACE SHIELDING.

784. Lead Development Association, London, Lead for Radiation Shielding, No. 4. Local Movable Shields (March 1964). Lead Development Association, 34 Berkeley Square, London.

Previous booklets in this series on Lead for Radiation Shielding have dealt, respectively, with the material and its properties, the design and construction of lead-shielded containers for storage and transport purposes, and space shielding.

Local Moveable Shields are required by hospitals and factories, and by laboratories which handle radioactive materials for research purposes, process and package them for sale to the medical establishments and industry, or treat them for safe disposal. The shields - which must be of a flexible design that is easy both to erect and dismantle - are needed to cover the removal of radioactive materials from their containers, and subsequent examination, manipulation or treatment by remote control. A lead-brick construction meets all these requirements, and this booklet describes the individual brick systems, and the advantages to be gained by using the standard units now available.

785. J. H. Tait, An Introduction to Neutron Transport Theory (1964) London, Longmans, Green and Co., Ltd.

The purpose of this book is to give an introductory account of the mathematical methods used in neutron transport theory. An attempt has been made to introduce the reader to the practical methods which are available, and where possible the physical significance of various steps in the mathematical arguments has been explained. A number of numerical methods are described which require the use of large digital computers. However, the book is not concerned entirely with a description of such methods, and in the energy independent case several analytical solutions are described. An appendix is devoted to an explanation of the variational method, and examples are given in the text of the use of this method in the calculation of boundary conditions for the diffusion approximation, and in the derivation of the diffusion equation from the transport equation.

786. J. W. Leake, and J. W. Smith, Calibration of a Neutron REM Counter, AERE-R-4524 (July 1964) Availability: Three shillings Net, H. M. Stationery Office.

Measurements have been carried out on a neutron counter assembly built to the general specification of Andersson and Braun. The assembly comprises a boron trifluoride proportional counter surrounded by polyethylene, with an intermediate perforated layer of high density boron-loaded plastic.

The neutron response of the counter has been accurately determined at thermal energies and for the range 7 keV to 16 MeV. The response has been shown to fit the U.K.A.E.A. code of practice, relating neutron fluence and dose equivalent, within ± 15% over the neutron energy range 30 keV to 6 MeV. The response per rem for neutrons outside this energy range does not match the required curve quite so well, but it is still possible to achieve a response within ± 50% of that required for a neutron energy range extending from thermal to 12 MeV.

It is shown that the optimum dimensions, of the various parts of the moderator system surrounding the BF_3 counter, are critical for all neutron energies. The moderator components were made from solid blocks of polyethylene, and the optimum dimensions required for a best fit to the required response characteristic were found to differ somewhat from those quoted by Andersson and Braun, who used a moderator system built up from thin polyethylene sheet. The dimensions quoted in this report have been adopted for the design of the A.E.R.E. Type 1940A neutron rem counter.

787. R. Nilsson and E. Aalto, Tests of Neutron Spectrum Calculations with the Help of Foil Measurements in a D_2O- and in an H_2O-Moderated Reactor and in Reactor Shields of Concrete and Iron, AE-155 (September 1964) Availability: Aktiebologet Atomenergi, Library Section, Box 43041, Stockholm 43, Sweden.

Foil measurements covering the fast, epithermal and thermal neutron energy regions have been made in the centre of the Swedish D_2O-moderated reactor R1, in the pool reactor R2-0, and in different positions in reactor shields of iron, magnetite concrete and ordinary concrete. Neutron spectra have also been calculated for most of these positions, often with the help of a numerical integration of the Boltzmann equation. The measurements and the calculated spectra are presented.

788. E. Aalto and R. Nilsson, Measurements of Neutron and Gamma Attenuation in Massive Laminated Shields of Concrete and a Study of the Accuracy of Some Methods of Calculations, AE-157 (September 1964) Availability: Aktiebologet Atomenergi, Library Section, Box 43041, Stockholm 43, Sweden.

Extensive neutron and gamma attenuation measurements have been performed in magnetite and ordinary concrete up to a depth of 2 meters in order to study the accuracy attainable by some shield calculation methods. The effect of thin, heavy layers (Pb) has also been studied. Experimental facilities and instrumentation, especially the foil detection methods used for thermal and epithermal neutrons, are described in some detail. Great weight is laid upon a thorough error analysis. The fluxes measured are compared to those calculated by an earlier version of the British 18-group removal method (RASH B_3), by an improved removal method (NRN) developed at AB Atomenergi, and by numerical integration of the Boltzmann equation (NIOBE). The results show that shielding calculations with the newer methods give fluxes that are generally within a factor of 2-3 from the true values. A greater accuracy seems to be difficult to obtain in practice in spite of possible improvements in the mathematical solution of the transport problem. The greatest errors originate in the translation between the true and calculation geometries in the uncertainty of material properties in the case of concrete, and in approximations and inaccuracies of radiation sources.

789. H. H. Van Tuyl, BREMRAD--A Computer Code for External and Internal Bremsstrahlung Calculations, HW-83784 (September 1964) Availability: $2.00, CFSTI.

A Fortran II program for an IBM 7090 computer was developed for calculating external bremsstrahlung by the method of Bethe and Heitler as extended by Elwert, and internal bremsstrahlung by the method of Knipp and Uhlenbeck. The thick target cross section for external bremsstrahlung from monoenergetic electrons was found to differ significantly at low energies from Wyard's approximation and from a linear approximation of bremsstrahlung intensity. Although both approximate methods have previously been used with reasonable success for calculating shielding requirements for beta emitters, much more satisfactory calculations result from Bethe-Heitler and Knipp-Uhlenbeck methods for determining the bremsstrahlung spectrum.

790. A. F. Avery and J. Butler, The Calculation of Fast Neutron Spectra for Pressure Vessel Damage Studies, AEEW-R-377 (May 1964) Availability: Four Shillings Net, H. M. Stationery Office, also from AEC as NP-14383.

Radiation damage in reactor environments is principally caused by fast neutrons and a knowledge of the neutron spectrum between about 0.01 and 9 MeV is an essential requirement for predicting damage rates in steel pressure vessels. At the present time numerical solutions to the transport equation for laminated shields in one dimension can be obtained by iterative methods and the neutron penetration problem in more complex geometries can in principle be tackled by Monte Carlo methods in which special sampling techniques are employed to reduce the variance in regions remote from the source. For design calculations the computing time requirements of both iterative and Monte Carlo methods is prohibitively large and their accuracy at deep penetration is limited by the fundamental cross section data. These disadvantages are to some extent overcome in the multigroup removal source method which has proved to be a valuable tool for secondary gamma source predictions in shields. An attempt has accordingly been made to extend the scope of this technique in order to calculate the fast neutron spectrum in laminated shields for the study of pressure vessel damage. The new model has been applied to graphite, water and D_2O moderated systems; it compares favourably with the more rigorous techniques and affords a simple and relatively accurate design method for predicting the neutron spectrum incident on a reactor vessel in regions which are not perturbed by ducts and voids.

791. T. H. Jones, N. E. Scofield, L. G. Haggmark, and W. J. Gurney, Experimental Determination of the Gamma-Ray Angular Dose Albedos of Concrete, Aluminum and Steel, USNRDL-TR-790 (October 6, 1964) Availability: AD-452958.

The angular distribution of dose albedo was measured for "semi-infinite" concrete, aluminum and steel slabs irradiated by plane-parallel beams of Co^{60} and Cs^{137} gamma photons. The photons were incident on the slabs at angles, measured from a normal to the slab, of arccos 1.00, 0.75 and 0.50. For the necessary sensitivity, the reflected dose rate was measured by a digital dosimetry system using a plastic scintillator as the detector.

An empirical formula for angular dose albedo was derived from the experimental data. Comparisons are made with the Monte Carlo calculations of D. J. Raso and the experimental results of C. E. Clifford (DRCL).

792. Charles M. Huddleston, A Simple Method for Calculation of Gamma-Ray Shielding Properties of Shelter Entranceways, USNCEL TN-651 (October 26, 1964). Availability: U. S. Naval Civil Engineering Laboratory, Port Hueneme, California.

An empirical equation is described for calculating the attenuation factor for gamma radiation within a ducted entranceway into a shelter. Some sample problems are worked, and a discussion is given of the accuracy of the formula.

793. M. Merrill, W. L. Cowell, Literature Survey of Concretes for Nuclear Radiation Shielding, USNCEL-TR-084 (AD-239871) (June 1960) Availability: Naval Civil Engineering Laboratory, Port Hueneme, California.

A literature survey was made to determine the extent of knowledge about shielding materials used as aggregates in concrete. Information is presented on (1) mix designs incorporating various types of heavy aggregates, (2) available cost data relating to construction methods and aggregates, (3) problems associated with heavy concretes during construction, and (4) nuclear shielding properties for concretes.

The use of heavy aggregates for increasing the density of concretes has been well explored by previous investigations. At present there are no materials that can compete economically with varite and iron ores. Of the iron ores, magnetite is the most popular. Of the high cross-section materials for thermal neutron absorption, boron is generally the most economical.

794. A. B. Chilton, D. Holoviak, and L. K. Donovan, Determination of Parameters in an Empirical Function for Buildup Factors for Various Photon Energies, USNCEL-TN-389 (AD-249195), (August 1960) Availability: Naval Civil Engineering Laboratory, Port Hueneme, California.

In the computation of gamma-ray attenuation, it is desired to use a simple expression for build-up factors. In this note a simple analytical expression was used for Dose Build-up Factors from a radioactive isotropic point source. The parameters of the expression were determined by the method of "least squares" to obtain an optimum fit to experimental data for build-up factors for aluminum at various photon energies.

795. R. L. French, M. B. Wells, and N. M. Schaeffer, Penetration of Neutron and Gamma Radiation through the Openings of Underground Structures, RRA-T41 (November 30, 1964) Availability: AD-613591 CFSTI $2.00 (fs) (mf) $0.50.

Calculations were performed of the radiation intensities in open concrete-lined cylindrical holes and in open concrete bunkers due to a fission source located in air at distances of 100 to 700 feet. The calculations are part of an analysis of experiments performed with the Tower Shielding Reactor II (TSR-II) at Oak Ridge National Laboratory which included numerous radiation measurements in the underground structures with and without shields over the openings. The approach followed in the calculations was to determine, from geometric considerations, the portion of the incident radiation intensities which could enter the structures. Methods were developed to account for effects of the air-ground interface and albedo models were devised to simplify calculation of wall-scattering and production of secondary radiation in the walls. In general the calculated and measured data agree within approximately 25 per cent.

796. J. I. Marcum, Neutron and Gamma Ray Dosages from a Source of 14 MeV Neutrons Near an Air-Ground Interface, RM-4190-PR (August 1964) (SECRET).

This report represents radiation dosage calculations for sources of 14 MeV neutrons at various altitudes above an air-ground interface. Several changes in cross sections have caused these results to differ substantially from previously obtained results for similar calculations. These changes and others are necessary for making up-to-date predictions of the radiation effects.

797. G. Aude and J. Laverlochere, Gamma Spectra of Radioelements Formed by Irradiation with 14-MeV Neutrons, ORNL-tr-348 (1963). Availability: Dep. (mc). (Translated from SAR-G-63-38)

In recent years 14 MeV neutron generators uncovered new analytical possibilities based on activation. Many radioelements formed by radiation in such neutron fluxes are different from those produced in reactors.

The objective of this work was to formulate a catalog of the γ spectra of these radioelements and to measure the dosage sensitivities that can be expected when this method is used. In order to do so they radiated about 50 elements and measured the γ ray spectra emitted by the formed radioelements. A 150 KV generator with a tritium target was used. The normalized irradiation and counting conditions thus made it possible to produce a catalog of the spectra and a comparative table of sensitivities. These are published in this report.

798. F. W. Kruger, Monte Carlo Calculations of Gamma-Ray Back-scattering by Air, ORNL-tr-440 (June-July 1964). Availability: Oak Ridge National Laboratory, Oak Ridge, Tennessee. (Translated from Kernenergie 7, 406-11 (June-July 1964)).

A point source of gamma-quanta with an angular distribution symmetrical to the surface normal on the interface is located on the absorbing interface of an air half-space. By means of the Monte Carlo method, the dose rate produced by back-scattering in the air half-space is determined on the interface for distances between 15 and 500 m from the source, for energies of the quanta emitted by the source between 0.2 and 1.5 MeV as well as for different zones of the angle of reflection. The results of the Monte Carlo calculations can be shown with sufficient accuracy by empirical equations and are discussed.

799. T. E. Northrup, "A GAM-NEU Viewing Window Design," Nucl. Struct. Eng. 1(1), 108-117 (January 1965).

Scientific and engineering experimenters are increasing their research activities in areas where equally potent amounts of gamma and neutron radiation occur. Shielded viewing windows have offered the best means for visually observing experiments being conducted in a single type of radiation environment. Multiple radiation does not change the usefulness of viewing windows, but it does add to their design problems. A discussion is presented on radiation sources, design criteria, neutron and gamma shielding equations, effective removal cross sections, window configuration, standard terminology, and the construction requirements for a single Gam-Neu window designed for installation at the Oak Ridge National Laboratory. A design approach for a Gam-Neu window is presented as a guide for the development of future shielded viewing window concepts.

800. G. Kunszt, "High Water Content Shielding Concrete with Iron and Bauxite Aggregates," Nucl. Struc. Eng. 1(1), 118-126 (January 1965).

Since limonite ores available in Hungary do not satisfy the general quality requirements for reactor shielding concrete aggregates because of insufficient crystallinic water and low density, the possibilities of using bauxite instead of limonite were explored. Though the density of bauxite is still lower than that of Hungarian limonites, there are types with more than 25% hydrate water content. Use of bauxite as concrete aggregate resulted in a hydrate water content exceeding that of best limonite concretes of abroad, while sufficient specific density can be provided by the addition of a due percentage of scrap iron. This new type of concrete has adequate mechanical and thermal characteristics, and a hydration water content to be considered stable in spite of chemical reactions between the $Ca(OH)_2$ set free of the cement on one hand, and bauxite on the other hand. This research concluded with the establishment of empirical formulae for the relationship between concrete composition and neutron attenuation.

801. A. B. Chilton, "Backscattering of Gamma Rays from Point Sources by an Infinite-Plane Concrete Surface," Nucl. Sci. Eng. 21(2), 194-200 (February 1965).

Backscattering factors, as fractions of the direct dose rate, are obtained for point sources of gamma radiation, specifically Cs^{137} and Co^{60}, placed near a plane interface between vacuum and concrete. The method is based on the application of albedo principles, using the Chilton-Huddleston formulation for albedo. The results are considered practically applicable to air-concrete or air-ground interface situations, provided the source-detector, source-interface, and detector-interface distances are within certain limits. The lower limit is in theory the order of a mean free path of the source radiation in concrete, although under certain circumstances the present results are valid for distances even less. The upper limit appears to be on the order of a few dozen feet, but further precise experimental work is needed to establish this.

802. M. A. Capo, Polynomial Approximation of Gamma-Ray Buildup Factors for a Point Isotropic Source, APEX-510 (November 1958). Availability: $1.50 CFSTI.

Polynomial expressions were derived for approximation of the gamma ray dose, energy, and energy absorption build-up factors tabulated for a point isotropic source in NYO-3075. The expressions represent the buildup factor as a function of gamma ray energy and the number of relaxation lengths for a specified material. Bivariant polynomial coefficients are presented in tabular form for water, aluminum, iron, tin, lead, tungsten, and uranium. In addition, coefficients are presented for approximation of buildup factors as a function of the number of relaxation lengths at specific energies.

803. Susan M. Miller and K. Parker, Neutron Cross Sections of Natural Titanium in the Energy Range 0.001 eV - 18 MeV -- Incorporation of United Nuclear Corporation Data in the UKAEA Nuclear Data Library, AWRE O-77/64 (October 1964). Availability: Dep. (mn) Micro Methods Ltd. E. Ardsley and Wakefield, Yorks, England.

UNC neutron cross section data for natural titanium has been modified as necessary and incorporated in the UKAEA Nuclear Data Library. Energy and angular distributions of secondary neutrons from elastic scattering and the (n,n') and (n,2n) processes are given so that the data are complete for the purposes of neutronics calculation. The appropriate nuclide identification number is 190.

804. F. J. Davis and P. W. Reinhardt, "Radiation Measurements Over Simulated Plane Sources," Health Physics 8(3) 233-243 (June 1962).

To better interpret aircraft measurements of fallout, a simulated plane source array was laid out with positions at 100-ft intervals in the form of a square each side of which measured 2000 ft. For ground measurements a smaller square array of 100 positions at 10-ft intervals was placed in the center of the large array. Two sets of sources were used in the array. Co-60 and Cs-137 sources of 4.5 and 15 mc, respectively, were available to be placed in the large array and sources one-hundredth these values in the smaller array. Measurements 3 ft above ground indicated 200 μr/hr for Co-60 and 225 μr/hr for Cs-137. Measurements were made in aircraft at 100-ft intervals above the array using scintillation counters and a 256-channel scintillation spectrometer. Flights for comparison of point sources were made over Co-60, Cs-137 and I-131. Relationships of the radiation intensity above point and plane sources as well as spectral data are given. For these sources the tests demonstrate that above certain heights the effect of build-up cannot be neglected but is adequately represented by a linear formula.

805. C. M. Huddleston and N. F. Shoemaker, A Mathematical Derivation of Contour Lines for Constant Albedo of Gamma Rays on Concrete, USNCEL TN-539 (October 31, 1963) Availability: AD-422861 $1.60 CFSTI.

The differential dose albedo of gamma rays on concrete depends on the energy of the incident gamma radiation, the polar angle of incidence, and the polar and azimuthal angles of backscatter. A mathematical method is developed to derive iso-albedic contours, i.e., contour lines on the surface of a hemisphere where backscattered radiation dose will be constant, when the angle of incidence is fixed and the point at which scattering occurs is at the center of the hemisphere. Graphs are presented showing iso-albedic contours for gamma radiation from Co-60 and Cs-137, incident at polar angles with cosines: 1.00, 0.75, 0.50, 0.25, and 0.10.

806. R. K. Thatcher, D. J. Hamman, W. E. Chapin, C. L. Hanks, and E. N. Wyler, The Effect of Nuclear Radiation on Electronic Components, Including Semiconductors, REIC Report No. 36 (October 1, 1964). Availability: CFSTI, $6.00 (cy).

This report presents information available in the REIC files that covers the state of the art of knowledge on the effects of nuclear radiation on basic electronic parts. It represents a summary of information accumulated within the past 3 years plus information previously reported in REIC Reports Nos. 1, 10, and 18. This report presents component results that are grouped as to family within each component class type. The results presented in the report are intended to provide a basis for judging the merits of the parts when they are to be used in circuity that will be exposed to a radiation environment. Some of the data included in the report can be considered as design oriented to the extent that they indicate radiation levels at which the parts would be expected to perform satisfactorily.

807. W. E. Chapin, D. J. Hamman, E. N. Wyler, and D. Jones, Space-Radiation Damage to Electronic Components and Materials, REIC Report No. 32 (October 30, 1963) Availability: Radiation Effects Information Center, Columbus, Ohio.

This report presents the results of research experiments conducted at various radiation facilities to determine the effects of proton and electron radiation on the electrical and physical properties of electronic components and their materials. Most of the investigations are concerned with radiation effects on such semi-conductor devices as solar cells, solid-state diodes, and transistors. This arises from the knowledge that these components will cause most of the problems in space applications. A limited amount of experimentation has been conducted for other commonly used electronic components because gamma- and neutron-radiation data indicate that they do not present problems comparable to solid-state devices. Metals are also discussed.

Some discussion concerning shielding is included with the hope that research in directions of improving devices and their operating conditions will be encouraged.

808. Jan Nilsson, "Geometrical Attenuation of Particle Streaming in Annular and Ordinary Ducts," Nukleonik, 6(6), 285-287 (October 1964).

Current annular and ordinary duct formulas have been generalized with respect to the angular source distribution function which was put proportional to an arbitrary non-negative power of cosine of the emission angle. Each of these generalized formulas is shown to be a special solution of a generally derived "attenuation-integral."

The frequently used formula approximating streaming attenuation in a straight annular duct overestimates in general the average flux at the duct and even at large distances. A more correct and even slightly simplified expression has been derived analytically for the average flux and a comparison has been made between the two expressions.

809. C. E. Clifford, "Effects of the Ground on the Gamma Dose from Distributed CS-137 Sources," Can. J. Phys., 42(12), 2373-2383 (December 1964).

The effects on the exposure dose of different ground materials and of a ground cover similar to snow were measured for a smooth surface contaminated with 0.66-MeV γ sources (^{137}Cs). Measurements were made also of the effect of a rough surface on the exposure dose. For uniform contamination of a smooth ground, agreement within 10% was obtained with theoretical dose distributions based on infinite-medium calculations. However, even moderate ground roughness reduced the dose at a detector height of 1 meter by a factor of two below its value for a smooth surface. Failure to specify the effect of ground roughness on smooth field calculations as a function of the surface roughness appeared to be the largest source of error in the calculation of exposure dose.

810. N. J. Broadway and S. Palinchak, The Effect of Nuclear Radiation on Elastomeric and Plastic Components and Materials, REIC Report No. 21 (Addendum) (August 31, 1964). Availability: AD-454056, Radiation Effects Information Center, Battelle Memorial Institute, Columbus, Ohio.

This report is an addendum to REIC Report No. 21 and presents the state of the art of the effects of nuclear radiation on elastomeric and plastic components and materials from 1961 to the present.

The mechanism of radiation damage and the effects of radiation in various environments are briefly discussed. Data summarizing the radiation-effects information on specific components and on the various types of elastomers and plastics are presented in detail. Areas in which additional work is needed are indicated. Radiation polymerization or volcanization are not covered in this report.

The report is intended to be sufficiently inclusive to make it valuable as a reference guide relative to radiation effects under varying conditions of temperature and vacuum on elastomeric and plastic components and materials.

811. K. M. Case, "Elementary Solutions of the Transport Equation and Their Applications," Annals of Phys. 9(1), 1-23 (January 1960).

A new method of treating problems involving the transport equation is discussed. Starting from Van Kampen's observation that it is sufficient that "solutions" be distributions, the elementary solutions of the homogeneous equation are considered. These are found to have completeness and, in some cases, orthogonality properties which lead to the solution of more interesting problems by a conventional eigenfunction expansion. While the method is illustrated here with the simplest examples of neutron diffusion, it seems to be generally applicable.

812. C. A. Bisselle, R. A. Karm, and J. Wethington, Jr., "An Experimental Study of γ-Ray Attenuation in Polyethylene-Lead Shields," International Journal of Appl. Rad. and Isotopes 15(9), 529-539 (September 1964).

The shielding efficiency of lead and of polyethylene-lead combinations was studied with Co-60 radiations. Four types of shields of equal surface density were used: (i) lead alone, (ii) polyethylene followed by lead, (iii) lead followed by polyethylene, (iv) lamination of polyethylene and lead. Pulse-height spectra of the radiations transmitted through these shields were taken and converted into their respective photon spectra by the inverse matrix technique. From the photon spectra, dose rates in air were calculated. It was found that lamination of polyethylene and lead gave the best shield. Energy deposition ratios in NaI(Tl) obtained directly from the pulse-height spectra agreed with the dose rates in air obtained from the photon spectra.

813. D. L. Broder and K. K. Popkov, "Methods for the Calculation of the Radiational Thermal Output in the Body and Shielding of a Nuclear Reactor," Soviet J. At. Energy (English Transl.) 15(5), 1113-1120 (November 1963).

The radiational thermal output in the body and protective shields of energy-producing reactors is very dangerous, since it can lead to the explosion of these reactors. In order to lower the amount of heat escaping into the body and the most strongly stressed structural sections of reactors, heat screens are usually used.

In the present article we describe methods for the calculations needed in the design of thermal shields. Expressions are given that can be used for the calculation of the thermal output due to the penetrating radiation in various shielding setups, including those with the most commonly used geometry.

814. B. M. Terent'ev, V. A. El'tekov, and D. I. Dolenko, "Calculation of γ-Ray Energy Absorption in Heterogeneous Macro-Systems," Soviet J. At. Energy (English Transl.) 15(5), 1127-1131 (November 1963).

By means of the Monte-Carlo method, a determination was made of the fraction of γ-ray energy absorbed in the elements of a complex heterogeneous system formed by a cylindrical, irradiated water-equivalent mass, cylindrical γ-ray sources located inside or outside the mass, and units of auxiliary equipment. From the results, various energy efficiencies (e.e) were determined depending on the γ radiation from particular systems which were models of radiochemical equipment. Results from corresponding experiments were in good agreement with the calculated data. The program enabled one to find the e.e. for arbitrary macro-systems of a similar type.

815. I. B. Keirim-Marjus, V.T. Korneev, V. V. Markelov, and L. N. Uspenskii, "Measurement of Neutron Tissue Dose Outside Reactor Shielding," Soviet J. At. Energy (English Transl.) 15(5), 1132-1139 (November 1963)

RUS instruments, developed by the authors, are described that enable one to measure the flux and tissue dose rate of intermediate neutrons, which make a significant contribution to neutron tissue dose outside reactor shielding. The neutron dose composition was investigated in experiments at the IRT-1000 reactor, and it was shown that it depends essentially on shielding composition. It was established that the neutron tissue dose computed from readings taken with the RPN-1 instrument were actually too low by a factor amounting to one and one half outside water shielding to five outside concrete shielding.

816. C. P. Altamirano and S. T. Perkins, Angular Distribution Coefficients for Elastic Scattering of Neutrons, AN-1321 (December 1964) Availability: TID-21629 CFSTI $5.00 (fs).

This report contains the results of an analysis which utilized the experimental differential cross section for elastic scattering of neutrons. The calculated quantities for all materials were the elastic scattering cross section, the average logarithmic energy decrement per collision and the Legendre expansion coefficients. The neutron energy range was approximately from 0.1 to 14 Mev. These parameters have been presented graphically as a function of neutron energy when sufficient data were available. For materials in which this was not the case, the results are presented in tabular form.

817. John L. Abbott and Robert L. Berger, Basic Neutron Transport Computation for Determining Neutron Effects on Nuclear Weapons Systems, NAVWEPS Report 8293 (July 31, 1964) Availability: AD-444375, Naval Weapons Evaluation Facility, Albuquerque, New Mexico.

Based on information from previous studies of neutron transport, this study develops basic neutron transport data, expressed as neutrons per square centimeter per original source neutron. Computation methods were developed for determining direct and scattered neutron flux which are combined to give the total basic neutron flux. These data were developed for co-altitude ranges from 100 to 100,000 feet at altitudes ranging from sea level to 200,000 feet.

818. E. R. Leach, B. P. Fairand, and L. H. Bettenhausen, The Space-Radiation Environment and Its Interactions with Matter, REIC Report No. 37, (January 15, 1965) Availability: AD-456854 Radiation Effects Information Center, Columbus, Ohio.

Space-radiation data are summarized and evaluated. Radiation-measurement techniques are discussed, and the apparatus used on several American space probed and satellites is described. Particle flux as a function of type of radiation and energy is summarized in graphic form, and basic concepts of the interaction of space radiation with matter are discussed.

819. R. Bjornerstedt, "Health Hazards from Fission Products and Fallout - II. Gamma Radiation from Nuclear Weapons Fallout (1)," Arkiv. for Fysik 16(28), 293-313 (October 1960).

In the evaluation of the health hazards from nuclear weapons fallout gamma-radiation as it is in general sufficient to consider fallout as equivalent to fission products formed in thermal neutron fission of U-235 as the total gamma-decay and the average gamma-energy are closely comparable most of the time. In the time interval 1.2-6 years an exception to this rule should be made if the fission modes (Pu-239, n_f), (U-238, n_{14}) and (U-235, n_{14}) are considered and an accuracy of better than 30% is desired.

Gamma-spectra given as number of gamma-quanta per minute and 0.1 MeV energy interval for 10^4 fissions have been calculated for the following times after fission: hours: 1, 2, 5, 10; days: 1, 2, 5, 10, 30, 100; years: 1, 3, 10, 30, 100. Fifty energy groups have been used, covering the range 0.0-5.0 MeV. Each group has a width of 100 MeV from e.g. 1.300-1.399 MeV (denoted by 1.3 MeV).

The difference between (U-235, n_{th}) and other fission modes has been analyzed in two ways. Firstly, the sum of gamma-quanta per minute as a function of time after fission is expressed as percentage deviation from (U-235, n_{th}). Secondly, the average gamma-energy (Σ MeV/Σ gamma) has been calculated for all six modes.

820. R. L. Henry, B. O. McCauley and S. C. Doney, Ordnance Corps Radiologically Protected Pod Test at ORNL-TSF - Interim Report, FZK-135 (August 1962) (SECRET)

At the Tower Shielding Facility of Oak Ridge National Laboratory, the shielding effectiveness of the Radiologically Protected Pod was tested in two radiation environments. These environments were provided by the leakage from two shield systems around the Tower Shielding Reactor. Both shields simulated a fission-weapon radiation environment, with the neutron spectrum of one being harder than the other. Both shields yielded approximately the same neutron-to-gamma ratio. Extensive free-field mapping measurements were made of both environments.

The RPP was tested as a function of source-RPP separation distance, angle of orientation, source elevation angle, RPP height above the ground, and auxiliary shielding. Measurements were also obtained inside seven geometric shields designed and fabricated by the Ballistics Research Laboratory.

The measured protection factors afforded by the RPP for neutrons agreed with predicted values; however, the measured protection factors for gammas were lower than predicted.

821. W. C. Farries and J. R. Stokes, Postanalysis of Ordnance Corps Radiologically Protected Pod Test at ORNL-TSF, FZK-145 (December 1962) (Secret).

The Radiologically Protected Pod (RPP) was tested in two simulated nuclear-weapons-burst environments at the Oak Ridge National Laboratory - Tower Shielding Facility (ORNL-TSF). The RPP was tested both by itself and when integrated into the tank mockup - i.e., the engine-compartment and weapons-housing mockup. This latter configuration has been designated the Radiologically Protected Tank (RPT). Data from the RPP tests were used to evaluate the calculational methods, whereas data from the RPT tests were used to evaluate the overall shielding effectiveness of radiologically protected vehicle.

Station 4, at a horizontal distance of 840 feet from the reactor, was the primary location for the RPP and RPT tests and was also the primary location for which the analysis was performed. Calculations of the fast-neutron and gamma dose-rate attenuation by the RPP and RPT were made as a function of vehicle rotation angle for each source environment. Neutrons reflected from the interior walls of the RPP (or RPT) comprise approximately 33 percent of the total neutron dose inside the RPP (RPT).

822. F. C. Engesser, W. E. Thompson, and J. M. Ferguson, Gamma Rays Resulting from Interactions of 14.7 MeV Neutrons with Carbon, Nitrogen, Oxygen and Silicon, USNRDL-TR-791 (October 23, 1964) Availability: AD-453072, Naval Radiological Defense Laboratory, San Francisco, California.

The prompt gamma rays resulting from the interaction of 14.7 MeV neutrons with the nuclei of the elements carbon, nitrogen, oxygen and silicon, have been investigated. Cross sections for the production of many of these gamma rays have been measured by observing them with a carefully shielded detector during neutron bombardment of a given elemental sample. It is found that the n,n'γ process predominates and that gamma rays resulting from the deexcitation of the lowest energy levels are generally most prominent. Other processed including n,pγ, n,$\alpha\gamma$ and n,n'$\alpha\gamma$ are also observed. Some of the most important gamma rays such as the 6.13 MeV gamma ray from O^{16} the 4.43 MeV gamma ray from C^{12} and the 1.78 MeV gamma ray from Si^{28} are found to have respective production cross sections due to the n,n γ process of 9.95 ± 1.0, 13.13 ± 1.3 and 19.02 ± 2.7 mb/sr at 90 deg.

823. C. K. Menkes, Calibration of High-Intensity Co^{60} Sources at Camp Parks and Some Related Field Measurements, USNRDL-TR-794 (November 1964) Availability: Naval Radiological Defense Laboratory, San Francisco as AD-454477.

Five high-intensity Co^{60} gamma sources at the outdoor Irradiation Range in Camp Parks were calibrated and some measurements made on dose distributions and scattering contributions to direct dose. These measurements utilized dosimeters both energy-dependent and independent to give an indication of the degree of energy degradation of the primary gamma beam with distance. The vertical dose distribution to 5 ft in height off the ground was examined as a function of horizontal distance from the source. Boundaries of the inverse square falloff region are delineated and some limitations set on extrapolating the results.

All five sources were found to be somewhat below their curie rating though the four track sources were uniform to within 5 percent of each other. The tower source, however, proved to have a curie rating of about 1050, or approximately one third that of the track sources at the time of calibration.

824. S. Pearlstein, Neutron Cross Section Evaluation Group - Analysis of (n,2n) Cross Sections for Nuclei of Mass A > 30, BNL-897 (December 1964) Availability: $3.00 CFSTI.

The ability of the statistical model of nuclear reactions to interpret measured (n,2n) cross sections has been tested. Good agreement was observed for 22 nuclei having mass numbers greater than 30. The statistical model was then applied to all nuclei in this mass range to estimate both mono-energetic and fission spectrum averaged cross sections.

825. R. W. King, N. J. Broadway and S. Palinchak, The Effect of Nuclear Radiation on Elastomeric and Plastic Components and Materials, REIC Report No. 21 (September 1, 1961) Availability: $5.00 CFSTI as AD-267890.

This report presents the state of the art of the effects of nuclear radiation on elastomeric and plastic components and materials from 1947 to the present. As such, it supersedes the following REIC documents: Reports 3, 3A, 3-2A, 9, and 13, and Memorandums 1, 3, 8, 15, and 17.

The mechanism of radiation damage and the effects of radiation in various environments are briefly discussed. Data summarizing the radiation-effects information on specific components and on the various types of elastomers and plastics are presented in detail. Areas in which additional work is needed are indicated. Radiation polymerization or vulcanization are included only if the data have a bearing on radiation effects on the finished polymer.

This report is intended to be sufficiently inclusive to make it valuable as a reference guide on the effects which can be anticipated from nuclear radiation on elastomeric and plastic components and materials.

826. D. G. Collins, R. L. French, L. G. Mooney, K. W. Tompkins and M. B. Wells, Simplified Methods for Calculating the Penetration of Neutron and Gamma Radiation into Underground Concrete Structures, RRA-M45 (November 25, 1964) Availability: AD-609308 CFSTI (fs) $2.00, (mf) $0.50.

The analysis of two experiments performed at Oak Ridge National Laboratory's Tower Shielding Facility provides a basis for the development and evaluation of simplified methods for predicting the prompt weapons radiation intensities inside underground concrete structures. Using the TSR-II in a special shield to simulate a nuclear weapon radiation environment, one experiment mapped the radiation inside two concrete bunkers with an interconnecting tunnel located in a hillside approximately 700 feet from the source. The other experiment mapped the radiation in concrete-lined cylindrical holes at distances of 100, 228 and 450 feet from the reactor. In both experiments the openings of the underground structures were shielded with various thicknesses of concrete. Albedo methods were developed for calculating neutron and gamma-ray reflection from the concrete walls of the structures and for calculating gamma dose rates due to thermal neutrons captured in the walls. Penetration calculation methods which consider the energy and angular distributions of the incident and emergent radiations were also developed. Most of the simplified methods are based on systematization of Monte Carlo data. Except for the problem of predicting thermal-neutron fluxes for use in secondary gamma calculations, the overall calculational approach appears adequate.

827. M. B. Wells, A Monte Carlo Study of Neutron Transport Through Aluminum, Polyethylene, and Carbon Slabs, RRA-T47, (December 30, 1964). Availability: Radiation Research Associates, Fort Worth, Texas.

Monte Carlo methods were used to perform a parameter study of the transport of fast neutrons through slabs of aluminum, polyethylene, and carbon. The calculations were performed for the purpose of providing data which could be used in further machine calculations of the neutron dose rate in space vehicles which results from proton-induced nuclear evaporation. Plane isotropic monoenergetic neutron sources were used in the calculations. The source plane locations and source energies were selected so that data for slabs to 30 cm in thickness and source energies between 0.5 and 14 Mev could be obtained by interpolation from the Monte Carlo calculations.

Infinite media data for plane sources in the same three materials were obtained from additional Monte Carlo calculations of the spatial distribution of the dose rate from point-isotropic-monoenergetic neutron sources in these infinite media. A comparison of the results of the slab and infinite medium calculations shows that there are significant boundary effects on the fast-neutron dose rates transmitted through aluminum and carbon slabs. Although the magnitude of the boundary effect depends on the source material, plane location, and slab thickness, large differences are noted between the transmitted dose rates for slabs and for infinite media.

828. Arthur B. Chilton, Backscattering for Gamma Rays from a Point Source Near a Concrete Plane Surface, Bulletin 471 (Engineering Experiment Station, University of Illinois) (1964) Availability: $1.50, Engineering Publications Office, College of Engineering, University of Illinois, Urbana, Illinois.

Backscattering factors, as fractions of a direct dose rate in a vacuum, are obtained for point sources of gamma radiation, specifically cesium-137 and cobalt-60, placed near a plane interface between vacuum and concrete. The method derived is based on application of albedo principles, using the Chilton-Huddleston formulation for albedo. Previously published values of parameters for the Chilton-Huddleston formula are not used: but a new set, believed to be better, is derived. The results are considered practically applicable to air-concrete or air-ground interface situations, provided the distances of source to detector, source to interface and detector to interface are greater than a mean-free-path of the source gamma radiation in the concrete or ground but substantially less than that in air. The basic methodology relates to infinite planes; however, factors for correcting these values for certain finite plane cases are derived. Limited experimental data are available and are compared with these calculations. The techniques for making the calculations for the specific radioisotopes indicated are generally applicable to any monoenergetic or mixed energy source having photons within the energy range from 0.2 MeV to several MeV.

829. Y. T. Song, Fast Neutron Streaming Through Two-Legged Concrete Ducts, USNCEL-R-354 (February 2, 1965) Availability: AD-457746, NP-14882, Naval Civil Engineering Laboratory, Port Hueneme, California.

As a part of the Laboratory's fundamental shielding studies for personnel shelters, fast neutron dose rates are calculated in the second leg of an air duct through concrete for neutron energies of 14 MeV and 2.5 MeV. The calculational technique is based on the albedo concept. Dose rates are also calculated by a Monte Carlo technique, and the results obtained by the two theoretical methods are compared with each other and with experimental measurements.

Comparison shows very good agreement among these three independent determinations.

830. National Bureau of Standards, Measurement of Absorbed Dose of Neutrons, and of Mixtures of Neutrons and Gamma Rays, NBS Handbook 75 (February 3, 1961) Availability: Superintendent of Documents, Washington 25, D. C., price 35 cents.

This Handbook presents a summary of currently available methods for determining energy absorption in matter as a result of its interaction with neutrons. Since neutrons are almost invariably accompanied by gamma radiation, mixtures of gamma radiation and neutrons are included. Such an endeavor is herein referred to as mixed radiation dosimetry, although the term absorbed dose is reserved to refer to only one of the quantities of interest, namely the specific absorbed energy in a specified medium (e.g., ergs per gram of water).

Discussions are general wherever possible; i.e., one is in principle just as interested in the application of dosimetry to radiation chemistry and materials damage as in its application to health physics and radiobiology. It is inevitable, however, that most of the detailed examples will be drawn from the latter categories since in these fields it has long been recognized that dosimetry plays an essential role. Prior to the discussions of methods and applications of dosimetry, useful concepts, units, and a brief survey of the fundamentals of the interaction of neutrons and gamma rays with matter will be presented.

831. J. A. Belcher, W. A. Flynn, and R. J. Thomson, SNAP 10A Radiation Shield Analysis, NAA-SR-9647 (October 15, 1964) Availability: $0.75 CFSTI.

The final analysis of the SNAP 10A flight system radiation shield has been completed. This report summarizes the results.

Preliminary analysis of the shield design indicated that the direct penetration dose at the 5-ft diameter reference dose plane 17.5 ft from the lower face of the reactor core would meet the flight design objectives of 10^{12} nvt and 10^{7} r for one year of operation. Subsequent analysis indicated that due to scattering effects from the edges of control drums, the NaK piping around the shield, and the thermoelectric converter system, some regions of the dose plane would be exposed to doses beyond the specified limits. However, through judicious placement and hardening of radiation sensitive instruments and the somewhat localized nature of the scattering effects an increase in shield size for the SNAP 10A flight test was not warranted.

This analysis employed conventional ray-tracing theory throughout. Technique, theory, computer codes, parameters, etc. are discussed in the report. In addition to computing radiation levels behind the shield, internal heat generation was determined.

The results show that for the particular combination of tolerance (10^{7} r), separation distance (17.5 ft), and reactor power (34 kw) that specific gamma shielding was not required. LiH was selected as the fast neutron shield material since for a given neutron attenuation it provides the lightest weight shield.

832. R. E. Maerker and F. J. Muckenthaler, "Calculation and Measurement of the Fast-Neutron Differential Dose Albedo for Concrete," Nucl. Sci. Eng. 22(4), 455-462 (1965).

Monte Carlo calculations have been carried out in great detail on the reflection properties of concrete for fast neutrons. This paper presents the results for the differential angular dose albedo as a function of emerging angle for various incident beam conditions, and touches upon some of the other distributions calculated. A comparison of the angular dose albedo calculations with the results of a series of experiments at the Tower Shielding Facility shows excellent agreement. A formula is presented for the differential angular dose albedo which reproduces the Monte Carlo results to better than 10%, and which extends the results to the entire hemispherical range for both incident and reflected neutron velocity vectors.

833. M. C. Clarke, E. W. Etherington, D. S. Greenaway and W. J. Patterson, The Escape of Scattered Gamma Radiation Through a Duct in a Shield, R and D/P/122/403 (1962). Availability: Atomic Power Constructions Ltd., London.

Photon intensity spectra and total photon intensities above 75 keV have been measured at several positions outside a duct in a concrete shield behind which a 16.8 millicurie cobalt-60 source was located. Dose rates have been calculated from the measured spectra and these are compared with predictions by two approximate theoretical methods. In some cases the measured spectra are compared with spectra predicted by the two theoretical methods.

834. T. W. DeVries, Fallout Radiation Energy Distribution as a Function of Altitude, FZK-187 (July 17, 1964). Availability: $6.00, CFSTI as AD 602492.

The gamma-ray spectra in the air above a plane containing fallout fission products have been calculated as a function of time and altitude. Similar calculations were made with the spectra of cobalt-60 and cesium-137 assumed as source terms. An experiment has been designed to measure the spectra calculated for cobalt-60 utilizing eleven cobalt-60 sources and a very sensitive sodium-iodide gamma-ray spectrometer.

The results of the calculations indicate that the relative spectra are constant above 800 feet for times after fission greater than one day and, furthermore, that the variation of the dose rate with altitude is constant for this time interval.

835. N. Hokkyo and M. Kitazume, Two-Group Calculation of Capture Gamma Rays from Infinite Slab, AEC-tr-5077 (July 1962). Availability: $0.50 CFSTI. (Translated from Nippon Genshiryoku Gakkaishi, 2, 474-477 (1960)).

The capture gamma rays generated in reactor shield frequently have primary importance in determining the leakage radiation level around the reactor. This report deals with the case in which the fast and thermal neutron currents incident on a face of infinite slab are given, and gives the general formulae with which to calculate the capture gamma currents and fluxes leaking from the said face and the rear face. The result with regard to the capture gamma currents is an extension of the well-known formulae of Iliffe to two-group approximation for the thermal neutron.

836. L. Albrecht, Tables of Gamma Ray and Beta Ray Spectra of Fission Products, ORNL-tr-27 (August 1963). Availability: Oak Ridge National Laboratory, Oak Ridge, Tenn. (Translated from Kernenergie 6, 390-391 (August 1963).

A compilation of beta- and gamma-spectra of the fission products is presented in tabular form. Fission products with half-lives from 15 sec to 100 years are included. The literature is covered up to about August 1962.

837. Martin Leimdorfer, "The Backscattering of Fast Neutrons from Plane and Spherical Reflectors," Transactions of Chalmers University of Technology, Gothenburg, Sweden, 1964.

The Monte Carlo method has been used for a study of neutron backscattering from plane and concave spherical reflectors of hydrogen, water, concrete, aluminum, iron, and uranium. A calculation technique, which was initially designed to be efficient for deep penetration calculations, is applied with only a simple alteration, reversing the bias to promote the migration of neutrons back toward the source.

838. Martin Leimdorfer, "On the Use of Monte Carlo Methods for Calculating the Deep Penetration of Neutrons in Shields," Transactions of Chalmers University of Technology, Gothenburg, Sweden, 1964.

The exponential transformation, previously applied by the author to problems concerning deep penetration of photons in plane geometry, is here applied to problems of deep penetration of neutrons in spherical geometry. The representation of nuclear data is briefly described and the extension of the exponential transformation technique to spherical geometry is discussed on a quite general basis, to show the proper way of derivation in other systems of coordinates. In a sample problem, concerning the dose rate versus distance (up to 210 cm) from a point fission source in concrete, the present method is compared with the Moments Method and excellent agreement is shown.

839. J. Nilsson and R. Sandlin, The Transmission of Thermal and Fast Neutrons in Air Filled Annular Ducts through Slabs of Iron and Heavy Water, AE-166 (December 1964). Availability: Dep. (mn)

An investigation has been carried out concerning the transmission of thermal and fast neutrons in air filled annular ducts through laminated Fe-D_2O shields. Measurements have been made with annular air gaps of 0.5, 1.0, 1.5 and 2.0 cm, at a duct length of half a meter. The neutron fluxes were determined with a foil activation technique.

The thermal flux was theoretically and experimentally divided into three components, a streaming, a leakage and an albedo component. The fast flux was similarly divided into a streaming component and a "leakage" component. A calculational model to predict the components was then developed and fitted to the data obtained by experiments.

The model reported here for prediction of neutron attenuation in ducted configurations may be applied to straight annular ducts of arbitrary dimensions and material configurations but is especially designed for the problems met with in short ducts.

840. G. T. Chapman, Determination of Thermal-Neutron Flux Distributions in the Bulk Shielding Reactor II by Copper Wire Activation Techniques, ORNL-3759 (March 1965). Availability: $3.00 CFSTI.

The thermal-neutron flux distribution within the ORNL stainless steel Bulk Shielding Reactor II has been determined by exposing copper wires at selected locations in the core during reactor operation. The gamma-ray activity resulting from thermal-neutron activation of the copper was counted as a function of position along the wire, and the results were converted to estimates of thermal-neutron flux by the methods described in this report.

The method of analysis was checked by independently exposing gold and copper wires at a common location in the reactor. The thermal-neutron fluxes obtained from analysis of both sets of data were in agreement within $\sim 3\%$. The method was further checked by comparing the results of a wire exposure with calculations of the distribution made by the PDQ reactor code. Agreement was good in statistically important regions.

Several CDC-1604A programs were written to facilitate processing of the wire data, and are included as an appendix.

841. W. C. Ingold and C. M. Huddleston, An Empirical Formula for Calculating Gamma-Ray Dose Attenuation in Concrete Ducts, USNCEL-R-349 (November 25, 1964) Availability: $2.00 CFSTI.

A survey is presented of the current status of the calculation of gamma-ray dose-rate attenuation in air ducts through concrete. A simple empirical formula is exhibited which shows satisfactory agreement with the results of more complicated computational techniques and with experimental results. This simple formula represents a large saving in computation time -- 2 seconds per case compared to 400 seconds by IBM-1620 computer. Its validity is established for a wide range of duct geometries and for gamma-ray energies up to 3 MeV.

842. P. D. Holt and J.A.B. Gibson, "High-Energy Gamma-Ray Dosimetry: Experimental Measurements at a Nuclear Power Station," J. Nucl. Energy, Parts A/B, 19(1), 17-31 (January 1965).

The γ-radiation dose rate and the variation of dose-rate spectrum has been determined at Chapelcross nuclear reactor station. The dose rate was measured with survey instruments and personal dosimeters, i.e. film dosimeters and quartz-fibre electroscopes. The spectrum was determined with γ-ray spectrometers using sodium iodide (thallium-activated) crystals in conjunction with multichannel analysers and portable single-channel analysers. The measurement of the dose-rate spectrum and determination of the proportion of the radiation above 5 MeV enables the response of various instruments and dosimeters to be assessed for use around nuclear power stations.

The results of the measurements show that the unscattered fraction of high-energy radiation (> 5 MeV) was responsible for from 20 to 60 per cent of the total dose. The survey instruments using Geiger-Muller counters as detectors gave an enhanced reading at all dose rates and could be corrected directly. The film dosimeters also gave results which were 50 percent too high but the results from the AERE/RPS plastic film dosimeter could be corrected approximately by measuring the ratio of the dose rates under two filters; the resulting error was about 10 per cent.

843. W. H. Henry, C. Garrett, "Scattering of Uncollimated Cobalt-60 Gamma Radiation by Concrete and Lead Barriers," Acta Radiol., NS2(5), 353-363 (October 1964).

The percentage contribution to exposure rate, in roentgens per units time, due to cobalt 60 gamma radiation scattered by concrete and lead barriers has been measured for a point isotropic source and an ionization chamber in lines normal and parallel to the barrier surface.

844. D. W. Henricks and C. W. Hoffman, Attenuation of Neutrons by an M-48 Tank, BRL Memorandum Report No. 1443 (AD-334924), November 1962. (SECRET).

The over-all attenuation of neutrons by tanks and other armored vehicles has been measured in field tests of nuclear weapons. Controlled measurement of the directional dependence of the incident radiation attenuation provided by the vehicle was lacking.

In this study a Hurst-type fast neutron dosimeter was used to measure the neutron transmission through an M-48 tank at the driver and commander positions. Measurements were made at a 4-meter distance with a 2-curie polonium 210-beryllium neutron source placed at 15-degree azimuthal intervals around the tank and at elevation angles of 0, 15, and 30 degrees.

Comparisons are made with field test data.

845. J. M. Hammersley and D. C. Handscomb, Monte Carlo Methods, John Wiley and Sons, Inc., 1964.

This monography surveys the present state of Monte Carlo methods. Although we have dallied with certain topics that have interested us personally, we hope that our coverage of the subject is reasonably complete; at least we believe that this book and the references in it come near to exhausting the present range of the subject. On the other hand, there are many loose ends; for example we mention various ideas for variance reduction that have never been seriously applied in practice. This is inevitable, and typical of a subject that has remained in its infancy for twenty years or more. We are convinced nevertheless that Monte Carlo methods will one day reach an impressive maturity.

846. R. M. Cowdroy, "Concrete Blocks for Radiation Shielding," Atomic Energy (Australia), 7(4), 27-30 (October 1964).

Two shielded enclosures were constructed to house linear particle accelerators at the Australian A.E.C.Research Establishment, Lucas Heights. One was made of precast concrete blocks and the other was of normal concrete poured on the job. It was thus possible to make a direct comparison between the two methods.

Ilmenite sand was used to increase the density of the blocks which were used in a certain application to avoid difficulties inherent in pouring concrete. Measurements and calculations showed that the attenuation of the block wall was approximately the same as that of the poured concrete and the cost was somewhat less.

847. C. R. Frost, The Radioactivity of Thermal Neutron Fission Products of U-235, U-233, and Pu-239, AAEC/TM-252 (June 1964) Availability: Dep. (mn).

The calculated individual and total β-radioactivity of thermal fission products of U235, U233, and Pu239 are given for irradiation periods of 200, 300, 600, and 1000 days, and cooling periods from 0.1 to 100 years, neglecting neutron capture by fission products. All fission products with half lives greater than two days are considered. The calculated total fission product radioactivity data of U235 and Pu239 for 200 days and 1000 days irradiation and cooling periods to 100 years are compared with other published data.

848. D. B. Crawley and O. R. Owen, Final Shielding Report for the Dragon Reactor Experiment, DP-Report 296 (September 1964). Availability: Dep. (mc); $3.30 (ph), $2.40 (mf) CFSTI.

The Final Shielding Report for the Dragon Reactor Experiment includes the revised contents of the Preliminary Report and examines the ultimate design of the Reactor Components.

Fission product retaining fuel proposed for the core charge allows some relaxation of shielding in the area of the fission product plant.

Radiation damage to Reactor materials, and the access to maintenance areas after shutdown are considered. The Appendix contains data required by the Mercury Computer shielding programmes.

849. J. I. Marcum, Neutron Fluxes from a Vertically Collimated Source of 14-MeV Neutrons at an Air-Ground Interface, RM-4120-AEC (August 1964) Availability: Dep. (mn); $1.00 (cy), 1(mn) CFSTI.

This report gives results for the flux and dosage from a vertically collimated upward-pointing source of 14 MeV neutrons at a simulated air-ground interface. The solutions were obtained by the use of the RAND Monte Carlo Neutron Transport Code, and are presented in such a way that they can easily be compared to results previously obtained for an isotropic point source.

850. W. A. Hehs and G. E. Miller, Measured and Calculated Nuclear Radiation Distribution in Liquid Hydrogen, FZK-182 (March 27, 1964). Availability: N64-18350 CFSTI $4.60 (fs).

The nuclear radiation measurements and calculations obtained from a series of experiments designed to investigate the effects of nuclear heating in liquid hydrogen are reported. The experimental arrangement simulated the radiation source and liquid-hydrogen propellant-tank geometry of a typical nuclear rocket system. The operating capacity of the tank was approximately 125 gallons.

Nuclear radiation measurements were made in and around the liquid-hydrogen tank with the tank both empty and full of hydrogen. The radiation detectors and techniques used in making these measurements are described, and the measurements are compared with calculated nuclear radiation intensities. The calculations include secondary-gamma-ray production in the liquid hydrogen as well as in other surrounding materials. Assumptions and methods utilized in performing these calculations are discussed. The measured and calculated radiation intensities are, in general, in quite good agreement.

851. W. A. Hehs, B. O. McCauley, G. E. Miller and D. M. Wheeler, Nuclear Radiation Heating in Liquid Hydrogen, FZK-186, Vols. I and II (June 30, 1964) Availability: 186-1, N64-25009, CFSTI, $11.50 (fs), 186-2, N64-25011, CFSTI, $28.50 (fs).

A comprehensive collection of data is given for evaluating methods to calculate the nuclear-radiation energy-deposition rates and the resultant temperature distributions in liquid hydrogen in a pressurized tank.

Nuclear measurements provide data to determine heating gradients in the liquid hydrogen and to determine the contributions of neutron interactions and gamma interactions to the total heating rates. Boil-off and self-pressurization measurements determined the total heat input into the cryogenic system.

Temperature distributions in the liquid hydrogen were monitored under variable flow conditions: pressurization, flow rate, heating gradient, and heat input. No analysis of the resultant temperature distributions in the hydrogen was performed. Background information and accuracy of the data required to perform and evaluate an analysis have been provided.

The results of nuclear analyses made during the study with the C-17 shield penetration code and with the COHORT Monte Carlo procedures compare favorably with the measured nuclear-radiation distributions and energy-deposition rates.

852. K. Ishimatsu, Distribution of Gamma Radiation from Co-60 Source in Water, ORNL-tr-199 (1960). Availability: JCL $1.60 (fs), $0.80 (mf). (Translated from Nikon Genshiryoku-Gakkai-Shi 4(3), 175 (1960)).

Measurements have been made of the distribution of gamma-rays from a point source of ^{60}Co in an infinite water medium.

Energy flux spectra were measured by a scintillation spectrometer for penetration distances between 5.49 mean free path and 12.8 mean free path, while dose rates were measured by ionization chambers for distances between 0.610 mean free path and 2.75 mean free path. The results were compared with calculated results obtained by the moments method. A satisfactory agreement was obtained between dose buildup factors given by the experimental results and those given by the calculated results. On the other hand, a slight discrepancy seemed to exist between the experimental energy flux spectra and the calculated ones.

853. H. Gotting, Baryte as Radiation Shielding Materials, ORNL-tr-505 (1962). Availability: Dep. (mn), $1.60 (fs), $0.80 (mf), JCL. (Translated from Atom und Strom. 8(10),67-70 (1962).

The use of Baryte ore as an aggregate for use in concrete radiation shields is discussed. The space saved compared to ordinary concrete (about 40%) is almost as great as for magnetite concrete and costs considerably less.

854. A. Honig, Dose Build-up Factors for Concrete, ORNL-tr-502 (1963). Availability: JCL, $1.60 (fs), $0.80 (mf). (Translated from Kerntechnik, 6, 393-399 (September 1963)).

The use of buildup factors in estimating shield thickness is reviewed and detailed tables of the buildup factor for ordinary concrete (0-300 cm) are presented. The Taylor fit coefficients are also given. The calculation of shield thicknesses with the use of a special shielding slide rule is demonstrated.

855. W. Heine, K. Sagel, and U. Zwicker, Lead and Lead Alloys as Shielding Materials, ORNL-tr-504 (1962). Availability: Not available except through Scripta Technica, 275 Madison Avenue, New York, New York, 10016. (Translated from Kerntechnik, 4(4), 150-155 (April 1962)).

Materials for radiation shielding mainly must satisfy physical, mechanical and corrosion-chemical requirements. Due to its high density and stability with respect to radiation influences, lead represents an effective shielding material against gamma- and x-rays. By means of additions of calcium and barium, it is possible to manufacture high-strength lead alloys which are suitable for the construction of brick walls consisting of lead shielding blocks due to their high fatigue strength. On account of their high corrosion resistance, a protective coating can be dispensed with in most cases.

856. Boris Rafailovic Bergelson, Reactor Neutron Shield Without Hydrogen, ORNL-tr-554 (1964). Availability: $1.60 (fs), $0.80 (mf), JCL. (Translated from Jaderna Energie 10, 281-287 (August 1964)).

The report deals with non-hydrogen biological shields for nuclear reactors. It contains an analysis of the concept, selection of materials, optimization of the composition of shield layers of this type, and the spatial and energetic distribution of neutrons in the shield. It studies structural and other aspects in the application of boral, graphite, and iron in technical and biological reactor shields from the viewpoint of their effectiveness.

857. F. F. Haywood, "Spatial Dose Distribution in Air-Over-Ground Geometry," Health Physics, 11(3), 185-92 (March 1965).

Spatial distributions of gamma-ray exposure and neutron dose were measured using a point 60-Co (800 c) and the ORNL Health Physics Research Reactor as a small fission source. The sources were supported on a 1527-ft tower, at the Nevada Test Site, at heights ranging from 27 to 1500 ft.

Neutron dose rate was measured (independently of the gamma-ray exposure rate) with a recoil proton proportional counter, and the gamma-ray exposure rate was measured (independently of the neutron dose rate) with a halogen-quenched geiger tube which was surrounded by a filter (tin and lead) such that the response was proportional to exposure in roentgens over a broad range of photon energies.

Measurements were made in a plane 4500 ft long and 100 ft high which extended eastward from the base of the tower. At large source-detector separations, a significant decrease in intensity was noticed at a detector height of 2 ft, relative to the intensity at ground level, for both neutrons and gamma-rays.

For a given slant range greater than about one relaxation length, there is an increase in neutron dose rate at the interface by a factor of 2.4 as the reactor is raised from 27 to 1500 ft above the ground; and, similarly, there is an increase in gamma-ray exposure rate by a factor of 2.1. These increases appear fixed for source-detector separations greater than about 400 yards.

An effect of the presence of the interface was observed by comparing experimentally determined values of the exposure buildup factor for 60-Co gamma rays to calculated buildup factors in an infinite medium. Boundary correction factors K(exp) were determined experimentally. K(exp) is defined as the ratio of dose in the finite air medium to dose in the infinite air medium at the same slant range.

The dependence of neutron dose rate on variation of detector heights and on variation of source heights is compared to theoretical models with satisfactory agreement.

858. L. C. Woods, Introduction to Neutron Distribution Theory, Methuen & Co., Ltd., London, and John Wiley & Sons, Inc., New York, 1964.

This brief account of neutron distribution theory is intended for applied mathematicians, engineering scientists and physicists who want an introductory survey, at a reasonably advanced level, of the basic equations of the subject, together with some of their more important applications to reactor theory. The book may also be useful to those nuclear engineers who wish to study reactor theory beyond the usual initial level imposed by elementary diffusion theory. The treatment should be well within the capacity of final-year honours classes in physics and mathematics.

While the emphasis is on the systematic derivation of the equations, starting from Boltzmann's transport equation, many of the more straightforward applications to problems in reactor physics have been included to illustrate these equations. There has not been space enough in the book to include any account of the applications of the theory to reactor kinetics or to heterogeneous reactors, but in any case such topics properly belong to the derived subject of reactor analysis.

859. G. D. Joanou and C. A. Stevens, Neutron Cross-Sections for Aluminum, NASA CR-54260 (GA-5884), (November 13, 1964) Availability: N65-15054 CFSTI $3.00 (fs),(mf) $0.75.

A set of neutron cross sections has been prepared for aluminum. These data are based on existing experimental data whenever possible. Gaps in the existing data have been filled by theoretical calculations and in some cases by "educated" guesses.

The compiled and evaluated data have been incorporated into the GAM-II slowing down program and into the GATHER-II thermalization program. The literature survey associated with this report is believed to be complete to November 1964.

860. G. D. Joanou and C. A. Stevens, Neutron Cross Section for Tungsten Isotopes Topical Report, NASA-CR-54261 (GA-5885) Novmeber 13, 1964. Availability: N65-15960 CFSTI $2.00 (fs) $0.50 (mf).

A set of cross sections for the isotopes of tungsten, as well as for the natural element, has been obtained. The cross sections are based upon experimental data wherever possible, complemented with the results of nuclear model calculations. In some cases, where neither measurement nor a suitable calculation was available, it was necessary to make an "educated" guess. The results have been incorporated into the library tapes of the GAM-II and GATHER-II computer programs.

861. G. D. Joanou and C. A. Stevens, Neutron Cross Section for Beryllium, NASA-CR-54262 (GA-5905) November 13, 1964. Availability: N65-16202 available without charge.

A revised set of neutron cross sections has been prepared for beryllium. These data are based upon existing experimental data wherever possible.

The compiled and evaluated data have been incorporated into the GAM-II slowing-down program and into the GATHER-II thermalization program. The literature survey associated with this report is believed to be complete to November 1964.

862. G. D. Joanou and M. K. Drake, Neutron Cross Sections for U^{235}, NASA-CR-54263 (GA-5944) December 10, 1964. Availability: $4.00 CFSTI.

A revised set of neutron cross sections has been prepared for U-235. These data are based upon experimental data whenever possible.

The compiled and evaluated data have been incorporated into the GAM-II slowing down program and into the GATHER-II thermalization program. The literature survey associated with this report is believed to be complete to November 1964.

863. T. E. Northrup, High-Density Concrete for Gamma and Neutron Attenuation, ORNL-3704 (March 1965) Availability: CFSTI $3.00.

The special aggregates needed to produce concrete shields capable of attenuating high and almost equal intensities of gamma and neutron radiation at a temperature of 100°C are described in this report. Material acquisition and concrete test information on ferrophosphorus-limonite concrete and hydrous-iron-ore-limonite concrete is presented. Special appendices cover high-density aggregate mining, iron-pyrite impurities in hydrous iron ore, and shielding aggregate data tables. The basic approach to high-density concrete mix design is reviewed to show that theoretical calculations for concrete mixes do give results that are reliable. However, laboratory tests with produced shielding aggregates are still required to confirm mix quantities, adjust workability, establish the need for additives, etc. before any concrete is actually placed.

864. D. K. Trubey, A Calculation of the Neutron Dose in Ordinary Concrete Due to Air-Scattered Neutrons from a Point Fission Source, ORNL-TM-973 (March 15, 1965). Availability: Oak Ridge National Laboratory, Oak Ridge, Tennessee.

A calculation of the fast-neutron dose in concrete due to a point fission source located in air at a distance of 100 g/cm^2 has been performed. The moments method code RENUPAK was used successively to produce the energy spectrum in air from the point source and the energy spectrum in concrete from the air-scattered source. Angular distribution conversion factors and data to convert to other amounts of air attenuation are given.

865. Keran O'Brien and James E. McLaughlin, The Boundary Effect on Gamma-Ray Transport from a Point Source, HASL-157 (January 1965). Availability: $2.00 CFSTI.

The density interface effect of γ-ray intensities resulting from a point source has been measured at 0.66 and 1.25 MeV. Source-detector separations from about 10 to 900 meters were employed. A phenomenological representation is suggested of this effect, based on a 2-dimensional diffusion equation, which reasonable describes the behavior both of our results and those of others.

866. M. A. Capo, R. N. Nassano, L. O. Ricks, L. D. Stephenson and H. C. Woodsum, Reactor Analysis of NRX-A-Radiation Analysis, WANL TNR-128, Vol. II. (September 1963) (Confidential).

To standardize radiation calculations a procedure designated the Shielding Standard Design Method was established. The Shielding Standard Design Method utilizes the two-dimensional diffusion code, CURE, and the one-dimensional diffusion code, P1MG to calculate few group neutron fluxes in two dimensions and multigroup neutron fluxes in one dimension, respectively. The P1MG fluxes are used in the HIC code to calculate secondary gamma spectra and neutron kinetic heating rates in one dimension. The HIC gamma spectra and the CURE two-dimensional fluxes are used in the NORMALIZER routine to obtain two-dimensional capture and inelastic gamma source distributions and secondary gamma ray source spectra. Gamma ray heating rate data are generated by Monte Carlo Program 18-0 and point kernel Program 14-0. This point kernel code is also used to obtain gamma dose rates and neutron spectra and dose rates outside the reactor.

The Shielding Standard Design Method was used to determine the reactor axial and radial power distributions, reactor core gamma ray source energy spectra, neutron spectra, and detailed spectrum and distribution of secondary gamma rays in region outside the reactor core. These data were used to generate radiation heating rates in all NRX-A components within the pressure vessel for use in thermal analysis.

Radiation environment data outside the pressure vessel are presented in graphical and tabular form. Included are fast neutron fluxes and dose rates, and gamma ray dose rates due to core gamma ray sources and secondary gamma ray sources outside the core. These data are presented for both the unperturbed environment and the environment perturbed by the test car privy roof shield. Radiation levels in the Test Cell neutronics tank are included. The above data will be used for comparison with NRX-A experimental data.

867. C. W. Watson and D. M. Peterson, Radiation Heating Rate Estimates for a Phoebus Reactor (U), LA-3158-MS, December 16, 1964, (Confidential).

Results of radiation heating rate calculations are presented for an operating Phoebus reactor. Also presented, in appendices, are computational details, auxiliary results (e.g., radiation dose rate estimates), and comparisons of the present estimates with scaled data from previous (Kiwi) radiation environmental studies.

Although the estimates presented here apply specifically to the reactor system as described, and will require revision as the Phoebus concept evolves, the heating rates given should be adequate for preliminary design purposes and, with some modification, should be applicable to a wide class of Phoebus reactors.

868. D. Spielberg, Design and Analysis of a Crew-Compartment Shield for a Radiation-Resistant Combat Vehicle, NDA-2117-1, October 15, 1959 (SECRET).

This is the final report describing the design and analysis of a crew-compartment shield intended to provide personnel with maximum radiological protection from a nuclear detonation, consistent with specified weight and volume limitations.

The neutron and gamma ray penetrations and the secondary gamma production in this shield have been determined using Monte Carlo techniques throughout. Considerable study and effort were expended to ensure that these calculations were based on the best available cross sections. Because of this, and because of the particular attention which was paid to establishing the proper energy and angular distribution of the radiation incident on the outer shield surface, we believe that the methods herein described represent a definite advance in the state of the art, and that the complex-geometry multilayer shield itself has been studied as accurately as it is now possible to do.

869. J. M. Norwood and R. E. Beissner, Formulation of Radiation Transport Solutions for Shielding Applications, Including Integral Transform Methods, Matrix Techniques, and Asymptotic Approximations, AFWL-TR-64-151, March 1965. Availability: AD-614250 $3.00 (fs) $0.75 (mf), CFSTI.

The objective of this study is to develop new and improved computational models for the determination of radiation intensity behind bulk shields and to provide further insight into radiation transport phenomena. A study of integral transform methods as applied to the neutron transport equation in finite geometrics under the spherical harmonics formulation is presented, representative problems are treated in slab, spherical, and cylindrical geometries. A matrix-integral-operator solution in slab geometry is included. The results of a study of an asymptotic method of obtaining angular distributions behind bulk shields are presented and the problem of electron transport in 1- to 5-MeV range is discussed.

870. O. A. Barsukov and V. S. Avzyanov, "Metal-Water Shielding for Point Neutron Sources," J. Nuclear Energy, Parts A/B, 19(2), 114-124 (February 1965).

The article is devoted to a study of neutron migration processes in metal-water shielding. Calculations are made of the spatial and energy distribution of neutrons in iron, water and multilayer metal and water shielding for a point Po-Be source. The twenty-group diffusion transport approximations used are checked experimentally. The work has made it possible to determine a number of particular features in neutron distribution in the aforesaid shielding.

871. A. Tsuruo, The Spectrum Matrix Method in γ-Ray Transmission Problems for Multiple Layers, NP-tr-1217 (July 1964). Availability: Dep. (mn); $1.60 (fs), $0.80 (mf) JCL. (Translated from Nippon Genshiryoku Gakkaishi, 6, 91-97 (1964)).

An approximate calculation method in γ-ray transmission problems is shown for multiple layers.

In this method, the differential energy flux of incident γ-rays is expressed by a vector, and the shielding property of each slab is represented by a matrix. The matrix is constructed by a relatively simple procedure, using the data of the effect of finite slabs calculated by the Monte Carlo method. The compound shielding property of multiple layers is also represented by the matrix which is obtained by the multiplication of the matrix corresponding to each slab. The differential energy flux of transmitted γ-rays is given by multiplying the incident vector by the matrix.

The results of the method are compared with those of the Monte Carlo calculation for Pb-H_2O and H_2O-Pb stratified slabs, and also with those of experiments for Pb, Fe and Al stratified slabs. The comparison shows that this method is useful for all combinations of materials for high incident energy, and when low energy γ-rays enter the heavy layer followed by a light or heavy layer.

872. R. L. Rexroad, and M. A. Schmoke, Scattered Radiation and Free Field Dose Rates from Distributed Cobalt-60 and Cesium-137 Sources, NDL-TR-2 (September 1960) Availability: AD-272959 CFSTI $6.00 (fs).

The purpose of this work was to verify theoretical calculations that will predict the radiation level above a uniformly contaminated plane of gamma radiation and serve as the open-field data for later shielding experiments.

Two sources, cobalt-60 and cesium-137, were used to simulate the uniformly contaminated plane. The dose rates at various heights above the source plane were measured with ionization-chamber dosimeters. Analytical expressions of the buildup factor to account for the increase in dose caused by scattered radiation were developed, and experimental results compared with theory.

The conclusions established were:

Buildup factors to account for ground and air-scattered radiation at a height, h, above an annular source of slant radius, r, may be expressed by

$B(h,r) = a + b\,\mu r + c_o e^{-c_1 \mu r}$, as follows:

Co-60

$$B(h = 1\text{ ft}) = 1.09 + 0.549\mu r + 0.189e^{-71.1\mu r}$$

$$B(h = 3\text{ ft}) = 1.11 + 0.529\mu r + 0.157e^{-86.1\mu r}$$

$$B(h = 6\text{ ft}) = 1.13 + 0.529\mu r + 0.150e^{-86.2\mu r}$$

Cs-137

$$B(h = 1\text{ ft}) = 1.03 + 0.629\mu r + 0.164e^{-18.3\mu r}$$

$$B(h = 3\text{ ft}) = 1.04 + 0.681\mu r + 0.158e^{-13.1\mu r}$$

$$B(h = 6\text{ ft}) = 1.07 + 0.677\mu r + 0.118e^{-10.7\mu r}$$

For radii greater than approximately 70 feet, experimental results agreed within ± 5% with theoretical calculations of the gamma dose rate above uniformly contaminated planes of cobalt-60 and cesium-137.

873. Office of Civil Defense, Design and Review of Structures for Protection from Fallout Gamma Radiation, PM-100-1 (February 1965) Availability: (Army) Defense Department, Washington, D. C.,

This manual is written for the use of Architects and Engineers when engaged in the design or review of structures for protection from fallout gamma radiation. It can be used to design protective features in new structures or to analyze existing structures for their protective capabilities when considering modifications. It assumes the user possesses a basic knowledge of modern physics, nuclear weapons effects, and engineering practice.

874. C. Eisenhauer, An Engineering Method for Calculating Protection Afforded by Structures Against Fallout Radiation, PM-100-1 (Supplement No. 1) (NBS Monograph 76) (January 1964) Availability: $0.20, GPO.

This report is a discussion of the technical assumptions underlying the methods currently recommended by the Office of Civil Defense (OCD) for calculating protection afforded by structures against fallout radiation. It discusses methods for calculating the contributions from radioactive sources on the roof and on the ground surrounding a simple one-storied building. It shows in detail how each technical chart in the OCD Professional Manual is derived from basic data on radiation penetration developed by Dr. L. V. Spencer. Charts from the Professional Manual and relevant curves from Spencer's work are included in this report in order to make it self-contained.

875. E. Aalto, "Comparisons of Measured and Calculated Neutron Fluxes in Laminated Iron and Heavy Water," Nucl. Sci. Eng. 22(1), 33-39 (May 1965).

Measurements of neutron fluxes have been performed in configurations depicting the regions extending radially and axially outwards from the core of a PHWR Reactor, in order to test the accuracy of the available methods in shield design on thin alternating laminae of Fe and D_2O.

The results are compared to fluxes calculated by the British 18-group removal-diffusion method and by the NRN method developed at AB Atomenergi. The results show that the values predicted may be expected to be within a factor of 2 from the true values in most cases. The predicted relative flux distributions follow the observed ones with fairly good accuracy in spite of the apparent misuse of diffusion theory for the thin regions in question.

It is also shown that the predicted change in the fast spectrum while penetrating these setups should be confirmable with certain threshold detectors.

876. J. V. Walker, "The Effects of Flux Anisotropy on Thermal-Neutron Flux Perturbation," Nucl. Sci. Eng. 22(1), 94-101 (May 1965).

The effects of neutron flux anisotropy upon thermal-neutron flux perturbations were found by experimentally determining the perturbations induced by foils placed in fluxes of known nonuniformities. Anisotropies in the magnitude of the vector flux were introduced by placing a 'black' cadmium absorber sheet in an isotropic flux produced by a uniform slowing-down source in water. The resulting angular and spatial distributions of the thermal-neutron flux were computed by using Yvon's method to solve the Boltzmann equation for an absorbing half space containing a uniform slowing-down source. Using indium foils with thicknesses from 14 mg/cm^2 to 184.9 mg/cm^2, the relative flux perturbations were measured to within ± 1% in regions which varied from a highly directional flux at the cadmium surface to essentially an isotropic distribution several mean free paths from the absorber. The experimental data indicated that the flux perturbations remained constant at all distances greater than about three mean free paths from the non-reentrant boundary, but that the flux depression decreased in the region near and vanished at the boundary. It was concluded, contrary to earlier predictions, that flux perturbation theories, based upon isotropic flux models, cannot be used to correct flux perturbations induced in all anisotropic fluxes. It was pointed out that, if this effect is overlooked, significant errors may exist in some relative or absolute flux measurements made in the region near a boundary or absorber.

877. IAEA Shielding Panel, Reactor Shielding, IAEA Technical Reports Series No. 34 (March 1964) Availability: National Agency for International Publications, Inc., New York, $3.50.

A panel of experts was convened from 9 to 13 March 1964 in Vienna at the Headquarters of the International Atomic Energy Agency and under the chairmanship of Mr. E. P. Blizard to discuss the present status of reactor shielding. The participants were prominent shielding experts from most of the laboratories engaged in this field throughout the world. They presented status reports describing the past history and plans for further development of reactor shielding in their countries and much valuable discussion took place on some of the most relevant aspects of reactor shielding. All this material is presented in this report, together with abstracts of the supporting papers read to the Panel.

878. J. D. Marshall, Analysis of Radiation Streaming Through Single-Bend Ducts, AFWL-TR-64-136 (January 1965) Availability: CFSTI $4.00 (fs) (mf) $0.75.

An experimental study has been performed to obtain data for verification of the Monte Carlo Multibend-Duct Procedure and to obtain data for a parametric study of radiation transported through a ducted shield. To accomplish these objectives, a 14-Mev neutron source and Co-60 gamma-ray sources have been used in conjunction with single-bend aluminum ducts in water shields. The duct bends were varied from 30° to 60° and the top-leg-to-bottom-leg ratio from 0.3 to 3.0. The calculation procedure has been verified within certain limitations, which were mainly a function of shield thickness. Experimental results were integrated over the exit plane of the ducted shields. The scattered component of the dose rate was obtained by subtraction of the calculated direct-beam component from the integrated experimental dose rates. The scattered component was 70 to 90 percent of the total radiation dose and increased with increasing shield thickness. No other general tendencies were noted over the range of the parameters considered.

879. M. A. Capo, Radiation Environment Around the Nerva Engine with NRX-A Reactor, WANL-TNR-058, June 1, 1962 (Confidential).

The purpose of this report is to present an analysis of the radiation environment around the NERVA engine with an NRX-A Reactor. Shielding computer program 14-0 was used to calculate the gamma ray energy flux and dose rate, and fast neutron (0.01 Mev ≤ E) flux and dose rate at fifteen different azimuthal angle positions around the NRX-A it distances of 5, 10, 15, 20, and 30 feet from the center of the reactor core. A one-dimensional multigroup P-1 code, "PiMG", was used to obtain the thermal neutron and fast neutron (E ≤ 0.01 Mev) flux along the core center line axially at the top and bottom, and radially on the line of symmetry. These data are presented in both tabular and graphical form.

880. H. Gronroos, Energy Dependent Removal Cross-Sections in Fast Neutron Shielding Theory, AE-184 (May 1965). Availability: Dep. (mn), Aktiebolaget Atomenergi, Stockholm (Sweden).

The analytical approximations behind the energy dependent removal cross-section concept of Spinney is investigated and its predictions compared with exact values calculated by Case's singular integral method. The exact values are obtained in plane infinite geometry for the two absorption ratios $\Sigma_a/\Sigma_t = 0.1$ and $\Sigma_a/\Sigma_t = 0.7$ over a range of 20 mfp and for varying degrees of forward anisotrophy in the elastic scattering. The latter is characterized by choosing a suitable general scattering function.

It is shown that Spinney's original definition follows if Grosjean's formalism, i.e. the matching of moments, is applied. The prediction of the neutron flux is remarkably accurate, and mostly within 50% for the spatial range and cases investigated. A definition of the removal cross-sections based on matching the exact asymptotic solution to the exponential part of the approximate solution is found to give less accurate flux values than Spinney's model. A third way to define a removal cross-section independent of the spatial coordinates is the variational method. The possible uses of this technique is briefly commented upon.

881. V. N. Abaev, G. A. Vasil'ev, A. P. Veselkin, Yu. A. Egorov, Yu. V. Orlov, and Yu. V. Pankrat'ev, "Attenuation of Pile Neutron Flux in Polyethylene," Soviet J. At. Energy (English Transl.), 15(1), 671-674 (July 1963).

The relaxation lengths of research reactor neutrons were measured in polyethylene. The findings are in excellent accord with data computed theoretically using the method of moments. The relaxation lengths are ~ 15% less than in the case of water.

882. V. N. Avaev, G. A. Vasil'ev, A. P. Veselkin, Yu. A. Egorov, Yu. V. Orlov, and Yu. V. Pankrat'ev, "Spectra of Fast Pile Neutrons in Passage through Polyethylene," Soviet J. At. Energy (English Transl.), 15(1), 675-677 (July 1965).

A single-transducer fast-neutron spectrometer was used to measure the spectra of fast pile neutrons after passing through polyethylene layers of different thicknesses. The results of measurements at $E_n > 3$ MeV show excellent agreement with data computed by the method of moments. At $E_n < 3$ MeV, the discrepancies are accounted for by the difference in the geometry and in the original spectra. Problems involving the correct procedures to be followed in setting up experiments of this type are discussed.

883. A. P. Veselkin, Yu. A. Egorov, Yu. V. Orlov, and Yu. V. Pankrat'ev, "Spectra of Fast Reactor Neutrons After Passing through Graphite, Lead, and Iron," Soviet J. At. Energy (English Transl.), 16(1), 34-40 (January 1964).

By means of a fast neutron scintillation spectrometer with one hydrogen-containing detector, the spectra of fast reactor neutrons after passing through various thicknesses of lead, graphite, and iron were measured in the range 0.7-11 MeV. The measurements were carried out in a water-moderated water-cooled experimental reactor in "barrier" geometry. The results of the experiments enabled us to determine the deformation of the neutron spectrum in relation to the penetration through the layers of material, and to calculate the relaxation lengths and the removal cross sections. These quantities were published earlier for fission spectrum neutrons in the energy range 0.7-3 MeV.

884. D. L. Broder, L. N. Zaitsev, B. S. Sychev, and A. M. Tugolukov, "Effect of Water Content in Concrete on Thickness and Cost of Reactor Shielding," Soviet J. At. Energy (English Transl.), 16(1), 26-33 (January 1964).

Results of biological shielding calculations for various types of concrete are compared in this paper. A graphite-moderated and graphite-reflector reactor is considered. The thicknesses and costs of biological shielding using concretes of different types and densities is analyzed as a function of the water content in the concretes. The goal of the authors is to determine, on the basis of the 46 most commonly used types of concrete in reactor construction throughout the USSR, the economic feasibility of increasing the amount of water contained in concrete.

885. P. LaFore, Some Experiments Concerning the Shielding of Gas-Cooled Graphite Reactors and Comparison with the Calculation, ORNL-tr-525 (1965). Availability: Dep.(mn), $1.10(fs), $0.80(mf), JCL. (Translated from Nucl.Struc.Eng. 1, 127-131 (1965).

Experiments designed to yield information on lateral shielding and in particular on the sources of capture-gamma rays in graphite reactors were carried out in NAIADE I, which used as a source a natural-uranium plate emitting 4×10^7 $n/cm^{-2}s^{-1}$. The experimental results obtained in different graphite, iron and concrete mock-ups were used to work out methods of calculation applicable to reactors. These methods were verified by means of measurements made in the lateral shielding of the EDF-1 reactor.

886. Rudolph Sher and Joan Felberbaum, Least Squares Analysis of the 2200 m/sec Parameters of U-233, U-235, and Pu-239, BNL-918 (March 1965). Availability: $1.00, CFSTI.

The "best" values of the 2200 m/sec cross sections of 233-U, 235-U, and 239-Pu published in 1962 (BNL-722) are revised on the basis of 22 additional measurements. For descriptions of the techniques used in the analysis, the reader is referred to BNL-722.

887. D. K. Trubey and M. B. Emmett, A Comparison of First- and Last-Flight Expectation Values Used in an 05R Monte Carlo Calculation of Neutron Distributions in Water, ORNL-RSIC-3 (May 1965). Availability: Dep.; Free (ORNL).

The distributions in water of fast neutrons from isotropic fission and 14-MeV point sources have been calculated with the 05R Code. Two methods of scoring the same case histories were used to estimate the energy distribution and the dose rate. The estimators employed were the "first-" and "last-"flight expectation, or statistical estimation, values. The first-flight estimator was found to be somewhat better due to the importance of the first flight in the total transport of the neutrons. The results are in good agreement with RENUPAK moments method results and with experimental results.

888. D. K. Trubey and M. B. Emmett, Some Calculations of the Fast-Neutron Distribution in Ordinary Concrete from Point and Plane Isotropic Fission Sources, ORNL-RSIC-4 (June 1965). Availability: Dep.; Free (ORNL).

Fast-neutron dose rates and energy distributions due to point and plane isotropic fission sources in an infinite medium of ordinary concrete have been calculated by the moments-method code RENUPAK. As a check on the method, one Monte Carlo calculation was performed which yielded results that are in good agreement with those for the corresponding RENUPAK case. The effects of varying the concrete composition, especially the hydrogen content, and the lower energy cutoff were examined for several cases and found to be significant. Detailed tables of the energy spectra are given at lethargy steps of 0.1 from 0.001 to 18 MeV.

889. D. K. Trubey, Tabulated Values of Scattered Gamma-Ray Fluxes in Iron Interpolated from Moments-Method Calculations, ORNL-RSIC-7 (May 1965). Availability: Dep.; Free (ORNL).

Tables of scattered gamma-ray fluxes in iron that are suitable for integration over source spectra have been generated on the IBM-7090 computer by Lagrangian interpolation of Goldstein-Wilkins moments-method results for point isotropic gamma-ray sources. Values of the fluxes were obtained for a consistent set of scattered gamma-ray energies taken in steps of 0.1 MeV from 0.2 to 1.0 Mev and in steps of 0.25 MeV from 0.25 to 10 MeV. One table was obtained for each scattered energy for various values of the source energy and of the distance from the source in mean free paths. Tables were then generated in which the distance from the source was converted from mean free paths to centimeters. The latter set may be integrated over any source spectrum to obtain the flux of scattered gamma rays in iron and, with small error, in any material in which the cross section is similar.

890. H. C. Claiborne, Survey of Methods for Calculating Gamma-Ray Heating, ORNL-RSIC-8 (June 1965). Availability: Radiation Shielding Information Center, ORNL - Free.

The engineering design of nuclear reactors is largely governed by the heat liberated in the fission process and the subsequent absorption of the associated radiation. External to the fuel region, the main contribution to the heat generation rate usually results from energy degradation and absorption of gamma photons escaping from the fuel region. For some design conditions, gamma radiation arising from neutron capture can contribute significantly to the heat generation rate.

This survey is concerned only with heat generation due to gamma rays, regardless of the source, interacting with the usual reactor materials. The purpose is to examine the various methods used in calculating gamma-ray heating for reactor design purposes. Comparisons of experiments with calculations are made where possible and recommendation are made as to the applicability of each method to reactor design.

The methods discussed that are adaptable to hand calculations include the buildup factor method and the straightahead-scattering method. Other methods require a high-speed digital computer. These include the differential energy spectrum method and the same methods used in neutron attenuation, namely, Monte Carlo, direct integration of the Boltzmann equation, moments method, S_n and P_n approximations to the Boltzmann equation, and diffusion theory. A listing of the available machine codes is made for each method.

891. D. R. Doty, An Absolute Measurement of Thermal Neutron Albedo for Several Materials, USNCEL-R-377 (May 1965). Availability: $2.00, CFSTI.

Albedo is a concept which has proved useful in the calculation of the penetration of nuclear weapons radiation through a shelter entranceway. The application of the albedo approach to describe the behavior of neutrons impinging upon various materials is the subject of this report.

Values of thermal neutron albedo for lead, iron, paraffin, aluminum, carbon, and high-density concrete were measured, using the nuclear reactor of the University of California at Los Angeles. The results are in reasonable agreement with theoretical calculations. Verification of the angular dependence of neutron scattering was attempted, but with only partial success.

Albedo as a function of thickness of the scattering material was measured for paraffin, carbon, and aluminum. The results essentially conform with theory.

The possibility of using a simple empirical formula for albedo calculations of thermal-neutron streaming through shelter ducts is apparent.

892. Z. G. Burson and R. L. Summers, Barrier Attenuation of Air-Scattered Gamma Radiation, CEX-63.3 (December 1964). Availability: Dep.; $1.00 (cy), 2(mn), CFSTI.

An experimental study was conducted to determine the attenuation provided by vertical and horizontal barriers exposed only to skyshine radiation from cobalt-60 and cesium-137 sources. Materials of steel, aluminum, concrete, and wood were used as barriers.

A sealed radioactive source was pumped at a uniform speed through a long length of flexible tubing to simulate a ring source. Point-source measurements were made by stopping the source at a given location. Dose measurements were made with ionization chamber detectors.

Measurements of the attenuation provided by a vertical barrier were made in a bunker where three sides and the top were of sandbags or lead; the fourth side was the barrier of interest for each experiment. The simulated ring source was set at a 100-ft radius. Just inside the 100-ft radius line surrounding the bunker, the ground was shaped to rise up a slight ramp to create a circular-shaped plateau. The angle of the ramp was designed to insure that direct radiation would pass between 1 and 2 ft above the top of the barriers-only air-scattered radiation reached the barriers.

Measurements of the attenuation provided by a horizontal barrier were made using detectors placed in a round hole in the ground, topped by the barrier material at ground level. A cobalt-60 point source was placed 100 ft from the hole. Sandbags or lead bricks were placed on the ground about 6 ft in front of the hole. The height of the lead bricks was the same height as the top of the shield. This assured that the shield was exposed only to air-scattered radiation.

Additional measurements at this same location were made to determine: (1) the skyshine dose rate as a function of distance from a cobalt-60 source placed on an air-ground interface; (2) the geometry air-ground interface from a point isotropic cobalt-60 source on the ground 100 ft away; and (3) lip scatter and wall backscatter corrections for a detector in an open hole exposed to skyshine radiation from cobalt-60.

The results apply to a variety of shielding problems. One practical application is the shielding provided by basement roofs and exposed basement walls from skyshine radiation originating from fallout.

893. R. M. Lessler and F. W. Guy, Gamma Dose Rates and Integrated Doses from Neutron-Induced Residual Radioactivity in Soil, UCRL-12339, Vols. I, II (unclassified), and III (classified), (1965). Availability: Vol. I, $3.00 (cy), Vol. II, $6.00 (cy), CFSTI.

Gamma dose rates and integrated doses above the soil resulting from neutron activation of the soil by a nuclear explosion have been calculated for various times after burst for several different initial neutron energies, heights of burst, soils, and atmospheric conditions. Methods of calculation are described and results presented. Examples are given of extrapolation of the results to situations not specifically covered in the data.

894. J. Adamson, A. M. Judd, and N. I. McNair, Neutron Flux in D.F.R. Rotating Shield. Measurements and Calculations, TRG Report-873 (October 14, 1964). Availability: Librarian, UKAEA, Risley, Warrington, Lancashire.

Neutron-induced reaction rates were measured through the inner rotating shield with the object of testing the validity of using standard bulk shielding calculation methods on D. F. R. It was found that the measurements did not provide an adequate test because streaming through control rod follower holes and other voids in the shield vitiates its bulk shielding properties. However, when the calculations of bulk attenuation and streaming are combined, and when allowance is made for the difficulty of applying the calculation methods to the awkward geometry of the system, the agreement with the measurements is acceptable.

895. Richard M. Lessler and Frank W. Guy, Neutron-Induced Activity in Earth and Sea Water from Buried and Surface Neutron Sources, UCRL-12407 (April 9, 1965). Availability: Dep. (mn), $3.00 (cy), 2 (mn), CFSTI.

This report presents results of computer calculations of activation of NTS soil, granite, and seawater by buried neutron sources of various neutron energy. Also presented are calculations of spatial distribution of neutron captures in the soil from a surface neutron source. The contribution to fallout from soil activation may be calculated from these data.

896. A. B. Chilton, "Two-Parameter Formula for Point-Source Buildup Factors," Nucleonics, 23(8), 119-122 (August 1965).

Parameters are given for the 2-parameter Berger buildup factor fitting function for all the materials and energies considered by Goldstein-Wilkins. A comparison with other empirical functions indicates that the Berger function is as accurate as any in general use and within the limits of accuracy of the data to which it was fitted. Data beyond 10 mfp were not considered.

897. Jean-Marie Paratte and Akbar Etemad, Neutron Flux Measurements in Bent Air Ducts through Water, EIR-BERICHT NR-78 (March 1965). Availability: Dep. (mn).

Experiments on neutron penetration in bent ducts through water, performed at the swimming pool reactor SAPHIR, are reported. These experiments consist of neutron flux measurements along the axis of the ducts by means of a thermal detector (dy) and three threshold detectors (Rh, In and P). The experimental facility and the detector activity measurement techniques are described. Results concerning ducts with two different diameters and two different bend-angles are presented. Some comments on these results are given.

898. Jean Kappes and John H. McNeilly, The Effect of Neutrons with Energies Above 6 MeV on Fission Foil Data, NDL-tr-46 (AD 611265) (October 1964). Availability: $1.00 (fs), $0.50 (mf), CFSTI.

This report describes a calculational correction to neutron-flux measurements that were made using fission foil detectors exposed to a radiation environment in which approximately 7.5 percent of all neutrons had an energy greater than 6 MeV.

This investigation has shown that in many cases it will be necessary to correct the fission foil results because of the increased cross section at high neutron energies. In the particular example presented, where 7.5 percent of the total number of neutrons had energies greater than 6 MeV, the correction for U-238 ranged between 21.9 and 27.3 percent and was marginal for Np-237 (7.2 to 10.6 percent). Although corrections existed for the remaining foils, they were all within the quoted experimental error of ± 10 percent for the threshold detector system.

It is expected that significant corrections will have to be applied to most foil data when they are exposed to neutron spectra contained larger percentages of neutrons with energies greater than 6 MeV.

This investigation describes an approach to the solution of the problem posed by high energy neutrons on threshold detector results, and emphasizes the fact that the problem must be considered in all future applications of this measurement technique.

899. Wilma Sonia Hehl, Calculation of the Neutron and Gamma-Ray Distribution in the EBWR Radial Shield at 100 MW Operation, IEA-74 (November 1964). Availability: Dep. (mn).

The present study refers to the radial shield of the EBWR on an assumed operational level of 100 MW.

The equal-volume sphere core, with a radius of 73 cm, was chosen for the study as indicated by calculations for the best core geometry. Shielding in the radial direction, as made up of water, boron, steel, lead, heavy concrete and light concrete, was analyzed.

The fast and thermal neutron flux calculations were done by two-group analysis, with the aid of computer for the latter. Determination of the gamma flux from the core, as well as the capture gammas, comprise the rest of the calculations.

The procedures followed and the calculations made in connection with the study are explained, and the results are tabulated and plotted.

900. Wilson K. Talley, Analytic Expressions for Calculating Neutron Distribution Functions by Monte Carlo (Thesis), (1964). Availability: University Microfilms, Inc., Ann Arbor, Mich., $5.00 as No. 64-9098).

In Monte Carlo calculations, the successive directions and energies of a particular history of a neutron slowing down in a homogeneous medium can be determined without reference to the locations of the collisions. For each set of such discrete directions, the analytic form of the spatial distribution of any particular collision is developed by integrating over that of the previous collisions. Not only is the statistical error of such a method less than that of ordinary Monte Carlo, but an analytic expression for the energy distribution of the last collision enables one to calculate a sample of the vector flux density at chosen positions and energies.

Previously, the use of analytic spatial distributions has provided scattering probability densities for plane source neutrons in an infinite homogeneous medium and for a plane source on the boundary of a finite slab. The solution to the more general problem of a plane source anywhere within the finite slab is presented here; however, the main portion of this thesis treats the more complicated three-dimensional problems involving point sources.

The scattering probability distribution functions for a point source in an infinite and a semi-infinite medium are developed, and the spatial moments of the former are given. The results for the plane source are shown to follow those from the point source. The extension of this method to finite bodies is discussed. The problem of round-off error (due to machine truncation) is examined, and a method to circumvent it is set forth.

901. G. D. Joanou and C. A. Stevens, Neutron Cross Sections for U-238, GA-6087 (Rev.) (NASA CR-54290) (April 16, 1965). Availability: NASA, Office of Scientific and Technical Information, Attention: AFSS-A, Washington, D. C.

A revised set of neutron cross sections has been prepared for U-238. The compiled and evaluated data have been incorporated into the GAM-II slowing down program and into the GATHER-II thermalization program. The literature survey associated with this report is believed to be complete to November 1964.

902. James H. Renken, Transmission of X Rays through Air, SC-RR-65-141 (April 1965). Availability: $3.00, CFSTI.

The transport equation describing the transmission of photons through a medium which both scatters and absorbs radiation is considered. After specializing to the case of a point source emitting isotropically into a homogeneous medium, a solution for the photon density as a function of various parameters is obtained by using the polynomial method of Fano and Spencer. Sample numerical results are given.

903. Nanette Chazan, Tabulated Neutron-Induced Gamma Production Cross Section for Primary Neutron Energies of 0.1 to 15 MeV, UCRL-14007 (January 13, 1965). Availability: University of California, Lawrence Radiation Laboratory, Livermore, Calif.

Gamma-production cross sections, both integrated and differential, are tabulated. The data are categorized according to whether the gammas produced are continuum, discrete, or total over all gamma energies.

904. J. A. Baran, Blockhouse Dosage Contributions Resulting from Window-Collimated, Ceiling Scattered Fallout Radiation, AD 443549 (1963). Availability: (Master's Thesis), Kansas State University, Manhattan, Kansas.

The theory of the penetration of fallout gamma radiation into structures has been extensively studied and formulated in the past few years. This monograph considers one aspect of this theory which has not been investigated--the contribution of radiation dose rate in a protective structure resulting from fallout radiation entering the structure through apertures in vertical walls and reflecting off the ceiling of the structure. This problem is known as the "ceiling shine" problem.

FORTRAN II digital computer programs are written whereby the differential and total (incident and reflected) aperture-collimated, dose rate contributions at the ceiling of an idealized concrete structure are determined. A program was also written for the aperture-collimated, ceiling scattered dose rate at any point in the structure. Results are graphically presented for the case of the concrete blockhouse presently erected at the Kansas State University Nuclear Engineering Shielding Facility (a 20' x 20' x 8' structure with basement and a window centered horizontally in one wall). This window covers approximately 5% of the area of one wall.

The average total dose rate albedo of the concrete blockhouse ceiling for incident cobalt-60 gamma photons was determined from the derived differential dose rate albedo to be fourteen per cent. The window-collimated radiation incident on the ceiling was just under one percent of the dose rate outside the structure at the standard position three feet above the contaminated plane. Approximately one tenth of one per cent of this outside radiation was thus reflected at a given ceiling point.

905. Donald S. Sasscer, Gamma Dosage in the Vicinity of Cylindrical Voids in Shields (Thesis), (1964). Availability: $6.20, University Microfilms, Inc., Ann Arbor, Mich., as No. 64-9283.

The Monte Carlo technique was used to determine the rate at which energy is absorbed in water at the surface of cylindrical ducts through an otherwise homogeneous shield assumed to be infinite in extent and 200 cm thick. The centerline of the ducts was perpendicular to the direction of infinite extent and a non-disintegration-per-second, isotropic, point source of cobalt-60 was located on the centerline at midthickness. The positions of the first collisions were determined by a programmed sequence of positions between 0.5 and 27.5 cm from the duct at which initial collisions were allowed to occur. The rate at which energy was absorbed at the surface of the ducts having length to radius ratios of 10, 26.25, 50, and 100 was obtained as a function of distance along the centerline of the ducts. The results were analyzed dimensionally by use of the Buckingham Pi theorem, which predicted that the basic variables could be grouped into six dimensionless Pi terms. The manner in which the energy absorption buildup factor (π_1) varies at the surface of ducts six mean free paths (mfp) in length (π_3) is determined as a function of the distance from the source (π_2) for ducts with various ratios of length of radius (π_4).

906. F. C. V. Worman and P. K. Lee, Integral Gamma and Neutron Mesurements on the NRX-A2 Reactor, LA-3296-MS (April 12, 1965). Availability: $2.00, CFSTI.

This report presents data obtained on integral neutron fluxes and gamma and neutron rad doses measured on Experimental Plans 2, 4, and 5 of the NRX-A2 reactor tests. Studies included a 2,700-foot freeline, a meridian ring, and an equatorial ring. Neutron fluxes and gamma and neutron rad doses exhibited a close approximation to an inverse square relationship, with air attenuation, along the freeline.

Since the NRX-A2 reactor was similar to the Kiwi B reactors, data were compared by Group H-8 of the Los Alamos Scientific Laboratory with earlier Kiwi B data. Information obtained from these studies indicated that the radiation leakage patterns observed for NRX-A2 were very similar to those of the reactors of the Kiwi B series.

907. R. L. French, A Comparative Study of Radioactive Source Arrangements for Simulating Fallout Gamma Radiation Fields, RRA-T45 (June 15, 1964). Availability: $4.00, CFSTI as AD 612032.

Monte Carlo techniques were used in a study of three basically different approaches to simulating the gamma radiation environment near the air-ground interface due to fallout uniformly distributed on the ground surface. The energy and angular distribution of the photon flux at a receiver 3 feet above the ground due to a Co-60 point isotropic source also three feet above the ground and at separation distances of 100 to 800 feet was computed, and the results were found to bear little resemblance to those from a uniform fallout field. Results computed for a receiver position three feet above idealized Ce-144, Cs-137, and Co-60 infinite plane sources on the ground indicated good simulation of the angular distribution of the gamma-ray dose and, except for energies greater than 1.25 MeV, reasonable simulation of the energy spectra from fallout. A third source arrangement, called the "compact simulator," was found to simulate the fallout gamma radiation environment equally well. The compact simulator utilizes Ce-144, Cs-137, and Co-60 in a finite plane source 60 feet in radius and in an elevated toroidal source which acts as a virtual source for the remainder of the infinite plane. In the compact simulator, an overhead slab of water acts as a reflector to simulate air scattering.

908. K. D. Lathrop and B. G. Carlson, Discrete Ordinates Angular Quadrature of the Neutron Transport Equation, LA-3186 (September 21, 1964). Availability: Dep., $3.00, CFSTI.

Methods of preparing discrete ordinates quadrature coefficients are described. Quadrature sets which satisfy complete symmetry conditions and various moment conditions are derived and tabulated, and critical thicknesses of one-dimensional slabs, spheres, and cylinders are calculated with these sets. Prescriptions for relaxing symmetry conditions and point location requirements are discussed, and orthogonal (Legendre-Tschebyschev) quadrature coefficients applicable in one-dimensional cylindrical or two-dimensional rectangular geometry are tabulated. Recipes are described for preparing biased direction sets, and a method of basing the bias upon material composition is outlined.

Preliminary computational results indicate that double Legendre and half-range moment satisfying quadrature sets are most accurate in one-dimensional plane geometry, while even-moment satisfying, completely symmetric sets are recommended for other geometries. At the present state of the art of discrete ordinates computations, results indicate that boundary condition treatment and, in curved geometries, the handling of the ray-to-ray transfer (streaming) terms can affect accuracy as much as further refinement of angular quadrature. Since computational results may depend upon all three of these quantities, further work is needed before an optimum quadrature method can be selected.

909. O. I. Leipunskii, B. V. Novozhilov, V. N. Sakharov, The Propagation of Gamma Quanta in Matter (English Edition), Pergamon Press (1965), $15.00.

This volume, translated by P. Basu and edited by K. T. Spinney, J. Butler and J. B. Sykes, is a detailed account of Russian work in gamma ray shielding up to 1959. The introduction contains sections on the interaction of gamma radiation with matter, multiple scattering and distribution functions. Subsequent sections deal with the theory of multiple scattering (several methods of solving the Boltzmann equation), and propagation of radiation for various geometrical configuration of the sources and absorbing media.

910. M. J. Berger and L. V. Spencer, Penetration of Gamma Rays from Isotropic Sources through Aluminum and Concrete, NBS Tech Note 11 (May 11, 1959). Availability: $0.50, CFSTI.

Semi-analytical expressions, with numerically specified paramters, are given which represent the gamma ray dose distribution in infinite aluminum or concrete media, for sources that are monoenergetic (with energies between 10.22 Mev and 0.0341 Mev), isotropic, and have the form of an infinite plane, point, disk or spherical surface.

911. C. M. Davisson, "Interaction of γ-Radiation with Matter," Chapter 2 and Appendix I, Alpha-, Beta- and Gamma-Ray Spectroscopy, Vol. 1, Kai Siegbahn (editor), North-Holland Publishing Company (1965).

A summary and brief description of most of the processes by which γ-rays interact with matter are given in Table 1. In the energy range being considered here, i.e. from about 50 keV to about 50 MeV, most of the interactions are due to one of three processes: (1) the photoelectric effect, in which a photon gives all its energy to a bound electron, which uses part of the energy to overcome its binding to the atom and takes the rest as kinetic energy. This predominates at low energies, (2) scattering by the atomic electrons, in which the photon is deflected from its original direction, with or without loss of energy. At energies much greater than the binding energies of the electrons, the photons are scattered as if the the electrons were free and at rest. This is the Compton effect, and around 1 MeV it is the dominant mode of interaction, (3) pair production, in which a photon, in the field of a nucleus or an electron, disappears with the creation of an electron pair. The total kinetic energy of the resultant particles is equal to the photon energy minus the mass energy of the two particles which have been created. It starts about 1 MeV and as the energy increases it becomes the dominant mode of interaction. Currently there is interest in detecting and studying Delbruck scattering, which is elastic potential scattering by the Coulomb field of the nucleus. Since it is elastic, coherent radiation, it combines with Rayleigh scattering, nuclear resonance scattering, and nuclear Thomson scattering. So to detect Delbruck scattering the contributions of the others to the elastic scattering must be known. Therefore the theoretical values and the experimental detection of these smaller effects are of considerable interest.

912. L. P. Abagyan, M. O. Bazazyants, I. I. Bondarenko, and M. N. Niklolaev, Group Constants for Nuclear Reactor Calculations, I. I. Bondarenko (editor), Consultants Bureau Enterprises, Inc., New York, $17.50.

Even though the basic contents of this book are devoted to systems of constants per se, the book also goes into several questions involving techniques in the compilation and use of such systems.

Fast and intermediate-spectrum neutrons play a crucial role in reactors of all types. Therefore, the brunt of the attention in this book is placed on constants for fast and intermediary neutrons. Their role is the predominant one in fast- and intermediate-neutron reactors. However, even in thermal reactors the processes of slowing down and of propagation of fast and intermediary neutrons play a primary role. Moreover, fast and intermediary neutrons determine to a significant degree the properties of the reactor radiation shielding to be used.

The book also provides constants for the thermal group of neutrons--but only by specifying the values of the cross sections with reference to free and fixed atoms in the case of 0.025-eV neutrons. At the same time, the questions related to the compilation of multigroup systems of constants for exact calculations of neutron thermalization go quite beyond the scope of this book.

913. J. Sharpe, Nuclear Radiation Detectors, Methuen and Co., Ltd.-London, John Wiley and Sons, Inc., New York (1964), $4.95.

In this book I have endeavoured to present basic information needed by those interested in the design and use of detectors for nuclear radiation. The plan of the treatment is outlined in the introduction and excludes detailed discussion of the technical aspects of instruments and of electronic equipment.

914. C. Eisenhauer, Scattering of Cobalt-60 Gamma Radiation in Air Ducts, NBS-TN-74 (October 1960). Availability: $0.75, CFSTI.

The exposure dose rates due to Cobalt-60 gamma radiation scattered in small air ducts in concrete has been measured for ducts with one and two right angle bends. The inside corner of a right angle bend has been found to be an important source of scattered radiation. Results are analyzed in terms of solid angle relationships and attempts are made to extrapolate experimental results to other duct configurations.

915. J. C. LeDoux and L. K. Donovan, Shielding Factors for Underground Shelters of Various Geometric Shapes, USNCEL-R-080 (AD 254344), (April 5, 1961). Availability: U.S. Naval Civil Engineering Laboratory, Port Hueneme, California.

This study investigates the additional nuclear shielding from an isotropic (plane) radioactive gamma source afforded by various shapes of underground curved-roof shelters compared to the basic slab shield. This additional shield is defined in the form of a dimensionless geometry factor which is a function of the physical dimensions and shape of the shelter.

Curves are presented from which the geometry factors for the underground shelter shapes of spheres, horizontal cylinders, ellipsoids, and vertical cylinders or silos can be obtained with minimum calculations using only the physical dimensions of the shelter and the depth of material above the crown of the shelter.

The geometry factors thus obtained are independent of material and photon energy except in the case of the silo and can be used as dimensionless factors once the attenuation for a slab shield has been calculated for a particular set of radiological conditions and materials.

916. F. C. Maienschein, J. L. Russell, Jr., R. J. Huber, E. E. Bennett, V. V. Verbinski, G. L. Brownell, H. M. Epstein, A. Sayres, W. R. Burrus, J. C. Courtney, Proceedings of the Special Session on Fast Neutron Spectroscopy (American Nuclear Society), edited by J. Greenborg, ANS-SD-2 (December 1964). Availability: American Nuclear Society, Hinsdale, Illinois.

(No abstract needed.)

917. B. Stŏces, Multiple Group Calculation of a Biological Neutron Shield with Layers of Steel and Graphite, ORNL-tr-507 (April 1964). Availability: Dep. (mn); $1.60 (fs), $0.80 (mf), JCL. (Translated from UJV-1032/64).

Multigroup computation is used in the search for an optimal ratio between the amount of iron and graphite in a neutron shield of a designated thickness, composed of two or more layers; the effect of the addition of an absorber is considered, either in homogeneous dispersion in graphite, or concentrated behind the shield.

918. A. Rudloff, The Use of Build-Up Factors in Calculating Gamma Radiation from Plane Sources behind Absorber of Finite Extension, ORNL-tr-656 (November 1964). Availability: Not available except through Scripta Technica, 275 Madison Avenue, New York, New York, 10016. (Translated from Atomkernenergie 9, 451 (1964).

The penetration of gamma radiation from plane sources through absorbers of finite extension is calculated by the use of a two parametric formula for the build-up factor. The parameters, depending on the quantum-energy of the gamma radiation and the atomic number of the absorber, are adapted to the values of Goldstein and Wilkins in the best way possible.

The calculation is made for circular and rectangular plane sources, each for the radiation of Co-60 resp. Cs-137. If the detector is placed vertically under the center of the plane source, the dose rate depends in first approximation only on the solid angle from which the detector faces the plane source. For rectangular plane sources, the side-lengths relation of which differs considerably from 1, it is necessary to introduce a corrective factor. The results are compared with experimental data found in a course of radiation protection in Manhattan, Kansas (USA).

919. M. B. Wells and J. M. Newell, Neutron Transport Data for Single and Multilayer Slabs of Aluminum, Polyethylene, and Carbon, RRA-T52 (May 28, 1965). Availability: Radiation Research Associates, Inc., Fort Worth, Texas.

Dose rate transmission data for 18 MeV plane isotropic neutron sources embedded in slabs of aluminum, polyethylene, and carbon are presented for slab thicknesses to 30 cm. These data as well as the neutron transmission data previously reported for monoenergetic sources between 0.5 and 18 MeV were integrated over the evaporation neutron spectrum to provide transmission data for use in evaporation neutron transport calculations. To illustrate the application of the transmission data, neutron doses behind slabs of aluminum, polyethylene, and carbon were computed for evaporation neutrons produced by normal incident protons with an energy spectrum defined by the equation

$$\Phi(E) = 3.49 \times 10^{8} E^{-3.12} \text{ (protons cm}^{-2} \text{ MeV}^{-1} \text{ flare}^{-1}\text{)}.$$

Simplified calculational methods are outlined for use in predicting the transmission of neutrons from plane isotropic sources embedded in multilayer slabs. The simplified methods, which make use of the results of the single-layer slab calculations, were found to produce results which are in good agreement with dose rate transmission data from Monte Carlo calculations for two and three layer slabs.

920. John C. Courtney, Fast-Neutron Scattering from Small Cylinders of Steel, Aluminum, and Graphite, ORNL-TM-1156 (July 1965). Availability: $4.00 (hc), $0.75 (mf), CFSTI as AD 618922.

Time-of-flight methods were used to measure the spectra of fast neutrons scattered at angles of 20, 45, 70, 110, and 160 deg from the incident beam by small (2-in.-diam, 2.5-in.-high) cylinders of steel, aluminum, and graphite. The measurements were made at the General Atomic Electron Linear Accelerator (LINAC), where a pulsed source of bremsstrahlung-produced neutrons was obtained by irradiating a lead target with 14-nsec bursts of 34-MeV electrons. The energy spectrum of the neutrons approximated a fission spectrum. Some of the neutrons struck the scattering sample, which was placed 13 in. from the source, passed through a system of collimators, down a drift tube, and into a liquid scintillator fast-neutron detector. Pulse-shape discrimination circuitry was used to eliminate pulses due to any gamma-ray background.

Measured source spectra were used as input for Monte Carlo calculations of the scattered neutrons, and the results were compared with measured scattered spectral intensities. The calculations and experiments, integrated over energy agreed within 16%. Since the spectral shape was satisfactorily reproduced, it is believed that the uncertainty in the measurement of the integrated fast-neutron flux by foil activation accounts for any discrepancies. These comparisons indicate that the ORNL cross-section library for these elements is adequate to handle neutron scattering calculations.

921. S. K. Mehta, A New Method for the Evaluation of Neutron Emission, Kerma, Kerma Equivalent, Dose Equivalent and the Spectrum Below 1 MeV for (α,n) Sources, Using Bonner Spheres, ORNL-TM-1171 (July 1965). Availability: Dep. (mn); $5.00 (cy); 3 (mn), CFSTI.

Neutron emission (F), kerma (K), kerma equivalent (Keq), dose equivalent (DE), the ratios K/F, Keq/F, and DE/F, and an upper bound to the neutron spectrum below 1 MeV have been obtained for Am-Be, Pu-Be, Po-Be, and Am-B neutron sources by using modified Bonner sphere neutron spectrometer. The spectrometer consisted of 0- to 12-in.-diam Bonner spheres and 0- to 3-in.-diam spheres modified by the addition of boron and cadmium covers. The results were computed by the OPTIMO linear programming code which synthesizes optimal combinations of the response of the various spheres. The known smoothness of the neutron spectra from the source is utilized, and an upper bound to the spectra above 5.5 MeV is taken from other measurements, since the response of the Bonner spheres does not have enough variation above 5.5 MeV. The response of the modified spheres to monoenergetic neutrons was calculated by assuming exponential absorption below 1 KeV and by S_n transport calculations above 1 KeV. The values of DE/F in units of rems/neutron-cm^2 for the Am-Be, Pu-Be, Po-Be, and Am-B sources are $(3.79 \pm 0.21) \times 10^{-8}$, $(3.72 \pm 0.28) \times 10^{-8}$, $(3.84 \pm 0.19) \times 10^{-8}$, and $(3.75 \pm 0.21) \times 10^{-8}$, respectively. The errors given refer to the 68 per cent confidence intervals and include all known sources of error.

922. W. E. Edwards, Gamma Ray Production Cross Sections (Suppl. No. 1), GEMP-360 (August 26, 1965). Availability: Dep. (mn); CFSTI, $2.00 (cy), $0.50 (mn).

Gamma ray production cross sections are presented in this report as a function of 15 neutrons lethargy groups and 11 photon energy groups for 15 elements and isotopes. They were computed by Program GAMMA-P and account for the production of gamma rays by neutron radiative capture, neutron inelastic scattering, neutron induced fissioning, and (n,α) reactions.

923. A. E. Profio and J. Kirkbride, Measurement of Fast Neutron Spectra in an Infinite Medium of CH_2, AFWL-TR-64-180 (April 1965). Availability: CFSTI, $4.00 (hc), $1.00 (mf).

Measurements of the fast neutron spectrum and angular distribution in a large assembly of paraffin wax (CH_2) were performed for the Air Force Weapons Laboratory under Contract AF 29(601)-6376. Two targets were irradiated with a pulsed electron beam from a Linear Accelerator, one an essentially point isotropic fission spectrum source and the other a narrow photoneutron source for small angle measurements. The time-of-flight technique was employed to measure the energy from approximately 0.3 MeV to approximately 15 MeV with an energy resolution of 10 percent. Estimated error in the spectrum is ± 10 percent from 0.5 to 8 MeV and ± 20 percent outside this range. The source spectra were measured in air for input to calculations. Data were obtained at 0° up to 15 cm penetration with preliminary results to 45 cm. Angular flux measurements were made up to 60° at 30 cm, with preliminary data at 8° and 45 cm. The results, within the stated accuracy and resolution, should be suitable for testing analytical techniques and corresponding shielding codes.

924. R. L. French, Simulation of Fallout Gamma Radiation Fields by Monoenergetic Plane Isotropic Sources, RRA-M51 (June 21, 1965). Availability: Radiation Research Associates, Inc., Fort Worth, Texas.

Monte Carlo techniques were used in a study of the gamma radiation environment three feet in air above plane sources of monoenergetic and fission-product gamma radiation. The scattered and direct-beam fluxes were computed in each of ten receiver energy intervals between 0.04 and 3.5 MeV and eight 10-degree receive angle intervals between 0° (nadir) and 180° (zenith). Source energies of 0.10, 0.67, and 1.25 MeV, which are representative of the gamma rays from ^{144}Ce, ^{137}Cs, and ^{60}Co respectively, were used in the calculations. Additional calculations were performed for source energies of 0.14, 0.25, 0.40, 0.85, 1.75, 2.50, and 3.50 MeV and the results were weighted with the gamma-ray spectra from fallout of various ages. Detailed comparisons of the results show that the angular distribution of the dose from a fallout field is simulated with reasonable accuracy over most angles by a plane source of ^{137}Cs or ^{60}Co and over all angles by a mixture of ^{144}Ce, ^{137}Cs, and ^{60}Co. The simulation of the fallout energy spectra is quite poor, however, even if combinations of the three isotopes are employed. Comparisons of the Monte Carlo data with measured data for monoenergetic plane sources show good agreement.

925. John T. Terrall, Particle Escape from the Ends of Cylindrical Tubes, APEX-525 (March 1958). Availability: $1.00, CFSTI.

This report is concerned with the fraction of those particles entering a finite cylindrical tube that escape out the ends of the tube without making a collision. The assumptions in the report are (1) that the entering angular distribution is a function only of the polar angle measuring deviation from the normal at the entering surface element, (2) that any material in the tube is distributed uniformly, and (3) that the tube has a circular cross section.

The theory is discussed in two parts, "Escape from an Empty Tube" and "Escape from a Nonempty Tube."

The main discussion centers on the calculation of the escape fraction for those particles entering in the neighborhood of a point on the side surface, since this is the crux of the theory.

926. R. L. French, M. B. Wells, and N. M. Schaeffer, Neutron and Gamma Radiation Penetration into Concrete-Shielded Underground Structures, RRA-T51 (May 31, 1965). Availability: Radiation Research Associates, Inc., Fort Worth, Texas.

The fast-neutron and gamma-ray dose rates in concrete-shielded and concrete-lined cylindrical holes and bunkers located at distances of 100 to 700 feet from the ORNL Tower Shielding Reactor (TSR-II) were computed and compared with measured dose rates. The calculations, which in most cases agree within 25 percent with the measurements, show the relative importance of the various radiation components including direct-beam, shield-scattering, wall-scattering, and secondary gamma rays. Wall-scattering was found to contribute up to 40 percent of the fast-neutron dose rate at positions deep inside the structures. Thermal-neutron capture in the concrete shields accounted for as much as 90 percent of the gamma-ray dose rates for thick shields. Capture gamma rays produced in the walls of the cylindrical holes and bunkers were also important in many cases. Simplified methods were developed for computing most of the radiation components. An appendix gives the application of these methods to a cylindrical hole with laminated iron and concrete shields.

927. B. Eriksson, On the Use of Importance Sampling in Particle Transport Problems, AE-190 (June 1965). Availability: Dep. (mn); Aktiebolaget Atomenergi, Stockholm, Sweden.

The idea of importance sampling is applied to the problem of solving integral equations of Fredholm's type. Especially Boltzmann's neutron transport equation is taken into consideration. For the solution of the latter equation, an importance sampling technique is derived from some simple transformations of the original transport equation into a similar equation. Examples of transformations are given, which have been used with great success in practice.

928. Leonard B. Gardner, Alan J. Mettler, Monte Carlo Calculation of Neutron Streaming Through Two-Legged Duct Entranceways, USNCEL-R-379 (NP-15094, AD 616521). Availability: $3.00, CFSTI.

Several techniques for calculating dose rates in the study of shelter entranceways are reported herein. Calculations have been verified by experiments. These calculations also determine neutron energy spectra, a parameter that is difficult to measure and which is very useful in gaining insight into the design of protective structures. Both the neutron flux spectra and dose rates are calculated for various positions along the centerline of a two-legged duct. Calculations were performed utilizing the ADONIS computer code which solves the Neumann approximation to the transport equation by Monte Carlo techniques. These Monte Carlo calculations were compared to second-leg calculations utilizing a semiempirical albedo formula, and were also compared to experimental measurements made with a tissue equivalent dosimeter. Included is a discussion of UNIGEOM, a generalized computer program for ADONIS geometry, designed to be run on the IBM 7094.

929. Dean H. Stoddard, Radiation Properties of Californium-252, DP-986 (June 1965). Availability: $1.00, CFSTI.

The radiation properties of californium (~72 wt% ^{252}Cf) produced by thermal-neutron bombardment of transplutonium isotopes are presented. The gamma and neutron radiations associated with this material were converted to dose rate, for evaluation by isotope users.

930. R. Schamberger and J. Celnik, MCR Small Source Experiments, GMAD-3078-28 (May 3, 1965). Availability: Dep.(mn); CFSTI, $4.00 (cy), $0.75 (mn).

Military Compact Reactor (MCR) shield design analytical techniques and cross sections were tested by comparing calculations with measurements for experiments emphasizing the secondary gamma-ray production in specific test materials.

Measurements were made with a gamma-ray spectrometer behind a configuration consisting of a layer of gamma-ray shielding material (the test sample) adjacent to a small polonium-beryllium (Po-Be) neutron source, followed by neutron and gamma-ray shielding material to reduce extraneous detector response. Calculations of spectrometer response for samples of lead, tungsten, and titanium hydride were performed with a three-dimensional Monte Carlo code, derived from the one-dimensional Monte Carlo codes used in the MCR preliminary design shield analysis.

The calculations generally gave satisfactory predictions of the measurements.

Differences between the measured and calculated responses indicated that some of the cross sections used, particularly for lead, were low and should be revised. It is estimated that these revisions would lead to a 10% increase in predicted total dose rate external to the MCR shield.

931. A. R. Larson, Use of DSN Code for Neutron Shielding Calculations, Pratt and Whitney TIM No. 792 (December 27, 1963). Availability: Dep. (mn); $1.00 (cy), 1(mn), CFSTI.

The Los Alamos DSN Code was applied to the calculation of the neutron attenuation kernel for a point fission source in hydrogen. Excellent results were obtained when use was made of the anisotropic option, n equal to 16 in the Sn approximation and a set of cross sections having a fine group structure between 0.1 and 14 MeV. The broad group structure in the high energy range of 16 group cross sections used for reactor calculations was shown to be inadequate.

932. J. O. Howard, G. M. Jacks, Advanced Test Reactor, Final Shielding Design Report, IDO-24467 (December 1963). Availability: $6.00, CFSTI.

This report has been prepared as a design reference document describing the calculation methods and engineering results for shielding analysis work done during the design of the Advanced Test Reactor. All calculations that are significant in the plant shield design are described. Also included are sections on heat generation rates, materials activation, and spent fuel and test element transfer.

The calculation descriptions are divided into two parts: The methods used to obtain the activity of the source are described in Section 3, and the geometry and the attenuation equations from source to dose point are described in the remaining sections.

933. B. Carlson, "The Numberical Theory of Neutron Transport," Methods in Computational Physics, Vol. I, Statistical Physics, Academic Press (1963), pp. 1-42.

The numerical theory of neutron transport is reviewed with an emphasis on modern computing methods, especially the S_n method. The topics considered are: coordinate systems and derivation, transformation, and solution of the difference equations.

934. J. H. Hubbell and M. J. Berger, Photon Attenuation and Energy Transfer Coefficients. Tabulations and Discussion, NBS Report 8681 (May 10, 1965). Availability: National Bureau of Standards, Washington, D. C.

This paper consists of two parts. The first part contains tabulations of the photon attenuation coefficient and the energy transfer coefficient. The values of the attenuation coefficient represent a revision which takes into account new theoretical data on the photoelectric effect and on radiative corrections to Compton scattering and pair production. Rayleigh (coherent) scattering and the photonuclear effect and on radiative corrections to Compton scattering and pair production. Rayleigh (coherent) scattering and the photonuclear effect are disregarded, but can be included by use of auxiliary tables. The main tables cover the energy range 10 keV to 100 MeV for 21 elements and four mixtures. This range is extended to 10 GeV for 16 elements by interpolation from experimental data. The energy transfer coefficient data were taken directly from tabulations in the literature, and cover the energy range 10 keV to 100 MeV. In the second part, information is given on the individual processes (photoelectric absorption, Compton scattering, pair production and other interactions of photons with matter) which combine to give the attenuation coefficient. Theoretical and experimental data are reviewed, and numerical tables and approximation formulas are given.

935. H. J. Amster and R. C. Gast, "The Analysis of Foil Activation Experiments in Infinite Homogeneous Media," Nucl. Sci. Eng., 11(2), 167-178 (October 1961).

A calculational and experimental procedure is developed to eliminate nearly all of the uncertainties in the source and foil corrections required in most age experiments. The calculations are rigorous and yield, in addition to flux ages, values of actual foil activation at the exact spatial positions of the measurements; comparisons with experiment therefore involve fewer assumptions and produce more information. Being required in the calculations, the angular sensitivity of Cd-covered in foils is determined theoretically and checked with measurements. Comparisons between new calculations and data from old activation experiments with fission and Na-Be sources in water are made; the results suggest that a new type of fission source experiment is still called for.

936. L. G. Haggmark, T. H. Jones, N. E. Scofield and W. J. Gurney, "Differential Dose-Rate Measurements of Backscattered Gamma Rays from Concrete, Aluminum, and Steel," Nucl. Sci. Eng., 23(2), 138-149 (October 1962).

The angular distribution of backscattered dose-rate ratio was measured for 'semi-infinite' concrete, aluminum and steel slabs irradiated by plane-parallel beams of Co-60 and Cs-137 gamma photons. The photons were incident on the slabs at angles, measured from the normal to the slab, of arccos 1.00, 0.75 and 0.50. For the necessary sensitivity, the backscattered dose rate was measured by a digital dosimetry system using a plastic scintillator as the detector.

An empirical formula for differential dose-rate ratio was derived from the experimental data. Comparisons are made with two other experiments and a semi-empirical formula fitted to a Monte Carlo calculation. The experiments generally agree to within 20%. The values based upon the Monte Carlo calculation are generally 20% to 35% lower than the experimental values.

937. E. Aalto, R. Fraki and K. Malen, "The Fine Adjustment of the Neutron Penetration in the NRN Method," Nucl. Sci. Eng., 22(4), 443-450 (August 1965).

Recommended values have been experimentally obtained for the angle (defined by its cosine μ_o in center-of-mass system) that determines the boundary between 'removal' and 'nonremoval' collisions and regulates the deep penetration of neutrons in the NRN method. Measured attenuations in three different, most common, shield materials: water and magnetite and ordinary concrete, give $\mu_o = 0.6$ ($\pm$ 0.1) for elements with $A > 1$. For hydrogen, $\mu_o = 0.45$ is recommended.

The results indicate, besides, that the neutron flux predictions are not overly sensitive to the changes in the removal source, caused by varying μ_o. The usual smoothing effect of the diffusion is material dependent, and the strength of the coupling between removal and diffusion parts is seen to decrease when going from water to magnetite and ordinary concrete.

938. A. W. Boyd, H. W. J. Connor, S. A. Kushneriuk, "Measurement and Calculation of Energy Deposition Due to Capture Gamma Rays Inside Annuli of Cadmium, Silver, and Stainless Steel," Nucl. Sci. Eng., 22(4), 494-495 (August 1965).

The energy deposition rates in graphite inside tubes of cadmium, silver, and stainless steel was measured using a graphite-CO_2 ion chamber. Calculated results agreed well with the measured values.

939. G. E. Hansen, H. A. Sandmeier, "Neutron Penetration Factors Obtained by Using Adjoint Transport Calculations," Nucl. Sci. Eng., 22(3), 315-320 (July 1965).

Adjoint transport theory is most widely used in perturbation theory. A most common problem here is the determination of the reactivity change in a self-multiplying system due to the insertion of an absorber in a small region.

There is, however, a class of problems of the source-detector type where adjoint transport theory proves to be a very effective and fast way of obtaining the desired results. In many practical source problems we want to evaluate the reaction rate, say fissions or absorptions, in a material surrounded by a moderator due to a neutron flux incident on the assembly. Here the main advantage of using the adjoint method as opposed to the conventional real-flux shell-source calculations is a significant reduction in computer time. The reactions induced by each group of source neutrons is obtained from one run of an adjoint problem.

To obtain the same information from real-flux calculations we need an individual run for every energy group g. Computer time savings ranging by a factor of 5 to 30 are representative.

The theory previously reported by one of us (H.A.S.) in the classified literature is derived and subsequently applied to the following problems.

a. The fissions induced in a spherical plutonium-detector foil separated by a moderating layer from an incident collimated neutron beam;

b. A neutron-dose-rate detector device consisting of a lithium iodide crystal to register absorptions surrounded by a sphere of polyethylene;

c. The theoretical evaluation of the neutronic coupling coefficient between two reactors, as one might visualize in a clustered-Rover nuclear-reactor rocket-engine system.

940. A. Mockel, "Reflection and Transmission by a Strongly Absorbing Slab," Nucl. Sci. Eng.,22(3), 339-349 (July 1965).

Reflection and transmission of monoenergetical particles with a known ingoing distribution by a strongly absorbing slab is studied from the numerical standpoint.

Various approximation methods based on known theoretical solutions are presented: in section III we proposed an approximation based on Chandrasekhar X and Y functions; in section IV we obtain the reflection and transmission by using a variational technique, and we show that a successive-collision technique gives identical results; and in section V we proposed a diffusion-like approximation, with adjusted coefficients, of the form

$$R = \frac{1-e^{-2\alpha x}}{1 - Ae^{-2\beta x}} R_{as} \quad ,$$

where

$$R_{as} = \frac{1-f(c)}{1+f(c)} + g(c) \quad ,$$

and

$$T = \frac{(1-B)e^{-\alpha x}}{1+Be^{-2\alpha x}} \quad .$$

The first approximation gives good results for low c values; the second one, for high c values.

The diffusion-like approximation, however, is accurate to more than 2% for all values of c between 0.1 and 0.9. Moreover, it is far easier to compute than any of the former ones.

941. G. L. Burrows and D. B. MacMillan, "Confidence Limits for Monte Carlo Calculations," Nucl. Sci. Eng., 22(3), 384-385 (July 1965).

Experience with the Shapiro-Wilk test of non-normality of results of Monte Carlo calculations is described. It was found that all bad results obtained by the author was flagged by the test. A test procedure is given for a sample size of 25.

942. S. W. Smith, Safety Standard for Non-Medical X-Ray and Sealed Gamma-Ray Sources, NBS Handbook 93 (January 3, 1964). Availability: GPO $0.30.

Basic protection recommendations pertaining to x- and sealed gamma-ray sources are given in this handbook. Numerous graphs and tables are given to facilitate the design of shields for x-ray sources and the following gamma-ray sources: cesium-137, chromium-51, cobalt-60, gold-198, iridium-192, radium-226, and tantalum-182.

943. R. S. Hubner, Analysis of Neutron Transmission through Slabs of Lithium Hydride, NAA-SR-MEMO-10853 (January 4, 1965). Availability: Dep. (mn); $1.00 (cy), 1(mn), CFSTI.

The results of the analyses of the ORNL Tower Shielding experiments on fast neutron transmission through lithium hydride slabs are presented and compared to the experimental data. Satisfactory agreement in the scattered dose is noted in all cases with the maximum deviation at the 30 inch thickness being only 25%. The calculated transmitted spectra and n^{th} collision doses are presented. The Monte Carlo techniques used in the analysis were described in NAA-SR-9418; modifications to these methods are presented in the appendix.

944. A. W. Casper, Data Report and Analysis of Measurements Made in Water Behind Arrays of LiH and Li^6H, XDC-59-10-69 (October 1959). Availability: Dep. (mn); $2.00 (cy), 1(mn), CFSTI.

This report presents the fast neutron dose rates, the gamma dose rates, and the thermal neutron flux measurements made in water behind arrays of LiH and Li^6H. The measurements were made in the pool at the GE-ANPD Source Plate Facility at Battelle Memorial Institute. Fast neutron dose rates, gamma dose rates, and thermal neutron flux measurements behind equal thicknesses of LiH and Li^6H are compared. The effect of varying the LiH and Li^6H thicknesses are also presented. Point kernel programs were used to compute the fast neutron and gamma dose rates and these values are compared with the measurements.

945. L. R. Glasgow and J. M. Haughian, Shielding Doors, UCRL-11810 (Conf-650306-42) (March 2, 1965). Availability: Dep. (mn); $1.00 (cy), 1(mn), CFSTI.

This paper describes general types of shielding doors and the drive systems for opening and closing them. It compares estimated costs of the different types of doors based on factors they have in common. It describes a number of shielding doors designed and built at Lawrence Radiation Laboratory. These include a vertically rising door actuated by a hydraulic cylinder in a pit beneath it, a rotating slotted-barrel door, a swinging door, and a number of more or less typical sliding doors. Some of the problems involved in the design and use of these doors are discussed. Actual costs for a number of doors are given.

946. James Terrell, Prompt Neutrons from Fission, LA-DC-6675 (1964). Availability: Dep. (mn); $2.00 (cy), 1(mn), CFSTI.

A survey is given of the present state of knowledge of the spectrum, angular distribution, and number of prompt fission neutrons, as functions of incident neutron energy and individual fragment mass, for low-energy fission.

The energy spectrum of prompt neutrons has been found to be of the same form, nearly Maxwellian, for many different types of fission. It has been shown that this type of spectrum is to be expected on the basis of evaporation from moving fragments, and theoretical predictions of the spectrum agree very accurately with experimental data. Some data are now available on the variation of the neutron spectrum with fragment mass and angle of emission.

Only recently has it become possible to take accurate data on the angular distribution of the neutrons. It appears that the neutrons have the angular distribution to be expected if emitted almost isotropically from the moving fragments, with a possibility that some small fraction are not emitted in this way, but directly from the fissioning nuclide.

Much work has been done on the variation of fission neutron number $\overline{\nu}$ with incident neutron energy for neutron-induced fission. The neutron number increases roughly linearly with energy, with a slope of the order of 0.15 neutron/MeV. There is now evidence that this slope changes somewhat with energy. This change must be associated with other changes in the fission process.

947. E. L. Slaggie and J. T. Reynolds, O^{16} Fast Neutron Cross Sections and Legendre Moments Below 15.0 MeV, KAPL-M-6452 (April 30, 1965). Availability: Dep. (mn); $2.00 (cy), 1(mn), CFSTI.

Fast neutron cross sections and Legendre moments for O^{16} are given at energies below 15.0 MeV. Below 2.6 MeV the elastic cross section and moments are obtained analytically from a fitting formula based on a combined resonance and potential well theory. Moments above 5.0 MeV are obtained from optical calculations. These calculations are described and detailed references are given for the data on which the library is based.

948. J. R. Seibold and A. J. Goodjohn, Shielding Data for the 10-MW EBOR, GAMD-3419 (August 15, 1962). Availability: Dep. (mn); $2.00 (cy), 1(mn), CFSTI.

The EBOR shield has been designed to satisfy the requirements imposed by operational procedures followed in the conduct of experiments in the reactor and in the safe handling of radioactive materials when they are removed from the reactor. Since no information of a developmental nature will be sought in the performance of the shielding, the design proceeded on the basis of adequate safety, ease of fabrication and minimum cost. These functions are served best by the use of ordinary concrete as the basic shield material. In regions of space limitations, a high density concrete or composite shield or iron and ordinary concrete was used.

This report includes discussion of the following main topics:

1. Shielding requirements during normal operation at 10 MW.
2. After-shutdown shield requirements and radiation levels encountered during normal fuel handling, maintenance and inspection.
3. Radiation levels for purposes of estimating radiation damage of critical components.
4. Radioactive contamination and activation of the internal and external coolant systems.
5. Storage facilities for radiactive materials such as fuel element control blades and loop gases, etc.
6. Activation levels of various materials, shield plugs, etc.
7. Auxiliary shielding such as casks, etc.

949. G. W. R. Endres, Neutron Spectrum Measurements at Hanford Work Locations, HW-SA-3525 (CONF-642-33), (May 15, 1964). Availability: Dep. (mn); $2.00 (cy), 1(mn), CFSTI; Gmelin, AED-CONF-64-082-40.

Hanford occupational exposures to neutron radiation are due mainly from two types of neutron sources: production reactors, and plutonium in chemical processing facilities. A knowledge of the incident neutron spectrum is necessary to accurately interpret data obtained from the current nuclear track type personnel neutron dosimeters.

The basic device used to measure the spectrum was a surface barrier sandwich detector with neutron converters of either Li^6 or He^3. The energy deposited in each surface barrier detector, by the reaction products, was added electronically to obtain the initial neutron energy.

The neutron spectra measurements obtained during this study are typical of PuF_4 isotopic sources, shielded plutonium production reactors, and plutonium metal with a significant neutron emission due to spontaneous fission of the Pu^{240} isotope. An explanation of the data analysis is given along with appropriate curves showing the results.

The plutonium fluoride spectrum has a broad peak at about 0.9 MeV, slightly lower than the theoretical peak of 1.2 MeV. The plutonium metal spectrum has a rather sharp peak at about 1.1 MeV. The peak is more pronounced than the Watt fission spectrum. The spectra inside and outside the reactor shielding show how the neutron spectrum is degraded in going through the shield.

950. S. B. Herdad, A. A. Suarez, O. Y. Mafra, D. C. DaCunha Reis, and L. P. DeMoura, Measurement of Fast Neutron Fluxes and Spectra at the IEA 5MW Reactor, IEA-83 (CONF-631104-6), (December 1964). Availability: Dep. (mn).

Fast neutron fluxes and energy distribution were measured in some irradiation facilities at the IEA 5 MW swimming pool reactor. The differential fast neutron spectrum in the energy interval 0.5 to 10 MeV was determined in beam hole number 7, utilizing a simplified proton recoil telescope and nuclear emulsion techniques. Threshold detector techniques are being used to specify fast neutron spectra inside the reactor. Magnesium Sulfate is suggested as a "double" threshold detector, through the reactions $S^{32}(n,p)\ P^{32}$ (E_{eff} = 2.9 MeV) and $Mg^{24}(n,p)\ Na^{24}$ (E_{eff} = 6.3 MeV), providing a simple and low cost way to measure fast fluxes and estimate the shape of the integral fast neutron spectrum. Some preliminary results are presented.

951. W. B. Green, Analysis of Shield Test Facility Experiments Using the Two-Component Method, NAA-SR-MEMO-10559 (October 13, 1964). Availability: Dep. (mn); $1.00 (cy), 1(mn), CFSTI.

The two component method of shield analysis is reviewed and discussed, and the experimental configuration is described; the study indicates that results obtained from removal theory are as good as those obtained using the two component method, for obtaining dose rate estimates at points within about 6 feet of the shield surface; other results are cited which indicate that the two component method is a valuable tool of analysis for dose points greater than this distance.

952. T. N. Krishnamurthi, V. K. Sundaram, and P. P. Damle, Radiation Dose from Volume Sources, AEET/HP/Th-9 (1962). Availability: Dep. (mn).

Exact integral expressions for gamma ray flux at different points from finite cylindrical and spherical sources, having uniform source distribution, are presented in this report. The type of geometry chosen for the evaluation of the flux simplified the integrand to a great extent. The integral expressions are computed for two specific cases of cylinder and sphere using TIFRAC Digital Computer for general values of the parameters involved and the curves are presented.

953. C. Rangarajan, U. C. Mishra, B. Y. Lalit, S. Gopalakrishnan, and S. Sadasivan, Fission Products Data and Its Application in Studying Fallout from Nuclear Weapon Tests, AEET-209 (1965). Availability: Dep. (mn).

Relevant data on the fission products released in nuclear weapons tests are tabulated in this report. A study of the relative concentration of the various fission products is useful in determining the mode of fission and the type of fissile material used.

Tables give the half-life and beta and gamma energies of the important fission products and the yields of the fission products for thermal, fission spectrum and 14-MeV neutrons for U-233, U-235, U-238, Th-232 and Pu-239. The yields as measured in debris from an actual bomb are also tabulated.

954. American Concrete Institute, Concrete for Radiation Shielding, Compilation No. 1 (2nd Edition), (1962). Availability: $4.50, American Concrete Institute, P. O. Box 4754, Redford Station, Detroit 19, Michigan.

This is a compilation of eleven articles reprinted from the American Concrete Institute Journal. The articles appeared in the period 1953-1959. A large amount of information is given concerning the use of concrete for radiation shields. Attenuation and physical properties of various concretes are described with special emphasis on high density concretes.

955. M. B. Wells, Differential Dose Albedos for Calculation of Gamma-Ray Reflection from Concrete, RRA-T46 (December 30, 1964). Availability: Radiation Research Associates, Inc., Fort Worth, Texas.

Differential dose albedos for gamma ray reflection from concrete were derived from Monte Carlo calculations of gamma ray scattering in air and concrete for source energies of 0.6, 1, 2, 4, and 7 MeV. Curves showing the albedo for reflection in the plane of incidence are presented for angles of incidence of $\theta_o = 0^o$, 30°, 45°, 60°, and 75°. At incident energies greater than 2 MeV the computed dose albedos are smaller than those reported by Raso (also based on Monte Carlo) and those given by the Chilton-Huddleston formula. However, comparisons with experimental data for Cs^{137} and Co^{60} gamma rays reported by Jones, Schofield, Haggmark, and Gurney of USNRDL, C. E. Clifford, of DRCL, and Barrett and Waldman of Technical Operations, Inc., indicate that the calculated albedos for these sources agree as well with the experimental data as do calculations based either on the Chilton-Huddleston formula or Raso's Monte Carlo calculations.

956. J. Adamson, A. M. Judd, and N. I. McNair, "Neutron Flux in the Shield of the Dounreay Fast Reactor," J. Brit. Nucl. Energy Soc., 4(2), 119-134 (April 1965).

Neutron-induced reaction rates were measured through the fast reactor shield with the object of testing the validity of using standard bulk shielding calculation methods on DFR. It was found that the measurements did not provide an adequate test because streaming through holes in the shield vitiates its bulk shielding properties. However, on combining the calculations of bulk attenuation and streaming, and making allowance for the difficulty of applying the calculation methods to the complicated geometry of the system, the agreement with the measurements is acceptable.

957. S. Popovics, "An Investigation of the Unit Weight of Concrete," Mag. Conc. Res., 16(49), 211-220 (December 1964).

Experimental results are presented which show the effects of cement content, air content, slump, aggregate grading and specific gravities on the unit weight of fresh and hardened concrete. With a knowledge of these effects an intelligent and resolute selection of the different proportions for a specified concrete density can be made. Formulae are presented which show the relations between the concrete composition and the unit weight of the fresh and hardened concrete. These formulae are applicable even when the mineral aggregate is a mixture of two or more materials of different specific gravities. The form of the equations presented is relatively simple considering the large number of variables involved. Values computed from the formulae show good agreement with experimental results. These formulae thus provide a new and easy method for selecting the most economical concrete composition for a given unit weight of concrete.

958. J. M. Chapman, The Variation of Dose Attenuation of Two-Legged Concrete Ducts with Incident Gamma-Ray Energy, USNCEL TN-707 (April 16, 1965). Availability: U. S. Naval Civil Engineering Laboratory, Port Hueneme, California.

Gamma-ray dose attenuation factors were measured in a 3-foot square and an 11-inch square duct using Au-198 (.412 Mev), Cs-137 (.662 Mev), and Co-60 (1.25 Mev) gamma-ray sources. Attenuation factors for given geometries were compared as a function of the incident gamma-ray energy. It was found that the attenuation factor decreased monotonically with increasing energy.

Measured attenuation factors were compared with values obtained using a computer code based on the albedo concept. It was found that calculated attenuation factors are high by as much as 78 percent for Au-198 in the 11-inch duct.

This work was sponsored by the Defense Atomic Support Agency.

959. L. W. Szymendera, "An Analytical Solution of Multi-group Removal-Diffusion Equations, in Plane Multi-Layered, Reactor Shield System," Nucl. Struc. Eng., 1(5), 476-491 (May/June 1965).

This paper gives the analytical solution of the set of differential equations describing, in the multi-group removal-diffusion approximation of Boltzmann's transport equation, the neutron flux distributions in the multi-layered plane shield of a nuclear reactor. The solution is obtained in closed form of a linear combination of some well-defined auxiliary expression which can be easily integrated and differentiated.

960. E. L. Kaplan, Monte Carlo Methods, UCRL-5120-T (Rev.) (September 1958). Availability: Dep. (mn); $4.80 (ph), $2.70 (mf), OTS.

The paper is concerned with the basic concepts and techniques of Monte Carlo, such as the generation of random numbers, the construction of samples from a given distribution, and techniques for reducing the variance of the estimates. The latter include the expected values method, systematic and stratified sampling, importance sampling, splitting and Russian roulette, conditional Monte Carlo, and correlation and regression. These are discussed with reference to two examples: a system comprising a roulette wheel and three urns, and a simplified neutron shielding problem. For the two-stage sampling in the first example, the inequalities among the variances of ten of the estimates obtained by various methods are investigated.

961. A. L. Frank, R. A. Taylor, Gamma Radiation Characteristics — Angular Distribution Over a Desert Terrain Fallout Field, USNRDL-TR-856 (June 11, 1965). Availability: U. S. Naval Radiological Defense Laboratory, San Francisco, California.

In the summer of 1962, a gamma-radiation spectroscopy experiment, in conjunction with Project 2.14 of the DOMINIC II Sun Beam test series, was carried out in a fallout field at the Nevada Test Site near Mercury, Nev. The main problem was to measure the gamma-ray spectra as a function of angle of incidence. The measurements were made at a height of 4 ft. Data were taken for six azimuthal and ten vertical angles of the detector at 3 days and 9 days after shot time, respectively. Since the azimuthal data were about the same, photon spectra were extracted for only the maximum and minimum intensity angles, which differed by about 10 per cent. Scofield's matrix iteration procedure was used to extract photon spectra from the pulse-height data.

The vertical-angle photon spectra were examined and compared in various ways. A ground spectrum, plotted against the calculations of Bjornerstedt and Nelms and Cooper for the gamma-ray spectra of fission products, showed some similarity with these data. The photon number and energy fluxes were calculated versus vertical angle, and the average energy per photon over the total solid angle was calculated to be 0.436 MeV per photon. Finally, per cent energy flux per steradian as a function of vertical angle was plotted and compared with the similar curve of Mather, et al. Some notable differences were found, especially in the skyshine contribution.

A calculation of ground roughness for the desert terrain was made by the absorbing overlayer model. An overlayer equivalent to about 24 ft of air was found to best account for the ground roughness.

962. L. Hjarne and M. Leimdorfer, A New Method for Predicting the Penetration and Slowing Down of Neutrons in Reactor Shields, AE-185 (NP-15161), (May 1965). Availability: Dep. (mn).

A new approach is presented in the formulation of removal-diffusion theory. The "removal" cross-section is redefined and the slowing-down between the multigroup diffusion equations is treated with a complete energy transfer matrix rather than in an age theory approximation. The method, based on the new approach, contains an adjustable parameter. Examples of neutron spectra and thermal flux penetrations are given in a number of differing shield configurations and the results compare favorably with experiments and Moments Method calculations.

963. A. F. Holloway and E. M. Campbell, "Scattering of Co-60 Gamma Rays Off Concrete," Radiology, 79(1), 126-131 (July 1962).

The scattered exposure rate at several points 90 cm above a concrete floor was measured. The Co-60 source was also 90 cm above the floor. The scattered exposure rate was also measured as a function of height above the floor.

964. C. McDonnell, J. V. Velletri, A. W. Starbird, and J. F. Batter, The Barrier Attenuation Introduced by a Vertical Wall, PSDC-tr-15 (September 1, 1964). Availability: CFSTI, $3.00 (cy), $0.75 (mn).

A series of tests designed to evaluate the attenuation introduced by a vertical wall adjacent to a horizontal field of contamination is described. The variation of the resultant attenuation with height is found to agree well with theory. Actual attenuation values measured are higher than those theoretically predicted by an amount equivalent to approximately eight percent of the wall thickness.

965. C. Jonemalm, "Nomogram for Determining Shield Thickness for Point and Line Sources of Gamma Rays," Nucl. Struc. Eng., 1(3), 324-331 (March 1965).

A set of nomograms is given for the determination of the required shield thickness against gamma radiation. The sources handled are point and infinite line sources with shields of Pb, Fe, magnetite concrete ($\rho = 3.6$), ordinary concrete ($\rho = 2.3$) or water. The gamma energy range covered is 0.5 - 10 MeV. The nomograms are directly applicable for source and dose points on the surfaces of the shield. They can easily be extended to source and dose points in other positions by applying a geometrical correction.

966. E. E. Morris and E. J. Moses, Calculation of Gamma Ray Dose from Rectangular Sources Using Harmonic Analysis, NBS Report 8695 (May 25, 1965). Availability: National Bureau of Standards, Washington, D. C.

Techniques are developed for evaluating the detector response due to radiation from rectangularly shaped surfaces. The angular distribution of the radiation is represented as a series of elementary functions dependent on azimuth and obliquity. Calculations are then performed for specific rectangle shapes and detector positions using these functions. Sample results are given for a plane monodirectional source of 1.25 MeV gamma rays incident on a concrete scattering medium. These results are derived from more extensive tabulations contained in the report.

967. A. J. Budka, A Method for Calculating Secondary Gamma Ray Transmission through Single-Material Slabs, BRL R 1284 (April 1965). Availability: AD 469654, Ballistics Research Laboratories, Aberdeen Proving Ground, Maryland.

A method is presented for calculating gamma ray transmission data for any depth distribution of secondary gamma radiation in plane single-material slabs. The method utilizes a weighting scheme and appropriate monoenergetic secondary gamma ray transmission data. The weighting scheme is based on a polynomial description of secondary gamma ray source strength as a function of slab depth.

968. P. K. Lee and F. C. V. Worman, Integral Gamma and Neutron Measurements on the Kiwi TNT, LA-3304 (May 3, 1965). Availability: $1.00, CFSTI.

Gamma and neutron doses and integral neutron fluxes produced by the Kiwi TNT nuclear destruct event were documented. Dosimetry stations were positioned close to the reactor for measurements of prompt and fission product radiation. In addition, gamma doses resulting from the effluent cloud were monitored at distances out to 50 miles downwind with dosimeter films.

Total rad doses of gamma and neutron radiation were 1.91 x 10^4 rads 50 feet from the reactor and decreased to 46.9 rads at 1,000 feet. Fission product gamma activity was selectively dispersed by the winds and produced an anisotropic gamma dose distribution even as close as 100 feet from the reactor. Gamma doses downwind from the reactor were 250 mR at 8,000 feet, 330 mR at 16,000 ft, 150 mR at 6 miles, and 10 mR at 12 miles.

969. P. J. Aline, Neutron and Gamma Streaming in Void Channels, GEAP-3114 (January 31, 1959). Availability: Dep. (mc); $4.80 (ph), $2.70 (mf), CFSTI.

An expression for the neutron or gamma flux which streams from a void channel parallel to the axis of a cylindrical reactor core has been obtained. Diffusion theory was used to determine the neutron current into the channel in terms of the channel radius and length, assuming an isotropic flux near the surface of the channel. The final expression for the streaming flux depends only upon the magnitude of the flux at the mid-plane of the channel and the channel dimensions. Evaluations of the final expression for typical cases are presented in curves permitting simple computation of specific problems.

For an axial cosine flux distribution the neutron current into the channel is positive over the central 80 percent of its length, becoming negative in the end regions. This effect tends to reduce the magnitude of the streaming flux. The neutron or gamma streaming flux emanating from the channel is characterized by a collimated beam. Its effect on shielding and gamma heating must be investigated in detail for a particular reactor design. The present method yields a streaming flux four to eight times smaller than previous methods.

970. R. B. Walton and R. E. Sund, Delayed Gamma Rays from Fission, AFWL-TR-65-95 (September 1965). Availability: $3.00 (hc), $0.75 (mf), CFSTI as AD 620510.

Studies of delayed gamma rays from fission at early times were conducted using the pulsed beams of neutrons and bremsstrahlung x rays from an electron linear accelerator to produce fissions. Measurements of the energy spectrum of isomeric gamma rays from the neutron fission of Pu^{239} and U^{235} were performed with a NaI detector at a number of time intervals between 50 μsec and 600 μsec. These measurements showed six prominent gamma rays which probably resulted from the decay of three different fission product isomers. The energies of the gamma rays range from 250 keV to 1260 keV and the half-lives from 29 μsec to 78μsec. The intensities of delayed gamma rays from the neutron fission of U-235 and Pu-239 were measured as a function of time between 60 μsec and 30 msec. The data were normalized at 20 msec to the intensities derived from the theoretical work of Griffin and the data of Fisher and Engle. Results of the spectrum and intensity measurements gave a complete description of delayed gamma rays from U-235 (n,F) and Pu-239 (n,F) for the time interval from 50 μsec to 30 msec. The intensities of delayed gamma rays from photo-fission of U-235, Pu-239, and U-238 were measured at early times, starting at 2 μsec after fission. An intense short-lived component with an apparent half-life of a few μsec was observed for all three cases of photofission. Although the photofission data for the various target nuclei are qualitatively similar in the time region extending from 2 μsec into the plateau region, some distinct differences in the shapes and magnitudes of the decay curves do exist. For both U-235 and Pu-239, the intensity of delayed gamma rays in the isomeric time region from photofission is almost the same as that from neutron fission but is appreciably lower in the plateau region.

971. J. Moteff, Introduction to Nuclear Propulsion, Lectures 5 and 6 - Shield Physics, N64-29787 (NASA-CR-52958, GEMP-190d), (March 1963). Availability: $6.60, CFSTI.

A brief introduction to the concepts and nomenclature of shielding technology is given. The emphasis is on problems encountered in nuclear propulsion of rockets.

972. A. Bergstrom and M. Leimdorfer, Gamma Dose Rates from Ground and Steel Activation Following Nuclear Fission Explosions in Air, FOA 4 RAPPORT A 4402-454 (NP-15146) (November 1964). Availability: Dep. (mn).

A series of calculations has been made to display the expected activation gamma dose rate at the air-ground interface induced by neutrons from a nuclear burst in air under different conditions. The removal-diffusion technique was used for the neutron slowing-down calculation. The nuclear explosive was surrounded by varying effective thicknesses of the remainder of the initiating explosive (TNT), ranging from 0 to 50 cm. The burst altitude as well as the ground water contents were also varied, the former from 100 to 1500 m and the latter from 0 to 50 per cent by weight. The results are split up to give separately the dose rate contributions from activated Na, Mn, and Al nuclei. Thermal neutron fluxes are also presented for a number of illustrative cases. A supplementary calculation was made to analyze the gamma dose rate contributions from a steel slab on the ground.

973. D. K. Trubey, S. K. Penny, and K. D. Lathrop, A Comparison of Three Methods Used to Calculate Gamma-Ray Transport in Iron, ORNL-RSIC-9 (October 1965). Availability: Radiation Shielding Information Center, ORNL (Free).

Calculations of the gamma-ray energy spectra, dose rate, and heating in a sphere of iron have been made using the Monte Carlo and S_n methods and the data from a moments method calculation (Goldstein-Wilkins). The source was taken to be isotropic and distributed uniformly within a sphere with an energy distribution proportional to $e^{-1.1E}$, where E is the energy in MeV. The results of all calculations were in very good agreement.

974. John R. Stehn, Murrey D. Goldberg, Renate Wiener-Chasman, Said F. Mughabghab, Benjamin A. Magurno, and Victoria M. May, Neutron Cross Sections, Volume III, Z = 88 to 98, BNL-325 (2nd Edition) Suppl. No. 2 (February 1965). Availability: $3.00, CFSTI.

This supplement to BNL-325 presents new information on neutron cross sections accumulated in the five years since the previous supplement was issued. It supercedes previous issues only when there is new information. This issue covers elements having the atomic numbers 88 to 98.

The format differs from preceding versions in several ways. The more normal size, arrangement of all data by nuclide, and fluid format is likely to make this supplement more convenient for the user.

975. F. C. V. Worman, P. K. Lee, Integral Gamma and Neutron Measurements on the Kiwi B4D-202 and Kiwi B4E-301 Reactors, LA-3286 (July 23, 1965). Availability: $4.00, CFSTI.

This report is an evaluation of the leakage radiations produced by the Kiwi B4D-202 and Kiwi B4E-301 reactors. Integral measurements of gamma dose, neutron flux, and neutron dose were made on all experimental plans that involved criticality. For reactor Kiwi B4D-202, these were Experimental Plan 2, a calibration run; Experimental Plan 3, a power profile or low power run; and Experimental Plan 4, a full power run. Kiwi B4E-301 involved a calibration run, Experimental Plan 3; a low power run, Experimental Plan 4; a full power rune, Experimental Plan 5; and a restart and second full power run designated Experimental Plan 6.

Gamma doses in air on Experimental Plan 4, Kiwi B4D, ranged from 1.12×10^5 rads at 60 feet to less than 20 rads at 1,800 feet. Neutron doses in air varied from 6.43×10^4 rads at 60 feet to 2.64 rads at 3,000 feet. Thermal neutron fluxes declined from 87.2 per cent of the total fast neutron flux at 60 feet to 24.6 per cent at 1,500 feet.

Comparable data for Experimental Plan 5 of Kiwi B4E show the gamma doses in air ranging from 5.93×10^5 rads at 60 feet to 1.93×10^2 rads at 1,800 feet. Neutron doses in air varied from 2.21×10^5 rads at 60 feet to 3.24 rads at 3,000 feet. Thermal neutron fluxes diminished from 123 per cent of the total fast neutron flux at 60 feet to 28.8 per cent at 1,500 feet.

Documentation of the radiation field, in close proximinity to the reactor core was accomplished on Kiwi B4D by using two meridian rings, an inner ring at a 50 inch radius from the core center of the reactor and an outer ring at 90 inches from the same point. On Kiwi B4E only the outer ring was utilized. The meridian rings were used only on the calibration runs.

Radiation measurements were also made at shielded positions in and around the test cell on experimental plans for both reactors.

976. Hironori Ono and Akira Tsuruo, "An Approximate Calculation Method of Flux for Spherical and Cylindrical Sources with a Slab Shield," J. Nucl. Sci. Tech., 2(6), 229-235 (1965).

An approximate calculation method for unscattered flux from spherical and cylindrical sources with a slab shield is presented. The method is based on modifying the source geometry to a form convenient for calculation. The sphere is approximated by a conical fragment of spherical shell, and the cylinder by a columnar fragment of cylindrical shell. The procedure for manual calculation is relatively simple, and the difference between results of approximate and exact calculations is less than + 5% for large sources ($\mu_s R_o \geq 4$), and $\pm$ 15% for small sources ($\mu_s R_o \approx 0.25$). Graphs of the functions for approximate calculation are presented. The method has several advantages over Rockwell's method.

977. A. Tsuruo, M. Shindo, and M. Kawabata, "A Systematic Approach to the Duct Streaming Problem and Some Results to Unscattered Flux Calculation," J. Nucl. Sci. Tech., 2(4), 121-126 (April 1965).

A systematic plan for calculating the flux distribution at the center and near the exit of a straight cylindrical duct is proposed.

An infinite plane source on one side of the shield barrier is assumed to have either an isotropic or a $\cos\theta$ angular distribution. To obtain the flux distribution on the other side of the shield barrier, calculation is performed to separately obtain

(a) the flux coming from the disk source at the entrance of the duct,

(b) the flux coming from the annular source surrounding the entrance of the duct.

Moreover, (1) unscattered flux, (2) single scattering flux, (3) multiple scattering flux are considered separately. These fluxes are compared with that for the case of infinite slab shield without duct.

The ratio of the unscattered flux ϕ_{duct} at the center of the duct exit to the unscattered flux ϕ_{slab} at a point behind the slab, and that of the unscattered flux $\phi_{annular}$ from the annular source to the unscattered flux ϕ_{disk} from the disk source are obtained for plane isotropic and plane cosine angular sources. The above mentioned technique of separate calculation provides a new approach to duct streaming problems.

978. I. Kataoka and K. Takeuchi, "A Method for the Numerical Integration of the Photon Transport Equation in Slab Geometry," J. Nucl. Sci. Tech., 2(1), 30-38 (January 1965).

A procedure for the direct numerical integration of the transport equation in slab geometry is introduced. The angular flux is expressed as a series of Legendre polynomials. The iteration process is utilized for the spatial integration of the transport equation. Prior to applying this method both to neutrons and to photons, a series of studies on the photon penetration problem has been performed to assess its validity. The first code, the EOS-1, was prepared for the NEAC 2206 computer to perform the above mentioned work and also to solve some practical problems of not very large scale.

A comparison of energy buildup factors obtained by the EOS-1 code with those from the moments method and from the Monte Carlo method indicates that the method presented in this paper provides a correct treatment of the attenuation of radiations in a slab. The results also agree well with the energy spectra from the moments method, with the energy spectrum measured in the Bulk Shielding Reactor and with the energy-angular flux distributions obtained with 60-Co and 137-Cs sources behind iron and aluminum plates, both with respect to spectrum pattern and to absolute values.

979. J. S. Cheka, F. W. Sanders, T. D. Jones, and W. H. Shinpaugh, Distributions of Weapons Radiation in Japanese Residential Structures, CEX-62.11 (August 1965). Availability: $1.25, CFSTI.

The Oak Ridge National Laboratory Health Physics Research Reactor, attached to a hoist platform on a 1500-ft tower at the Nevada Test Site, was used in a continuation of studies to evaluate radiation doses of persons exposed to nuclear weapons. As a part of the operation, distribution of radiation in radiation analogs of Japanese residences was studied. Measurements of neutron and gamma radiation were made in single houses and in group configurations. Formulae describing the radiation patterns with respect to house parameters were derived by the use of multiple regression analysis. Nine parameters were found to predict the radiation exposures at the measured points so that 50% of the predictions were in error by less than $\pm$ 6% and 90% of the predictions were in error by less than $\pm$ 17%. These formulae can be used to evaluate individual exposures in Hiroshima and Nagasaki.

980. K. Shure and N. R. Williamsen, Application of Diffusion Calculations to N-16 Specific Activity, WAPD-BT-31, p. 37-42 (1964). Availability: $3.00, CFSTI.

The flux distribution of high-energy neutrons within and close to a reactor core, required for calculation of N-16 specific activity in pressurized water-cooled and moderated reactors, can be determined efficiently through the use of one-group diffusion procedures. Results obtained compare favorably with those from earlier methods. The diffusion procedure is readily adaptable to an arbitrary spatial fission variation in one or more dimensions.

981. T. E. Northrup, "High Density Slag Concrete," Nucl. Struc. Eng., 1(3), 347-351 (March 1965).

A cheap high-density waste slag material with a specific gravity of 3.80 is derived from the lead smelting industries. This material was investigated to evaluate its availability, quality variations, price stability, handling methods and equipment required, and shipping methods and limitations. The slag costs $9.50 per short ton f. o. b. Oak Ridge, Tennessee.

Physical property experiments were performed on the slag to determine its mechanical gradiation, specific gravity, absorption, unit weight, and percentage of voids.

On a pound-for-pound-of-shielding basis, the materials for waste lead slag concrete are 30% cheaper, f. o. b. Oak Ridge, Tennessee, than materials for barytes and magnetite concretes.

982. H. Kakimoto, T. Saigusa and S. Shimizu, "Streaming of Gamma Rays through Metallic Pipes," Bull. Inst. Chem. Res., Kyoto University, 43(1), 22-44 (1965).

The streaming of the 662 keV gamma rays from Cs-137 through steel and lead pipes of various sizes was observed using a NaI(Tl) scintillation detector. The scintillation pulse-height spectra of the streamed rays were measured with different pipes. The whole course of the measurement was performed out of doors with the pipes placed horizontally 2 m above the earth's surface to get rid of the unfavorable scattered rays from the surroundings. To interpret the observed spectra the theoretical calculations were carried out by taking account of the single-scattering of rays in the wall of the pipe. The general features of the observed phenomena could be explained qualitatively by the theoretical results. The effect of the streamed rays scattered twice by the wall of the pipe and also evaluated theoretically and found to be negligibly small. The experimental results and the theoretical considerations are presented in details in the present paper.

983. I. C. Roberts and D. A. Goodwin, Analysis of ASTR Radiation Heating in Aerospace Structural Materials, AFWL-TR-64-152 (February 1965). Availability: $4.00 (cy), $0.75 (mn), CFSTI, as AD 613471.

The heat generated in the material used for shielding and structure constitutes an important problem associated with the use of nuclear energy in aerospace missions. The calculation of the heat generated involves the use of computer codes based on the differential energy spectra obtained by a moments-method solution of the Boltzmann transport equation. This solution yields a modified spectrum at a specified point from a given source point after penetration of a multilayer shield. A summation of the spectra from all the source points gives the total spectrum. Integration of the spectrum over energy, with the use of the proper energy-absorption coefficients gives the heat deposition rate in the test materials.

To evaluate the calculational methods, some typical shielding and structural materials were irradiated by the Aerospace Systems Test Reactor (ASTR) at the Nuclear Aerospace Research Facility (NARF). The materials tested in this experiment were Inconel-X, aluminum, and 6% borated polyethylene. The difference between the calculated and measured data is less than 20% of the measured value for Inconel-X and aluminum but is larger for the borated polyethylene.

984. A. Tsuruo, The Spectrum Matrix Method in Gamma Ray Transmission Problems for Multiple Layers, NSJ-TR-21 (Translated from Nippon Genshiryoku Gakkaishi, 6(2), 91-97 (February 1964)). Availability: Dep. (mn); $1.60 (fs), $0.80 (mf), JCL.

An approximate calculation method in γ ray transmission problems is shown for multiple layers. In this method, the differential energy flux of incident γ rays is expressed by a vector, and the shielding property of each slab is represented by a matrix. The matrix is constructed by a relatively simple procedure, using the data of penetration in infinite media calculated by the moments method and the data of the effect of finite slabs calculated by the Monte Carlo method. The compound shielding property of multiple layers is also represented by the matrix which is obtained by multiplication of the matrix corresponding to each slab. The differential energy flux of transmitted γ ray is given by multiplying the incident vector by the matrix.

The results of the method are compared with those of the Monte Carlo calculation for Pb-H_2O and H_2O-Pb stratified slabs, and also with those of experiments for Pb, Fe, and Al stratified slabs. The comparison shows that this method is useful for all combinations of materials for high incident energy, and when low energy γ rays enter the heavy layer followed by a light or heavy layer.

985. W. A. Coleman, A Determination of Intermediate-Energy Neutron Albedo Data for Concrete Using Monte Carlo, ORNL-TM-1205 (September 1965). Availability: Dep. (mn); CFSTI, $3.00 (cy), $0.75 (mn).

The problem of radiation streaming through multibend air-concrete ducts may be treated by determining the reflected energy-angle spectra for a particular concrete composition, and utilizing these data in an albedo-random walk for a particular geometry. The author has calculated differential energy-angle albedo data for a semi-infinite steel reinforced concrete medium using a series of incident angle and energy conditions. The energy range from 0.5 eV to 200 keV was divided into ten groups ΔE_o^i equally spaced in lethargy. The incident angles were 0°, 45°, 60°, 75°, and 85°. A general purpose Monte Carlo neutron transport code, O5R, was used to obtain a collision distribution for each incident energy band ΔE_o^i and angle θ_o^j. The distribution of reflected neutrons in energy and angle was determined by computing, at each collision, the probability of escape along each of 54 directions (defined by angles θ_ℓ and ϕ_m).

In addition to the differential energy angle albedo the following studies were made for intermediate energies:

1. The radial dispersion of reflected neutrons from the point of incidence.
2. The distribution in depth and energy of intermediate energy captures.
3. The distribution in polar angle θ_ℓ and depth of those neutrons which were degraded below 0.5 ev before escaping.

The reflected singly scattered spectra calculated analytically agreed (within statistics) with the Monte Carlo estimate of the singly scattered spectra. A comparison study showed the Monte Carlo results to be in good agreement with those obtained by using the S_n method.

986. J. P. Nichols, Calculations of Concrete Shields for Fission Sources, ORNL-TM-1167 (June 17, 1965). Availability: Dep. (mn); $1.00 (cy), $0.50 (mn), CFSTI.

Calculations were made of the spatial radiation dose rates from point isotropic fission sources in normal concrete and seven concretes with high iron ore content. The basic tools used were four IBM-7090 computer codes: RENUPAK for fast neutrons, DTK for neutrons of low energy, SDC for primary gammas, and SAGE for capture gammas.

987. H. Wagner and H. Henssen, Shielding Calculations for the OM-Ship Reactor, ORNL-tr-548 (1964). Availability: Dep. (mn). (Translated from EUR-1888.d)

A shielding arrangement with a minimum weight is to be determined for a given core, where the requirement must be satisfied that the total dose rate at the outside of the shield amounts to about 100 mrem/h. Iron-water combinations are to be considered exclusively, the division of which into layers of various thicknesses can be varied. In addition to the gamma-radiation produced by (n,γ)-processes in the shield, the contribution of the core gamma-radiation is also to be taken into consideration. Furthermore, the heat production by absorption of gamma-quanta and inelastic neutron radiation in the individual shielding layers are to be calculated.

988. A. Etemad, Penetration of Thermal Neutrons in Empty Cylindrical Canals Surrounded by a Hydrogenous Medium, ORNL-tr-569 (December 1963). Availability: Dep. (mn); $7.60 (fs), $2.45 (mf), JCL. (Translated from EIR-Bericht Nr. 60).

The thermal neutron flux in empty cylindrical canals surrounded by a hydrogenous medium is the object of a theoretical study resulting in the development of a method of calculation. This method is based, on the one hand, on the hypothesis that equilibrium exists between thermal and fast neutrons, and on the other hand, on the application of diffusion theory in the diffusing medium, and of transport theory at the interior of the canal.

Measurements are given of the longitudinal thermal flux at the interior of two canals of different dimensions within the SAPHIR reactor pool and opposite its core. A comparison is then made between these measurements and the calculated values obtained by the method proposed. The agreement obtained is most satisfactory; the maximum difference between the measured and calculated value is of the order of 22% for one of the canals and of 14% for the other.

Measurements of the radial distribution of the thermal neutron flux both at the interior and at the exterior of the canals are also given, along the axis of both the epithermal and fast neutron fluxes.

989. F. W. Krueger and U. E. Michaelis, Radiation Streaming through Ducts and Voids in Shields, ORNL-tr-815 (1965). Availability: $3.00 (fs), $0.75 (mf), CFSTI. (Translated from Kernenergie 8(3), 137-162 (Mar. 1965)

A summary of the simplified schemes for the calculation of radiation transport through openings, ducts, voids and inhomogeneities in shields is presented.

990. F. M. Tomnovec and J. M. Ferguson, Neutron-Field and Induced-Activity Measurements — Operation BREN, CEX-62.50 (September 1965). Availability: $1.00, CFSTI.

Operation BREN (Bare Reactor Experiment, Nevada) was an experiment that used a 1500-ft tower and a bare (no shielding) fast nuclear reactor patterned on "Godiva" to simulate a nuclear weapon detonated at various heights above the ground. Certain characteristics of the neutron field from this unshielded reactor were measured. The activation of gold, manganese, sulfur, cadmium-covered gold, and cadmium-covered manganese was determined as a function of distance from the reactor and of depth in the ground. The data show how those parameters, important for calculating neutron-induced activity in the soil, vary as a function of the height of the source and the slant range from the source.

991. E. J. Aalto, "A Comparison of Shielding Calculations with Absolute Measurements in Massive Shields," Nucl. Appl., 1(4), 359-374 (Aug. 1965).

Extensive neutron and gamma attenuation measurements have been performed in magnetite and ordinary concrete up to a depth of 2m to collect reference data for a study of the accuracy attainable by shield calculation methods. Experimental facilities and the configurations studied are described accurately enough to enable the reader to test his own methods against these measurements. Great weight has been laid upon absolute accuracy and a thorough error analysis.

The fluxes measured are compared to those calculated by an earlier version of the British 18-group removal method (RASH B_3), by an improved removal method (NRN) developed at AB Atomenergi, and by numerical integration of Boltzmann equation (NIOBE). Calculations with the newer methods give fluxes that are generally within a factor of 2 of the true values. Any greater accuracy seems to be difficult to obtain in practice in spite of possible improvements in the mathematical solution of the transport problem. The greatest errors originate in going from the true geometry to that used for calculation, in the uncertainty of material properties in the case of concrete, and in approximations and inaccuracies of radiation sources.

992. E. A. Dukleth, J. M. Raphael, B. Wedell, P. D. Bush, and A. P. Donaldson, Design and Construction of the High Density Concrete Shielding for the ETR, AECU-3655 (May 1957). Availability: Dep. (mc), $9.30 (ph), $3.60 (mf) CFSTI.

Economic studies showed that high density concrete made with magnetite aggregate was the most feasible choice from a cost and construction standpoint. Except for a few precautions, this concrete can be placed in essentially the same manner as conventional concrete.

993. H. M. Epstein, D. A. Dingee, and J. W. Chastain, A Study of the Primary Shield for the PRDC Reactor, BMI-APDA-622 (April 15, 1957). Availability: $6.30 (ph), $3.00 (mf), CFSTI.

Temperature distributions, irradiation effects, stacking arrangements, voidage, and economics for the borated-graphite shield of the PRDC reactor were investigated.

Of the shield systems considered, four are reported here. System I contains 30 in. of 1 per cent borated graphite, with either ordinary graphite or a cement as a filler for the remainder of the volume. The maximum temperature at the flex plate in this sytem was calculated to be 500 F. Systems 2 and 3 consist of 2 in. of 5 per cent borated graphite near the core vessel and 1/2 in. of Boral at the primary-shield tank. A filler material of carbon blocks is used in System 2 and graphite in System 3. The calculated maximum temperatures were 700 F and 350 F, respectively. System 4 consists of a laminated structure of Boral and graphite near the primary-shield and carbon-block filler. It was calculated to have a maximum temperature of 600 F at the flex plates. The maximum temperature at the flex plates recommended by APDA is 500 F.

Energy storage and radiation damange were found to be within permissible limits in all four systems. However, these conclusions are based on experimental data from the Hanford reactor in which the neutron-energy spectrum differs considerably from the PRDC spectrum.

A porosity of less than 740 cu ft is required in order that a sodium leak from the core vessel does not expose the core. The voidages in any of the systems mentioned above is about 400 cu ft excluding absorption effects. These are believed to be small.

The systems containing Boral were found to be less expensive than the ones using only borated graphite. Over-all material costs range between $230,000 for Boral systems and $350,000 for borated-graphite systems.

994. G. T. Western, R. C. Baird, F. L. Gibbons, and J. R. Williams, The Interactions of 14.6 MeV Neutrons in Iron, Lithium-7, and Carbon, AFWL-TR-64-140 (March 1965). Availability: $4.00 (cy), $1.00 (mn), CFSTI, AD 614451.

A time-of-flight spectrometer for the measurement of fast-neutron energies has been constructed and used to measure the energy and angular distributions of 14.6 Mev neutrons scattered from Fe and C. The neutron differential cross sections for inelastic scattering from Fe were measured at $\theta_{CM} = 50^{\circ}$, 70°, and 115°. The neutron differential cross sections for elastic scattering from C and for inelastic scattering from the first level in C were measured at $\theta_{CM} = 47^{\circ}$ and found to be 49.6 ± 3.6 and 21.8 ± 1.7 mb/ster, respectively. Total cross sections for Fe and C were measured as 2.49 ± 0.05 and 1.30 ± 0.02 barns, respectively.

The energy and angular distributions of inelastically produced gamma rays from Fe and Li^7 were measured using two different types of gamma spectrometers. The differential scattering cross sections for production of the 0.845- and 1.24-MeV gamma rays and for gamma rays in the energy interval $1 < E\gamma < 10.5$ Mev from Fe were measured for nine angles of emission. The differential cross section for inelastic scattering from the 0.477-Mev level of Li^7 was measured as 7.73 ± 0.54 mb/ster at $\theta_{CM} = 90^{\circ}$.

The calculated energy levels for Li^7 and Fe^{56}, using the collective model, are in good agreement with experimental values. The strong-coupling approximation is shown to be valid in both nuclei.

995. W. E. Thompson and F. C. Engesser, Gamma Rays Resulting from Interactions of 14.7 MeV Neutrons with Magnesium, Aluminum, Calcium, Titanium and Iron, USNRDL-TR-861 (June 1965). Availability: U. S. Naval Radiological Defense Laboratory, San Francisco, California.

The prompt gamma rays resulting from the interaction of 14.7 MeV neutrons with the nuclei of the elements magnesium, aluminum, calcium, titanium, and iron, have been investigated. Cross sections for the production of many of these gamma rays have been measured by observing them with a carefully shielded detector during neutron bombardment of a given elemental sample. It is found that the $n,n'\gamma$ process predominates and that gamma rays resulting from the deexcitation of the lowest energy levels, with one or two notable exceptions, are generally most prominent. Other processes including $n,p\gamma$, $n,\alpha\gamma$, and $n,d\gamma$ are also observed. Some of the most important gamma rays are found to have the following elemental production cross sections (in millibarns per steradian at 90 deg.): 1.37 MeV gamma ray from Mg, 30.9 $\pm$ 3.1; 1.81 MeV gamma ray from Al, 13.7 $\pm$ 2.0; 3.73 MeV gamma ray from Ca, 9.0 $\pm$ 1.4; 0.99 MeV gamma ray from Ti, 51.9 $\pm$ 5.2; and 0.85 MeV gamma ray from Fe, 57.6 $\pm$ 5.8.

996. A. Horsley, Neutron Cross Sections of the Proton in the Energy Range 0.0001 eV - 20 MeV, AWRE O-23/65 (July 1965). Availability: 4s. 0d. HMSO.

Recommended values are given for neutron total and partial cross sections of H-1 in the energy range 0.0001 eV to 20 MeV. The available experimental data are supplemented where necessary by estimates based on nuclear theory. Details of energy and angular distributions are given so that the data are complete for the purposes of neutronics calculations.

The recommended data in this report have been written on punched cards (3 A/H data, data file number DFN-211, for the cross sections of free hydrogen atoms, and 3 B/H data, data file number DFN = 212, for the cross sections per hydrogen atom of a water molecule) as part of the UKAEA Nuclear Data Library.

997. S. O. Moore, Optical Model Analysis of Elastic and Inelastic Scattering of Neutrons by Aluminum in the Mev Region, BNL 927 (June 1965). Availability: $1.00, CFSTI.

This report is a continuation of a previous study by E. H. Auerbach and S. O. Moore of the elastic and inelastic scattering of neutrons by heavy elements in the MeV region. Here we extend the work in the same energy range to a study of aluminum, a light mass deformed nucleus. The Hauser-Feshbach formalism using transmission coefficients determined for a complex, spin-dependent, diffuse-edge potential with surface absorption has been used in the calculations.

998. P. Zigman and J. Mackin, "Early Time Decay of Fission Product Mixtures — II. γ-Energy Release and Ionization Rates Following Thermal Neutron Fission of U-235," Health Phys., 5(1/2), 79-84 (April 1961).

Photon spectra at various times from 1 sec to 5 hr after instantaneous thermal neutron fission of U-235 are listed. From these are computed total γ-ray energy release rates and theoretical air ionization rates 3 ft above an infinite plane uniformly contaminated with fission products. Equations which provide a good fit to the rate curves within the estimated accuracy of the data are (t in sec): $\Gamma(t) = 2.78t^{-1.23} - 2.41t^{-1.45}$ MeV/sec per fission and $I(t) = [1.41t^{-1.23} - 1.23t^{-1.45}]\ 10^{-8}$ r/hr per fission/ft^2. Comparisons of integrated doses calculated from the photon spectra with doses derived from the familiar $t^{-1.2}$ function show substantial differences for times after fission less than 1000 sec.

999. Philip C. Fisher and Leon B. Engle, "Delayed Gammas from Fast-Neutron Fission of Th-232, U-233, U-235, U-238, and Pu-239," Phys. Rev., 134(4B), B796-B816 (May 25, 1964).

Results of an experiment are presented in which the energy and time dependence of gammas emitted after instantaneous fast-neutron-induced fission of Th-232, U-233, U-235, U-238, and Pu-239 were measured. Gammas having energies between 0.12 and 6.5 MeV were detected during five short time intervals between 0.2 sec and 45.0 sec after fission occurred. The measured spectra and seventeen bin distributions of the absolute number of photons/fission-sec-MeV and intercomparisons and integrations over energy and time of these distributions are given. The rate of energy emission from the products of Th-232 + n was found to be 4.5 times higher than that from Pu-239 + n during the first time interval after fission. Delayed gamma spectra from the products of Th-232 + n are noticeably harder than those from Pu-239 + n during all five time intervals measured. The several uranium isotopes' behavior is intermediate to that of the products of Th-232 + n and Pu-239 + n in both rate of energy emission and hardness of energy spectra. The significant differences in the measured characteristics of the delayed gammas from the five fissionable isotopes permit the isotopes to be arranged in a sequence. Approximate position in the sequence depends on $\bar{z}$, the average displacement of the initial fragment distribution from fission. It appears possible to predict approximate values of delayed neutron yield, which Keeping has shown to also depend on $\bar{z}$, of other fissionable isotopes on the basis of delayed gamma measurements.

1000. R. E. Maerker and F. J. Muckenthaler, Differential Fast Neutron Albedos for Concrete, Vol. I. Descriptions of Monte Carlo Calculation and TSF Experiment and Comparison of Calculated and Experimental Results, Vols. II-VI. Results of Monte Carlo Calculations for Incident Angles of 0, 45, 60, 75, and 85 Degrees, ORNL-3822, Vols. I - VI (October 1965). Availability: Vol. I - Dep.mn; CFSTI $2.00 (cy), $0.50(mn); Vol. II-VI - Dep. mn; CFSTI, $6.00 (cy), $1.25 (mn) each.

This report is one of a series of six reports presenting experimental and calculated differential fast-neutron albedos for concrete. The experiment was conducted at the ORNL Tower Shielding Facility and consisted of measuring neutrons reflected at various angles from a concrete slab which had been placed in a highly collimated beam of neutrons from the TSR-II (four angles of incidence). The calculations were performed with the 05R Monte Carlo computer code and covered six different source energy bands for each angle of incidence of a "gun barrel" source of monodirectional neutrons. Volume I describes the experimental and calculational procedures, compares experimental and calculated results, and gives a general description of the machine output from the calculations. Volumes II-VI present the machine output in detail, each volume representing a different angle of incidence as follows: Vol. II, 0 deg; Vol. III, 45 deg; Vol. IV, 60 deg; Vol. V, 75 deg, and Vol. VI, 85 deg. If the nature of the output in Vols. II-VI is not clear, reference should be made to the output description in Vol. I.

These results are also reported in Nucl. Sci. Eng., 22(4), 455-462 (Aug. 1965).

1001. S. Tomoeda, W. E. Kreger, M. B. Hastings, and W. G. Miller Gamma-Ray Penetration into the Compartments of a Light Aircraft Carrier, USNRDL-TR-343 (July 7, 1959). Availability: U. S. Naval Radiological Defense Laboratory, San Francisco, California.

The penetration of gamma rays into compartments of an aircraft carrier from three different radioisotopes has been measured. Co-60 (1.25 Mev), Cs-137 (.66 Mev), and Ir-192 (.34 Mev) sources were exposed at numerous points on the Flight Deck and Landsverk air-wall equivalent dosimeters were used to measure dose in two general areas of the ship and at three different deck levels. Average dose rates and transmission factors have been calculated by integration of the data to represent the case of uniform contamination at 1 curie per square foot. The average dose rate for the different deck levels range from 28 r/hr to 0.93 r/hr for Co-60; 12 r/hr to 0.20 r/hr for Cs-137; and 4.4 r/hr to 0.06 r/hr for Ir-192. Two types of transmission factors have been computed. The ratios of the average dose rate found 3.5' off each deck to the computed dose rate in air 3.5' above the Flight Deck run from 0.242 to 0.01 for the 3 isotopes in the "A" Section and 0.196 to 0.0017 for the "B" Section. The ratios of the average dose rate found 3.5' off each deck to the computed dose rate in air 3.5' off the same deck run from 0.307 to 0.037 for the 3 isotopes in the "A" Section and 0.459 to 0.008 in the "B" Section.

1002. S. Tomoeda, M. B. Hastings, and W. G. Miller, Experimental Data of Gamma-Ray Penetration into the Compartments of a Light Aircraft Carrier, USNRDL-TR-412 (April 6, 1960). Availability: U. S. Naval Radiological Defense Laboratory, San Francisco, Calif.

The penetration of gamma rays into compartments of an aircraft carrier from three different radioisotopes has been measured. Co-60, Cs-137, and Ir-192 sources were exposed on the Flight Deck and dosimeters were used to measure dose in two general areas of the ship at three different deck levels. Data of dose distribution are presented for the three sources used and for the different source points considered. Diagrams indicating source positions, detector positions, deck and compartment wall thicknesses and main structural member specifications are also presented.

1003. B. W. Shumway, S. Tomoeda, G. E. Plummer, W. G. Miller, and M. B. Hastings, The Dose Distribution Within an Aircraft Carrier Exposed to Uniform Co-60 Contamination on the Flight Deck, USNRDL-TR-466 (October 4, 1960). Availability: U. S. Naval Radiological Defense Laboratory, San Francisco, California.

Experimental measurements of specific dose rate were made at 242 stations on six decks of a light aircraft carrier with its Flight Deck "uniformly contaminated" with Co-60. From these data general rules regarding dose distribution were drawn, and the effects of various structural elements were studied. Transmission factors were obtained which agreed fairly well with values computed from earlier data collected with point sources of Co-60. Depth dose measurements at eight locations showed the radiation to maintain its highly penetrating character.

1004. L. G. Haggmark, Ship Shielding Factors - Computational Method Compared to Experimental Results, USNRDL-TR-514 (June 7, 1961). Availability: $1.60 (fs), CFSTI as AD 259677.

A previously developed computational method is used to obtain the ratios of the gamma-radiation dose rates at given interior points of a light aircraft carrier to the dose rate 3.5 feet above the center of the Flight Deck which is assumed to be covered uniformly with Co-60. The results of the computations are compared with values determined by an experiment on the ship where the Flight Deck was uniformly contaminated with Co-60.

The comparison is made for two portions of the ship which have different structural components. The results of the comparison show the computed values to be higher than the experimental values and generally within 28 percent in the relatively "open" portion of the ship and generally worse than 50 percent in the more heavily compartmentalized portion of the ship.

1005. E. L. Kaplan, Monte Carlo Schemes for Neutron Migration Using Importance Sampling and Splitting, UCRL-5918-T (1961). Availability: Dep. (mc), $1.60 (fs), $0.80 (mf) CFSTI.

Monte Carlo methods using importance sampling and splitting are proposed for estimating the effectiveness of neutron shields. As usual, the aim is to increase the number of particles that go through or around the shield, this increase being compensated by a decrease in the weights of these particles. With suitable modification, the ideas are applicable to other Monte Carlo problems.

One way of using splitting and Russian roulette is to require that the weight of each particle be approximately equal to a specified function of the coordinate; one such function is given in equation (3). The proposed form of importance sampling uses a cross section for particle collisions that is the product of the neutron cross section (or its square root) with a linear function of the cosine of the angle between the particle direction and some preferred direction. The probability density of this cosine itself is taken proportional to the reciprocal of this same linear function. This function is adjustable on both a regional and an overall basis.

1006. E. Tochilin and B. W. Shumway, Flux and Spectrum of Simulated Fission Neutrons in the Vicinity of an Opening in a Neutron Shield, USNRDL-TR-448 (July 19, 1960). Availability: U. S. Naval Radiological Defense Laboratory, San Francisco, California.

Neutron dosimetry relating to a partial-body exposure facility is described in detail. The spectrum was similar to that from U-235 fission. Measurements were made with sulfur threshold detectors, pocket ionization chambers, photographic emulsions, and a modified long counter. Presence of the partial-body shield increased the neutron dose in the exposure region by about 16 per cent. Spectral measurements showed that thicknesses of up to 12 in. of paraffin moderator produced no changes in the spectrum above 0.6 Mev, the lower limit to which measurements were made.

In contrast with physical measurements, biological studies indicated that the presence of the partial-body shield did not increase the dose level. In view of the consistent physical results, further biological experiments should be performed.

1007. G. T. Western, Experimental Determination of Neutron Flux Distributions in Slabs and of Emergent Secondary Gamma Rays, NARF-62-6T (FZK-9-176) (June 1962). Availability: $9.60 (fs), CFSTI as AD 282753.

Neutrons from the Aerospace Shield Test Reactor, positioned within the Outside ASTR Tank, were deflected 90° with a carbon reflector onto shield materials of linear and borated linear polyethylene, $ZrH_{1.95}$, Inconel X, BeO, Be, and stainless steel (type 309). The emergent flux of various monoenergetic capture gamma rays were experimentally determined for several thicknesses of each material with a special 3-crystal NaI(Tl) anti-coincidence total-absorption gamma-ray spectrometer. Also, the spectrum of neutrons incident upon the shielded materials and the flux distribution of thermal neutrons within the material were measured.

Neutron spectral and flux distribution data were also obtained for the shield materials positioned in the direct beam of neutrons from the ASTR.

Additional data on the relative significance of (n_{th},γ) and $(n,n'\gamma)$ gamma rays produced in various materials, when exposed to the direct beam of neutrons from the ASTR, were accumulated.

1008. G. A. Luce, B. S. Burton, and D. M. Wheeler, Multilayer Shield Experiment II, NARF-58-29R (June 1958). Availability: General Dynamics, Fort Worth, Texas.

The first tests of multilayer shield mockups in the Outside Tank Facility have been completed. The tests, part of a continuing joint investigation by Convair and General Electric, were designed to provide information for design of the shield for the XMA-1 power plant.

In the experiment, slabs of various shielding materials were assembled in different arrangements into shield-sector mockups. These assemblies were than placed in the Outside Tank Facility and subjected to radiation from the Ground Test Reactor. Three types of measurements were taken:

1. external measurements, with dosimeters and BF_3 counters;

2. internal measurements, with activation foils and chemical dosimeters placed between shield layers; and,

3. external measurements of gamma spectra emitted.

Results of the internal measurements indicate that Hevimet is quite opaque to thermal neutrons, and that gamma rays from thermal and resonance neutron capture in Hevimet may contribute significantly to the external gamma dose rate. Hevimet capture gamma rays were found, in some cases, to cause a significant number of photoneutrons in mockups containing beryllium as well as those containing beryllium oxide.

The effect of hydrogenous backing of non-hydrogenous neutron-shielding materials was found to be relatively small. For fast neutrons and gamma rays, the external dose rates were found to be independent of the location of the hydrogenous material within the non-hydrogenous column. The external thermal neutron flux was reduced most when the non-hydrogenous material was placed at the outer end of the mockup.

Many of the dose measurements indicate that gamma rays from capture in the OTF structural materials contribute significantly to the external dose. This conclusion is substantiated by the observed gamma-ray spectra. Typical gamma spectra are shown in the appendix.

1009. D. G. Anderson, Co-60 Gamma Dose Measurements Inside Cylindrical Shields, NARF-57-60T (December 1957). Availability: General Dynamics, Fort Worth, Texas.

The transmission and attenuation of air and ground scattered gamma radiation through cylindrical crew shields has been investigated to provide information for the evaluation of calculational techniques. Gamma dose rates were measured along the centerline of two cylindrical crew shields of identical shapes, one half the size of the other. The larger cylinder corresponded in size to the engineered crew shield.

Various side-wall rubber thicknesses, shadow shield angles, and source-detector separation distances were investigated for both cylinders. Measurements were made for air and ground scattered radiation.

The results of this investigation indicate that cavity size is not an important factor in crew shield penetration.

1010. H. B. Vanfleet, F. B. House, and M. L. Coffman, Analysis of Penetration of Co-60 Gamma Rays Through Rubber Slabs, NARF-56-40T (November 1956). Availability: General Dynamics, Fort Worth, Texas.

The data on the penetration of gamma radiation through rubber and lead is of importance in the design calculation of crew shields. This report analyzes the available data on the penetration of gamma radiation through square slabs of rubber and lead. Penetration is examined as a function of thickness of rubber, with and without lead backing, and as a function of angle of incidence.

The mean free path for Co-60 gamma rays in rubber has been obtained. In addition, buildup factors have been calculated for over two mean free paths in rubber.

The penetration for normal incidence has been calculated by the use of both single-scattered theory and buildup theory. This provides a comparison of the two methods of calculation. Buildup theory seems to hold the greatest promise for shielding calculations.

1011. B. E. Morris, N. D. King, and H. G. Carter, Measurement of Radiation Heating in Aerospace Structural Materials, NARF-63-8T (FZK-9-187) (September 16, 1963). Availability: $6.60, CFSTI.

Radiation heating measurements were obtained for several aerospace structural materials at the Nuclear Aerospace Research Facility. The Aerospace Systems Test Reactor, positioned in the Outside ASTR Tank, was the source of radiation.

Heating measurements were obtained throughout the following five configurations: (1) 4 inches of T-61 weldable aluminum, (2) 4 inches of Inconel-X, (3) 4 inches of 6% borated polyethylene, (4) 4 inches of 6% borated polyethylene preceded by 1 inch of lead, and (5) 4 inches of 6% borated polyethylene preceded by 1/4 inch of boral preceded by 1 inch of lead. Gamma-ray and neutron dosimetry measurements were taken by means of a 4-cc ion chamber, gamma-ray chemical dosimeters, sulfur pellets, and gold foils. Heating rates were calculated for Configurations 1 through 4, and both measured and calculated data for all configurations are presented in graphic and tabular form.

A method of correcting for heat leakage in the borated polyethylene transducer is outlined, and the required empirical parameters, obtained from a bake-oven experiment, are given along with suggested changes which should be incorporated in future experiments of this type.

1012. R. L. Mather, Neutron Distributions Near an Air-Soil Boundary as a Function of Soil Water Content, USNRDL-TR-530 (June 9, 1961). Availability: U. S. Naval Radiological Defense Laboratory, San Francisco, California.

A computational method developed in an earlier report, USNRDL-TR-478, applies to the neutron field from an air-burst weapon. In this report the neutron flux as a function of energy and depth in the vicinity of the air-soil boundary at large distances from ground zero is investigated for several soil-water contents. Several aspects of the results are presented which conform with expectations. The results are briefly compared with the results from cyclotron experiments.

1013. G. E. Plummer and W. G. Miller, Geometry and Barrier Attenuation Generated by a Vertical Slab Exposed to a Plane Co-60 Source, USNRDL-TR-677 (September 27, 1963). Availability: $4.60, CFSTI.

A solution to the problem of how one may determine the dose received by a detector that has a vertical barrier interposed between it and a horizontal infinite radiation field has been attempted. Steel barriers of effective mass thickness 0 to 73 lb/ft^2 were employed. The detector was collimated so that regardless of its distance from the barrier it always saw a constant circular area on the barrier.

The infinite radiation field was simulated by a traveling Co-60 source that was pumped through an array of plastic tubing that covered a 100-ft semicircular area. An extrapolation to the infinite field case was attempted. The final results are expressed as ratios of dose received by a collimated, shielded detector relative to the free-in-air dose received by an unprotected detector 3 ft above the plane. The experimental points are compared to a family of theoretical curves.

1014. R. L. Mather, Residual Gamma Radiation Fields from Induced Soil Activities, USNRDL-TR-240 (April 9, 1958). Availability: U. S. Naval Radiological Defense Laboratory, San Francisco, California.

The residual gamma radiation from neutron-induced activities in the soil around ground zero following a nuclear detonation is an important hazard in some situations. Two empirical profiles of activation vs depth in the soil are assumed and the characteristics of the direct gamma radiation field above the soil are computed. Scattered radiation is estimated to contribute about 86 percent as much hazard as the direct radiation but is not computed.

The characteristics for the induced radiation field are contrasted with the characteristics of a fallout radiation field and marked differences are found in the angular distribution (more uniform), variation of total intensity with height (less), radius of the importantly contributing area (less), and the radiation protection afforded by a hole (much less).

1015. F. M. Tomnovec, The Activation of Sodium, Manganese, and Aluminum in Nevada Test Site Soil, USNRDL-TR-500 (February 21, 1961). Availability: $2.60 (fs), CFSTI as AD 259991.

In a laboratory experiment employing 12-MeV protons on beryllium as a neutron source, the neutron activation and resultant distribution of radioactive Na^{24}, Mn^{56}, and Al^{28} in Nevada Test Site soil is shown to be similar to actual data taken at a nuclear weapon field test. This establishes the laboratory experiment as a valid predictor for the neutron activation of soil elements from atomic weapon neutron fields. This report indicates that large amounts (10 to 20 percent) of the activation of Na^{24}, Mn^{56}, and Al^{28} are produced by neutrons above the cadmium cutoff threshold. From our data we were able to calculate the resonance integrals of sodium, manganese, and aluminum.

1016. J. Adamson, N. I. McNair, and A. M. Judd, Gamma Dose Rate in the D. F. R. Rotating Shield Calculation and Measurement, TRG Report 845(D) (September 26, 1964). Availability: UKAEA, Dounreay Experimental Reactor Establishment, Thurso, Scotland.

This report describes a study of the gamma radiation along the central axis of the rotating shield of D. F. R. and is complementary to the earlier report on the neutron flux in the shield. The object was to find the most important sources contributing to the gamma radiation at steady power and to test the application to the Fast Reactor of the recommended methods of calculation described by Avery *et al*. As in the case of the neutron shielding problem the complicated structure of the shield makes the calculation difficult and somewhat uncertain, but reasonable agreement is found between calculation and experiment.

1017. S. S. Rosen, APPR-1 Research and Development Program; Shielding Experiments and Analysis, APAE-35 and Supplement 1 (October 15, 1958). Availability: $3.00 CFSTI.

This report describes a shielding evaluation program conducted on the APPR-1 at Fort Belvoir, Virginia. An attempt was made to cover all the major areas of shield design involved in a reactor system of the APPR type.

The important conclusions are summarized below.

(1) The APPR-1 shielding was conservatively designed and there are many places where it can be reduced with a saving in both cost and weight.

(2) The neutron distribution through the APPR-1 shield is described with sufficient accuracy by modified two group theory to warrant its use in future shield designs.

(3) The IBM 650 code which was written to calculate the gamma flux distribution through the primary shield gave results sufficiently close to the measured values so that it can be used with confidence in future design work.

(4) Calculation of the N^{16} activity in the primary coolant with the use of the activating flux above 10 MeV and a 16.3 mb cross section gives results which are within the experimental error of the measured value.

(5) Calculation of the dose rate due to the N^{16} activity was done by an available IBM 650 code. It gave results that were considerably below the measured values within the vapor container but that agreed sufficiently well with the measured values outside the vapor container. This code can be used for secondary shield design with confidence.

(6) All the mechanisms involved in the accumulation of corrosion product activity in the primary coolant cannot be adequately described analytically. The prediction of the demineralizer activity is only within a factor of three.

1018. F. G. Moote and C. H. Obrist, Shielding Measurements at the SM-1 Reactor (June 1961), APAE-35 (Supplement 2) (March 16, 1962). Availability: Dep. (mc); $4.60 (fs), $1.55 (mf), CFSTI.

Neutron flux and gamma radiation measurements through the SM-1 primary shield were made at the startup of Core II in June 1961. They extend previous measurements (APAE-35) both vertically and horizontally in the primary shield and in the rod drive pit. Dose rate measurements on spent fuel elements under water are also reported.

1019. R. H. Peterson and H. W. Ege, Use of the Discrete Ordinate Method for Describing the Transport of Fast Neutrons Through Iron Slabs, BRL-Memo-1374 (October 1961). Availability: $3.60 (fs), CFSTI as AD 269831.

The use of the discrete ordinate method for solving a simplified form of the neutron transport equation as a semi-empirical description of dose transmission through iron slabs was investigated. The basic information to which the model was fitted consists of a detailed set of Monte Carlo results which takes into account the incident energy and direction, the slab thickness and the exit direction. For 0.16-MeV neutrons, the fit was found to be excellent in all respects. At energies of 0.85, 1.2, 2.0, and 4.0 MeV, the total dose transmission was found also to give a good fit but its angular density and the total dose reflection were somewhat in error.

1020. C. L. Schlemm, A. E. Anthony, Jr., and Z. G. Burson, Scattered Gamma Radiation Measurements from a Co-60 Contaminated Field, AFSWC-TN-59-6 (AD-210485) (January 1959). Availability: $19.80 (ph), $6.80 (mf), CFSTI as PB-149987.

A 90° sector of a circle of 604-foot radius was contaminated with Cobalt-60 point sources to an approximate level of 1.72 millicuries per square foot, as a means of simulating an area approaching an infinite contaminated field.

Radiation measurements were made in an open hole with a PDR-27A radiac instrument and Dupont 553 film to determine the percentage of the total radiation which is due to scattered radiation (skyshine). It was found to be approximately 10 ± 4 percent. These measurements were made at selected points from approximately 36 inches above to 25 inches below ground level.

Data were obtained in a miniature basement to determine the attenuation of the skyshine by various thicknesses of concrete on the roof.

Additional data were obtained to determine the energy of the scattered radiation as a function of the angle above the plane, using a total absorption gamma spectrometer. The unshielded effective energy for dose considerations ranged from 0.90 to 0.12 MeV for elevation angles from 10° to 80° above the plane.

All the instrumentation was placed in the vicinity of the apex of the radiation field.

1021. S. M. Ermakov, V. G. Zolotuklin, and V. A. Kon'shin, "Monte Carlo Calculations of the Energy and Angular Distributions of Gamma Quanta Penetrating a Plane-Parallel Layer of Finite Thickness," Theory and Methods of Nuclear Reactor Calculations (G. I. Marchuk, Editor), Consultants Bureau, pp. 119-122 (1964), New York.

The space-energy distribution of γ quanta escaping from a layer of polyethylene, CH_2, was investigated. Computations were carried out for γ quanta with initial energies $E_0 = 0.5$, 1.25, 2.5, and 7 MeV incident at angles $\alpha = 0^\circ$, 15°, 60°, 75°, 90°, measured from the normal, on a layer of polyethylene with thicknesses of 10, 15, and 25 cm.

1022. Sh. S. Nikolaishvili, "Space-Energy Distributions of Fast Neutrons in Hydrogen," Theory and Methods of Nuclear Reactor Calculations (G. I. Marchuk, Editor), Consultants Bureau, pp. 53-58 (1964), New York.

The problem of the energy distribution of fast neutrons at various distances from a point isotropic source in an infinite hydrogenous medium is one of the simplest problems of transport theory. It plays an important role in the development of various approximate methods for the solution of the more complicated problems in neutron physics.

This paper is devoted to a solution of this problem by the moments method. Each of the integral equations for finding the space-angle moments of the function for the differential neutron flux is solved by a numerical method based on a representation of the kernel of the equation in a degenerate form. The desired function is determined approximately by means of the first three even moments. In addition, calculated results are given for monoenergetic sources with energies of 2, 4, 5, 8, 10, 12, and 14 MeV at distances up to 90 cm. The density of hydrogen nuclei is assumed equal to the partial density of hydrogen in water.

1023. M. J. Schumchyk, R. E. Rexroad, M. A. Schmoke and W. C. Hampton, Fallout Shielding Characteristics of Combat Vehicles, M60 and M48A2 Tanks, M59 and M113 Armored Personnel Carriers, NDL-tr-45 (AD 473334), (September 1965). Availability: Nuclear Defense Laboratory, Edgewood Arsenal, Maryland.

The object of this work was to determine residual gamma radiation protection factors for the M60 and M48A2 tanks and the M59 and M113 armored personnel carriers.

A residual radiation field with uniform density of contamination was simulated by means of a cobalt-60 point source circulation system. Mannequins of laminated masonite and wood were placed in the crew and passenger positions of the vehicles. The gamma-exposure rate was measured with ionization chambers at the surface of these mannequins and in the free field. The ratio of the free-field exposure rate to the exposure rate within the vehicle is the protection factor.

1024. Radiation Effects Information Center, Radiation-Effects State of the Art, 1964-1965, REIC Report No. 38 (June 30, 1965). Availability: Radiation Effects Information Center, Battelle Memorical Institute, Columbus, Ohio.

Developments in the field of radiation effects on electronic components including semiconductors, polymeric materials, lubricants, flotation fluids, hydraulic fluids, structural metals and alloys, ceramics, space radiation environment, dosimetry, and ceramic and metallic fuel materials are reviewed. Programs currently being conducted in radiation effects are briefly given for each section of the report.

1025. E. Aalto, R. Sandlin, and R. Fraki, "Measured and Predicted Variations in Fast Neutron Spectrum when Penetrating Laminated Fe-D_2O," Nukleonik, 7(7), 421-26 (August 1965).

Variations of the fast neutron spectrum in thin regions alternating Fe and D_2O have been studied using threshold detectors (In (n,n′), S(n,p) Al(n,α)). The results have been compared to those calculated by two shielding codes (NRN and Rash D) of multigroup removal-diffusion type.

The absolute fast spectrum calculated in our rather complicated configurations was found to agree with measurements within the same accuracy (a factor of two) as did the thermal flux. The calculated spectrum is slightly harder than the measured one, but the detailed variations (covering the range 1:5) in the form of the spectrum when penetrating Fe agree with observations to within 15 - 20%. In and Al activities were found to be proportional to the integrated flux over 1 MeV throughout the whole configuration, while S showed the least proportionality.

1026. R. L. French, "Gamma-Ray Energy and Angular Distributions Above Fallout," Health Physics, 11(5), 369-83 (May 1965).

Detailed data on the gamma radiation environment in air 3 ft above fallout uniformly distributed on a smooth ground surface were calculated using Monte Carlo techniques. The basic data give the energy and angular distributions of the direct-beam and the scattered photon flux above monoenergetic plane isotropic gamma-ray sources for ten different source energies from 0.10 to 3.5 MeV. For each source energy, the flux arriving at the receiver in each of ten energy intervals between 0.04 and 3.5 MeV and eighteen equal receiver angle intervals between 0° (nadir) and 180° (zenith) was calculated. The dose arriving through each 10-degree interval of receiver angle was also computed. To illustrate their application, the results of the monoenergetic calculations were weighted and combined to obtain energy and angular distribution of the photon flux from fallout of four different ages: 1.12 hr, 23.8 hr, 4.57 days and 9.82 days. Comparisons with measured and previously calculated data, both for actual fallout fields and for simulated infinite plane sources, show favorable agreement.

1027. A. Tsuruo, "Unscattered Flux from Spherical and Cylindrical Sources with Shell-Shaped Shields," J. Nucl. Sci. Tech., 2(7), 261-269 (July 1965).

The effect of shell-shaped shield surrounding a spherical or cylindrical source is studied. The shapes of different shields with a common attenuation parameter μt is classified into six types according to the positions of the inner and outer surfaces with respect to the source and the detector. The exact equations and their approximate formulas for unscattered fluxes from sources with shell-shaped shields are derived. The difference between the results from the approximate and the exact calculations is less than ± 2% for dense and/or large sources ($\mu_s R_o \gtrsim 4$), and ± 20% for small and/or thin sources ($\mu_s R_o \approx 0.25$). Functions expressing the effect of shield shape are also derived. This effect is very large, and sometimes reaches a factor of from 2 to 10. The results of calculation for spherical sources are compared with those by Ascoli, and it is found that his approximate method has a large error for large sources and/or dense shields ($\mu R_o \gtrsim 4$). The results of calculation for cylindrical sources agree well with those by Taylor and Obenshain, and also with Osanov's experiments.

1028. W. G. Miller, A Semi-Empirical Determination of Dose Rates Within an Aircraft Carrier Enveloped by a Cloud of Co-60, USNRDL-TR-906 (September 15, 1965). Availability: U. S. Naval Radiological Defense Laboratory, San Francisco, California.

Dose rates are calculated for seven detector locations aboard an aircraft carrier enveloped by a cloud of Co-60. The assumed cloud is a hemisphere of 500-ft radius which is centered at a detector and lies upon the horizontal plane through the detector. It contains 1 curie/ft^3 of Co^{60}. The bases of the calculations are dose rates obtained at the detector locations by irradiation of the ship from many distant positions with a point source of Co^{60}.

1029. V. K. Daruga, I. I. Lazutkin, A. N. Nikolaev, V. K. Sakharov, B. I. Sinitsyn, and S. G. Tsypin, "Passage of Neutrons through Carbon and Through an Iron-Carbon Mixture," Soviet J. Atomic Energy (English Transl.), 17(1), 754-757 (July 1964).

In planning nuclear reactors, an important problem is the construction of radiation shielding. For most shielding materials theoretical calculation is difficult, owing to the lack of full and detailed information on the interaction cross sections with neutrons. The theoretical and experimental data on the passage of neutrons through carbon, obtained with small thicknesses, are contradictory. The present paper studies the spatial distribution of neutrons of various energies after passage through carbon layers from 10 to about 130-150 cm thick. Passage through an iron-carbon mixture is also studied. The latter is comparatively cheap, efficient enough for gamma-ray or neutron shielding, possesses good thermal stability, and may find wide use as shielding for projected reactors. The shielding is composed of alternate layers of iron and carbon. The iron constitutes 37.4% of the volume.

1030. R. M. Derbowka and D. V. Cormack, "Spectra of the Multiple-Scattered Radiation in a Medium Irradiated with Cobalt 60 Gamma Rays," Brit. J. Rad., 38(453), 653-60 (September 1965).

A scintillation spectrometer with a 2 x 2 in. NaI(Tl) crystal has been used to measure the spectrum of the scattered radiation in a "hardboard" phantom irradiated with the beam from a ^{60}Co teletherapy unit. Spectral measurements were made of the radiation travelling at various angles to the primary beam at depths of 2, 5, 10 and 15 cm and for beam areas of 25 and 100 cm^2. At each angle the spectrum of once-scattered photons from a thin scatterer was also measured. These spectra of once-scattered radiation were fitted to the peaks of the spectra from the phantom and subtracted to obtain the spectra of multiply-scattered photons only. The resulting multiple-scatter spectra were corrected for distortions in the crystal and integrated with respect to the scattering angle. These integrated spectra were then compared with spectra derived from the calculations of Bruce and Johns (1960). Although the measured and calculated curves show the same general features there appear to be differences in the width and position of the low-energy peak of the spectrum. Uncertanities in both the measurements and the calculations appear to be great enough to account for the differences.

1031. D. A. Kozhevnikov, "Sapce-Energy Distribution of Neutrons in a Homogeneous Moderator," Soviet J. Atomic Energy (English Transl.), 17(1), 722-735 (July 1964).

The moderation of neutrons in a homogeneous medium with an arbitrary isotope composition is investigated using the method of spherical harmonics. The dependence of the mean free path of neutrons on lethargy is taken into account approximately. At short distances from the source, the distribution function does not depend on the approximation order in using the method of spherical harmonics, while it coincides with the results of the age approximation; a more accurate estimate of the limits within which the age approximation can be applied is given. At large distances, the spatial dependence of the distribution function is completely defined by the spatial distribution of primary neutrons, while the angular distribution and the energy spectrum of moderated neutrons do not depend on the coordinates. The results obtained by calculating the spatial distribution of indium neutrons in water for monoenergetic and polyenergetic sources are given.

1032. L. K. Zoller, Neutron Activation Foil Data OTT II and III, XDC 60-1-123 (December 22, 1959). Availability: General Electric, Cincinnati, Ohio.

This report presents the foil measurements made inside configurations tested during OTT II (3/24/58 to 4/11/58) and OTT II (5/5/58 to 6/6/58). These data, presented both graphically and in tabular form, include aluminum, sulfur, and copper responses. The graphic presentation of the data is given as foil activity at reactor shutdown time (counts/min-gm-watt) as a function of detector location within the test configuration (shown schematically). The tabular presentation gives each foil activity at reactor shutdown (count/min-gm-watt) and the corresponding neutron flux at reactor shutdown (neutrons/cm^2-sec-watt) for each foil location (card number) in the configuration.

The configurations were various arrangements of LiH, Be, BeO, boral, Hevimet, plastic, stainless steel, and uranium.

1033. E. White, OTT IV and V Foil and Chemical Dosimeter Data, XDC 60-6-61 (May 4, 1960). Availability: General Electric, Cincinnati, Ohio.

The material presented in this report represents neutron activation foil and chemical dosimeter measurements taken during OTT IV (10/20/58 to 11/19/58, 1/6/59 to 1/15/59) and OTT V (5/1/59 to 6/15/59) test sequences. The particular configuration geometry and foil activity (counts/min-gm-watt), corrected to reactor shutdown time, are shown graphically as a function of detector (foil) location within the test configuration (card number). The gamma ray dose rates (mr/hr-watt) are similarly shown as a function of detector (chemical dosimeter) location.

The configurations were various arrangements of lead, plastic, LiH, BeO, Be, Mo, stainless steel, ZrH, Hevimet, Inconel X, and borated materials.

1034. C. M. Cialella and N. D. King, The Shielding Properties of Simple Enclosure Shields, BRL-MR-1663 (June 1965). Availability: Ballistics Research Laboratory, Aberdeen Proving Ground, Maryland.

The transmission of gamma rays and neutrons through several stratified cubic and spherical shielding enclosures was experimentally determined. The sources included gamma rays from cesium-137 and cobalt-60 and monoenergetic neutrons of energies 1, 5, and 16.9 Mev. The shields included 3 cubes of different overall sizes but with the same wall thicknesses and materials and 2 spheres, one with the same wall thickness and materials as the cubes.

Preliminary comparisons of the results from the suite of shields is included.

1035. Thomas Jaeger, Principles of Radiation Protection Engineering (Translation by Lawrence Dresner), McGraw-Hill Co., $12.50 (September 1965).

This survey of radiation protection engineering presents the theory and data involved in shielding ionizing radiation and preventing the spread of radioactive contaminants. The book is a brief resumé through 1959 to many topics.

The book details:

Atomic and nuclear behavior including the interaction of ionizing radiation with matter,

Radiation detection with explanations of detectors and scintillation counters,

Biological consequences of radiation emphasizing man's exposure limit,

Sources of particle emission such as radioactive isotopes and reactor cores,

Application of the above to protective apparatus.

Final chapters cover separation facilities, particle accelerators, reactor accidents, containment of reactor systems, and other topics.

The preface states that the principles are presented from the view of the civil engineer.

(RSIC)

1036. Division of Radiological Health, Radiological Health Handbook, PB 121784R (September 1960). Availability: $3.75, CFSTI.

This handbook is a collection of data in tabular, graphical, and nomogram form. Along with a glossary and physical, chemical, and mathematical data, there is a great deal of radioisotope, decay, and radiation protection data. A disintegration scheme table is included.

1037. E. Aalto, R. Fraki, and R. Sandlin, "Measured and Predicted Variations in Fast Neutron Spectrum in Massive Shields of Water and Concrete," Nucl. Struc. Eng. 2(2), 233-242 (July 1965).

The absolute magnitude, and the variations in form, of the fast neutron spectrum during deep penetration (0.8 - 1.1 metre) in massive shields of water, ordinary and magnetite concrete have been studied by using threshold detectors (In (n,n′), S(n,p) Al(n,α)). The results have been compared with predictions by two rigorous (NIOBE, Moments method), and two non-rigorous (multigroup removal-diffusion) shielding codes (NRN, RASH D).

The absolute results predicted were in general within 50% of the measured ones, i.e., showed as good or better accuracy than thermal and epithermal flux predictions in the same small-reactor configurations. No difference in accuracy was found between the rigorous and non-rigorous methods.

The changes in the relative form of the spectrum (indicated by variations in the (Al/S) and (In/S) reaction rate ratios and amounting to factors up to 3 - 4 during a one metre penetration in water) were rather accurately (within 10 - 30%) predicted by all of the methods.

The photonuclear excitation of the 335 keV level used for detecting the In (n,n′) reaction was found to distort completely the In results in water at penetrations > 50 cm.

1038. E. Aalto, "Measured and Predicted Neutron Fluxes in, and Leakage through, a Configuration of Perforated Fe Plates in D_2O," Nucl. Struc. Eng. 2(1), 142-150 (July 1965) (AE-198).

Detailed neutron flux measurements have been performed in a 48 cm thick configuration of thin regions of Fe and D_2O (30% of total thickness Fe), both when the Fe regions are massive and when they are penetrated by 15 cm dia. D_2O channels. It was found that the total leakage of neutrons through the configuration was increased by 25% in the latter case.

It is shown that this increase and the detailed flux distribution on a duct axis can be satisfactorily predicted by a one-dimensional shielding code when the duct is calculated as a pure D_2O layer and a radial buckling term is used for the < 1 eV neutrons when penetrating the Fe regions. Another calculation through the massive part of Fe is to be performed in the usual way. It is believed that this 2-component method is usable in a wider range of similar configurations.

A calculation with regions where the Fe and D_2O ducts have been homogenized into a single material overestimates the increase in leakage, and the relative error is greater than that in the 2-component calculation.

1039. L. H. Scherer, OTT-II Chemical Dosimeter Data, XDC 60-3-33 (February 11, 1960). Availability: General Electric, Cincinnati, Ohio.

This report contains gamma ray data taken with chemical dosimeters placed at several positions inside the slab configurations tested in the OTT during test series III, June 5, 1958, through July 6, 1958.

1040. E. A. King, Limited-Energy Neutron Sources Produced by the Decay of Excited Nuclei, IDO-17115 (September 1965). Availability: $3.00, CFSTI.

Three phenomena are considered as sources of neutrons. These are:

Radionuclide delayed neutron emission

Nitrogen-17 and fission product delayed neutron precursors are considered and several systems for their use in MTR are proposed.

Photo-neutron reactions

Radionuclide gamma ray bombardment of beryllium and deuterium is reviewed. Several reactions not previously considered in the literature are included.

Neutron-capture gamma rays are considered as gamma ray sources. Several (n,y) and (γ,n) targets are considered. Systems for their use in MTR are proposed.

Alpha-neutron reactions

Reactions using alpha particles emitted from radionuclides are considered as sources. Some usable radionuclide alpha particle sources are listed, nuclides with usable reaction thresholds are tabulated, and a bibliography of alpha-neutron sources is included.

1041. G. R. Crocker, D. T. Wong, Gamma-Emission Data for the Calculation of Exposure Rates from Nuclear Debris, Volume II. Induced Activities, USNRDL-TR-888 (August 13, 1965). Availability: Dep. (mn), CFSTI, $3.00 (cy), $0.75 (mn).

Photon energies and photon abundances have been compiled and summarized for some induced activities which may result from nuclear events. The data are presented in tabular form, including photon energies and abundances for the gamma rays, X rays, and beta particles emitted per disintegration. A list of multipliers is also presented for converting activities of the radionuclides to infinite-plane exposure rates.

1042. M. R. Fleishman, P. S. Mittelman and E. S. Troubetzkoy, Optimum Shield Design, UNC-5017 (Part B),(October 15, 1962). (Confidential).

The shield optimization techniques described in Part A of this report are applied to design a shield for the MCR reactor.

1043. R. H. Ritchie and V. F. Anderson, Some Monte Carlo Results on the Penetration of Neutrons from Weapons in an Air-over-Ground Geometry, ORNL-3116 (July 12, 1962). (Secret).

Monte Carlo calculations have been carried out in order to obtain neutron distributions at various distances in air from monoenergetic neutron sources. These results have been golded with theoretical leakage spectra from various devices. The dose and activation fluxes resulting have been compared with experimental data obtained during the detonation of these devices.

1044. C. Cialella, D. Hendricks, and J. Miller, Attenuation of Gamma Rays by an M-48 Tank, BRLM-1176 (November 1958). (Secret).

The over-all attenuation of gamma rays by tanks and other armored vehicles has been measured in field tests of nuclear weapons. Controlled measurement of the directional dependence of the attenuation of the incident radiation was lacking.

In this study four ionization chamber detectors (placed in the driver, gunner, loader, and commander positions) simultaneously measured the transmitted radiation for each position of the radioactive source. Measurements were made with a two-curie cobalt-60 gamma source placed at fifteen degree azimuthal intervals around the tank and at fifteen degree intervals of elevation from zero to ninety degrees. A means of extrapolating these data for comparison with data obtained in nuclear weapons field tests is submitted.

1045. E. P. Blizard, V. R. Cain, F. H. Clark, L. B. Holland, R. E. Maerker and H. E. Stern, Determination of Radiation Intensities in Missile Silos Due to a Weapon Burst: Calculations and General Calculations, ORNL-TM-253 (August 30, 1962). (Secret).

The results of calculations of the neutron and gamma-ray dose rates in shielded missile silos under certain weapon burst conditions are given.

1046. C. F. Johnson, Radiological Armor Program Methods and Data Compilation, Volume 2. Radiation Environments, FZK-200-2 (August 31, 1965). (Secret).

This document is Volume 2 of an initially projected seven-volume series which compiles data, method, and techniques useful under the Radiological Armor Program for the U. S. Army Tank Automotive Center (ATAC), Warren, Michigan. The methods and data included in this compilation were extracted from documents which, since 1959, report the results of work performed at the GD-FW Nuclear Aerospace Research Facility (NARF). This volume deals with the three disparate radiation environments which are associated with the explosion of a nuclear weapon and which must be defined before efficient radiological shields can be designed. These three environments are (1) the radiation at the source, i.e., the leakage spectra from particular weapons and from various experimental and theoretical mock-ups; (2) the radiation in an infinite air medium at various distances from fission and 14-Mev sources; and (3) the radiation in an air medium perturbed by the vehicle and the ground in the vicinity of the vehicle. Also included in the third environment is the residual ground radiation in the blast vicinity due to ground activation and fallout. Analyses of the three environment types are presented as well as the measured and calculated parametric data and a discussion of the methods utilized to obtain these data. The data are given in 138 tables and 24 figures.

1047. W. A. Hehs, R. J. Prevost, D. M. Wheeler, Radiological Armor Program Methods and Data Compilations, Volume 3. Slab Penetration and Reflection Data, FZK-200-3 (July 30, 1965). (Secret).

This document is Volume 3 of an initially projected seven-volume series which compiles data, methods, and techniques useful under the Radiological Armor Program for the U. S. Army Tank Automotive Center (ATAC), Warren, Michigan. The methods and data included in this compilation were extracted from documents which, since 1959, report the results of work performed at the GD-FW Nuclear Aerospace Research Facility (NARF). This volume gives slab transmission and reflection data generated by computation and experiment. The presentation is organized in the three general categories of radiation sources: (1) broad-beam sources, (2) monodirectional sources, and (3) secondary-gamma-ray sources. Data in the first category are presented in a format that readily permits evaluation of shield-material thickness deviations and/or substitutions on shield effectiveness. Measured data in the second category are included in this volume to validate data produced by the Slab Penetration and Reflection Calculation (SPARC procedure). Transmission probabilities are presented from approximately 700 SPARC problems. Secondary gamma rays produced in shield materials by thermal-neutron capture and fast-neutron inelastic scattering are considered in the third category. A tabulation and discussion of secondary- gamma-ray data from various experiments and calculations are presented, and application and evaluation of the data are given when possible.

1048. B. E. Morris and J. R. Stokes, Radiological Armor Program Methods and Data Compilation, Volume 4. Theoretical and Experimental Analyses of Shielded Crew Compartments, FZK-200-4 (August 31, 1965). (Secret).

This document is Volume 4 of an initially projected seven-volume series which compiles data, methods, and techniques useful under the Radiological Armor Program for the U. S. Army Tank Automotive Center (ATAC), Warren, Michigan. The methods and data included in this compilation were extracted from documents which, since 1959, report the results of work performed at the GD-FW Nuclear Aerospace Research Facility (NARF). This volume presents the initial design methods and procedures and the pertinent results of the Radiologically Protected Pod (RPP) test and analysis program. The relaxation-length concept applied to the line-of-sight attenuation model was used in all the initial shield-design and evaluation computations. A description is given of this method and a variation thereof (shield optimization) that were used to design the RPP, although this method has, for the most part, been replaced by computer procedures which provide a more rigorous physical simulation of the radiation transport phenomena. The remainder of this volume (the largest part) describes the RPP simulated-weapons test at the Oak Ridge National Laboratory Tower Shielding Facility and the actual-weapons test during Operation SUNBEAM (Shot Small Boy) at the Nevada Test Site. The data, measured and calculated, are presented in 76 tables and 44 figures.

1049. J. Greenborg and M. J. Berry, Analysis of the N Reactor Primary Shields, BNWL-135 (September 1965). (Confidential).

Neutron flux distribution in three groups (fast, epithermal and thermal) and gamma ray dose distribution in the N reactor reflector and primary shield were measured and compared to calculations. Agreement was within 10% for all neutron flux in the reflector, fast neutron flux and gamma ray dose in the shield. Agreement was within about a factor of two for thermal and epithermal flux in the shield. These distributions extend over a reduction in flux of 10^{-8} for gamma radiation and 10-12 for fast neutron flux.

Power normalized neutron fluxes were measured by absolute counting of irradiated sulfur, gold and cadmium covered gold foils. Gamma dose was measured with calibrated ion chambers. These detectors were spaced from the core-reflector interface to outside edge of the primary shield and were irradiated during the course of the reactor startup and power ascension tests. All power levels were known during irradiation providing absolute power normalization.

The shield calculations were accomplished with the removal-diffusion theory code MAC (18 removal groups, 31 diffusion groups). Calculations were normalized to calculated core-reflector interface fission density based on measured core power distribution. Experimental data reduction and analytical procedures are discussed.

1050. L. Hjarne and M. Leimdorfer, "A Method for Predicting the Penetration and Slowing Down of Neutrons in Reactor Shields," Nucl. Sci. Eng., 24(2), 165-174 (1966).

A new approach is presented in the formulation of removal-diffusion theory. The removal cross section is redefined and the slowing down between the multi-group diffusion equations is treated with a complete energy-transfer matrix rather than in an age theory approximation. The method, based on the new approach, contains an adjustable parameter. Examples of neutron spectra and thermal flux penetrations are given in a number of differing shield configurations, and the results compare favorably with experiments and Moments-Method calculations.

1051. A. B. Chilton, "A New Variant to the Exponential Transformation Technique in Monte Carlo Shielding Calculations," Nucl. Sci. Eng., 24(2), 200-208 (1966).

Monte Carlo calculations of deeply penetrating radiation fields are improved by use of a transformation involving the factor e^{-cz}. The optimum value of c generally appears to be at or near the value for total attenuation coefficient for the source radiation, but there are some difficulties in choosing such a value of c. This work describes and illustrates an approach that allows c to be made equal to μ_o, and the resulting transformed equation is handled in a simple fashion. The technique is encoded into a FORTRAN program called FETMOC, and illustrative problems are solved. Results are given in terms of buildup factors. Two-way energy spectrum and directional distribution at arbitrary points are obtained, also. All results are compared with previously reported calculations to the extent possible.

1052. D. K. Trubey, A Survey of Empirical Functions Used to Fit Gamma-Ray Buildup Factors, ORNL RSIC-10 (1966). Availability: Radiation Shielding Information Center, Oak Ridge National Laboratory, P. O. Box X, Oak Ridge, Tennessee.

The use of simple functions to estimate gamma-ray buildup factors is reviewed on the basis of simplicity and accuracy. Analysis shows that the best fitting form is the 4-term polynomial fit with coefficients published by Capo. For most purposes, however, a function proposed by Berger is recommended since it is as simple to use as the linear form and is essentially as good as the more complicated forms.

Tables of coefficients are given for the linear, quadratic, Berger, and Taylor fitting functions. Some typical graphs of the functions are also given.

Use of the Berger and polynomial functions is demonstrated for the case of exponential source distributions in plane geometry and the use of buildup factors in finite geometry is briefly described.

1053. Francis H. Clark, The Exponential Transform as an Importance-Sampling Device -- A Review, ORNL-RSIC-14 (1966). Availability: Radiation Shielding Information Center, Oak Ridge National Laboratory, P. O. Box X, Oak Ridge, Tennessee.

The exponential transform is reviewed, with emphasis on its use as a guide to effective importance sampling in the solution of the Boltzmann equation by Monte Carlo methods. Contributions of various workers are assembled, along with numerical results. Special consideration is given to approximate forms and to effective practical methods. Problems related to negative effective cross sections, tracking across discontinuities, directional biasing in inhomogeneous media, and high variance in back-scattered components are specifically treated.

1054. K. D. Lathrop, "Use of Discrete-Ordinate Methods for Solution of Photon Transport Problems," Nucl. Sci. Eng., 24(4), 381-388 (1966).

To permit numerical solution of photon transport problems by the method of discrete ordinates, an anisotropic scattering approximation and a multigroup cross-section preparation recipe are selected. The incorporation of the anisotropic scattering approximation in a discrete-ordinates transport-theory code is described. Results of discrete-ordinate calculations are compared to Monte Carlo and moments-method computations in three test problems. Flux values and leakage percentages in the different methods of solution are found to be in excellent agreement, even when a relatively low-order (four to six terms of a Legendre polynomial expansion) anisotropic scattering approximation is used in the discrete-ordinates method. In the test problems considered, the discrete-ordinates method is (computationally) nearly an order of magnitude faster than the other methods.

1055. V. K. Daruga, A. N. Nikolaev, V. K. Sakharov, S. G. Tsypin, I. I. Lazutkin, D. M. Pinkhasik, and B. I. Sinitsyn, "Investigation of the Space-Energy Distribution of Neutrons from the BR-5 Reactor in an Iron-Ore Medium," Soviet J. At. Energy, English Transl., 17(1) 757-760 (1964).

The present article discusses the possibility of using the relatively inexpensive iron-ore medium with a high content of iron and light elements (oxygen) for the shielding of nuclear reactors. This medium, which can readily be installed, is capable of withstanding high temperatures without considerable changes in its properties.

Standard beneficiated iron-ore concentrate with the following composition was used as the initial material: ~ 60% by weight Fe; ~ 30% by weight O_2;8-10% by weight Si, Mg, Ca, and Al, ~ 1% by weight Mn, Pb, Cu, Ti, and C. Small amounts of binding additions were introduced in the concentrate in order to endow it with suitable structural properties. The final product was characterized by good thermal conductivity and thermal stability.

1056. V. P. Mashkovich, B. I. Sinitsyn, A. N. Nikolaev, and S. G. Tsypin, "Attenuation of the Flux of Fission Neutrons in Iron-Water Mixtures," Soviet J. At. Energy, English Transl., 17(1), 760-761 (1964).

The present work examines attenuation of fast, thermal, and intermediate neutrons in iron-water mixtures with more than 75 vol. % iron. The measurements were made with the B-2 shield-testing assembly of reactor No. BR-5. The unidirectional disc source had a total collimation angle of about 5 deg. The fast-neutron spectrum was measured photographically. It was found to coincide, within the experimental error, with the fission spectrum at energies > 3 MeV, and to be softer than the latter at lower energies. The results show that, at a concentration of 18% water, the attenuation coefficients for fast, thermal, and intermediate neutrons are equal.

1057. V. K. Daruga, D. S. Pinkhasik, S. G. Tsypin, A. N. Nikolaev, and B. I. Sinitsyn, "Investigation of Fast-Neutron Penetration Through Sodium," Soviet J. At. Energy, English Transl., 17(2), 855-856 (1964).

In the present study we have determined the relaxation lengths of neutrons with energies of more than 0.5 MeV in a 1360 x 1860 mm sodium prism. The values determined vary from 28 cm ($\geq$ 7 MeV) to 40 cm ($\geq$ 0.5 MeV).

1058. A. Tsuruo, M. Kawabata, and M. Shindo, "Single Scattering Streaming of Gamma Rays Through Cylindrical Ducts," J. Nucl. Sci. Technol., 2(9), 325-330 (1965).

An exact equation, for use in the calculation of the gamma ray streaming after single scatterings in a cylindrical duct, and its approximate formula, are presented. The dose of single scatterings at the exit of a cylindrical duct from a disk source at the entrance can be approximated by empirical functions.

The coefficients k_1 and k_2, as well as the constants s_0 and s_1 vary with source energy and wall materials. The values for lead, iron, concrete, tin, magnetite concrete, graphite and water are given in graphical representation for an energy range from 0.5 to 10 MeV.

A study is made of the distribution of the arrival probabilities of photons, corresponding to the positions of scattering points for the photons from the source to the duct exit. It was found that the effective scatterings contributing to the flux at the exit occur mainly in the shallow part of the duct wall, near its surface.

1059. A. Tsuruo, "Multiple Scattering Streaming of Gamma Rays Through Cylindrical Ducts," J. Nucl. Sci. Technol., 2(10), 406-409 (1965).

A semi-analytic Monte Carlo calculation has been applied to the duct streaming problem. The first positions of collision of photons are selected systematically in a certain volume surrounding the duct, and further scattering positions determined by the random walks of the photons. The expectation value that the photons arrive at the duct exit from their scattering point are integrated over the volume.

The multiple scattering doses through ducts surrounded by a lead and water shield are presented. The ratio between scattered and unscattered dose is about 40% for duct in water when $E_0 = 1$ MeV and $L/R_0 = 10$; the ratio descreases with increase of E_0 and L/R_0. It also decreases with increasing atomic numbers of the shield materials. A large part of the scattered dose consists of once scattered photons. The statistical deviation of the calculated dose is about 5% for 6,000 ~ 8,000 histories, when the relative dose attenuation is of the order of 10^{-6}.

The buildup factors of gamma rays from a point isotropic source for a slab shield are also obtained using the above method; the results agree well with those by the moments method.

1060. C. B. Braestrup, "Shielding Design for Cobalt-60 Teletherapy Installations," Nucl. Struc. Eng., 2(1), 134-141 (1965).

Discussion of the principles of radiation shielding design for cobalt-60 teletherapy installations. The requirements for radiation protection are outlined, and the influences of general layout and operational aspects on the design of the shielding for teletherapy rooms are indicated. Shielding design examples are given.

1061. K. W. Foster, Gamma Shielding Requirements for Plutonium-238 and Polonium-210 (Revised), MLM-1315 (1966). Availability: $1.00, CFSTI.

Gamma shielding requirements were computed for plutonium-238 and polonium-210. Literature values were used for the gamma energies, their relative abundances, and for the mass absorption coefficients of these decay components in various shielding materials. A point source and isotropic radiation were assumed; self-absorption was not considered. For plutonium-238 the greatest shielding thicknesses were required to attenuate the 760-keV component. The following shielding thicknesses are required to reduce the dose rate from a bare 1-kg plutonium-238 source to less than 2.5 mr/hr at 1 m: 0.85 cm lead, 2.30 cm iron, 0.45 cm uranium, and 10.6 cm concrete.

The gamma radiation intensities were computed for 100-w, 1-kw, and 10-kw polonium-210 heat sources. The thicknesses of uranium required to attenuate the 0.803-MeV gamma radiation from each of the above sources to 1000, 100, 10.0, and 1.0 mr/hr at 1 m were calculated.

1062. H. Borella, J. Jacovitch, and Z. Burson, Evaluation of the Fallout Protection Afforded by Brookhaven National Laboratory Medical Research Center, CEX-60.1 (1961). Availability: CFSTI, $1.75.

An experimental study was made to determine the protection against fallout radiation provided by the Medical Research Center at Brookhaven Laboratory. Shelter areas in the basement which could be used as emergency hospital wards were found to offer satisfactory shielding during a fallout situation.

This study also added data to the nuclear energy civil effects research being conducted by the Civil Effects Test Operations, Division of Biology and Medicine, United States Atomic Energy Commission, on the radiation shielding provided by structures.

A fallout radiation field was simulated by pumping a sealed Co-60 source through a long length of evenly distributed tubing. Radiation measurements were made inside the Medical Center by dose-integrating ionization chambers.

In general, the protection factors (ratio of open-field exposure dose rate to structure exposure dose rate) varied from 200 to 400 throughout the basement and from 12 to 20 on the first floor. Two isolated areas in the basement indicated much higher protection factors (1400 and 4000). Since this was a large one-story structure with a flat roof, fallout on the roof would probably contribute more than 90% of the total exposure dose rate at most points within the building during a fallout situation. Methods of significantly increasing the protection at most points of interest are limited to increasing the shielding material between the shelter areas and the roof or removing the contamination from the roof.

1063. Z. Burson, H. Borella, and D. Parry, Experimental Evaluation of the Fallout-Radiation Protection Afforded by a Southwestern Residence, CEX-60.5 (1960). Availability: CFSTI, $0.50.

An experimental study was conducted to determine the fallout-radiation protection afforded by a residence representative of a type of construction much in favor in the Southwest: a single-story stucco and frame house with a heavy shake roof and no basement. This study was one of many such studies sponsored by Civil Effects Test Operations, Division of Biology and Medicine, U. S. Atomic Energy Commission, for the purpose of evaluating the protection presently afforded by ordinary homes and structures against the dangers of fallout radiation.

The protection afforded by the home was determined by simulating a fallout-radiation field above and immediately surrounding the house and measuring the radiation level within. The radiation field was simulated by pumping a sealed Co-60 source through a long length of tubing evenly distributed over the test area. Highly sensitive dose-integrating ionization chambers were used to measure the radiation level inside the structure. The test was performed rapidly, easily, and safely. Valid statistical data were obtained even though the radiation level was of such low magnitude that it was unnecessary to evacuate any of the neighboring homes.

The protection factors within the house (ratio of exposure dose rate in the open field to exposure dose rate in the structure) ranged from 2.8 to 4.4, depending on the location. The results compare favorably with those found in previous exercises under similar conditions.

1064. Z. Burson and H. Borella, Experimental Evaluation of the Radiation Protection Provided by an Earth-Covered Shelter, CEX-60.6 (1962). Availability: CFSTI, $1.00.

A study was undertaken to determine the protection against fallout radiation provided by an earth-covered shelter. The study indicated that the shelter offered excellent protection from fallout radiation deposited on and around the shelter. This study also added additional data to the research in radiation shielding provided by various structures which is being conducted by the Civil Effects Test Operations, Division of Biology and Medicine, U. S. Atomic Energy Commission.

A fallout radiation field was simulated by pumping, at constant speed, a sealed Co-60 source through a long length of tubing which was evenly distributed over an area. Radiation measurements were made inside the shelter by dose-integrating ionization chambers.

In general, the results indicated that the protection factor (ratio of the open-field exposure dose rate to the exposure dose rate in the structure) was approximately 5000 in the center of the shelter, increasing to 10,000 to 15,000 along the sides, and decreasing to about 3000 near the ends. Directly below vents the protection factor was found to be approximately 2500 at the 3-ft level.

The shelter was a half-round corrugated-steel structure covered by a minimum of approximately 2 ft of earth on the top and 5 to 7 ft of earth on the sides.

1065. M. G. Silk, Attempts to Determine the Fast-Neutron Spectrum in a Thermal Reactor by Means of Li-6 and He-3 Semiconductor Spectrometers, AERE-M-1590 (1965). Availability: HMSO, 3s.6d.

Helium-3 and lithium-6 semiconductor neutron spectrometers have been used to determine the fast-neutron spectrum in GLEEP, a thermal reactor with a graphite moderator. A fast electronic circuit of resolving time 100 nsec has been used to eliminate pile-up effects as far as possible, and by combining the results obtained from both spectrometers the neutron spectrum has been determined over an energy range from 0.8 MeV to 6.4 MeV. The experimental spectra are shown to be in agreement with theoretical spectra derived by Monte Carlo calculations. The results suggest that silicon-neutron reactions are relatively unimportant in this case. A small peak observed in the lithium-6 spectra may confirm that there is a discontinuity in the cross section for the Li-6(n,α)H-3 reaction in the neutron-energy region between 2 MeV to 2.5 MeV.

1066. D. F. Shook, D. L. Alger, D. Bogart, and R. A. Mueller, Thermal Neutron Perturbation by Dysprosium-Aluminum Foils in Water and Uranyl Fluoride-Water Solutions, NASA-TN-D-3237 (1966). Availability: CFSTI, $1.00.

Thermal neutron flux perturbation factors have been measured for dysprosium-aluminum alloy foils in water and uranyl fluoride-water solutions. The foils are representative of a type widely used for thermal neutron flux surveys. The measurements provide additional data for comparison with analytical calculations of foil perturbation factors in water. The uranyl fluoride-water solutions used comprise absorptive diffusion media different from water or graphite that have been studied exclusively in the past. The edges of the thicker dysprosium-aluminum foils used are a significant part of the total foil surface so that the data provide a test for several proposed edge corrections to perturbation factor calculations. The experimental perturbation factors are in good agreement with the predictions of published calculational methods except for the dimensionless generalization of Dalton and Osborn.

1067. R. Sanna, K. O'Brien, M. Alberg, S. Rothenberg, and J. McLaughlin, Nuclear Emulsion Spectrometry at Low and Intermediate Neutron Energies, HASL-162 (1964). Availability: CFSTI, $4.00.

The representation of Pu-Be and reactor spectra are entirely satisfactory indications that within the limitation of the smoothness requirement, this method of nuclear emulsion spectrometry can be generally relied upon. The accelerator neutron spectra, are quite hard, as we shall expect on theoretical grounds. Where comparison with theoretical expectation is possible, agreement is good.

1068. C. Eisenhauer, "An Image Source Technique for Calculating Reflection of Gamma Rays or Neutrons," Health Physics, 11(11), 1145-1154 (1965).

A simplified representation for the problem of a point source and point detector located above an interface is discussed. It is shown that the geometrical effects can be taken into account by a simple scaling function. For a given material and radiation the remaining effects depend primarily on one angular variable with only a secondary dependence on the ratio of source distance to detector distance from the interface. Applications are made to neutrons, gamma rays and particles whose energy does not change with scattering (one-velocity case). The analogous representation for problems in transmission of radiation through slabs is also discussed.

1069. R. E. Malenfant and G. A. Graves, Ground Test Radiation Experience at the KIWI B-4 Tests, AIAA Paper No. 65-593 (1965). Availability: $1.00, American Institute of Aeronautics and Astronautics, New York.

An extensive series of radiation level measurements from the Kiwi B-4 tests at NRDS have provided the detailed information on the radiation environments from ground operation of propulsion reactors which is necessary for the prediction and evaluation of future problems. The complexity of test site areas almost precludes obtaining this information by calculation alone, particularly post-shutdown, the condition of greatest practical interest. The components (direct beam, air scatter, and activation) of the total dose rates post-shutdown were separated and identified by comparing measurements taken before and after reactor removal from the test area. These values were well-confirmed through cross-check measurements on reactors free of activation fields. The isolated reactors exhibited lower dose fields than expected. No evidence of local fission product contamination was found. Area activation was relatively high -- at some post-shutdown times and distances, activation and reactor dose rates were comparable -- but the activation decayed fairly rapidly with effective half-life groupings near 2 1/2 hours, 15 hours, and 30 days. Reactor shields will be needed to control activation and permit sustained area occupation after tests of large energy release -- e.g., 10^7 megawatt-sec -- but with them, operations like dewar refilling can proceed steadily after about three days.

1070. R. Dehn, "Measurement of Gamma Heating in D.F.R.," Nucl. Instr. Methods 36(1), 141-152 (Sept. 1965).

A series of experiments has been carried out in the Dounreay Fast Reactor at power levels from 1 to 60 MW to measure the gamma heating (total energy deposition) in materials such as tungsten and stainless steel.

Most of the measurements were made in the first row of the breeder and in this position the mean value for the gamma heating in stainless steel was found to be 28 Mw/g.MW ±15%.

Several different calorimeter designs are discussed, the electrical calibration procedure is outlined; one standard type and the noval twin bulb differential calorimeter are described in detail.

1071. T. E. Stephenson, The Neutron Cross Section of Sodium Below 40 keV, BNL-961 (December 1965). Availability: $2.00 CFSTI.

The neutron cross section of sodium from 1 eV to 50 keV is calculated using the single level formula and BNL-325 resonance parameters for the 2.85-keV level. A good fit to the data is obtained with the spin-dependent scattering radii $R_+' = 5.85$ f and $R_-' = 5.18$ f. A good fit to the experimental data throughout the region was not found using a spin-independent scattering radius. The excess absorption resonance integral is calculated to be 0.072 barns and is in good agreement with experimental values. The scattering resonance integral is calculated to be 72 barns which compares favorably with an evaluated experimental result. The low energy potential scattering is described by spin-dependent square potential wells.

1072. A. N. Komarovsii, Shielding Materials for Nuclear Reactors, Pergamon Press, 1961.

This important and topical translation from the Russian deals with the shielding and technological properties and with the characteristics of building materials used for constructing radiation shielding for nuclear reactors and accelerators. Particular consideration is paid to specially heavy and hydraulic concretes. In ordinary concretes only the shielding properties, means of ensuring maximum density and homogeneity are investigated.

This book collates, classifies and, as far as possible, presents results of a series of Russian and foreign investigations and practical methods for constructing radiation shielding for nuclear reactors and accelerators. After summarising the hazards arising from nuclear radiations and the methods of defining the dose rates, the book gives a wealth of information on the variation in attenuation which results from varying the type of concrete. Very full details are provided on the different types of concrete which have been used, together with their properties and strengths, and the technology of shield construction as applied to concrete. The economic considerations involved in the choice of concrete and numerous details of costs in the Soviet Union are given.

1073. G . Pomraning, A Method of Solution for Particle Transport Problems, NASA CR-GA-6497 (July 23, 1965). Availability: National Aeronautics and Space Administration, Office of Scientific and Technical Information, Attention: AFSS-A, Washington, D.C. 20546

A new method of solution for the linear transport equation is presented. While the method is basically numerical in that the final equations are well suited for solutions by a large digital computer, a good deal of analysis has been introduced to keep the numerical work at a minimum. That is, in most instances one has a qualitative idea of the nature of the solution to a given problem. The method reported here contains sufficient flexibility so that these qualitative notions can, for a given problem, easily be incorporated into the final equations. The motivation for this work was the desire to treat all linear transport problems, including such extremes as reactor calculations (almost isotropic distributions) and shielding calculations (highly anisotropic distributions), by a unified formalism.

In developing this method, the following four distinct areas were studied: (1) Representation of the scattering kernel, (2) Treatment of the energy variable, (3) Treatment of the angular variable, (4) Treatment of the spatial variable. The formalism was developed for a general three-dimensional geometry, and explicit equations were derived for the three-one-dimensional geometries. The resulting approximation should be useful in a wide variety of physical problems, including fissionable assembly calculations, and charged particle, gamma ray, and high energy neutron penetration calculations.

1074. M. N. Thompson and J. M. Taylor, "Neutron Spectra from Am-α-Be and Ra-α-Be Sources," Nucl. Instr. Methods 37(2), 305-308 (December 1965).

The neutron energy spectra obtained from the sources Am-α-Be and Ra-α-Be have been measured using a proton recoil spectrometer with P-ray discrimination. Agreement with the calculations of Anderson is found.

1075. W. S. Hehl, Calculation of the Neutron Flux Distribution in the EBWR Core Shielded by Water, Iron, and Concrete, IEA No. 85 (December 1964). Availability: Dep. (mn).

In this study the operation of the EBWR at 100 Mw when shielded by 28.6 cm of water, 10 cm of iron, and 300 cm of concrete was considered. The equal-volume sphere core, with a radius of 73 cm was chosen for the study as indicated by calculations for the best core geometry.

The fast and thermal neutron flux calculations were done by two-group analysis. Calculations for the latter were made both by hand and by computer.

The procedures followed and the calculations made in connection with the study are explained, and the results are tabulated and plotted.

1076. A. Prince, Neutron Cross Sections for Tantalum and Tungsten, GEMP-388 (Nov. 12, 1965). Availability: Dep. mn; CFSTI $5.00 cy, $1.00 mn.

Neutron cross sections for Ta-181, natural tungsten and its four main isotopes have been calculated from 1 keV to 14 MeV. The total, elastic, inelastic, and differential scattering cross sections were computed using ABACUS II in which the equivalent local potential to the non-local potential of Perey and Buck were employed.

The radiative capture cross sections were derived by the method due to Lane and Lynn where the low energy experimental data on resonance parameters were used in determining the average level spacings and strength functions.

1077. Z. G. Burson, Experimental Radiation Measurements in Conventional Structures Part III, The Attenuation of Air-Scattered Radiation in a Basement, CEX-59.7B (November 1965). Availability: Dep. mn; CFSTI $1.00 cy, $0.75 mn.

An experimental study was conducted to determine the air-scattered radiation contribution to a basement structure from a simulated plane source of ^{60}Co on the ground outside and to determine its attenuation through various structural components.

A sealed radioactive source was pumped at a uniform speed through a long length of flexible tubing to simulate both an area and a ring source. Exposure measurements were made with ionization chamber detectors.

A bare, open basement was used as the basic structure. A series of radiation measurements was made in this order:

In the open basement.

Within shelter walls constructed in one end of the basement.

Within these shelter walls surrounded by sandbags.

In the fully constructed shelter, no sandbags.

In both the basement and shelter after a wood roof was constructed over the entire basement.

In the basement and shelter after an additional 4-in. (total of 8 in.) layer of concrete planks was placed over the wood roof.

The top of the materials (basement walls, wood roof, or concrete planks) was flush with ground level. This assured that radiation reaching the detectors had been scattered at least once in the air.

The results for all stages of construction compared well with prior theoretical calculations and with other experimental data in similar configurations. Comparisons were also made with data taken in a similar basement below a two-story brick house.

The results are useful for design of basement shelters and for the analysis of protection provided by basement structures from fallout radiation.

1078. J. W. Webster, Some Conclusions Regarding the Use of the Sn Method in Shielding Calculations, ORNL TM-1196 (Sept. 15, 1965). Availability: Dep. mn; CFSTI, $3.00 cy, $0.75 mn.

This study was initiated so that further numerical experience could be gained with the Sn method as a tool for shield analysis. The need for this type of study was made known by the results of comparative shield calculations reported by the Shielding Division of the American Nuclear Society in report ANS-SD-1, titled "Neutron Attenuation in Optically Thick Shields." The fundamental problems chosen for the investigation were neutron penetration in hydrogen and in water resulting from a point fission source.

Some of the results and conclusions of this study are:

1. The Sn method, as applied by means of the DTF code, gives excellent agreement with the moments method for fast-neutron penetration over the distances from a point source studied here, i.e., up to 90 cm in hydrogen and 120 cm in water. It seems reasonable to conclude that satisfactory agreement would be obtained at even greater penetration.

2. An angular segmentation of flux of at least $n = 12$ should be used for fast-neutron penetration and even greater segmentation is necessary for accurate computation of the spatial distribution of slow neutrons.

3. Taking into account the terms of the Legendre expansion of scattering cross section up to order 2 gives good results; however, the effect of inclusion of higher terms was not quantitatively evaluated.

4. A spatial mesh (Δr) of 5 cm or less is necessary for fast-neutron penetration problems and 2 cm or less if the distribution of slow neutrons is of interest.

5. A 28-group structure with a lethargy width per group of 0.2 for $E > 1$ MeV was used in all the calculations. It is concluded that this is quite satisfactory.

1079. Georgia Nuclear Laboratories, Gamma P, N65-27373 (NASA CR-63600) (ER-7448)(ND-5001) (September 1964). Availability: GPO; $1.00, $0.50 (mf).

The gamma spectra of the Radiation Effects Reactor was measured with three external shield configurations, i.e., 16 inches of Lithium Hydride, 16 inches of Lithium Hydride plus 8 inches of water, and 8 inches of water. These measurements were made with a collimated single crystal sodium iodide scintillator (4" x 5" diameter). Histogram photon and dose distributions were then computed by means of a 33 x 33 inversion matrix.

These measurements show considerable hardening of the basic reactor spectrum, i.e., the U-235 fission spectrum plus capture gammas from the reactor coolant and structural materials. Data reveals that Lithium Hydride is not unique in shaping the resultant filtered spectra, but instead demonstrates effects similar to those produced by water. The measured dose distribution shows qualitative agreement with the calculated SNAP-8 environment.

1080. G. D. Trimble, G. K. Houghton, and J. H. Audas, Measurement of Neutron Spectra in Liquid Hydrogen, N65-19701 (NASA CR-54230) (GA-5750) (Sept. 3, 1964). Availability: CFSTI, $6.00 (hc), $1.25 (mf).

Differential fast neutron spectra are presented for the neutron energy range 0.5 to 15 MeV in both water and liquid hydrogen as a function of space and angle. Measurements were made for 13.0, 10.5, 7.0, 4.5, and 2.5 inches of liquid hydrogen at angles of 0, 15, 37, 53, and 78°; water measurements were made for 1.5 and 6.4 inches at the same angles.

A description of the liquid hydrogen experimental facility is given, including design considerations. Also included is a discussion of the development of the isotropic fast neutron source and differential neutron spectra taken with this source.

1081. F. W. Woods, W. A. Hough, D. O'Neal, and J. Barnett, "Gamma Ray Attenuation by Loblolly Pine Wood, An Investigation of Integral Counting," Forest Sci. 11(3), 341-345 (1965).

Within limits, attenuation by wood of Pinus taeda L. by either integral or differential counting was equally satisfactory. Because the former requires simpler equipment and is more adapted to field work, it was further explored. Linear attenuation coefficients were measured for the wood (specific gravity .485 g/cm^3) at various moisture contents using an integral narrow beam of gamma photons from a mock iodine-131 source. Half-thickness values and mass attenuation coefficients were also calculated for each moisture content. Results indicate that the wood has a mass attenuation coefficient very similar to that of water, greater than that of concrete or aluminum, and smaller than that of lead or tin. Half-thickness values are about twice as high for oven dry wood as for water. However, at a moisture of 50 - 60 per cent (by volume) wood has a half-value almost identical to that of water. At this moisture content about 6 centimeters of wood between the detector and a point source will reduce the county rate by half.

1082. B. D. O'Reilly and R. C. Lewis, The Calculation of Slowing-Down Transfer Matrices in the Spherical Harmonics Representation, NAA-SR-11530 (March 25, 1966). Availability: $3.00, CFSTI.

This report discusses the calculation of transfer matrices for use in multigroup spherical harmonics calculations. The defining equations are derived from considering the slowing-down source of neutrons in the spherical harmonics representation. Both elastic and inelastic scattering processes are accounted for. Two complementary procedures are used to calculate the matrix elements for inelastic scattering: (1) the discrete level model, and (2) the Weisskopf evaporation model. Elastic scattering is accounted for by the discrete level model with zero threshold energy.

Particular attention has been given to describing the differential angle-energy cross section for neutrons exciting a discrete energy level of the target nucleus. The usually neglected double-valued relation between incident and emergent energies in the laboratory frame, which exists when the incident energy approaches closely to the threshold energy, is taken into account. Neutron balance conditions are found for the group-averaged slowing-down current and for the group averages of the higher-ordered Legendre coefficients of the slowing-down distribution, as well as the usual balance condition for the scalar slowing-down flux. It is found that the transitions of neutrons incident in the double-valued region contribute negligibly to the overall neutron balance. A sample calculation of the P_0 matrix for C^{12} in a 16-group structure is also given. The sample calculation is carefully analyzed with respect to the effects of numerical approximations on the accuracy of computed matrix elements. A number of numerical procedures are recommended for improving the accuracy of transfer matrix calculations. Methods are discussed for obtaining estimates of spectra appropriate to critical reactors, reactor kinetics, and shields.

1083. J. M. Chapman, Dose-Attenuation Variation with Incident Gamma-Ray Energy in Two-Legged Concrete and Steel Ducts, USNCEL-TR-442 (NP-15926) (April 1966). Availability: CFSTI, $2.00.

Gamma-ray dose-attenuation factors were measured in concrete and steel ducts. For concrete, 3-foot-square and 11-inch-square ducts were used with Au^{198} (0.412 MeV), Cs^{137} (0.662 MeV), and Co^{60} (1.25 MeV) gamma-ray sources. For steel, an 11-inch-square duct was used with Cs^{137} and Co^{60} sources. Attenuation factors for given geometries were compared as a function of incident gamma-ray energy. The relative effectiveness of steel and concrete ducts of a given geometry was determined.

It was found that the attenuation factor decreases monotonically with increasing energy in concrete ducts. However, in the 11-inch steel duct the attenuation factor for the high-energy source (Co^{60}) was greater than for the low-energy source (Cs^{137}). In comparing the 11-inch concrete and 11-inch steel ducts, it was found that dose rates in the concrete duct were higher by a factor of about 2. Measured attenuation factors were compared with values obtained using a computer code based on the albedo concept. It was found that calculated attenuation factors agree to within $\pm 30\%$ of the measured attenuation factors.

1084. I. Umeda and T. Hyodo, "Transmission of Gamma Rays Through Pipe Walls," Nucl. Struc. Eng. 2(3), 335-340 (September 1965).

Transmission of gamma rays from 60-Co and 137-Cs line isotropic sources through a steel pipe wall was measured as a function of the wall thickness. The measurement was carried out with two alternative arrangements: a point isotropic source and a long cylindrical ionization chamber, and also a moving point isotropic source on the pipe axis and a small spherical ionization chamber.

Comparing the build-up factor obtained by this experiment with that for similar geometry calculated from the results obtained by Goldstein and Wilkins for an infinite homogeneous medium using the method of moments, the ratio of the former to the latter was 0.90 for 60-Co and 0.87 for 137-Cs, respectively, for the wall thickness larger than 1.5 cm.

1085. J. J. Broerse and F. J. van Werven, "Shielding Measurements for 3 MeV and 15 MeV Neutrons," Health Phys. 12(1), 83-93 (January 1966).

With a view to determine the minimum thickness of shielding required for a neutron generator, the absorption characteristics of various materials were measured for neutrons produced through the $T(d,n)^4He$ and $D(d,n)^3He$ reactions. The shielding thickness required to reduce the transmitted dose to permissible levels of exposure was found to be determined by the reduction of the dose rate of the 15 MeV neutrons.

Since the shielding is required for the protection of the operating personnel, the quantity of interest is the dose equivalent. This quantity was measured with a neutron rem detector for various shielding materials as a function of the shielding thickness. Attenuation curves are reported for water, concrete, paraffin and borated laminated wood.

After the final shielding of the neutron generator was constructed, the radiation levels outside the shield were found to be below the maximum permissible values, as indicated by the instrumentation used.

1086. W. G. Bingham, "Low Activation Shielding Materials for Nuclear Reactor Environmental Test Chambers," Nucl. Struc. Eng. 2(2), 243-247 (August 1965).

This reports the results of a testing program and study to find low-activation shielding materials for nuclear reactor environmental test chambers. Economic trade-offs associated with low-activation shielding materials are considered. Activation dose rates in existing test chambers are compared with study results.

1087. J. Nilsson and R. Sandlin, "A Method for Prediction of the Neutron Attenuation in Annular Ducts," Nucl. Struc. Eng. 2(3), 323-326 (September 1965).

A calculational procedure for prediction of the attenuation of thermal and fast neutrons in straight air-filled annular ducts is described. The underlying theory, not presented here, takes into consideration arbitrary angular distributions of the source flux and treats both collided and uncollided contributions to the air gap flux. The procedure works also at short distances from the duct mouth.

1088. S. Sasse, "Method for Weight Optimization of a Two-Material Shield for Reactor Radiation," Nucl. Struc. Eng. 2(3), 327-334 (September 1965).

Method of analysis of the problem of designing a homogeneous two-material spherical reactor shield of minimum weight, treating radiation attenuation in plane geometry. The optimization procedure is illustrated by the outline of an example calculation for a reactor shield consisting of a hydrogeneous material (organic plastic) loaded with lead where the space-variable volume fraction of lead is optimized to give a shield of minimum overall weight.

1089. H. H. Naumann and K. H. Waechter, "Shielding for Radioisotope Bremsstrahlung Sources Sr-90 Plus Y-90," Int. J. Appl. Radiat. Isotop. 16(10), 613-15 (October 1965).

The exposure attenuation curve for a Sr-90 and Y-90 source was measured for a lead shield and found to agree quite well with the theoretical values of Florkowski but in disagreement with the experimental values of Florkowski. The difference is attributed to the detectors used.

1090. I. A. Kazanskii, S. P. Belov, and E. S. Matusevich, "Angular and Energy Distributions of Gamma Rays Scattered in Iron and Lead," Soviet J. At. Energy (English Transl.) 5(4), 1354-1357 (October 1958).

In this investigation, measurements were made of angular energy distributions of scattered γ-rays from Co-60 (E = 1.33 and 1.17 MeV) and Au-198 (E = 0.411 MeV) sources. The γ-rays travelled through lead ($\mu_o r$ = 2.2, 4.1, 6.3, 8.2, where μ_o is the coefficient of absorption of the γ-rays, r is the thickness of the lead or iron filter), and iron ($\mu_o r$ = 2, 4, 6, 8.5, 9.8).

1091. B. P. Bulatov and O. I. Leipunskii, "The Albedo of Gamma Rays and the Reflection Buildup Factor," Soviet J. At. Energy, English Transl. 7(6), 1015-1017 (April 1961).

The energy albedos and reflection buildup factors (flux albedo) were determined experimentally for Co-60 and Au-198 gamma rays reflected from light materials (iron). The angular distribution could be represented by a power of the sine of the angle. Dependence on the radius and thickness of the scatterer was also determined.

1092. V. A. Vasilev and V. A. Shishkina, "Back-Scattering of Gamma Radiation in Aluminum," Soviet J. At. Energy, English Transl. 4(2), 283-284 (February 1958).

The backscattered flux density was measured as functions of distance from the source (Cs-137 or Au-198) to the Al scatterer and of scatterer thickness.

1093. B. P. Bulatov, "Gamma-Ray Albedo of Co-60, Cs-137, and Cr-51," Soviet J. At. Energy, English Transl. 7(4), 847-850 (March 1961).

The albedo of gamma rays reflected from carbon, aluminum, iron, and lead was measured as functions of angle and atomic number. The sources (Co-60, Cs-137, and Cr-51) were in direct contact with the surface of the scatterer.

1094. U. A. Ulmanis and N. A. Dubinskaya, "An Investigation of Gamma-Ray Backscattering," Soviet J. At. Energy, English Transl. 3(7), 804-806 (1957).

The reflected intensity of Cr-51, Cs-137, and Co-60 gamma rays was measured as functions of reflector thickness and atomic number.

1095. U. Moller, Subprograms for the Solution of Problems in Neutron Physics by Means of the Monte Carlo Method, ORNL-tr-876 (February 1965). Availability: Dep. (mn), JCL $3.60 (fs), $1.37 (mf). (Translated from KFK-298)

Within the scope of the formulation of a general Monte Carlo code for studies in reactor physics, this report contains the descriptions of the subprograms utilized in the form of two separate parts.

1096. J. DeFelice, Shielding Analysis for the Fast Flux Test Facility, COO-1430-1 (June 30, 1965). Availability: Dep. (mn), $2.00 (cy), $0.50 (mn).

The shielding requirements for the Fast Flux Test Facility were investigated and an initial specification of required types and thicknesses of shield materials was obtained.

The radial shield specification was 3.5 ft of iron plus 5 ft of graphite. The axial shield was specified to consist of 3 ft of iron above the 30' sodium pool. A number of alternate materials for the shield were also investigated. The best of these alternate materials was found to consist of a 5' layer of borated Fe_2O_3 in place of the 5' radial graphite shield. The borated Fe_2O_3 produced a secondary sodium activation about a factor of 8 less than the graphite shield. Furthermore, the use of borated Fe_2O_3 could very possibly result in a cost saving of approximately one million dollars by the elimination of the graphite and its associated inert gas systems.

Complete neutron flux traverse calculations for the axial shield and the radial shield are presented both for the reference design and for a number of alternate materials. The results of parametric studies on a number of alternate shield materials are also presented. The results of variations in shield thickness for the reference design are also given in the report.

1097. U. Moller, Solution of the Transport Equation by Means of the Monte Carlo Method, ORNL-tr-875 (February 1965). (Translated from KFK-297

This report contains the theoretical and practical fundamentals of a Monte Carlo code for the treatment of problems in neutron physics in the form of two complete parts.

1098. H. A. Sandmeir, G. E. Hansen, R. B. Lazarus, and R. J. Howerton, Computation of Fast Neutron Penetration in Air by the "Sn-Method" with Special Emphasis on the Use of "Multitable-Multigroup Cross Sections Sets", LA-3415 (November 17, 1965). Availability: CFSTI $3.00 (cy), $0.75 (mn).

Much theoretical work has been done in the past to represent the angular dependence in the scattering source term of the Boltzmann equation by means of Legendre or other series expansions. However, relatively little work has been done to feed this information into our present day Sn codes. The Sn-transport codes at LASL allow a representation of anisotropy in the scattering-source term by means of multitable cross sections sets, and we develop here a formalism to generate these sets. The numerical evaluation of Sn cross section entries for these multitable sets has been computerized by one of us. (R.B.L.)

Deep penetration calculations (3000 meters) of high energy neutrons in homogeneous air are presented. These calculations used recently evaluated basic cross section data for nitrogen and oxygen from the LRL-LASL Library (Howerton tape).

The variations in the neutron flux due to different approximations are discussed as are also the expected variations in the flux due to inaccuracies in cross section data.

1099. R. Schamberger and J. Celnik, Shield Small Source Experiments, Report No. 2, UNC-5118 (May 6, 1965). Availability: Dep. (mn) CFSTI $4.00 (cy), $0.75 (mn).

MCR shield design analytical techniques and cross sections were tested by comparing calculations with measurements for experiments emphasizing the secondary gamma-ray production in specific test materials.

Measurements were made with a gamma-ray spectrometer behind a configuration consisting of a layer of gamma-ray shielding material (the test sample) adjacent to a small Po-Be neutron source, followed by neutron and gamma-ray shielding material to reduce extraneous detector response. Calculations of spectrometer response for samples of lead, tungsten, and titanium hydride were performed with a three-dimensional Monte Carlo code, derived from the one-dimensional Monte Carlo codes used in the MCR preliminary design shield analysis.

The calculations generally gave very satisfactory predictions of the measurements, the agreement being within twice the uncertainty in the experiment.

Differences between the measured and calculated responses indicated that some of the cross sections used, particularly for lead, were low and should be revised. It is estimated that these revisions would lead to a 10% increase in predicted total dose rate external to the MCR shield.

1100. W. L. Zijp, Review of Activation Methods for the Determination of Fast Neutron Spectra, RCN-37 (May 1965). Availability: Dep. (mn).

A review is given on the possibilities for determining flux densities, fluences and spectra of fast neutrons from the counting results obtained with threshold detectors. After an introductory chapter on the response of a threshold detector, on the definitions of fast flux density and on effective threshold and effective cross sections, some threshold reactions of interest are considered. A chapter is devoted to the determination of fast flux density and fast fluence. A following chapter reviews the various methods mentioned in literature for the determination of the spectral distribution from the response integrals. The semi-empirical spectrum representation proposed by Genthon is reviewed. After the mentioning of the activities of the Euratom Working Group on Dosimetry the results of the discussions in the IAEA Panel on In-Pile Dosimetry are mentioned. Tables with nuclear data and an extensive bibliography are included at the end.

1101. R. M. Lessler and F. W. Guy, Gamma Dose Curves from Neutron-Activated Soil and Seawater Compared with Fission Product Decay, UCRL-14332 (August 1965). Availability: Lawrence Radiation Lab., University of California, Livermore, California.

Curves are presented showing total γ-MeV (to 50 years) and γ-MeV/hour resulting from decay of granite, NTS soil, and seawater which have been activated by one mole of 14-MeV or 0.5-MeV neutrons. Curves are graphed as a function of time from 1 hour to 10 years. Fission-product decay curves are shown for purposes of comparison.

1102. D. F. Schweiger and C. U. Steichen, Monte Carlo Evaluation of Fast Neutron Fluxes and Heating in the S8DS Neutron Shield, NAA-SR-MEMO-9916 (August 27, 1965). Availability: Dep. (mn), CFSTI $1.00 (cy), $0.50 (mn).

Fast neutron fluxes were calculated for a nominal separation distance of 26 inches between the center of the active core volume and the top of the lithium hydride casting. The maximum fast flux (0.092 to 10.56 MeV) was found to be 2.9×10^5 neut/cm^2-sec-watt. At an operating power level of 600 kwt this corresponds to a flux level of 1.74×10^{11} neut/cm^2-sec. Fast fluxes were computed down to a depth of 28.75 cm below the top of the casting.

A comparison was made between the Monte Carlo results, a removal theory calculation by the QAD program, and a reactor leakage equivalent point source calculation. All three calculations agreed within a factor of three. The Monte Carlo and point source calculation agreed closely.

Fast neutron heating rates were computed from the Monte Carlo flux levels. The maximum heating load was estimated to be 2.7×10^{10} MeV/cm^3-sec. This corresponds to 0.0042 watt/cm^3 or 419 btu/hr-ft^3.

1103. J. L. Russell, Jr. et al., Neutron Spectra in Bulk Media by Time-of-Flight, GA-6313 (April 9, 1965). Availability: Dep. (mn); $1.00 (cy), 1 (mn) CFSTI.

Time-of-flight neutron spectroscopy dates back almost to the discovery of the neutron. But only recently have we begun to use this method to measure fast and intermediate neutron spectra in bulk media with the objective of evaluating and improving methods for calculating neutron spectra in reactors and shields. The advent of very intense, nanosecond scale pulsed neutron sources has made practical the measurement of deep penetration neutron spectra with good resolution.

In the past year at General Atomic, several experiments have been completed in geometries designed to be calculable by specific computer codes. In this paper, some of the results of these experiments are discussed and compared with calculation, the experimental equipment is described, the specific experimental results are presented and compared with calculation, and some tentative conclusions are discussed.

1104. F. Zerdin and V. Dimic, Experimental Verification of a Removal Age-Diffusion Multigroup Program in a Thermal Column, NIJS-R-437 (September, 1964). Availability: Dep. (mn).

A removal age-diffusion multigroup program, with only four energy intervals in the removal group, for a bulk shield was experimentally verified in the thermal column of the heavy water experimental 6,5 MW reactor in Vinca. Thermal and epithermal neutron fluxes were measured in a usual manner by using bare and cadmium covered Au and In foils. The measured cadmium ratio is accurate to the high flux region close to the reactor core. In the low flux region the experimental accuracy was not sufficient to get reliable results. The measured and calculated values of R_{Cd} are not in quite good agreement. The relative values of the calculated and measured thermal fluxes agree satisfactorily all the way through the thermal column, except at the outer face. The absolute values disagree due to normalization uncertainties.

1105. T. E. Dudley and M. R. Mendelson, Infinite Medium Spectrum-Weighted Group Constants for Gamma Heating Calculation, CONF-651101-41 (November, 1965). Availability: Dep. mn, CFSTI $2.00 (cy), $0.50 (mn).

In the calculation of gamma-ray heating, the required group constants are usually obtained by energy-averaging gamma cross sections over some assumed spectrum. Usually no attempt is made to obtain equilibrium gamma-ray spectra, but rather some convenient prescription, such as a fission spectrum, is used.

In this paper a method is described for computing gamma-ray group constants that is analogous to neutron procedures (e.g., MUFT, GAM). An infinite-medium spectrum is obtained by trapezoidal rule integration of the infinite-medium slowing down equation. Since the equations are coupled downward in energy only, the solution is easily obtained.

Such a procedure for calculating spectra and group constants has been coded into a FORTRAN-II computer program IMPS (Infinite Medium Photon Spectrum). Spectrum-averaged Legendre moments of group-to-group transfer cross sections (to P_{10}) are calculated from the Klein Nishina formula for transport-theory applications, and group-diffusion coefficients are computed for diffusion-theory problems.

It has been found that the spectrum shape is approximately maintained except at low energies so that the use of the source spectrum for group averaging does not lead to much error except at these low energies. Gamma-ray cross sections are, in comparison with neutron cross sections, smooth functions of energy and do not vary greatly in magnitude. Group constants thus tend to be spectrum insensitive, at least in regions where smooth sources (such as from fission) predominate. In special problems where rapidly varying sources are present (such as discrete capture spectra), or where low-energy photons are important, greater attention must be paid to the choice of the equilibrium spectrum. The group structure and the group-constant definitions can be important. Comparisons with IMPS of solutions using the GAM-I averaging prescription indicate that large spectral errors can be incurred with the latter method.

1106. K. G. Golliher and K. G. Randen, Capabilities of the 1 MW Shield Test and Irradiation Reactor, NAA-SR-11528 (November 10, 1965). Availability: Dep. mn CFSTI $2.00 (cy), $0.50 (mn).

As a prime contractor to the USAEC for the SNAP program, Atomics International has operated the Shield Test and Irradiation Reactor (STIR) at its Santa Susana Nuclear Development Field Laboratory since initial criticality in October 1961. The reactor facility is designed to provide a flexible source of neutrons and gammas for experimental work on shield development, for irradiation damage studies on electronic systems, and for the nuclear qualification of SNAP components. The first core of the reactor was fueled with SNAP fuel elements. This core operated at a power level of 50 kilowatts. The second core, which achieved criticality in December 1964, uses MTR-type fuel elements and operates at a power level of 1 megawatt.

This report describes the physical layout of the facility together with the results of measurements of the neutron fluxes and gamma ray dose rates.

1107. C. A. Anderson, Jr., Fission Product Yields from Fast (~ 1 MeV) Neutron Fission of Pu-239, LA-3383 (December 30, 1965). Availability: Dep. mn CFSTI $2.00 cy, $0.50 mn.

The sixteen measured yields of fission products from fast fission of Pu-239 published through June, 1965, are given. Information is presented which indicates the restricted range of mass numbers within which mirroring, or reflection, of data is applicable. Use of this information enables generation of fairly reliable guesses for six reflected data points. A curve fit to these data provides interpolated estimates of unmeasured mass chain yields. A modified equal-charge-displacement hypothesis is used to estimate independent and cumulative fission yields of nuclides.

It is hoped that future measurements of fission product yields from fast fission of Pu-239 will make the speculative portions of this report unnecessary.

1108. J. C. Albergotti and J. M. Ferguson, Total Neutron Cross Sections of Copper, Iron, and Aluminum Near 14 MeV, USNRDL-TR-929 (AD-625747) (November 8, 1965). Availability: CFSTI $1.00 (cy), $0.50 (mn).

Total neutron cross sections of Mg, Al, Fe, and Cu have been measured using T(d,n) neutrons in the energy range 12 to 14 MeV. The accuracy (counting statistics) is better than 1%. The energy resolution (two standard deviations of the energy resolution function) varies from 35 to 150 keV. Fluctuations of amplitude 2 to 3% and width of the order of 100 keV are found in the cross sections of Mg and Al. No such variations are found in the cross sections of Fe and Cu; a χ^2 test agrees with a straight line fit to the Fe and Cu data.

1109. L. V. Spencer, J. A. Diaz, and E. Moses, Neutron Penetration in Cylindrical Ducts, NBS Report 8542 (September 1, 1964). Availability: National Bureau of Standards, Washington, D.C.

The long, straight, cylindrical duct problem is treated in the approximation of no neutron displacement in the wall. A general transport equation is derived for fast or slow neutrons, and simplified by expansions in the angle variables, and triangularization. The equation is solved numerically for thermal neutrons in concrete, with an isotropic source at the duct entrance. The solutions are interpreted and compared with Monte Carlo data. Throughout, both conceptual and numerical use is made of albedo angular distributions which follow those of the Chandrasekhar theory for isotropic single scattering.

1110. J. M. Chapman and C. M. Huddleston, "Dose Attenuation in Two-Legged Concrete Ducts for Various Gamma-Ray Energies," Nucl. Sci. Eng. 25(1), 66-74 (May 1966).

Methods are outlined for computer calculation of first- and second-order contributions to gamma-ray dose rates in two-legged rectangular concrete ducts of personnel shelters. Four computer programs, based on values of differential dose albedo derived from Monte Carlo calculations, are used to calculate corner-lip inscattering, backscattering from surfaces visible to both source and detector, and combinations of these two scattering events. The first program computes the dose contribution from corner-lip inscatter and from combinations of corner-lip penetration and wall backscatter. The second calculates first-order backscatter from the basic scattering areas in the corner of the duct. The third computes the second-order backscatter contribution from two successive backscatters from the duct walls. The final program calculates the contribution due to combinations of one wall backscatter and one corner-lip inscatter. The results of the calculations are compared with experimental results obtained by several investigators. The experimental studies used 11 x 11-in., 3 x 3-ft, and 6 x 6-ft ducts; the gamma-ray sources were, variously, ^{198}Au, ^{137}Cs, ^{60}Co, and ^{24}Na. Since the difference between calculated and measured values of gamma-ray dose in the second leg is generally less than 30%, results obtained by calculation can be used to predict the attenuation factors of two-legged concrete ducts and entranceways leading into personnel shelters with an expected error of less than 30%.

1111. Y. Furuta, A. Tsuruo, S. Miyasaka, K. Tamura, and Y. Kanemori, "Dose Buildup Factors of Plane Parallel Barriers for 60-Co Plane Monodirectional Source," Nucl. Sci. Eng. 25(1), 85-92 (May 1966).

Dose buildup factors for a plane monodirectional source of ^{60}Co were obtained experimentally for plane parallel barriers composed, respectively, of water, graphite, ordinary glass, aluminum, ordinary concrete, heavy concrete, iron, and lead. For water, aluminum, ordinary concrete, iron, and lead, comparisons were made between the experimental and the theoretical values that were calculated by a method to obtain dose buildup factors of finite barriers. The results showed these values to be in good agreement for all materials except water. For water, further calculation by the Monte Carlo method supported the theoretical value.

1112. F. M. Tomnovec, J. M. Ferguson, and D. M. Weldon, The Effect of a Changing Gamma-Ray Fallout Spectrum on the Ground Roughness Factor, USNRDL-TR-915 (September 22, 1965). Availability: U. S. Naval Radiological Defense Laboratory, San Francisco, California.

It has been suggested that observed variations in ground roughness factors for desert terrain could be explained by the changing gamma-ray energy spectrum. Data taken at Operation DOMINIC for events Sedan, Johnie Boy, Danny Boy, and Small Boy indicated there could be relatively large changes in the gamma-ray spectrum depending on type of weapon, location and position of the device, fallout particle size, neutron-induced activity in weapon and soil, and the decay of the radioactive fallout.

Calculations were made for an effective ground roughness factor over a period of time to determine how the ground roughness factor varied when the energy of the gamma rays of the radiation field changed. The calculations indicated that the product of attenuation times the buildup factor is virtually a constant over all energies; therefore, it is concluded that there is little variation in the ground roughness factor with a change in the energy of the gamma-ray spectrum.

1113. M. J. Schumchyk, J. H. McNeilly, W. C. Hampton, J. C. Saccenti, I. Murschel, and J. Jacobson, Measurements of Gamma Radiation and Gamma Spectra Versus Height Above a Fallout Field Simulated with Co-60, NDL-TR-70 (AD-474963).(November 1965). Availability: Army Nuclear Defense Laboratory, Edgewood Arsenal, Maryland.

This experiment determined the free-field dose rates, gamma radiation buildup factors, and air-to-ground dose-rate correlation factors at various heights above a fallout field simulated by a Co-60 source exposed at several source-to-detector distances. The experiment also provided sufficient data to permit interested investigators to determine the energy spectrum and angular distribution of the Co-60 gamma radiation above such a fallout field. Dose rates at detector heights from 3 feet to 1200 feet, for horizontal source-to-detector distances to 1200 feet, were measured with ionization chamber dosimeters. The uncollided radiation or narrow-beam dose rate at experimental conditions were calculated, and the ratio of the experimental dose rate to the narrow-beam dose rate was determined to yield the buildup factors for various source-to-detector distances. Gamma spectral measurements were obtained at various heights and angular directions with a collimator designed to accommodate a 3- by 3-inch NaI(Tl) crystal. Data was read out on a multichannel analyzer.

1114. W. P. Krauter, Radiological Armor Program - Methods and Data Compilation Vol. 1. Historical Summary and Compilation Description, FZK-200-1 (AD-468729) (July 24, 1965). Availability: Nuclear Aerospace Research Facility, General Dynamics, Fort Worth, Texas.

This document is Volume 1 of an initially projected seven-volume series which compiles data, methods, and techniques useful under the Radiological Armo Program for the U. S. Army Tank Automotive Center (ATAC), Warren, Michigan. The methods and data included in this compilation were extracted from documents which, since 1959, report the results of work performed at the GD/FW Nuclear Aerospace Research Facility (NARF). This first volume of the compilation presents a history of the radiological Armor Program at NARF, along with a complete description of the compilation methods and an outline of the contents of each volume in the series.

1115. J. D. Marshall, Measured and Calculated Data for Radiation Transmitted Through Ducts and Voids in Shields, AFWL-TR-65-164 (AD-477802) (December 1965). Availability: Air Force Weapons Laboratory, Kirtland Air Force Base, New Mexico.

Measured and calculated fast-neutron and gamma-ray dose rates are given for radiation transported through ducted shields. The shields were subjected to broad-beam radiation from the 10-Mw Aerospace Systems Test Reactor. A similar source model was employed in the calculations performed with the Monte Carlo Multibend-Duct Procedure (L05). The measured data were used to evaluate the calculational procedure and to establish criteria for the design of radiological armor containing either void or filled ducts. To aid in the comparison of various ducted-shield configuration, experimental data were integrated over exit surfaces. The measured-to-calculated dose-rate ratios for gamma rays vary from 1.1 to 2.2; for neutrons, from 0.5 to 5.6, with one isolated data point of 12. For the design of radiological armor it was found that (1) tolerance of simulated mechanical linkages passing through ducts is not critical unless personnel are in line with the transmitted radiation, (2) electrical cables "potted" with a low-viscosity epoxy may penetrate a shield without adverse effects, (3) wall material dose not have an appreciable effect on radiation streaming through an open duct, (4) a shield with several vision blocks may be as susceptible to radiation penetration as the same shield with one open duct, (5) high-density glass and/or indirect optical connections should be provided, and (6) the most effective position of a glass plug in a straight duct is nearest the source.

1116. B. W. Shumway, S. Tomoeda, and A. L. Frank, Shielding Effectiveness of Compartmented Structures in a Fallout Field, AD-423239 (POR-2220) (WT-2220) (April 29, 1963). Availability: U. S. Naval Radiological Defense Laboratory, San Francisco, California.

The objectives of Project 2.14 were: (1) to measure the penetration of fallout gamma radiation into six compartmented steel structures, and (2) to determine the free-field radiation characteristics of the fallout field.

Ionization chambers and film packets were placed inside steel compartments, which were then stacked to form six cubic structures measuring three compartments along each dimension. Each structure contained 27 compartments. The wall thickness of each structure was uniform. Thicknesses ranged from 1/4 to 1 1/2 inches. Dose measurements were made both inside and on the outside surfaces of the structures during various periods from II +74 to II + 262 hours.

Additional measurements were made with photographic film within steel slabs of thicknesses to match those of the compartmented structures. Comparison of measurements within the slabs with those within the compartmented structures gave a direct indication of extra dose attenuation produced by compartmentalization.

The free-field radiation characteristics of the fallout field were determined by three different means: (1) a survey meter to determine the dose rate at 49 equally spaced stations in a square array, 600 feet on a side, around the compartmented structures, (2) a truck equipped with a scintillation spectrometer for scanning the ground and measuring pulse-height spectra at various azimuthal and vertical angles, and (3) dose measurements taken at different heights above the terrain up to 45 feet.

1117. T. Sakai, M. Tanaka, T . Hirowatari, K. Nishimaki, Y. Nagahata, M. Shindo, and J. Miyakoshi, "Gamma-Ray Shielding Property, Strength, and Workmanship of Buildup Type Lead Slab Structure," ORNL-TR-843 (transl. from Hitachi Zosen Giho 25(4), 24-32 (November 1964). Availability: Dep. mn; JCL $3.60 fs, $1.19 mf.

Full-size buildup type lead slab structure as a part of secondary shielding for the marine nuclear reactor was constructed, and following items were investigated on the structure.

(1) Its workmanship

(2) Local deformation of the lead slabs and the bolts due to the vibration.

(3) Leakage of gamma ray through the slits at the part of joint and bolting.

Summaries obtained are as follows:

(1) The end face of joint part was not required special complicated geometrical form and the most simple, flat surface, was convenient in practical use from the point of view of the shielding and the strength.

(2) The lead slab structure being subjected to a vibration up to 500 Gal, (210 c.p.m., 10^6N), showed little local deformation. So this structure has no question against the vibration of ship.

(3) Also, semi-experimental formulas for the approximate calculation of leakage of gamma ray were derived against the slits in the part of joint and the bolting.

1118. K. Fahrmann, Determination of Fast Neutron Spectra in Iron with Threshold Detectors, ORNL-tr-1002 (June 1964). Availability: Dep. mn JCL $6.60 fs, $2.27 mf.

The present study serves for the preparation of spectrum measurements in the fast assemblies of the Rossendorf ring-zone reactor.

As shown by approximate calculations for the systems of interest, a knowledge of the neutron spectrum in the energy region of < 1 MeV is of special importance. Consequently, measurements were conducted in iron, where inelastic moderation of fast neutrons extensively raises the "soft" part of the spectrum. The measurements furthermore are of practical interest, since iron plays an important role as a shielding material of reactors.

Various inelastic scattering processes, in which low-level isomeric nuclei (< 350 keV) with half-lives of 10 min $< \tau_{1/2} <$ 12 d are formed, are investigated for their suitability for neutron spectrometry. The neutron spectra are determined in an iron block (50 cm ϕ x 45 cm) with incident fission neutrons by means of the reactions $Rh^{103}(n,n')Rh^{103m}$, $In^{115}(n,n')In^{115m}$ and the usual threshold reactions $S^{32}(n,p)P^{32}$, $Al^{27}(n,p)Mg^{27}$, and $Al^{27}(n,\alpha)Na^{24}$. In order to avoid absolute measurements -- particularly in the case of the isomer detection -- the detectors were calibrated in the fission spectrum of a thin enriched uranium converter. A perturbed fission spectrum was assumed as the spectrum model, where the perturbation function was represented as a superposition of exponential functions. The measured spectra are compared with those calculated by a multigroup diffusion calculation. The energy sensitivity of the selected reactions is compared to that of fission reactions (Np^{237}, U^{234}, U^{236}, U^{238}), in which the previously customary characterization by "effective threshold energies" was avoided.

1119. R . Nicks, N. Papmehl, H. Penkuhn, G. Perlini, C. Pleinevaux, C. Manduchi, and G. Zago, Experimental Study of the Propagation of Monoenergetic Neutrons in Water, ORNL-tr-1049 (a transl. of EUR-2161-F) (1964). Availability: Dep. mn; JCL $6.60 fs, $2.24 mf.

Neutron propagation in water has been measured with monoenergetic sources through a contract between EURATOM and the University of Padua. The 5.5 MeV Van de Graaff accelerator of the Physics Institute of Padua served to bombard neutron producing targets.

Measurements taken have yielded for water and oxygen a set of experimental values of the removal cross sections as a function of energy.

1120. J. Frippiat and R. Nicks, Gamma Radiation Curves Relative To Steel, Aluminum, and Uranium, ORNL-tr-1053 (Transl. from EUR-103.F) (1962). Availability: Dep. mn; JCL $1.60 fs, $0.80 mf.

The report presents the gamma doses due to thermal neutron activation of steel, aluminum, and uranium; the doses are normalized to 1-gram emitters and to distances of 1 cm.

1121. A. Honig, Computation of the Thickness of Lead Shielding Against Gamma Radiation, ORNL-tr-1059 (Transl. from Inzenyrske Stavby 7, 303-309 (1965). Availability: Dep. mn; JCL $1.60 fs, $0.80 mf.

Values of the gamma-ray dose buildup factor are given for lead in tabular and graphical form. They were computed from the Taylor fitting function and given for 23 energies between 0.5 and 10 MeV as a function of distance (cm) using an assumed density of 11.34 g/cm^3.

1122. A. Honig, Computation of the Thickness of Steel Shielding Walls Against Gamma Radiation, ORNL-tr-1060 (Transl. from Inzenyrske Stavby 8, 351-355 (1965). Availability: Dep. mn; JCL $1.60 fs, $0.80 mf.

Values of the gamma-ray dose buildup factor are given for steel in tabular and graphical form. They were computed from the Taylor fitting function and given for 23 energies between 0.5 and 10 MeV as a function of distance (cm) using an assumed density of 7.86 g/cm^3.

1023. A. Honig, Computation of the Thickness of Water Shields Against Gamma Radiation, ORNL-tr-1061 (Transl. from Inzenyrske Stavby 9, 399-404 (1965). Availability: Dep. mn; JCL $1.60 fs, $0.80 mf.

Values of the gamma-ray dose buildup factor are given for water in tabular and graphical form. They were computed from the Taylor fitting function and given for 20 energies between 0.5 and 10 MeV as a function of distance (cm) using an assumed density of 1.0 g/cm^3.

1024. A. L. Kaplan, J. Waldman, J. L. Jones, W. E. Barch, and R. L. MacNeil, Structure Shielding from Simulated Fallout Gamma Radiation, TO-B 65-27 (AD 629180, AD629217) (November 1965). Availability: AD-629180 (Final Report phase 1, CFSTI: HC$12.60 mf $1.00) AD-629217 (Final Summary Rept. Phase 1, CFSTI HC$1.60 mf. $0.50.

Experiments were performed to evaluate the application of the presently-used structure shielding theory to structures of simple geometry. Steel structures with cylindrical configurations and with rectangular configurations of different width-to-length ratios were used in these experiments to verify the theory, to study the effects of interior partitions and outer-wall apertures, and to evaluate theoretical predictions of in-and-down scattering below ground.

The shape factor verification experiments yielded results which the theory predicted to within 15%. For the interior partitions experiments, the theory predicted the experimental results to within 10% in most cases and 20% in all cases. For the structures with outer-wall apertures and no interior partitions, theory and experiment agree to within 15%. Introduction of the interior partitions, however, causes theoretical predictions to be up to 50% lower than experimental results at detector locations opposite the aperture, apparently caused by the failure of the theory to account for radiation scattering effects by the interior partition. The results of the basement shielding experiments showed agreement within 30% between theory and experiment for structures with no floor. However, the theory was a factor of 3 lower than the experimental results for structures with a 1/2-in. (20 psf) steel floor, indicating the requirement for a change in the floor barrier reduction factor currently used in the theory.

1125. R. H. Karcher and J. H. Wilson, Structural and Shielding Considerations in the Design of Hardened Facilities, HN-183 (Revised) (June 30, 1965). Availability: Holmes and Narver, Los Angeles, California.

Structural requirements and initial radiation shielding requirements have been calculated as a function of weapon yield and separation distance for fully buried domes, arches, and rectangular structures. The two criteria have been superimposed to determine the range and magnitude of the shielding problems attendant to the design of hardened structures. The single-degree-of-freedom spring-mass system was used to describe the response of structures to blast loading. Neutron penetration of structures was calculated by means of removal-diffusion theory while gamma-ray penetration was calculated using point kernel techniques. The interaction of structural and shielding criteria are analyzed in considerable detail and initial radiation shield requirements are given as a function of weapon yield for a number of discrete overpressure levels.

1126. J. M. Norwood, E. E. Jones, D. A. Goodwin, and P. L. Kleinjan, Investigation of Analytical Methods, Solution Sensitivity, and Structural Scattering in Neutron Transport, AFWL-TR-65-209 (March 1966). Availability: Air Force Weapons Laboratory, Albuquerque, New Mexico.

A limited investigation has been made of the effect of cross-section measurement error and of cross-section approximation upon the transport-equation solution at deep penetrations. The most serious error in deep penetration calculations appears to arise from inadequate treatment of anisotropic scattering; a P_1 scattering kernel is not adequate. Other aspects of the problem are included in the study. It is recommended that calculations be made to deep penetrations with transport codes capable of handling anisotropic scattering. An analytical approach to the solution of the multigroup formulation of the transport equation has been studied and the method yields the correct functional form of the vector flux in terms of the penetration distance and neutron direction of motion. Complete solution requires the inversion of large matrices. A method of inclusion of ϕ-dependence in the multigroup transport equation applicable to slab geometry has been investigated. The method should be practical for crew-compartment shielding calculations. The scattering of fast neutrons from structures has been subjected to a limited investigation by means of Monte Carlo methods, and approximate analytical formulas applicable to materials with strongly peaked forward scattering have been derived.

1027. A. E. Profio, Verification of Analytical Techniques (GAPLSN--Transport Theory and 05R -- Monte Carlo Theory) by Utilization of Measured Fast Neutron Spectra in Infinite Paraffin and Spherical Paraffin Shields, AFWL-TR-65-193 (GA-6765) (AD-478895) (March 1966). Availability: Air Force Weapons Laboratory, Air Force Systems Command, Kirtland Air Force Base, New Mexico 87117, Attn: WLDN.

High resolution measurements of the angular flux spectrum of fast neutrons in CH_2 have been made in point-source, infinite medium and point-source, finite medium, sphere geometries. Angular flux spectra were calculated with the GGSN and GAPLSN S_n transport codes and the 05R Monte Carlo code. Comparison with experiment established that P_3 anisotropic scattering and S_{16} angular mesh in the S_n codes were required to calculate neutron penetration to 30 cm. Results were relatively insensitive to the method of obtaining group average cross sections in the GAM-II code, but discrepancies above 8 MeV and below 1 MeV may arise from cross section errors. The S_n code did not calculate the zero-degree flux properly and future work should include small-angle calculations with a variable-angle S_n code. The Monte Carlo results were in good agreement with comparable S_n results, but disagreed with experiment above 8 MeV. Below 1-2 MeV statistical fluctuations were large at 30 cm penetration, even for 100,000 neutron histories and with source-energy biasing to reduce variance. Small-angle spectra were not calculated too well, particularly at the lower energy. However, part of this discrepancy is due to neglect of the uranium target in the 05R calculations. Recommendations are given for further investigations to resolve the remaining problems in the sphere measurements, and to extend the work to a two-dimensional geometry.

1128. G. T. Chapman, Gamma-Ray Attenuation Coefficients for Germanium, ORNL TM-1237 (March 2, 1966). Availability: Dep. mn CFSTI $1.00 cy, $0.50 mn.

An interpolation method based on the values listed in NBS-583 for five elements adjacent to germanium was used to determine the gamma-ray attenuation coefficients of germanium. If the region of the atomic numbers, Z, is relatively small and the gamma-ray energy is not too close to the k-shell energies for the elements in this range of Z, the attenuation coefficients will vary smoothly and not too drastically as a function of Z.

1129. H. Goldstein, Some Recent Calculations on the Penetration of Fission Neutrons in LiH, NDA-42 (August 7, 1957). Availability: Nuclear Development Corporation of America, White Plains, New York.

Fast neutron penetration from a point isotropic fission source in an infinite medium of LiH has been recalculated by the moments method using recently measured values of the non-elastic cross section as the absorption cross section. The newly computed doses fall faster than the previous theoretical values up to penetrations of about 60 gm/cm^2. At this distance, the dose found here is about one-half that predicted earlier. For larger penetrations, the new calculations decrease less rapidly than the old results, and the two dose curves cross about 105 gm/cm^2. Up to 60 gm/cm^2, the relaxation length for the dose curve is closely 5.44 gm/cm^2; from 60 gm/cm^2 to 120 gm/cm^2, it is 7.60 gm/cm^2. From the slope of the dose curve up to 70 gm/cm^2 a removal cross section for Li of 1.07 b can be deduced, which is only .06b larger than the experimental value, but the magnitude of the dose, up to 90 gm/cm^2, is better given by a Li removal cross section of 0.88 b. Number spectra have also been calculated and show the expected "hardening" with distance. The calculations could be further improved by taking into account secondary neutrons and by including revisions in the elastic scattering angular distributions. It is hoped these calculations can be undertaken in the near future.

1130. L. L. Lepisto and H. J. Tushar, Analysis of OTT Configuration 43A Using the Two-Component Method, DC 60-12-148 (December 27, 1960). Availability: General Electric, Cincinnati, Ohio.

The two-component method of shield analysis described in DC 60-4-87, was used to calculate neutron dose rates outside the OTT-43A arrangement. The results of the two-component method employing a $\cos^1$ and $\cos^3$ angular distribution for diffuse neutrons leaving the slab surface are compared to existing experimental data at the 9, 13, and 50 foot detector positions.

Good agreement was attained between the two-component method and experimental data. The best results were achieved with a $\cos^3$ angular distribution for the case of a hydrogenous shield material.

The two-component method using a $\cos^3$ angular distribution predicted a neutron dose rate 8.7% higher than the experimental value at the 50-foot position.

1131. T. R. Herold, Energy Spectra of Neutrons and Gamma Rays from Spontaneous Fission of Cm-244, DP-949 (July 1965). Availability: Dep. mn CFSTI, $2.00 cy, $0.50 mn.

The energy spectrum of the neutrons from the spontaneous fission of ^{244}Cm was measured from 0.5 to 6 MeV with a solid state ^{3}He spectrometer. The spectrum shows a maximum at about 0.75 MeV and decreases exponentially above 2 MeV.

The energy spectrum of the gamma rays that were emitted within 180 nanoseconds of the spontaneous fission neutrons was measured from 0.1 to 2.5 MeV with a gamma-ray spectrometer that included the ^{3}He spectrometer in a coincidence circuit. The gamma-ray spectrum shows a dual-peak maximum at about 200 keV and decreases exponentially above 800 keV.

Both measured spectra are similar to the corresponding spectra from the spontaneous fission of ^{252}Cf.

The measurements were made with a purified sample of 95.5 wt% ^{244}Cm that was produced in a Savannah River reactor.

1132. M. Kitazume, Some Considerations of Buildup Factors in Gamma-Ray Penetration for Multiple Layers. ORNL-tr-1134 (Transl. from J. At. Energy Soc, Japan 7, 9, 496-499 (1965). Availability: Dep. mn; JCL $1.10 fs, $0.80 mf.

A comparison is made of various formulas used to represent the buildup factor for multiple layers.

1133. E. Bagge, Shielding Experiments in the Geesthacht Shielding Reactor (Annual Report 1963), ORNL-tr-1048 (Transl. from EUR-2197-D) (December 1964). Availability: Dep. mn; JCL $11.00 (fs), $4.64 (mf).

Shielding experiments with the Geesthacht research reactor, started in the scope of a contract between Euratom and the Gesellschaft fur Kernenergieverwertung in Schiffbau und Schiffahrt mbH., were continued in 1963.

For the investigations the following experimental equipment, which had been described in detail in the annual report of the shielding group for 1962, were available:

Equipment for radiation shielding rests with a collimated beam (ESTAKOS)

Equipment for radiation shielding tests with large plates (ESTAGROP I)

A special work group developed and tested a number of measuring instruments and equipment required for these experiments and continued with the further development of methods for the measurement of neutron and gamma spectra.

In parallel to the experiments, a group of theorists conducted investigations with the research reactor of Geesthacht for the calculation of the shielding effect of shield arrangements.

1134. H. G. Fendler, A. L. Sellen, and K. Werner, Shielding of a Boiling Water Ship Reactor, ORNL-tr-887 (Transl. from Atomkernergie 10, 1-2, 11-22) (1965).

In the introduction, the paper deals with the ship project of a tanker with nuclear propulsion. It is followed by a discussion of the demand to be fulfilled by the shielding of the reactor and turbine, since these are operated with active steam and the manner of satisfying these requirements is described. A further part of the study presents some suggestions for the primary shielding. For the shield selected in this project, three-group calculations for neutron- and 10-group calculations for gamma-radiation are discussed and the results are presented.

1135. I. Umeda, Measurement of Dose Buildup Factors of Iron Slabs for a Line Isotropic Source, ORNL-tr-1133 (Transl. from J. At. Energy Soc. Japan 7, 9, 480-484 (1965). Availability: JCL $1.60 (fs), $0.80 (mf).

The purpose of this paper is to obtain experimentally dose buildup factors for transmitted gamma rays from an infinite line isotropic source through an infinitely extensive slab placed in parallel with the source.

In the measurements, an alternative arrangement was adopted, giving results equivalent to the line isotropic geometry with a point isotropic source (about 1 Ci ^{60}Co and ^{137}Cs), an air equivalent cylindrical ionization chamber (3 cm in diameter by 200 cm long) and iron plates (182.9 cm x 91.4 cm x 0.55 ~ 5.00 cm thick).

Up to 1 mean free path slab thickness, the dose buildup factor of the combination of infinite line isotropic source with infinitely extensive iron slab is the same as that with infinite iron medium. For thicknesses of 1 ~ 3 mean free paths, it is smaller with the former combination by about 7%. In excess of 3 mean free paths, the dose buildup factor of a point isotropic source in an infinite iron medium can be used instead of the infinite line source -- infinite slab combination without causing errors beyond 5%.