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ABSTRACTS

OF THE LITERATURE EXAMINED BY THE RADIATION SHIELDING INFORMATION CENTER

(REACTOR AND WEAPONS RADIATION SHIELDING)

RADIATION SHIELDING INFORMATION CENTER

ORNL-RSIC-6 (Vol. II)

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Neutron Physics Division

ABSTRACTS OF THE LITERATURE EXAMINED BY THE RADIATION SHIELDING INFORMATION CENTER

(Reactor and Weapons Radiation Shielding)

JANUARY 1969

NOTE:

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A B S T R A C T S

OF THE LITERATURE EXAMINED BY THE RADIATION SHIELDING INFORMATION CENTER

(Reactor and Weapons Radiation Shielding)

The abstracts included in this loose-leaf binder cover all the reactor and weapons radiation shielding 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, 1968), the latest edition of the bibliography is ORNL-RSIC-5 (Vol. II), Bibliography, Subject Index, and Author Index of the Literature Examined by the Radiation Shielding Information Center (Sept. 1968), which is being issued along with these 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 which is synchronized with this volume. That is, ORNL-RSIC-5 (Vol. II) and ORNL-RSIC-6 (Vol. II) both cover accession numbers 1136-1700.

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 Nucl. Sci. Abs. 22, No. 15, (1968).

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 index entries show which reports are on file in the depository libraries, which reports have been published in open-literature journals (journal citation given), and which reports can be purchased (source and price). Whenever a report is identified by more than one number, cross references from secondary to primary report numbers are cited.

The report number codes apply to AEC reports unless designated otherwise. Non-AEC reports are identified by parenthetical notes after the report codes; e.g., (non-AEC), (British), (Canadian), (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-85(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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(3) Microcard Editions in Microcard and micronegative form. Orders and inquiries should be sent to: Order Processing Department, 365 South Oak Street, West Salem, Wisconsin 54669.

(4) Clearinghouse for Federal Scientific and Technical Information, U. S. Department of Commerce, Springfield, Virginia 22151.

(5) Superintendent of Documents, U. S. Government Printing Office, Washington, D. C. 20402.

Public Availability of Non-AEC Reports

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*.

The availability of reports from Canada, Denmark, Norway, Sweden and the United Kingdom is indicated under CAN, DENM, NORW, SWED and UK in the section "Abbreviations Used in Availability Column."

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'Etude de l'Energie Nucléaire (C.E.N.), Studiecentrum voor Kernenergie (S.C.K.), Technical Information Dept., MOL-DONK (Belgium).

CERN

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

EURATOM

Reports from the Commission of the European Atomic Energy Community are available from: Office central de vente des publications des Communautés Européennes, 2, place de Metz, Luxemburg.

FRANCE

Commissariat à l'Energie Atomique. Inquiries for reports numbered 1-2199 should be addressed to Service Central de Documentation du C.E.A., Centre d'Etudes Nucléaires de Saclay, Boîte postale n° 2, 91. GIF-sur-YVETTE, FRANCE. Reports numbered 2200 and above are available from La Documentation Française, 31 quai Voltaire, 75, PARIS (7e), 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° 78 Le Vesinet (France) Telex 25667.

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GERMANY

Deutsches Elektronen-Synchrotron DESY. Copies are available from Deutsches Elektronen-Synchrotron DESY, Dokumentation, 2000 Hamburg 52, 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 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 Israel Atomic Energy Commission, Soreq Nuclear Research Centre, Yavne, Israel.

ITALY

Centro Informazioni Studi Esperienze (CISE). For sale at CISE Press Office, Casella Postale 3986, Milano, Italy. Copies are deposited in the libraries of various nuclear centers in Italy and abroad.

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), Tokai-mura, Naka-gun, Ibaraki-ken, 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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Institute of Nuclear Sciences. Copies are available from Librarian, N. Z. Institute of Nuclear Sciences, D.S.I.R., Private Bag, Lower Hutt, New Zealand. Copies are deposited in other countries.

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 free of charge from Nuclear Energy Information Center, Palace of Culture and Science, Warsaw, Poland.

SPAIN

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

SWITZERLAND

Eidgenössisches Institut für Reaktorforschung. "EIR" reports are available from Eidgenössisches Institut für Reaktorforschung, c/o Bibliothek, 5303 Würenlingen/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.

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

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Reports issued by organizations outside the United States will be sold by CFSTI *only to purchasers within the United States.*

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DENM Reports of the Atomic Energy Commission Research Establishment, Risö, are available on exchange from the Library, Danish Atomic Energy Commission, Risö, Roskilde, Denmark. Also for sale at the price shown from Jul. Gjellerup, Booksellers, 87 Sölvgade, Kobenhavn K., Denmark.

Reports of Danatom are available on loan from Research Establishment, Risö, Roskilde, Denmark. Copies are available from Danatom, Aurehojvej 2, Hellerup, Denmark.

Dep. AEC reports so designated are available at the AEC depository libraries in the United States and overseas. (See inside of front and back covers of *NSA*.) Prior to July 1, 1965, reports were deposited either in printed copy form or in microform. After that time, all reports have been in the form of micronegatives.

Reports from other countries so designated are available in the AEC depositories in the United States.

DTI Extension Available from the U. S. Atomic Energy Commission, Division of Technical Information Extension, P. O. Box 62, Oak Ridge, Tennessee 37830. (Conference papers are available on loan only.)

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 37830, contains information on all the CAPE- packages for sale.

Gmelin Available at a nominal charge and under the AED- number shown from Atomic Energy Documentation Center, Gmelin Institute, 40-42 Varrentrappstrasse, Frankfurt (Main), West Germany.

GPO For sale by the Superintendent of Documents, Government Printing Office, Washington, D. C. 20402.

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.

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

NORW Reports from Forsvarets Forskningsinstitutt are available from Norwegian Defense Research Establishment, Library, P. O. Box 25, Kjeller, Norway.

"KR" reports are available on an exchange basis from the Library, Institutt for Atomenergi, Box 40, Kjeller, Norway.

Reports of the Norwegian Reactor School are available from the Library, Institutt for Atomenergi, P. O. Box 40, Kjeller, Norway.

NRC For sale by the National Research Council of Canada, Ottawa, Ontario, Canada.

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UKAEA, London Available from United Kingdom Atomic Energy Authority, 11 Charles II Street, London, S. W. 1.

UN Mail orders should be directed to the Sales Section, Room 1059, United Nations Headquarters, New York, New York 10017.

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

ZLDI Zentralstelle Für Luftfahrtdokumentation Und-Information, Maria-Theresia-Strasse 21, Munich 27, Germany.

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1136. C. F. Ksanda, E. Laumets, and M. A. Hogan, Gamma-Ray Energy Pseudospectra for U^{235} Fission Products and for Various Times After Fission, USNDRL-TR-942 (AD-628982) (December 16, 1965). Availability: CFSTI, $3.00 cy, $0.75 mn.

The gamma-ray energy spectrum of U^{235} fission products following a nuclear explosion is made up of hundreds of different photon energies whose relative contributions to the spectrum change with time after fission. Gamma-radiation attenuation and shielding calculations can be greatly simplified by the use of "pseudospectra" that closely approximate the attenuation obtained with the "true" spectra. A technique for computing pseudospectra weighing fractions to obtain such simplified spectra composed of only 5 representative photon energies is described. Pseudospectra weighing fractions and pseudospectra for air and iron attenuation are presented. A flow chart and an IBM-704 computer program of the technique are also presented, along with the input data and examples of the printed output values. Throughout the time range considered (from 1.7 sec to 301 days after fission), the maximum attenuation error is less than $\pm 5\%$. The technique is valid for resolving almost any spectral attenuation data into five or fewer energy components, and may be used to solve problems involving five simultaneous equations in five unknowns with prescribed constraints.

1137. R. D. Richtmyer, A Non-Random Sampling Method, Based on Congruences, for "Monte Carlo" Problems, NYO-8674 (September 26, 1958). Availability: Dep. (mc); $7.80 (ph, CFSTI); $3.30 (mf, CFSTI).

The theory of systematic sampling, a possible competitor to random sampling for problems of Monte Carlo type, is discussed. It is based on the asymptotic distribution, in the unit cube in a space of many dimensions, of a sequence of points generated by congruences (mod 1) of the multiples of a vector with independent irrational components. Theorems based on algebraic number theory, and ones based on a statistical treatment, concerning the rate of decrease of the error with increasing sample size, are given. The method is illustrated by application to a simple, multiplicative, stochastic problem. The error is seen to be substantially smaller than when the sequence of points in the unit cube is a random one. This is strongly suggested by the algebraic theorems and confirmed by the statistical theorems; however, it is likely that, in practical Monte Carlo problems of the usual complexity, the improvement over random sampling will be evident only for very large sample sizes.

1138. S. A. W. Gerstl, "An Improved Double P_L Method Applied to Gamma Transport," Nucleonik 8, 2, 101-108 (1966).

A method for calculating the penetration of gamma rays in a finite slab by solving the one-dimensional energy and angular dependent transport equation is outlined. Using half-range spherical harmonics together with the usual full-range Legendre polynomials, we are able to formulate the boundary conditions exactly and to take the strong anisotropy of the scattering kernel fully into account. This could not be achieved by the usual spherical harmonics technique. In addition to the well known jump discontinuity in the energy spectra at the single-scattering cutoff, a new jump effect is produced with our method, which permits an elementary physical interpretation. Numerical results are shown for a double P_1 approximation. Its accuracy in the energy spectra surpasses that of an ordinary P_3 approximation inside the slab. Furthermore this method allows the presentation of spectra of the transmitted and reflected flux, and the discussion of their dependence upon the slab thickness.

1139. L. Szymendera and A. Strupczewski, "A Method of Approximate Evaluation of Initial Values for Shielding Computations," AEC-TR-4907 (pp. 537-547) translated from Nukleonika, 7, 7-8, 445-453 (1962). Availability: $12.95, CFSTI (entire publication).

The paper describes a method of approximate evaluation of neutron flux values at the core/reflector interface having no data from criticality computations. In this method it is sufficient only to know the basic data of designed reactor, such as reactor power, expected external surface of the core and multiplication coefficient. The method is based on the multigroup removal-age-diffusion approximation commonly used in shielding computations.

1140. Frank H. Attix, "The Meanings of 'First Collision Dose'," Health Phys. 12, 6, 793-99 (June 1966).

The term "first collision dose" has been used in several senses which physically differ from one another in subtle ways.

(1) First collision dose = absorbed dose in an isolated small mass.
(2) First collision dose = that part of the absorbed dose contributed by "primary" radiation only, in a small mass located anywhere.
(3) First collision dose = kerma.

Users of this term have been not always aware of the fact that it conveys different pictures to different people. The term "kerma" cannot be indiscriminately substituted wherever first collision dose is now employed.

1141. IAEA, Neutron Dosimetry, Vols. I and II, International Atomic Energy Agency, Vienna (1963). Availability: $13.00/set.

Volume I emphasizes energy discrimination and spectra. Volume II puts a greater weight on dose measurement, surveying, etc. These volumes, two especially, are an excellent summary description of neutron measurement instrumentation now in practical application.

1142. J. H. Thorngate and E. T. Loy, Post Pulse Gamma-Radiation Spectrum - Operation Bren, CEX-62.13 (June 1966). Availability: $1.25, CFSTI.

In order to understand better the gamma-ray field existing after the detonation of a nuclear weapon, a series of gamma-ray spectrum measurements were made as a function of time after a reactor pulse during operation BREN. Measurements were made with a 3-in. by 3-in.-NaI(Tl) detector located in a collimator which was aimed alternately at the reactor or 60° away. Although elaborate reduction of the data was not performed, the results indicate the gamma field was produced about equally by the fission products and by nuclear interactions between the delayed neutron and air.

1143. J. H. Thorngate, D. R. Johnson, and P. T. Perdue, Neutron and Gamma-Ray Leakage from the Ichiban Critical Assembly, CEX-64.7 (June 1966). Availability: $1.00, CFSTI.

This experiment was a continuation of studies begun in 1956 to determine the radiation doses received by the survivors of the Hiroshima and Nagasaki nuclear detonations. The "Ichiban" critical assembly was designed and constructed at Los Alamos Scientific Laboratory to provide a simulant for the Hiroshima bomb with respect to radiation "leakage" and spectrum. It was operated at steady-state power levels while the neutron and gamma-ray dose measurements were made. The results were corrected for room scattering and normalized to dose per fission. The first collision neutron dose measured at a distance of 9.5 meters was $7.7 \pm 0.3 \times 10^{-17}$ tissue rads per fission. Coupled with the mean dose per neutron of 1.14×10^{-9} rad/neutron/cm^2, this measurement represented a total fast neutron leakage of 0.77 ± 0.03 neutrons per fission.

1144. B. L. Berman, R. L. Bramblett, J. T. Caldwell, R. R. Harvey, and S. C. Fultz, "Photoneutron Cross Sections of Li-6," Phys. Rev. Letters 15, 18, 727-30 (UCRL-12376) (November 1965).

The photoneutron cross sections of Li-6 were measured as a function of photon energy from 5.7 to 32 MeV, using the monoenergetic photon beam achieved by the annihilation in flight of fast positrons from the Livermore Electron Linear Accelerator.

1145. D. Nachtigall and R. Rohloff, Sphere Techniques for Measuring Flux Densities and Dose Rates of Thermal, Intermediate and Fast Neutrons, SLAC-TRANS-23 (September 1965). Availability: Dep. (mn).

From the point of view of radiation protection, it is important to know the following quantities in a neutron field: flux density, energy dose rate, and dose-rate equivalent. The measurement of these quantities over the entire energy range of interest (10^{-2} to 10^{7} eV) has hitherto been a problem. The Bramblett, Ewing, and Bonner neutron scintillation counter with an LiI(Eu) crystal and polythene spheres as moderators is very useful for this purpose.

By means of a three-sphere combination (2, $3\frac{1}{2}$, and 11 inches in diameter), a neutron flux density can be obtained over the range of 0.5 eV to 7×10^{6} eV in a neutron field by summing the count rates obtained with the three spheres.

With a two-sphere combination (2-1/4 and 11-3/4 inches in diameter) and weighing factors, the dose rate curves for neutrons in the energy range of 10^{-2} eV to 7×10^{6} eV can be approximated.

A further three-sphere combination (5, 7, and 12 inches in diameter) and weighing factors in suitable for measuring the neutron dose-rate equivalent over the range of 0.5 eV to 7×10^{6} eV.

The biological dose curve, depending on the quality factor of Bond and Bateman, can be approximated over the range of 10^{-2} eV to 7×10^{6} eV by a three-sphere combination (2, 4, and 7 inches in diameter) and weighing factors.

Later changes of the generally used quality factors can be allowed for in the measurement techniques by other sphere combinations.

If one is interested in all four of the quantities mentioned, nine measurements with nine different spheres are necessary. The procedure can however be optimized. All four quantities can be measured over the energy range of 10^{-2} eV to 10^{7} eV as well as the average neutron energy, by five measurements with five different spheres (2, $3\frac{1}{2}$, 5, 7, and $11\frac{1}{2}$ inches in diameter) and weighing factors.

Besides the LiI(Eu) crystal detector, all other detector types based on the $^{6}Li(n,\alpha)$ reaction can be used, e.g., LiF thermoluminescence detectors.

1146. G. A. Ausman, Jr. and F. N. Wimenitz, Time-Dependent Neutron Transport from a Point Isotropic Source at an Altitude of 100,000 Feet, HDL-tr-1305 (AD-624041) (September 10, 1965). Availability: CFSTI, $7.00 (cy), $1.50 (mn).

Results of a Monte Carlo calculation of the neutron fluence and flux density in the vicinity of a point isotropic source at an altitude of 100,000 ft (30.48 km) are presented. The fluence and flux density were obtained at 20 detector locations as functions of time and energy for 12 monoenergetic neutron sources. The variation of atmospheric density with altitude is considered in these calculations. A brief discussion of the dependence of the results on detector location is given.

1147. R. Aronson and D. Yarmush, "Transfer Matrix Method for Gamma Ray and Neutron Penetration," J. Math. Phys. 7, 2, 221-237 (February 1966).

The problem of radiation transport is formulated in terms of a transfer matrix H which is a 2 x 2 matrix of operators. H is simply related to the more intuitive transmission and reflection operators T and R. An explicit expression for H is derived in slab geometry for radiation distributions that depend on the angle with the slab normal and on energy. H for a multilayer slab is the matrix product of the transfer matrices for the individual layers. A formal expression for H for a homogeneous slab of finite thickness is found in terms of the T and R appropriate to an infinitesimally thin slab. These in turn are related to the single-scattering distribution and therefore can be computed from the microscopic cross section. For purposes of computation, finite matrix representations for the operators must be introduced corresponding to the finite vectors which approximate the distributions. Expansions of the distributions in the cosine of the angle and group representations in energy were chosen in the present work. Some numerical results are presented for gamma rays on aluminum. The extension to problems with internal sources and to nonplanar geometries is outlined.

1148. S. P. Belov, V. A. Dulin, Yu. A. Kazanski, and S. G. Tsipin, "Spatial Distribution of 3 and 15 MeV Neutrons in Beryllium," J. Nucl. Energy, Pts. A/B, 20, 176-178 (1966).

No abstract.

1149. E. W. Webster, "Experiments with Medium-Z Materials for Shielding Against Low-Energy X Rays," Radiology, 86, 146 (January 1966).

No abstract.

1150. E. R. Epp, H. Weiss, "Experimental Study of the Photon Energy Spectrum of Primary Diagnostic X-Rays," Phys. Med. Biology, 11, 2, 225-238 (1966).

The photon energy spectra of primary diagnostic x-ray beams have been studied experimentally over the kilovoltage region 45-105 kvp using scintillation spectroscopy techniques. A method was devised which allowed the photon beam to be collimated without perturbing it and which permitted close reproduction of measurements. Correction for spectrometer distortion of spectra was done by means of a digital computer. Results are presented in terms of photon fluence per energy interval versus photon energy. Mean energies and half-value-layers were computed from the spectra. The half-value-layers were compared with independent experimental determinations using ionization chambers with essentially energy-independent responses over the photon-energy region of interest.

1151. NCRP, Measurement of Neutron Flux and Spectra for Physical and Biological Applications, NBS Handbook 72 (July 15, 1960). Availability: Superintendent of Documents, Washington, D.C. - $0.35.

The measurement of neutron flux and spectra is discussed, various methods are compared, and results of intercomparisons are given. Methods of measurements are discussed for the emission rate of radioactive neutron sources, thermal neutron flux, intermediate neutron flux, fast neutron flux, and neutron energy spectra. Neutron radiation instruments for area survey and personnel monitoring involving flux and spectrum measurements are included. Typical spectra of various neutron sources are shown.

1152. K. G. Randen and K. G. Golliher, Startup and Operation of the One-Megawatt Shield Test and Irradiation Reactor, NAA-SR-11175 (March 25, 1966). Availability: $2.00, CFSTI.

This report describes the experiments performed in connection with the startup of the Shield Test and Irradiation Reactor (STIR) and operating experience gained during its first six months of operation. The Shield Test and Irradiation Reactor (STIR) is a 1-Mw pool-type reactor used primarily to conduct the Basic Shielding Experimental Program under the AEC-sponsored General Supporting Technology Program at Atomics International. The reactor was first made critical on December 7, 1964 and was taken to full power for the first time on January 6, 1965. Some of the experiments described in this report were completed subsequent to the first full power run.

1153. R. H. Chesworth, Army Gas-Cooled Reactor Systems Program - ML-1A Shield Optimization Study, AGN TM-412 (December 1965). Availability: $4.00, CFSTI.

An evaluation of the results of this study indicates that it will be impossible to shield the ML-1A power plant in a way which will attenuate sufficient radiation to satisfy the dose rate criteria specified for the study without exceeding the package size or weight specifications. Since these latter criteria are considered to be fundamental to the mobility concept, it appears that the most promising method for reducing dose rates during and following reactor operation is to combine an optimization of the internal reactor shielding with an additional, external shield. Several external shield concepts were evaluated to select the design which involved the least compromise to plant shipping weight and startup time while retaining the maximum flexibility for maintenance activities.

The optimum shielding arrangement to satisfy the dose rate criteria specified for this study appears to consist of the following:

Modification of the reactor skid internal radial shield to the 3 in. tungsten-lead concept discussed herein.

Modification of the reactor skid internal axial shield to the conical, reduced height concept discussed herein.

Provision of a self-supporting membrane-type external water shield 44 in. thick as discussed herein.

Use of 1.5 ft of sandbag expedient shielding to reduce the dose levels to 4 mrem/hr at 100 ft during reactor operation.

1154. G. A. Vasil'ev, A. P. Veselkin, Yu. A. Egorov, V. A. Kucheryaev, and Yu. V. Pankrat'ev, "Attention of Reactor Radiations by Serpentine Concrete," J. Nucl. Eng., Pts. A/B, 20, 5, 390-97 (May 1966).

Experiments to study the shielding properties of concrete made with serpentine aggregate are described. It is shown that the shielding properties of such concrete in relation to neutrons are slightly better, and in relation to gamma radiation no worse, than those of limonite concrete. At the same time, such concrete withstands temperature considerably better and it can be recommended for use in shielding reactors up to at least 450°C.

1155. Kiyoshi Takeuchi and Iwao Kataoka, "A Method for the Numerical Integration of the Neutron Transport Equation in Slab Geometry," J. Nucl. Sci. Technol., 3, 5, 209-212 (May 1966).

No abstract.

1156. M. J. Berger and E. E. Morris, Dose Albedo and Transmission Coefficients for Cobalt-60 and Cesium-137 Gamma Rays Incident on Concrete Slabs, NBS Report 9071 (July 1966). Availability: National Bureau of Standards, Washington, D. C.

A tabulation is given of dose and reflection and transmission coefficients for gamma rays incident on concrete slabs. The coefficients are given as functions of: (1) the source energy (1.25 and 0.66 MeV); (2) the obliquities of incidence and emergence (both ranging from normal to grazing); (3) the relative azimuth of the directions of incidence and emergence; (4) the slab thickness (semi-infinite for reflection, up to four mean free paths for transmission). The results were obtained by a Monte Carlo calculation according to the collision density method, supplemented by analytical evaluation of the dose due to singly-scattered radiation.

1157. Georgia Nuclear Laboratories, Gamma Spectrum Measurements SNAP-8 Radiation Effects Test Program Volume I, NASA CR-54386 (ER)7448) (1966). Availability: CFSTI, $0.30.

Gamma-ray spectra resulting from Radiation Effects Reactor fission gamma rays and thermal neutron capture gamma rays filtered by water and lithium hydride have been obtained.

This work, originally issued as Georgia Nuclear Report ER-7448, was done under subcontract for the Aerojet General Corporation (NASA Contract No. NAS 5-417) in support of the SNAP-8 Radiation Effects on Materials and Components Program. The contract and subcontract were under the technical management of H. O. Slone and A. W. Nice, respectively, of the NASA Lewis Research Center.

1158. REIC, Nuclear Radiation Effects on Resistive Elements, REIC Memorandom 31 (July 15, 1966), Availability: Radiation Effects Information Center, Battelle Memorial Institute, Columbus Laboratories, Columbus, Ohio 43201.

This memorandum contains both steady-state (neutron fluence) and transient (gamma rate) effects information for resistive elements. Graphs of general trends of neutron-fluence effects as well as tabular summaries of the results of radiation studies are included.

Radiation effects in resistors generally are small in comparison to those experienced by semiconductors or capacitors. Generally, resistors can be used at neutron fluences as high as 10^{16} n cm^{-2} (fast) with resistance changes due to radiation being no more than would be expected from normal use.

An equivalent circuit for gamma-rate effects, data for the equivalent circuit, and a discussion of factors influencing gamma-rate effects on resistors are given. Although gamma-radiation effects are relatively small in resistors, these effects may be significant in some circuits at exposure rates as low as 10^7 R/sec.

1159. D. R. Doty, Differential Neutron Albedo for Thin Slabs, USNCEL-R-465 (August 1966). Availability: $3.00, CFSTI.

Data giving the experimental angular distribution of differential albedo for thermal neutrons incident upon thick scatterers of lead, high density concrete, and iron have been collected. Differential albedo of thermal neutrons on aluminum, iron, and polyethylene as a function of the material's thickness for a given incident and exit angles is shown. These data are compared with a theoretical formula.

Experimental verification of the theoretical formula is quite good for the angular variables. Asymptotic limits of the formula for thick and thin scatterers are as would be expected physically. The exponential dependence of the formula upon thickness agrees roughly within experimental errors. Reasons for inaccuracies in the magnitude of differential albedo are discussed.

An extension to general albedo measurements for thin scatterers at any energy is described. This method, employing elemental scatterers, is shown to yield reliable results in the thermal region when the thickness of the material is less than one mean free path. The method also does away with the need for a collimated detector in differential albedo measurements for thin scatterers.

1160. A. G. Lucas, Z. G. Burson, and R. E. Larquist, Design of a Shielded Source for the Irradiation of Natural Animal Populations, CEX-63.10 (EGG-1183-2054) (August 1966). Availability: $1.00, CFSTI.

This report discusses the design of a radiation source and shield that provides dose rates in the range 0.38 to 0.75 R/hr over a 20-acre circular plot. Calculations of the primary and secondary radiations contributing to the dose distribution at the soil surface were utilized in predicting the amount of shielding necessary to provide a uniform dose rate.

The source is placed on a 50-foot tower in the center of the irradiation site. A manually operated winch is used to raise and lower the source.

1161. F. P. Haywood, T. G. Provenaano, and J. A. Auxier, Operations Plan - Operation HENRE, CEX-65.03 (September 1965). Availability: $1.00, CFSTI.

A series of experiments, designated Operation HENRE (High Energy Neutron Reactions Experiment), will be conducted at the Nevada Test Site (NTS) in the summer of 1965. Basic and applied radiation studies will be incorporated in the operation which will utilize the 1527-ft BREN tower and a small, intense 14-MeV neutron source. The studies will be aimed at solving the problems of calculating the neutron and gamma-ray distributions originating from a point isotropic source. A comparison of experimental results with existing calculations and theories is expected to show which parameters contribute to the discrepancies now observed between calculations and experimental results.

The neutron source was designed to produce an average of 10^{13} n/sec for an operational period of at least four hours for each target. There will be approximately 30 targets available for the operation, thereby assuring approximately 120 hours of operating time. A description of the neutron source components, methods of operation, and safety plan is given.

The objectives and methods of measurement of the nine planned experimental programs are discussed in detail.

1162. NCRP, Shielding for High-Energy Electron Accelerator Installations, NBS Handbook 97 (July 1, 1964). Availability: U. S. Government Printing Office, Washington, D. C. $0.30.

The material contained in this Handbook has been under study for some five years. The long delay in issuing a report was caused mainly by the paucity of established data on protection from the radiations from high-energy electron accelerators. While many such accelerators are under design or construction for industrial, medical, and research needs, relatively few are in actual use, and these are mostly operated at moderate energies and limited outputs.

The problem of personnel protection is further complicated by the necessity of considering shielding or protection from electrons, bremsstrahlung, neutrons, and toxic gases -- all of which may be present at the same time. Many of the data presented represent extrapolations from known conditions and hence must be used with caution. It is believed that the formulations developed in this report are adequately presented, but their use presupposes that the user will have a substantial knowledge of the field of high-energy physics. It was hoped that a number of relatively straightforward protection recommendations could be made, and that specific examples of their use given. This has proven to be impractical, mainly because of the wide variations in the design and use of each installation.

For the reasons given above this report should be regarded as preliminary, in the sense that it is recognized as being far from as complete as desirable. Nevertheless it gives the best information on the subject available today, and points up clearly areas where more data are needed. As more machines come into operation it is hoped that more detailed protection recommendations can be developed.

1163. NCRP, Medical X-Ray Protection up to Three Million Volts, NBS Handbook 76 (February 9, 1961). Availability: Superintendent of Documents, Washington, D. C., $0.25.

This handbook contains recommendations for protection from x-rays generated in medical applications. Minimum requirements are given as well as information which should enable one to calculate the protection needed for situations not covered in the handbook. (RSIC)

1164. M. E. Anderson, An Elementary Guide to the Measurement of Fast Neutron Fluxes, MLM-1326 (July 17, 1966). Availability: $2.00 CFSTI.

This report is an introduction to the basic physics and mathematics involved in the measurement of fast neutron fluxes. It describes methods for detection of neutrons and the factors which must be taken into consideration when the measurements are being made. A glossary of some of the significant terms is also included.

1165. L. Stolyarova, "Use of Spectrometric Methods in the Physics of Gamma-Ray Shielding," AEC-tr-6698; Applied Spectrometry of Ionizing Radiation; Part Two - Applied Problems of Spectrometry; Chap-I, pp. 343-420 (1964). Availability: $8.25, CFSTI.

Some preliminary description of the transport of gamma rays is given as well as brief accounts of popular methods of calculation of gamma-ray transport. The experimental apparatus generally used in the measurement of gamma-ray spectra, namely the scintillation counter-multichannel analyzer, is described along with many of the problems of reducing pulse-height spectra to a gamma-ray spectrum. The discussion includes the usual experimental corrections, "unscrambling," and collimator effects. (RSIC)

1166. A. J. Ahlquist and F. W. Sanders, Gamma Dose Rate Measurements on the Phoebus 1A-321 Reactor, LA-3448-MS (May 10, 1966). Availability: $2.00, CFSTI.

Gamma-ray dose rate measurements were made on Experimental Plans 2, 3, and 4 of the Phoebus 1A-321 reactor testing series. Extrapolation of data from 180 feet gave a dose rate of 36.0 R/min-MW at the 60 foot station during the full power run (Experimental Plan 4). The average gamma-ray dose rate of the eight Kiwi reactors at the 60 foot station was 35.1 R/min-MW. Instrument difficulties necessitated using Xylene (gamma-neutron sensitive) detectors as gamma detectors. Integral doses show that they give a reasonable approximation of gamma dose rates.

1167. F. C. V. Worman and A. J. Ahlquist, Integral Gamma and Neutron Measurements on the Phoebus 1A-321 Reactor, LA-3393 (June 22, 1966). Availability: $3.00, CFSTI.

This report is an evaluation of the leakage radiation produced by the Phoebus 1A-321 reactor. Integral measurements of gamma dose, neutron flux, and neutron dose were made on all experimental plans that involved criticality. These were Experimental Plans 2, 3, and 4. A total of 69 stations were implemented on the three plans.

Gamma doses in air on Experimental Plan 4, the full power run, ranged from 1.15×10^6 rads at 60 feet to 2.50×10^1 rads at 2,100 feet. Thermal neutron fluxes declined from 96.5 percent of the total fast neutron flux at 60 feet to 28 percent at 3,000 feet.

Included in the report are additional gamma results obtained from film badges placed on arcs around the reactor with radii ranging from 1,000 to 128,000 feet.

1168. N. C. Dyer and R. L. Mundis, Experimental Shielding Study of the Argonne 12.5 MeV Tandem Van de Graaff Installation, ANL-7139 (January 1966). Availability: $1.00, CFSTI.

The radiation hazards associated with operation of the 12.5 MeV Tandem Van de Graaff Accelerator at Argonne National Laboratory were determined experimentally. The operating conditions of the accelerator during this study were such as would produce the greatest possible radiation hazards. The levels of penetrating radiation, both gamma and neutron, were measured outside the shield walls. With few exceptions, the radiation levels outside the shielding were far below 2.5 mRem/hr, which is the average level that would result in an exposure equal to the RPG (radiation protection guide) of 5 Rem/year. In areas adjacent to the accelerator that normally are occupied by personnel, such as the control room and data rooms, the radiation levels never exceeded 0.02 mRem/hr. A brief description of the Tandem Van de Graaff installation is given.

1169. E. W. Bennett, Graphs of Calculations of Energy Deposition and Air Fluorescence from Radiation Produced by a Nuclear Explosion in Space, LA-3439-MS (April 22, 1966). Availability: $3.00, CFSTI.

Results of calculations of energy deposition in the atmosphere and air fluorescence produced by x rays from a hypothetical nuclear explosion at an altitude of 10^5 km are presented in graphical form. The calculations were done for many parameters with the intent that the results would be useful in the detection of nuclear explosions in space, for example in systems design and signal interpretation, and in the prediction of atmospheric effects brought about by a high-altitude nuclear explosion.

1170. F. B. Golley and F. J. McCormick, Development of a 137-Cesium Facility for Ecological Research, TID-22499 (January 1966). Availability: $2.00, CFSTI.

This report describes the criteria, construction, and subsequent testing and operation of a 9200 curie 137-cesium source for ecological studies. The source was designed to be used as both a fixed radiation experiment and a portable facility. The cask and stand is approximately 47 inches high and 55 inches wide and weighs 1500 pounds. The cesium is exposed by opening two doors with handles attached to flexible shafts. These shafts extend through a concrete wall shield. The cask can be lifted onto a truck and transported to the field for *in situ* exposure of ecosystems. Safety precautions, shielding requirements and examples of dosimetry are also discussed.

1171. J. K. Warkentin, Gamma Energy Release from Fission Product Decay Following a Short Reactor Transient, IDO-17161 (April 1966). Availability: $2.00, CFSTI.

This report presents the results of an investigation to determine the rate of gamma ray energy release from fission product decay following a short power transient. The investigation involved several experiments in which the energy release rate as a function of time from 1 to 10^4 seconds after fission was obtained by measuring the decay gamma dose rate near the essentially unshielded SNAPTRAN-1 reactor following a transient of less than 300 milliseconds duration. The results obtained are compared with both experimental and theoretically computed data found in the literature. For decay timess less than 200 seconds, the results are in good agreement with the referenced experimental data but are considerably higher than the computed data. For longer decay times, agreement is good with all the referenced data.

1172. M. Alberg, K. O'Brien, J. McLaughlin, and A. Lazonoff, An Experimental Determination of Differential Energy and Angle Spectra for 137-Cs Gamma Rays in Sand, HASL-167 (February 1966). Availability: $2.00, CFSTI.

Differential energy and angle spectra from a point isotropic ^{137}Cs source in an effectively infinite medium of sand have been determined for gamma ray penetrations of 1, 2, and 3 mean free paths. The spectra were obtained from scintillation spectrometer measurements by an analytic method based on the Scofield iteration scheme. An integration of the results over all angles yielded differential energy spectra which are consistent with interpolated values of Goldstein and Wilkins' moments method calculation for aluminum.

1173. Z. G. Burson, Experimental Radiation Measurements in Conventional Structures - Part I. Radiation Measurements in Two Two-Story and Three One-Story Typical Residential Structures Before and After Modification, CEX-59.7B (July 1966). Availability: $1.50, CFSTI.

An experimental study designed to provide a basis for estimating protection against fallout radiation was performed on two two-story structures with basements and on three one-story structures without basements. These structures, located at the Nevada Test Site, were studied before and after modifications. This study, sponsored by the Civil Effects Branch, Division of Biology and Medicine, U. S. Atomic Energy Commission, added data to the nuclear energy civil effects research on radiation shielding provided by structures.

An idealized fallout-radiation field was simulated with the use of the Mobile Radiological Measuring Unit. This unit used a sealed radioactive ^{60}Co 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 experiments were conducted simultaneously with an architectural-engineering study to develop and demonstrate practical means of improving fallout-radiation protection by economical modifications of existing typical houses. Experimental data were gathered at each house before and after shelters were constructed or modified. The effects of various practical modifications of the existent structures, such as adding transite panels to the walls, building planters, or adding baffle walls, were also considered.

1174. F. W. Sanders, Gamma Dose Measurements -- Kiwi Transient Nuclear Test, LA-3446 (March 29, 1966). Availability: $2.00, CFSTI.

Group H-8 of the Los Alamos Scientific Laboratory's Health Division measured gamma-ray dose rates after the destruction test of a prototype nuclear rocket reactor which was designated the Kiwi Transient Nuclear Test. Dose rates were measured around the test point out to 500 feet and in the downwind direction out to 8,000 feet. It was found that even as close as 100 feet from the test point dose rates were affected by wind direction. The passage of the effluent cloud was observed at 4,000 and 8,000 feet. Insodose rate contours were determined at several post-event times, and estimates of integrated dose to personnel making emergency entries were prepared.

1175. T. E. Dudley and M. R. Mendelson, Group Constants for Gamma Ray Heating, KAPL-M-6497 (November 1965). Availability: Dep. mn; CFSTI, $3.00 (cy), $0.75 (mn).

A trapezodial rule quadrature solution of the infinite medium gamma ray slowing down equation is described, using the Klein-Nishina scattering kernel and cross sections computed from bivariate polynomial expansions. The resulting spectrum is used to weight few group constants for gamma ray heating calculations. Comparisons are then made between the modified energy slowing down treatment used in KARE gamma ray heating calculations with a more accurate few group treatment, for several infinite medium problems. It is found that discrepancies of ten to twenty percent in groupwise heating rates may exist in the infinite medium KARE calculations, indicating that spatial calculations might exhibit similar effects and suggesting further research in that area. The derivation of a consistent few-group formulation of gamma ray transport, as well as a variational approach based upon the use of two overlapping spectra is developed. The accuracy and usefulness of these methods will be investigated in the next phase of this work.

1176. O. L. McDermed and T. W. DeVries, Expansion of Fallout-Radiation Number-Spectra Calculations to Low Altitude, FZK-291 (AD 634534) (July 1, 1966). Availability: CFSTI, $4.00 (cy), $1.00 (mn).

Gamma-ray number and dose-rate spectra in the air above a smooth, quasi-infinite ground uniformly covered with fallout fission products are calculated. The spectra are calculated for 11 decay times, from 1.12 hours to 97.3 days, and at 7 altitudes, from 3 ft to 200 ft. The spectra are analyzed for the effects of altitude and decay time on spectral shape, with trends illustrated by computer tabulations and selected graphs.

Results show that the spectra soften with increasing altitude and that they soften with decay time for about 2 days, become slightly harder for about 3 weeks, then resume softening. Reduction of total dose rate with altitude is nearly independent of decay time (below 400 feet). A related study, to determine the effects of ground roughness, is recommended.

1177. W. G. Davey, "An Analysis of the Fission Cross Sections of 232-Th, 233-U, 234-U, 235-U, 236-U, 237-Np, 238-U, 239-Pu, 240-Pu, 241-Pu, and 242-Pu from 1 keV to 10 MeV," Nucl. Sci. Eng. 26, 2, 149-169 (October 1966).

The published cross sections of ^{232}Th, ^{233}U, ^{234}U, ^{235}U, ^{236}U, ^{237}Np, ^{238}U, ^{239}Pu, ^{240}Pu, ^{241}Pu, and ^{242}Pu from 1 keV to 10 MeV have been carefully studied to select best cross sections for fast reactor analysis. Frequently, the measurement is not of the absolute cross section but of the ratio between the unknown and some reference cross section; thus, in comparing measurements, it is important to determine if the same reference data were used in all cases. In this study, emphasis has been placed on determining the reference data used and, where necessary, the published data have been revised to accord with more accurate, currently accepted cross sections. Thus, it is believed that a consistent set of cross-section data has been derived.

Some cross checks have been made by comparing calculations based on the selected cross sections with integral measurements in broad fast-neutron spectra.

The study shows the great importance of the ^{235}U fission cross section in deriving other cross sections and emphasizes the necessity of re-evaluating nearly all fission cross sections, if it proves necessary to revise the ^{235}U data.

1178. L. V. Groshev, A. M. Demodov, V. N. Lutsenko, and V. I. Pelekhov, Atlas of Gamma-Ray Spectra from Radiative Capture of Thermal Neutrons, Pergamon Press, Inc., New York (1959). Availability: $20.00.

The main part of the book consists of gamma spectra measured with our magnetic Compton spectrometer, since they give a more complete idea of the total gamma radiation of the nucleus than do the data of other authors. Tables of gamma ray energies and intensities, taken from various sources, and gamma transition diagrams of nuclei formed by neutron capture, are also given.

As well as data already published, the Atlas includes gamma spectra of 19 elements which we have recently obtained with the magnetic Compton spectrometer, and information on low-energy gamma lines in the nuclei Cu^{64}, Cu^{66}, In^{116}, Ag^{108}, and Ag^{110}, obtained with the scintillation spectrometer by V. V. Sklyarevskii, E. P. Stepanov, and B. A. Obinyakov.

The Atlas begins with a table of stable isotopes, their thermal neutron capture cross sections, and the binding energies of the neutron in the resulting nuclei. This table also includes certain other data needed in the analysis of gamma spectra and the construction of gamma transition diagrams.

There are two appendices, which contain data on conversion electrons emitted in thermal neutron capture and on gamma rays from radioactivity which appear in gamma spectra from the (n,γ) reaction.

Data on gamma rays emitted in resonance neutron capture are not included.

1179. Alfred J. Budka and Thomas J. Dolce, Secondary Gamma Ray Transmission Through Plane Slabs of Concrete, BRL-1307 (AD-633216) (September 1965). Availability: CFSTI $4.00 (hc), $1.00 (mf).

This report contains curves to be used for estimation of number or dose leakage of gamma rays from distributed sources gamma ray in concrete slabs. The curves are for distributions which can be sufficiently well represented by a fourth order expansion in the space variable. Energy dependence is taken as discrete. A fuller explanation of the data is to be found in BRL-1284 and BRL-1223.

1180. D. E. Wood and D. E. Heagerty, Hardening Requirements and Neutron Effects Studies, Vol. II. A Comparison of Monte Carlo Calculations and Threshold Foil Measurements of 14-MeV Neutron Attenuation in Borated Polyethylene, KN-684-66-29A (AD-480476) (January 20, 1966). Availability: Kaman Nuclear, Colorado Springs, Colorado.

A study of 14 MeV neutron attenuation in borated polyethylene has been performed. Theory and experiment have been compared for both attenuation and detailed energy spectra. Calculations of the neutron fluxes were done with a Monte Carlo code which treats the collimator (a hole in a concrete slab) and the test shield (a finite slab of borated polyethylene) together. Measurements were made with several threshold detectors in the form of foils whose activity was determined by standard beta and gamma counting techniques. The neutron spectrum was determined from the foil activities by use of the Lagrange Interpolation Method of unfolding spectra from multiple threshold detectors.

The threshold detector method was proven applicable to spectra from 14 MeV neutrons penetrating a shield, although additional detectors would be desirable. The agreement between theory and experiment for attenuation and the higher-energy part of the energy spectrum shows that the Monte Carlo method and cross sections are reasonably accurate. Disagreement between the spectra below 5 MeV could be caused either by uncertainty in cross section data or by inadequate threshold detector data at the lower energies.

1181. R. Schamber, G. Grochowski, and R. Hanfling, MCR Critical Assembly Shield Experiments, GMD 3078-26 (EDR-4193) (May 3, 1965). Confidential.

MCR shield design and analysis techniques were tested by comparing calculations with measurements for experiments simulating major portions of the MCR shield. A split-table critical assembly of the MCR was the radiation source.

The analytical methods used were one-dimensional Sn transport theory and both one- and three-dimensional Monte Carlo techniques.

Experiments on a cylindrical mockup of the inner shield regions showed that all of the techniques adequately predict neutron activations and gamma heating in the MCR fixed (after-shut-down) shield.

Experiments in which rectangular slabs simulated nearly the full radial configuration of the MCR shield yielded neutron and gamma-ray dose rates closely predicted by the three-dimensional analysis but significantly underestimated by the one-dimensional spherical methods.

Further experiments and analyses are required to establish the validity of one-dimensional methods as applied to the near-cylindrical MCR shield and to test engineering shield features not addressed in experiments to date.

1182. R. Schamberger, S. Kellman, and M. Silberstein, MCR Shield Spectrum Generator Design Report, GMAD 3078-22 (EDR-4188) (May 3, 1965). Confidential.

This report describes the design of the Shield Spectrum Generator (SSG), a new device proposed as the basis for the principal experimental program for developing a sound MCR shield design. In the SSG, a subcritical depleted uranium cylinder transforms a 14-MeV neutron source, produced by a commercially available neutron generator, into a leakage source whose geometry and energy spectrum are similar to those of small fast reactors similar to the MCR. Reflector and shield regions constructed from building-block materials around this source are capable of representing the important geometric features of the MCR, including coolant ducts and other inhomogeneities. Design calculations are presented which show that the significant sources of radiation dose from the MCR are also present and measurable in a simulated MCR type shield exposed to the SSG. A program of cylindrical, slab, and duct experiments is outlined which, in conjunction with analytical calculations, should materially reduce the present undertainties in the design of MCR-type shields.

1183. F. H. Welch, Lithium Hydride Technology II. Fabrication of Shadow Shields for SNAP Application, NAA-SR-9400, Vol. II (September 30, 1965). Availability: $1.05 USAEC, DTIE, Oak Ridge, Tennessee.

The technology of lithium hydride casting is described as a method for fabricating SNAP reactor neutron shielding. The background, development, and current status are presented. The casting process and the shield casting hardware are described in detail.

All SNAP system shields and test specimens produced to date by the casting process are described individually. The method is capable of repetitively producing high-purith ($>$ 97.5 wt% LiH), high-density ($\sim$ 94% of theoretical density) lithium hydride shields rapidly and economically.

1184. Atomics International, SNAP Systems Capabilities, Vol. 2. Study Introduction, Reactors, Shielding, NAA-SR-11685, Vol. 2 (December 1, 1965). Availability: $3.45, USAEC, DTIE, Oak Ridge, Tennessee.

This report details the development status of the SNAP reactor shield designs. (RSIC)

1185. R. K. Hunter, Jr., GUMDROP Neutron and Gamma Ray Radiation Backgrounds, AFWL-TR-65-108 (December 1965). Availability:

The radiation background from the GUMDROP shot is given.

1186. D. L. Coffey and K. F. Wylie, Gamma Ray Dose Rates from Shielded Pu-238 Sources, MLM-1267 (February 14, 1966). Availability: $0.35, USAEC, DTIE, Oak Ridge, Tennessee.

Theoretical gamma ray dose rates from shielded, cylindrical plutonium-238 sources were computed and compared to measured dose rates. The measured and theoretical gamma ray dose rates agreed when suitable instrumentation and procedures were used and when certain theoretical effects were taken into consideration. The agreement between the values indicated that the published mass attenuation coefficients for plutonium and the elements used in the shielding materials are sufficiently accurate for computing dose rates. The agreement also validated self-absorption corrections which assumed a spherically distributed source; the corrections were valid when the length to diameter ratio of a cylindrical source was three or less. This method of computing dose rates could be applied to any gamma emitting source.

1187. L. Harris, Jr., G. Sherwood, and J. S. King, "Fast Neutron Spectra in Water and Graphite," Nucl. Sci. Eng. 26, 4, 571-573 (Dec. 1966).

Measurements of the space-dependent neutron spectra in reactor shielding materials have been initiated at the University of Michigan Ford Nuclear Reactor. The ultimate purpose is to provide data for accurate comparison with shielding codes, such as NIOBE, DTF II, and O5R for deep-shield penetrations. A relatively simple solid-state proton-recoil telescope has been developed that can measure neutron energies above 2.0 MeV, with modest ($\approx$ 12%) resolution, in the presence of the intense ($\approx 10^7$ R/h) gamma levels from a reactor source. The present note reports initial measurements on water and graphite between 2.0 and 12.0 MeV for a flux direction normal to the core face and shield penetrations up to 60 cm. Careful comparisons have been made with multigroup NIOBE calculated spectra.

It appears that for this most simple case of normally directed flux, the one-dimensional NIOBE program gives a very good prediction of differential intensities. Additional measurements are in progress on other materials and at oblique scattering angles. It remains to be seen whether the effects of large inelastic cross sections and oblique flux angles will introduct substantial disagreement between measured and calculated spectra.

1188. K. Preiss and A. B. Chilton, A Comparison of Theory and Experiment for Shielding by a Structure Against Fallout Radiation, AD-636787 (University of Illinois Nuclear Radiation Shielding Studies Report No. 1) (June 1966). Availability: CFSTI, $3.00 (cy), $0.50 (mn).

A comparison is made between experiment and theory for the calculation of shielding by a structure against fallout radiation.

Experimental results for exposure penetrating a roof slab, and for reduction factor, are found to agree with moments theory calculations, often to better than 10% when the geometry factor $L_a(\omega,X)$ was used.

A comparison between experiment and theory may be inaccurate due to anisotropy of the experimental source, to lack of source reflection in roof experiments, and due to error in estimating the thickness of the roof. The magnitudes of the errors due to these effects are investigated, and found to be small, but not necessarily negligible.

Detailed results of penetration in iron and concrete, due to plane isotropic sources of various energies are given, and a review of ground contribution to fallout radiation penetration through simple structures is appended.

1189. A. B. Chilton and C. Y. Fu, Exposure Field from a Point Isotropic Source Located at the Surface of a Thick Concrete Slab, AD-637125 (University of Illinois Nuclear Radiation Shielding Studies Report No. 2) (July 1966). Availability: CFSTI, $3.00 (cy), $0.75 (mn).

This report gives results of computations for the exposure field from point isotropic sources of Cobalt-60 and Cesium-137 near the surface of a semi-infinite concrete slab. A modification of the Chilton-Huddleston differential albedo formula developed for the vacuum/concrete interface situation is used. To modify the results to include the air effect, computation was made to account for the single-scattered contribution. The higher order air scattering contribution, which is smaller, was estimated approximately from infinite buildup factors in air. The final results, with horizontal source-detector distances out to 50 feet, are presented in comparison with various experimental results. The computational results are estimated to have an uncertainty of about 1%. Agreement with previous work is good in some cases, but under some conditions appreciable discrepancies exist. The preponderance of evidence supports the theoretical results and they are recommended for future use.

1190. V. S. Sychevm, V. V. Malkov, M. M. Komochkov, and L. N. Zaitsev, Transmission of High Energy Neutrons in Heavy Concrete, ORNL TR-1221 (translated from JINR-P-2479) (1965). Availability: Dep mn.

An experimental investigation was made of the neutron penetration curve in magnetite concrete with different H_2O contents. Curves were determined for neutrons above 20 MeV, above 2.2 MeV and above 1.44 eV.

1191. R. L. Witcofski, F. C. Watts, L. L. Engstrom, and C. E. Blackwell, "The Gamma-Ray Evaluation of Disgnostic Shielding Barriers," Radiology 85, 6, 1123 (Dec. 1965).

A source consisting of Ce-144 and Pr-144 is used to mockup disgnostic x-rays. With it, a transmission curve for lead is measured.

1192. S. Buscaglione and R. Manzini, On the Buildup Factors for Concrete: Coefficients of the Polynomial Representation, ORNL TR-1211 (translated from RT/FIMA(62)2) (June 1965). Availability: Dep mn.

Polynomial fit coefficients are computed for the buildup factors for concrete published in ANL-643.

1193. M. Kangilaski, The Effect of Nuclear Radiation on Dispersion Fuel Materials, REIC Report No. 40 (January 1, 1966). Availability: Radiation Effects Information Center, Battelle Memorial Institute, Columbus, Ohio.

A complete summary of the irradiation performance of the various fuel materials dispersed in different matrices has been collected. The irradiation performance of the following dispersions is discussed in the report: $BeO-UO_2$, $BeO-UO_2-ThO_2$, $Al_2O_3-UO_2$, $MgO-UO_2$, UO_2, UC, UC_2, and $(U,Th)C_2$ dispersed in graphite; UBe_{13} and $(U,Th)B_{13}$ dispersed in beryllium; UO_2, U_3O_8, U_3O_8 glass, and UAl_3 dispersed in aluminum; UO_2, UC, and UN dispersed in stainless steel; UO_2 dispersed in nickel-chromium alloys; UO_2 dispersed in biobium, UO_2 dispersed in molybdenum, UO_2, UN, and UC dispersed in tungsten. The data which is summarized includes percentage of fuel by weight and volume, specimen shape and size, fuel particle size, cladding material and clad thickness, irradiation temperature, heat generation rates, irradiation times, fuel burnup, fission gas release, dimensional and density changes resulting from irradiation.

1194. W. R. Burrus, W. J. McCool, J. M. Miller, D. R. Otis, and J. Smolen, Study of Advanced Shielding Materials, ORNL-2106, Parts 1-5, pp. 269-73 (August 29, 1956). Availability: Dep, $18.00 fs, $8.75 mf, CFSTI.

An extensive mockup experiment was initiated in which combinations of advanced shielding materials, such as lithium hydride, depleted uranium, zirconium, and tungsten, are being investigated. In the initial tests the shielding materials being studied are combinations of lithium hydride and other shielding materials immersed in transformer oil. The particular configurations used have been those of immediate interest in the GE-ANPD program, but such general interest surrounds the use of these materials that the more reliable data obtained thus far are presented here. This report is preliminary, however, in the sense that little analysis work has been performed.

1195. Southern California Edison Company, San Onofre Nuclear Generating Station, Unit I - Shielding, NYO-3270-2, Vol. II, Sec. 5.2 (1966). Availability: Dep. mn CFSTI, $6.00 cy, $1.50 mn.

The purpose of the Final Engineering Report and Safety Analysis for the San Onofre Nuclear Generating Station, Unit 1, is to describe and analyze the physical and operational characteristics of the facility as specifically related to safety of the general public. This report is intended to reflect the standards of engineering employed and to show the integration of safety requirements within the station design.

In order to demonstrate the adequacy of the final design to meet specific safety criteria, analyses have been made of a variety of faul conditions. Of the faults considered, those selected encompass the range of possible departures from normal operating modes which could result in a radiological hazard to station personnel or the general public. These analyses are included in this report.

Radiation shielding is designed for operation at rated power of 1,347 Mwt, 450 Mwe, based on radiation protection criteria established in 10 CFR 20.

Five categories of shielding are incorporated in the plant. The categories are named for their functions: primary shield, secondary shield, fuel handling shield, radwaste shield, and auxiliary shields.

1196. P. J. Hemmings and Susan M. Offord, Photon Interaction Data in the U.K.A.E.A Nuclear Data Library, AHSB(S) R 109 (1966). Availability: HM Stationery Office, Seven Shillings Net.

Photon interaction data for 25 elements and water have been collected for use in the UKAEA Nuclear Data Library. The methods of photon interaction are discussed with reference to transmission calculations. Tabulations are given of cross sections for each element, and the angular distribution for Compton scattering is given at 23 energies. The energy range considered is 0.01 to 20 MeV.

1197. H. E. Hungerford and R. F. Mantey, "Shielding the Enrico Fermi Fast Breeder Reactor, Nucleonics 16, 11, 120-125 (1958).

Shielding a fast reactor is tougher than shielding a thermal reactor; not only is the neutron spectrum harder, but neutron leakage is greater. Here's how those problems were solved for the Enrico Fermi Reactor.

1198. Wilson Durward Leggett, III, The Effect of Pipes on Gamma-Ray Dose Rate in a Nuclear Reactor Shield, Thesis (1964). Availability: University Microfilms, Inc., Ann Arbor, Michigan, $4.80.

The experimental investigation described herein was initiated to supply data and their interpretation to use in the analysis of the shield penetration problem created by pipes in a nuclear reactor shield.

A part of the shield of North Carolina State Reactor-3 (NSCRC-3) is a tank, roughly a five-foot cube, which can be rolled out of the reactor shield, filled with a shielding material penetrated by ducts, and rolled back in to become a portion of the reactor shield again. Experiments were conducted with void 5 in., 8 in., and 10 in. iron pipes in wet sand (ρ = 1.5 g/cm^3).

Pipe geometries investigated were: (1) straight pipes with varying length-to-diameter ratios, wall thicknesses, and angles to the reactor core; and (2) pipes with two right-angle offsets with varying pipe diameters. Supporting experiments measured gamma-ray dose at several positions in the sand shield and angular distributions at two locations. For all experiments, the gamma-ray detection device was a 1 in. by 1 in. NaI(Tl) scintillation crystal which, with appropriate electronic equipment, counts gamma rays and measures their energy. In the pipe studies, the crystal was positioned in the center of the pipe mouth toward the shield exterior, the position which can be expected to determine the maximum effect of the pipe.

1199. R. E. Maerker and F. J. Muckenthaler, "Measurements and Single-Velocity Calculations of Differential Angular Thermal-Neutron Albedos for Concrete," Nucl. Sci. Eng. 26, 339-346 (1966).

Single-velocity Monte Carlo calculations and measurements have been performed to determine the differential angular thermal-neutron albedos for a reinforced concrete for monodirectional beams of incident thermal neutrons. Preliminary calculations using a statistical estimation technique indicate up to 50 scatterings should be followed for each neutron to produce good estimates of the differential albedos, and up to 100 scatterings to produce good estimates of capture gamma-ray differential dose albedos. Deviation between experiment and calculation can be reduced to an average of 5.1% for 72 points of comparison if an anisotropic scattering law for water deduced from earlier Argonne National Laboratory measurements is assumed.

1200. D. B. MacMillan, "Comparison of Statistical Estimators for Neutron Monte Carlo Calculations," Nucl. Sci. Eng. 26, 366-372 (1966).

Several statistical estimators that may be used in neutron Monte Carlo calculations are compared by means of idealized problems. The objective is to evaluate these estimators for use in the study of neutron thermalization and thermal neutron behavior by Monte Carlo techniques. It is concluded that the minimum variance linear combination of the analog estimator and the modified path length estimator (proposed recently by Spanier) is the best of these estimators for such applications. Where computer limitations require the use of a simple (rather than a combination) estimator, the modified path length estimator is recommended.

1201. V. A. Dulin, V. G. Dvukhsherstnov, Yu. A. Kazanskii, and I. V. Shugar, "Angular and Energy Distributions of Neutrons at the Boundary of Two Media," J. Nucl. Energy, Parts A and B, 19, 10, 810-818 (October 1965).

Results are given of measurements of the angular and energy distributions of fast neutrons (0.4-3.4 MeV) emerging from water, graphite, aluminum, iron, nickel and lead at the boundary of the medium with water following their passage through thicknesses equal to 1.5-4.6 free path lengths. The neutron source was the $^{2}H(d,n)^{3}He$ reaction. The measurements were made with a single-crystal fast-neutron scintillation spectrometer provided with gamma-ray discrimination. The results of these measurements are compared with theoretical calculations.

1202. Karl-Heinz Muller, "Particular Solutions of the Energy-Dependent Boltzmann Equation," Nukleonik 8(1), 38-40 (January 1966).

The Carleman-Kupradse method for solving a coupled set of integral equations with sectional holomorphic kernels combined with a method based on a variation procedure, of removing the energy from the field of free variables, allows the construction of particular solutions of the neutron transport equation. The nuclear data enter the calculation only after they have been weighted and integrated over the energy variables (E,E'). Thus, the procedure allows the use of cross sections in a numerical form (e.g., the usual energy group representation). An approximate solution can be found from a linear system of algebraic equations.

1203. A. L. Kaplan, J. Waldman, N. B. Koepp-Baker, W. E. Barch, and R. L. MacNeil, Final Report - Phase II, Structure Shielding from Simulated Fallout Gamma Radiation, TO-B 65-123 (USNRDL TRC-30) (AD-641134) (June 1966). Availability: CFSTI, $3.00 cy, $0.50 mn.

Two experiments were performed in this second phase of a study designed to evaluate the validity of the current structure shielding theory. The first experiment completed measurements begun earlier of in-and-down basement radiation. Here, the floor barrier reduction factor presently used to predict basement reduction factors theoretically was found to produce calculations lower, by a factor of between 1.5 and 7, than the experimental results. A new theoretical floor barrier reduction factor, which is a function of both the floor thickness and the solid angle fraction subtended at the detector by the floor, was constructed within the formalism of the existing structure shielding theory. This new function agrees quite well with both experimental results and Monte Carlo calculations, over the entire range of wall and floor thicknesses used in the experiment. The second experiment investigated the assumption that in evaluating the barrier reduction factor an inhomogeneous barrier can be treated theoretically as a homogeneous barrier of appropriate thickness. For the wall thicknesses, rib spacing, and rib size studied, this assumption was found to be quite satisfactory at distances behind the barrier larger than one-half the rib spacing.

1204. M. L. Eaton and C. M. Huddleston, A Random Walk Treatment of Neutron Diffusion in Slabs, USNCEL R482 (September 1966). Availability: CFSTI, $3.00 (cy), $0.50 (mn).

The concept of a Markov chain has been used to treat the random processes of scatter and absorption which occur when neutrons are incident on a slab of finite thickness, such as a shield or an inside wall of a shelter entranceway. It is assumed that scattering is isotropic in the laboratory system and that the scattering and absorption cross sections do not change during a neutron-scattering history. The encouraging results obtained to date indicate that for thin slabs the analytical random walk method may have important advantages over Monte Carlo calculations (which require lengthy computer runs to obtain acceptably small statistical variances) moments method calculations (which are actually appropriate only for infinite medium cases), and numerical solutions of the neutron transport equation (which are lengthy, tedius, and necessarily approximate).

1205. L. Jung, Experimental Verification of a Geometrical Shielding Transformation, ORNL TM-1603 (September 1966). Availability: CFSTI $3.00 cy, $0.75 mn.

An experimental verification of the validity of the attenuation kernel in lid-tank-type geometry has been performed at the Lid Tank Shielding Facility at the Oak Ridge National Laboratory. The theory, based upon the attenuation kernel concept, proposed for this investigation states that the axial dose rates of the Lid Tank 28-in.-diam source is predictable from measurements with smaller sources. The axial thermal-neutron fluxes in a water medium were measured from sources of 7, 9.33, 14, and 19.80 in. diameters. The results obtained from the smaller disk sources showed a maximum deviation of 10.6% with an agreement of 2.7% in the average of the deviations.

1206. W. E. Thompson and F. C. Engesser, Gamma Rays Resulting from Interactions of 14.7-MeV Neutrons with Sodium, Sulfur, Chlorine, Potassium and Lead, USNRDL-TR-1043 (AD-638104) (July 14, 1966). Availability: CFSTI $2.00 cy, $0.50 mn.

The prompt gamma rays resulting from the interaction of 14.7-MeV neutrons with the nuclei of the elements sodium, sulfur, chlorine, potassium, and lead, 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 generally predominates and that gamma rays resulting from the deexcitation of the lowest energy levels, with one or two notable exceptions, are usually most prominent. Gamma rays attributable to $n,p\gamma$, $n,\alpha\gamma$, and n,d process are also observed. In chlorine and potassium, gamma rays attributable to the $n,d\gamma$ process are unusually preponderant. Some of the most important gamma rays are found to have the following elemental production cross sections (in millibarns per steradian at 90 deg): 0.44 MeV gamma ray from Na, 39.5 $\pm$ 4.5; 2.24 MeV gamma ray from S, 15.3 $\pm$ 1.7; 2.15 MeV gamma ray from Cl, 17.1 $\pm$ 2.0; 2.17 MeV gamma ray from K, 19.3 $\pm$ 2.5; 2.62 MeV gamma ray from Pb, 20.3 $\pm$ 2.7.

1207. E. Bagge and E. Fischer, Shielding Experiments at the Reactor Station Geesthacht - Annual Report, 1964, ORNL TR-1225 (Translated from EUR-2494.d) (1965). Availability: Dep. mn JCL $16.00 (cy), $7.49 (mf).

The shielding experiments, started within the scope of the contract between EURATOM and the Gesellschaft fur Kernenergieverwertung in Schiffbau und Schiffahrt (GKSS), effective as of 1-1-1961, were continued in 1964.

In addition to the experimental installations "ESTAKOS" and "ESTAGROP I", described in the annual report of the shielding group for 1962 (EUR 534), the large irradiation channel of the FRG with the "ESTAGROP II" installation was available for the first time for these investigations.

A special working group developed and tested a part of the measuring instruments required for the experiments as well as the equipment and was occupied with the further development of methods for the measurement of neutron and gamma spectra.

Parallel with the experiments, studies for the calculation of the shielding effect of shield configurations were performed by the group of theorists with the use of FDR.

The studies reported in the present compilation were carried out at the Institute of Reactor Physics of the GKSS in Geesthacht-Tesperhude.

1208. M. J. Condat, P. LaFore, and J. Radtoin, Special Concretes for Protection in Reactors, Risley-trans-646 (translated from CEA-2312) (1963). Availability: Dep. mn.

Many of the factors involved in the use of concrete as a radiation shield are discussed. The topics treated are as follows:

Main factors influencing choice of shielding concretes,

General aspects of radiation attenuation,

Qualities and weaknesses of ordinary concretes,

Methods of placing the concrete,

Behavior in the presence of radiation (stability),

Physical, mechanical, and thermal characteristics of the principle concretes.

(RSIC)

1209. Ye. L. Stolyarovy, Instruments and Methods of Radiation Analysis (Translation), FTD-MT-64-179 (AD-617098) (March 4, 1965). Availability: $4.00 (fs), $0.75 (mf), CFSTI.

This is a machine translation of a collection of published works of the staff, post graduate and senior students of the Moscow Engineering Physics Institute (MIFI), reported in 1959 and 1960 at scientific conferences of teachers at MIFI. A total of 19 papers on theoretical and experimental aspects of shielding are presented.

The table of contents is as follows:

Conversion from Point and Other Very Simple Sources to Cylinder, by V. I. Popov
Research in Spectral-Angular Distributions of Scattered Gamma Radiations in Protective Barriers for the Case of a Two-Dimensional Unidirectional Source, by Ye. L. Stolyarova, S. G. Chukhin, I. Ye. Konstantinov, and A. I. Miskevich
Angular and Energy Distributions of Gamma Rays Co-60, Scattered in a Heterogeneous Medium of Al + Pb, by A. V. Larichev and L. F. Klimanov
Research on Sensitivity Single-Crystal Scintillation Gamma Spectrometer with a Crystal of NaI(Tl) 80 x 80 mm, by A. V. Larichev and G. A. Cherevatenko

1210. Shun-ichi Miyasaka and Akira Tsuruo, "Dose Buildup Factors of Multi-Layer Slabs for a Point Isotropic Source," J. Nucl. Sci. Technol. 3, 9, 393-400 (September 1966).

Dose buildup factors for multi-layer slabs composed of lead and graphite were obtained experimentally for a point isotropic source of ^{60}Co and ^{137}Cs. The experimental results are compared with those calculated by the Monte Carlo method, and show a good agreement.

The buildup factors for the second material behind the first increase -- or decrease -- very markedly for heavy-light -- and light-heavy -- layers. From these experimental results, it would be possible to assume that the dose buildup factor for the second material of a double layer slab is composed of two parts, namely asymptotic and transient. The transient part is that which increases or decreases very rapidly within about 1 mean free path from the interface of the first and second layers. The asymptotic part increases proportionally with the buildup factor for the second material. The rise or fall of the buildup factor in the second material can be explained from the change of the spectra in the second material.

From the experimental results, the dose buildup factors for multi-layer slabs may be expressed as follows:

$$B_{1,2,\cdots N}\left(\sum_{i=1}^{N} x_i\right) = B_{1,2,\cdots N\ asymp}\left(\sum_{i=1}^{N} x_i\right) + \left\{B_{1,2,\cdots,N-1}\left(\sum_{i=1}^{N} x_i\right) - B_{1,2,\cdots,N\ asymp}\left(\sum_{i=1}^{N} x_i\right)\right\} \exp(-b_{1,2,\cdots,N\times N})$$

$$B_{1,2,\cdots N\ asymp}\left(\sum_{i=1}^{N} x_i\right) = k_{,2,\cdots,NBN}\left(\sum_{i=1}^{N} x_i\right).$$

1211. Murrey D. Goldberg, Said F. Mughbghab, Benjamin A. Magurno, and Victoria M. May, Neutron Cross Sections, Vol. IIA, Z=21-40, BNL-325 Sec. Ed., Sup. 2, Vol. 2A (February 1966). Availability: CFSTI, $4.00.

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 elements whose atomic numbers are in the range 21 to 40.

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.

1212. F. H. Clark and N. A. Betz, Results of the Analysis by Monte Carlo of Fast-Neutron Shielding Properties of Compound Cubical and Spherical Shields, ORNL-4009 (AD-639572) (Sept. 1966). Availability: CFSTI, $3.00 (cy), $0.75 (mf).

An extensive program of calculations was performed to check certain fast-neutron dose-rate measurements made at Project BREN in 1962. The response characteristic of the modified long counter was computed. Nitrogen and oxygen cross sections were reviewed and used in calculations of the transport in air of neutrons from a Godiva source to a distance of 66 g/cm^2. Transport results obtained by Monte Carlo were compared with those obtained by the moments method. A spectrum at 80-g/cm^2 penetration was determined. The perturbating effect of a structure on the neutron field in its neighborhood was estimated. The effect of variations of the spectrum with penetration distances on doses computed with different weight functions was also examined; the Snyder-Neufeld multicollision physical dose was found to vary with distance in a way markedly different from others examined. Work was also done on importance sampling techniques, especially the exponential transform, and the sampling problem was formulated in a simplified, approximate expression to facilitate statistical interpretations. Finally, a number of shield penetration calculations were performed to compare with the measurements and to determine the effects of shield shape, shield cavity size, and source angular distributions incident on the shields. The agreement between the calculations and the measurements was within the uncertainty in the detector response characteristic.

1213. F. H. Clark and N. A. Betz, Importance Sampling Devices for Selecting Track Lengths and Directions after Scatter in 05R, ORNL TM-1484 (Aug. 5, 1966). Availability: Dep. mn CFSTI, $1.00 cy, $0.50 mn.

Several importance sampling schemes are discussed in some detail. A path length stretching scheme and an exponential transform scheme, both available in 05R, are described analytically and by means of listings and flow charts. A scheme to adapt the two preceding to "never escape selection," not available in 05R, is described analytically only. A scheme for biased selection of Laboratory direction after scatter, available in 05R, is described analytically only. This last scheme, unlike the others, has not led to good results in any test.

1214. W. E. Unger, 238-PuO-2 Neutron Shielding Computation Comparisons, ORNL TM-1632 (August 1966). Availability: Dep. mn, CFSTI, $1.00 cy, $0.50 mn.

Calculations by ANISN, an S_nP_ℓ code, are compared with those by Monte Carlo and RENUPAK, a moments code.

$^{238}PuO_2$ neutron dose rates in various shielding materials calculated by ANISN are presented.

1215. E. A. Straker, Calculations of the Transport of Neutrons from Fission and 14-MeV Point Sources in an Infinite Medium of Air, ORNL TM-1547 (Aug. 9, 1966). Availability: Dep. mn CFSTI, $1.00 cy, $0.50 mn.

Calculations by various investigators of first-collision doses and energy spectra due to neutrons from fission and 14-MeV point sources in an infinite medium of air are compared with each other and with Monte Carlo and discrete-ordinate calculations performed by the author. In the Monte Carlo calculations for the 14-MeV source, the author investigated the sensitivity of the dose and spectral results to the manner in which inelastic scattering was treated in the calculations. It is concluded that the significant differences in the calculated doses and energy spectra for both fission and 14-MeV sources cannot be traced to differences in the calculational techniques alone, but that they may be due to variations in the cross sections used in the calculations. Varying the treatment of inelastic scattering led to wide variations, indicating the importance of obtaining accurate inelastic cross-section data.

1216. J. H. Wilson and R. H. Karcher, Shielding Effectiveness of Soils Against Initial Radiation, HN-186 (August 1966). Availability: Dep. mn. CFSTI, $5.00 cy, $1.00 mn.

The shielding properties of soils subjected to initial radiation are analyzed, and values of shielding effectiveness are determined including the extremes in behavior which are introduced by such factors as non-hydrogenous chemical composition, bound water content, and free water content. All soils for which complete chemical and mechanical analyses are available were considered. The sources of initial radiation included primary neutrons from the weapon and secondary gammas generated in the soil shield. For dry soils it was found that the bound water content is the major factor influencing shielding effectiveness. As the amount of free water in soil is increased, bound water becomes a less important factor. For wet soils it was found that those with high iron content were the least effective shields and those with low iron content, and coincidentally, with high silicon content, were the most effective. It was concluded that, for the purposes of describing shielding effectiveness, soils can be considered in three groups: (1) soils with high bound water and iron content, (2) soils with low bound water and iron content, and (3) soils with intermediate bound water and iron content.

1217. R. L. French and L. Olmedo, Ground Roughness Calculations for Fallout Gamma Rays, RRA-T61 (June 30, 1966). Availability: CFSTI, $3.00 hc, $0.75 mf.

A study was made of four models for calculating the effect of ground roughness on the gamma-ray energy and angular distribution in air above a fallout field. One model, called the "buried-source model" assumes that a uniform thin layer of soil covers the fallout with the thickness of the layer corresponding to the degree of ground roughness. Another, called the "mixed-source model" assumes that the fallout is mixed with a thin layer of soil. A third model, called the "collimated-source model," assumes that the fallout is deposited on a smooth ground surface but that the gamma-ray emission is constrained to certain angular sectors. The last model, called the "furrowed-surface model," assumes that the fallout is distributed over a ground surface which has uniform concentric circular furrows. Calculations of the energy and angular distributions of the uncollided and the singly scattered fluxes three feet above sources with one or more energies were performed using each of the models. Comparisons with measured data and consideration of the convenience of application indicate that the buried- and mixed-source models are preferable to the other two. The results indicate that the magnitude and the angular distribution of the uncollided flux is strongly influenced by ground roughness. The singly scattered flux, which can actually be increased by ground roughness, is much less sensitive to ground roughness than is the uncollided flux.

1218. R. L. French, J. H. Price, and K. W. Tompkins, An Analysis of the Performance of Enclosure Shields in a Real Fallout Field, RRA-T55 (Dec. 31, 1965). Availability: CFSTI, $4.00 hc, $1.00 mf.

State-of-the art shield calculational techniques were applied in the analysis of an experiment in which spherical and cubical multilayer iron and water enclosure shields were exposed in a real fallout field. The calculations were divided into three steps: 1) specification of the time-dependent energy spectrum of the gamma rays emitted by the fallout, 2) determination of the energy and angular distribution of the resultant gamma rays incident upon the shields, and 3) calculation of the penetration of the gamma rays into the shields. A variety of theoretical and measured data were used for the fallout emission spectra. Monte Carlo air- and ground-scattering data along with a limited amount of measured data were employed in establishing the radiation distribution incident upon the shields. Monte Carlo procedures were used to compute the penetration into the shields. The results were expressed as dose transmission factors (inside dose divided by outside dose) and compared with corresponding factors from the experiment. Both sets of results indicated that the shields were least effective during the first few hours and that the dose transmission factors decreased by as much as 50 percent by the end of the first day. However, the calculated results were found to be consistently higher than the measured results by as much as a factor of two depending upon the particular source term. After investigating several possible causes, it was concluded that the discrepancy was probably the cumulative effect of a deficiency of low energy photons in the calculated source terms, of support structure inside the enclosure shields, and of neglecting the effects of ground roughness in the calculations.

1219. W. D. Brown, X-Ray Attenuation and Absorption Coefficients, D2-125065-1 (July 8, 1966) (RESTRICTED DATA). Availability: The Boeing Company, Seattle, Washington.

In this report processes taking place when X radiation passes through matter are reviewed. Theoretical methods for calculating X-ray attenuation coefficients in the energy range from 1 keV to 100 MeV and the theoretical work performed up to the present time are described and evaluated. Several semiempirical methods for determining X-ray attenuation and absorption coefficients are described and compared with theoretical and experimental results. Graphs and tables of the attenuation and absorption coefficients for X-ray energies ranging from 1 keV to 10 MeV and for atomic numbers from 1 to 100 are contained in the Appendix. The data and method used in constructing these tables and graphs are discussed.

1220. D. M. Wheeler and R. J. Prevost, Transmission and Reflection of Nuclear Radiation for Various Incident Angles on Shield Materials, AFWL-TR-65-182 (February 1966) (RESTRICTED DATA). Availability: Air Force Weapons Laboratory, Kirtland AFB, New Mexico. Attn: RCS.1.

The adequacy of calculational methods used in predicting radiation reflection and transmission for shielding materials was investigated with emphasis on reflection of slant-incident radiation and of secondary gamma rays produced in the materials.

Experimental data, obtained by exposing slabs of shielding materials to narrow-beam radiation from the Aerospace Systems Test Reactor, were used to evaluate the Slab Penetration and Reflection Calculation (SPARC) procedure. These data include: (1) reflected and transmitted fast-neutron and gamma-ray dose rates for several thicknesses of steel and 6% borated polyethylene; (2) reflected fast-neutron and secondary-gamma-ray dose rates from a concrete slab and a sample of Nevada Test Site soil; and (3) secondary-gamma-ray reflection from several thicknesses of steel and polyethylene.

In general, calculational procedures produce results of adequate accuracy. With normal-incident radiation, calculated values of fast-neutron reflection fractions are less than measured values, ranging from 12% in the case of concrete to 30% in the case of polyethylene. With slant-incident radiation, the fast-neutron reflection fractions are a factor of 1.6 greater in the case of multilayer slab and about a factor of 2 greater for polyethylene.

For fast-neutron transmission fractions, the calculated values ranged from 22% less than experimental values for the multilayer slab to 25% greater for the steel slabs.

Recommendations are made that a 14-MeV neutron source be used to generate transmission and reflection data to confirm further the SPARC procedure.

1221. I. Bergqvist, B. Lundberg, L. Nilsson, and N. Starfelt, "Gamma-Ray Spectra from Inelastic Neutron Scattering," Nucl. Phys. 80, 198-208 (1966).

The gamma-ray spectra emitted after inelastic scattering of 5.8 and 7.5 MeV neutrons in Fe, Ni, Cu, Zr, Ag, I, Ta, W, Au, Hg, Pb, and Bi have been studied. The measured spectra were compared with gamma-ray spectra emitted in the capture of 0.1-MeV neutrons. This comparison strongly indicates that a large fraction of the high-energy gamma rays observed in neutron capture is due to a direct or semi-direct capture mechanism.

1222. M. Alberg, K. O'Brien, and J. McLaughlin, "An Experimental Determination of the Energy Distribution of Gamma Rays Scattered in Sand," Nucl. Sci. Eng. 25, 303-305 (1966).

The differential energy spectra of gamma rays from a point isotropic ^{137}Cs source placed in an effectively infinite medium of sand have been determined by scintillation spectrometer measurements that were interpreted with an analytic method based on the Scofield iteration scheme. The results are consistent with interpolated values of Goldstein and Wilkins's moments method calculations for aluminum. This study of gamma-ray transport in sand will form the basis for laboratory studies of the effect of material heterogeneities on the energy and angular distributions of gamma rays scattered in other media.

1223. H. Wagenfeld, "Normal and Anomalous Photoelectric Absorption of X Rays in Cyrstals," Phys. Rev. 144, 1, 216-244 (April 1966).

The photoelectric absorption of x rays in atoms (K-, L-, and M-shell) is discussed for the nonrelativistic region, taking retardation effects into account. This is done by expanding the retardation factors in the transition probabilities in a Taylor series and this corresponds to an expansion in terms of electric multipoles. This expansion converges very rapidly and even the octupole transitions may be neglected. In the case of anomalous absorption of x rays in perfect single crystals (Borrmann effect), the absorption cross sections become angular-dependent. This angular dependence allows in principle a direct experimental determination of the quadrupole term of the atomic photoelectric absorption cross section. Measurements by several authors indicate the existence of the electric quadrupole transitions.

1224. D. Taylor and M. Kansara, "Measuring Density with the Nuclear Back-Scatter Method," Nucleonics 24, 6, 54-56 (June 1966).

Since its introduction more than ten years ago strong claims have been made for the efficiency and sensitivity of the back-scatter gaging method, especially for measuring the density of soils. In brief, the back-scatter gage (Fig. 1) is both far more convenient to use and much faster than the direct-radiation gage (Fig. 2). It is clear, however, that back-scatter instruments are affected by soil type and by moisture content. Here we propose to review the sort of performance that is possible, and how the effects of soil type and moisture content can be minimized.

1125. A. Shimizu and H. Mizuta, "Application of Invariant Imbedding to the Reflection and Transmission Problem of Gamma Rays," J. Nucl. Sci. Technol. 3, 2, 57-65 (February 1966).

Equations were derived for the reflection and transmission functions of gamma rays through slabs of finite thickness, making use of the invariant imbedding principle. Numerical solutions were obtained for the reflection function (albedo) of semi-infinite medium. The results agreed well with the solutions by Monte Carlo method. Comparisons were also made with experimental results. Fair agreement was obtained except for the angular distribution of the reflected photons from light elements.

1226. V. V. Verbinski, "Angular Distributions of Low-Energy Neutrons Leaking from Various Scattering Materials," Nucl. Sci. Eng. 27, 1, 67-79 (January 1967).

Experiments in which a wide range of scattering materials in the form of slabs were bombarded by reactor neutrons showed that the angular distribution of low-energy (< 5-eV) neutrons leaking from the opposite side of a slab is independent of the source term and of the slab thickness for thicknesses greater than some minimum thickness z_{min}. In the case of pure lead, pure water, and mildly poisoned water, the resulting distributions are in agreement with the Fermi expression $\Phi(\mu) = 1 + \sqrt{3}\,\mu$. The results for pure lead are also in excellent agreement with one-velocity calculations. An imperfect experiment with poisoned lead is in qualitative agreement with one-velocity calculations. The angular distribution for LiH is described by $\Phi(\mu) = 1 + A\mu$, where A is less than $\sqrt{3}$ for subcadmium neutrons and greater than $\sqrt{3}$ at 1.5 and 5 eV. For energies above 5 eV, a Monte Carlo calculation on LiH showed that A continues to rise to a peak value of about 2.5 at 30 eV, after which it decreases to a value of $\sqrt{3}$ above 10^3 eV, where the absorption cross section of lithium becomes negligible. The applicability of two neutron transport codes that numerically integrate the Boltzmann transport equation was tested in additional calculations for LiH and water. Although the two codes have been used successfully in other types of shielding calculations, they yielded angular distributions for the same material that disagreed with each other, as well as with some experimental data. This suggests that the development of neutron transport codes should include angular distribution tests.

1227. V. V. Verbinski, "Slowing Down Spectra of Neutrons in Lithium Hydride," Nucl. Sci. Eng. 27, 1, 51-66 (January 1967).

Measurements of the spectra of neutrons moderated in LiH were made in the energy range of about 0.01 to 600 MeV, and the results were compared with calculated spectra obtained from a Monte Carlo calculation, a direct numerical integration of the Boltzmann equation (NIOBE code), a moments numerical calculation, and three infinite-medium thermalization calculations, each utilizing a different scattering kernel. The measurements were carried out by irradiating slabs of LiH with neutrons having a near-fission spectrum. The spectra of the leakage flux, of the forward-directed flux, and of the scalar flux within the slab were obtained at neutron penetrations of 2.5 to 10 cm. Below 30 eV, the leakage flux and scalar flux attained an asymptotic spectral shape at a penetration of 2.5 cm, and the forward-directed flux at about 5 cm. The shapes of the calculated spectra agree with the shapes of the measured spectra for all energy regions in which each calculation is valid. A large discrepancy between the NIOBE code predictions and the measurements below 0.08 eV is caused by upscattering and molecular binding effects, which are neglected by NIOBE. These effects were included in a neutron thermalization calculation for an infinite medium with a constant source density; however, good agreement with measurement was obtained only for the case in which the measurement had been made in a nearly gradient-free region. In a region of strong flux gradients, the spectrum of the forward-directed flux is shown to be related to that of the scalar flux with good accuracy by the Purohit expression, according to a NIOBE code calculation which yielded both spectra.

1228. J. Greenborg, "Neutron Flux and Gamma-Ray Dose Distributions in the NPR Shield," Nucl. Appl. 2, 5, 430-439 (October 1966).

Extensive neutron and gamma-ray measurements were performed in the reflector and primary shield of the NPR, a large power and plutonium production reactor. The measurements yielded fast, epithermal, and thermal-neutron flux and gamma-ray dose distributions through approximately 9 ft of reflector and shield assembly. Emphasis was placed on obtaining absolute flux and dose measurements with respect to reactor power.

The measured fluxes and dose rates were compared to those calculated by the removal diffusion theory computer program MAC; a calculation in 18 removal groups and 31 diffusion groups. Agreement is excellent for fast-neutron flux and gamma-ray dose rate in the concrete shield and for thermal and epithermal flux in the graphite reflector. Calculations of thermal and epithermal fluxes in the concrete shield are in lesser agreement with measured values; generally within a factor of 2.

1229. J. S. Hurst, Machinability of Some Infrequently Used Materials, Y-1550 (September 21, 1966). Availability: $2.00 CFSTI.

This report contains a compilation of methods which have been used for the successful machining of twenty-four material types which are infrequently encountered on a production basis. Methods reflect the craftsman's empirical determinations rather than on exhaustive study of the subject, since the variability of engineering materials precludes an absolute machining procedure unless large quantities of data over long runs are available. However, the data presented can serve either as enabling information for the one-of-a-kind job, or as the starting point for additional research.

1230. J. W. Smith, "Distribution of Neutron Dose with Depth," Kerntechnik 7, 6, 266-268 (1965).

Results of measurements of neutron depth dose are compared with flux to dose conversion functions. Best agreement is found with the Snyder-Neufeld function.

1231. M. V. Blinov, N. M. Kazarinov, and A. N. Protopopov, "Angular and Energy Characteristics of the Emission of 235-U Fission Neutrons," J. Nucl. Energy, Pts. A/B, 20, 5, 381-89 (May 1966).

Using ^{235}U fission by thermal neutrons, simultaneous measurements were made of the velocities of a fragment and a neutron moving in the same direction employing the time-of-flight method. As a result, the emission spectrum of the fission neutrons was obtained. The angular distributions and energy spectra of neutrons emitted at different angles to the direction of motion of the fragments were also measured. These distributions were compared with calculations in which the emission spectrum obtained was employed. The comparison showed that the data were in good agreement, with the exception of small discrepancies in the angular distribution.

An analysis of the results shows that the emission spectrum agrees with data calculated according to the statistical theory of evaporation, and also that not less than 90 percent of the ^{235}U fission neutrons are emitted in the process of isotropic neutron evaporation from fully accelerated fragments.

1232. A. Shambon, Spectral Energy Distribution of Unfiltered X-Rays from a 250 kV Constant Potential Machine, HASL-179 (August 1966). Availability: $1.00 CFSTI.

A representation of the energy distribution of four unfiltered beams from a 250 kV constant potential X-ray machine is presented. This representation is consistent with the inverse square law for attenuation of radiation with distance and with the dose rates obtained with a free air ionization chamber.

By using a filtered beam and air chamber measurements at two positions in the beam, the location of the X-ray source was accurately established. It was then possible to apply the inverse square law to partial dose rates in 10 kV intervals found analytically at one position and determine attenuated partial dose rates at two other positions of the unfiltered beam. The sum of these partial dose rates was compared with free air chamber measurements of the total dose rate. The total dose rate of the filtered beam was used to calibrate a Victoreen stray radiation chamber. This calibration was compared with one obtained with NBS calibrated sources. All comparisons showed good agreement.

A monitoring system was set up to provide a means to correct for small variations in the output of the machine.

1233. J. A. Kish, I. T. Myers, and J. J. Smithrick, Experimental Neutron Flux Albedo for Concrete, NASA TM X-1287 (September 1966). Availability: $1.00 (cy) CFSTI, $0.50 (mn).

Thermal neutron flux albedo for concrete due to an incident fast-neutron beam is of interest in predicting activation of a nuclear-rated space chamber. Some measurements of this quantity were made for limestone concrete by using plutonium-beryllium and reactor-beam neutron sources. In both cases, measurements were made by using a moving-detector technique to perform the integration of emergent thermal-neutron flux. The accuracy of results for the reactor source was limited by lack of beam-spectrum data, but a value of flux albedo for the PuBe source was determined to be 0.13 ± 0.04.

1234. J. H. Ray, G. Grochowski, and E. S. Troubetzkoy, Neutron Cross Sections of Nitrogen, Oxygen, Aluminum, Silicon, Iron, Deuterium, and Beryllium, UNC-5139 (Nov. 15, 1965). Availability: $6.00 cy, $1.25 mn CFSTI.

Neutron cross-section sets have been prepared for N, O, Al, Si, Fe, D, and Be for neutron energies from 0.037 eV to 18 MeV. The cross sections tabulated include the total, elastic, inelastic, (n,2n), and cross sections for charged particle emission. Information is also given on the angular distribution of elastically scattered neutrons and on the energy distribution of neutrons and gamma rays following nonelastic reactions.

1235. M. D. Goldberg, S. F. Mughabghab, S. N. Purohit, B. A. Magurno, and V. M. May, Neutron Cross Sections, Vol. 2B, Z = 41 to 60, BNL-325, Sec. Ed., Sup. 2, Vol. 2B (May 1966). Availability: $4.00 CFSTI.

This supplement to BNL-325 presents new information on neutron cross sections accumulated in the four years since the previous information. This issue covers only the elements whose atomic numbers are in the range 41 to 60.

The format differs from preceding versions in several ways. The more normal size, arrangement of all data by nuclide, the fluid format is likely to make this supplement more convenient for the user.

1236. K. Low and K. Edvarson, "Gamma Spectra from Mixed Fission Products of Ages Between 2 Hours and 7 Days," Ark. Fys. 29, 4, 359-365 (1965).

The gamma spectra of mixed fission products of ages between 2 hours and 7 days have been measured for thermal neutron fission of U-235 and Pu-239 and fast neutron fission of U-235. The results are summarized in diagrams, where the ratios between the intensities in a number of energy bands in the measured spectra are given as a function of age.

For comparison the photon spectra for MFP from U-235, thermal fission, have been calculated for the same time period.

1237. H. Conde and G. During, "Fission Neutron Spectra. Part II. Fission Neutron Spectra of U-235, Pu-239, and Cf-252," Ark. Physik 29, 4, 313-319 (1965).

Measurement of the fission neutron spectra from the neutron induced fission of U-235 and Pu-239 and the spontaneous fission of Cf-252 have been made. The spectra were measured with a time-of-flight technique using a plastic scintillator as the fission neutron detector. The low-energy part of the Cf-252 fission neutron spectrum (below 0.3 MeV) was measured with a Li^6-glass detector replacing the plastic scintillator. The fissile materials were mounted in xenon-gas scintillation chambers. The primary neutrons were produced by a 2.5 MeV Van de Graaff accelerator with top terminal pulsing. The resolving time of the time-of-flight equipment was better than 3 nanoseconds. Results fitted to Maxwellian distribution $\sqrt{E}\ e^{-E/T}$, are presented for neutron induced fissions of U-235 at 40 keV and 1.5 MeV and for Pu-239 at 40 keV of incident neutron energy. Results are also given for the fission neutron energy spectrum of Cf-252.

1238. R. L. Summers and Z. G. Burson, Experimental Evaluation of Techniques for Improving Fallout Protection in Home Basements, CEX-65.5 (December 1966). Availability: $1.50, CFSTI.

Radiation measurements were made in the basement of an existing two-story wood frame house in Area 1 at the USAEC Nevada Test Site to evaluate and verify some simple and convenient techniques for improving fallout radiation protection in homes with basements.

Fallout radiation was simulated by hydraulically pumping a sealed radioactive source of ^{60}Co at a uniform speed through tubing placed over the area of interest. Measurements were made by integrating ionization chambers.

The project was divided into six phases consisting of the following:

Phase 1 Bare basement with and without shielded basement windows.

Phase 2 Overhead barrier shield provided by supporting bricks placed between the ceiling joists. Several variations to this technique were tested.

Phase 3 Small shelters consisting of a concrete block sit-down shelter, a lean-to shelter and a workbench.

Phase 4 A tilt-up storage unit shelter and workbench variations.

Phase 5 An alternate variation to the overhead barrier shield technique in Phase 2.

Phase 6 Supplemental investigations: (1) three incremental layers of concrete blocks placed on the floor of the first floor, (2) a simulated brick veneer, and (3) typical first-floor furniture distributed realistically and then stacked in one corner.

The protection factors ranged from 11 to 25 in the bare basement. Shielding the basement windows had very little effect on the protection.

The overhead barrier technique provided the desired protection in the two-story house, but not for the simulated one-story house. The vertical position of the overhead mass is relatively more significant than its thickness. Consequently, using this technique would not achieve the desired protection for basements where a significant portion of the basement wall is exposed above ground.

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1238. (Continued)

The small shelters and tilt-up shelter provided very good protection. The lean-to and tilt-up shelters were considered difficult to construct by unskilled labor. The sit-down shelter and workbench were considered easy.

Household furniture stacked in one corner of the first floor provided only a small amount of increased protection in the same corner of the basement.

Engineering calculations of the bare basement were about 30% higher than corrected experimental data for roof contributions, and up to 50% lower for ground contributions.

The protection factors, derived from corrected experimental data, are expected to be accurate to within a standard deviation of $\pm 8\%$.

1239. K. H. Larson, J. W. Neel, H. A. Hawthorne, H. M. Mork, R. H. Rowland, L. Baurmash, R. G. Lindberg, J. H. Olafson, and B. W. Kowalewsky, Distribution, Characteristics, and Biotic Availability of Fallout, Operation Plumbbob, WT-1488 (July 26, 1966). Availability: $3.00, CFSTI.

This report includes the significant findings of CETO Program 37, related to the distribution, characteristics, and biological availability of fallout debres originating from the Plumbbob Test Series (1957) at the Nevada Test Site.

The use of aerial radiometric survey was adapted to routine radiation surveys and greatly increased the detail, accuracy and distance of fallout pattern delineation. Isodose rate and time-of-arrival contour maps are presented for seven tower mounted and four balloon mounted shots along with the predominant particle size fraction on several arcs along each fallout pattern.

Particles less than 44 microns in diameter contained about 30 percent of the fallout radioactivity from tower mounted detonations as compared to about 70 percent from balloon mounted detonations within distances at which fallout arrived by H + 12 hours. Balloon mounted detonations produced fallout debris of higher water and acid solubility than did tower mounted detonations.

The percentage of $Sr^{89,90}$ and $Ru^{103,106}$ was higher in fallout particles less than 44 microns in diameter than in larger fallout particles. There was a higher percentage of these radioelements in fallout debris from balloon detonations than from tower detonations. The amounts of water-soluble Ba^{140} and $Sr^{89,90}$ deposited by a balloon mounted and a tower mounted detonation of similar yield and height of burst were estimated to be similar despite relatively large differences in the total amounts of these radioisotopes deposited. Within distances at which fallout occurred at H + 12 hours, balloon mounted detonations deposited a maximum of 0.13 percent of the theoretical total of Sr^{89} produced while tower mounted detonations deposited a maximum of 2 percent. Tower mounted detonations deposited a maximum of 7.2 percent of the theoretical total amount of Sr^{90} produced.

Decay of beta radiation approximated the $T^{-1.2}$ decay expression from H + 12 to H + 6000 hours; decay of gamma radiation deviated to the extent that doses calculated by the observed decay values were 1.8 to 2 times greater than those calculated by the $T^{-1.2}$ relationship. There were also significant differences in the gamma energy spectrum with time after detonation.

In the environment, fallout radioactivity was apparently confined to the upper 2 inches of soil unless the soil was mechanically distributed.

1240. REIC, Radiation-Effects State of the Art - 1965-1966, REIC Report No. 42 (June 30, 1966). Availability: Radiation Effects Information Center, Battelle Memorial Institute, Columbus, Ohio 43201

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.

1241. C. L. Hanks and D. J. Hamman, The Effect of Nuclear Radiation on Capacitors, REIC Report No. 44 (December 30, 1966). Availability: Radiation Effects Information Center, Battelle Memorial Institute, Columbus, Ohio 43201.

This report is a compilation of the information available in the Radiation Effects Information Center on the effects of steady-state and transient nuclear radiation on capacitors. The capacitors are classified according to their dielectric materials into six classifications. These include glass- and porcelain-dielectric, mica-dielectric, ceramic-dielectric, paper- and paper/plastic-dielectric, plastic-dielectric, and electrolytic capacitors. The information presented is for use in making "ballpark" or broad estimates of the effects or damage that might be expected, and should be applied in the early design stages for this purpose. Detailed information concerning a specific environment should be requested from the REIC for the later or final design effort. Details concerning results of various test programs and experiments are summarized in tabular form for the various capacitor classifications. Details of the more extensive studies where larger sample sizes were used are also discussed in the text.

1242. W. E. Chapin, J. E. Drenna, and D. J. Hamman, The Effect of Nuclear Radiation on Transducers, REIC Report No. 43 (TIC Report No. 3) (October 31, 1966). Availability: Radiation Effects Information Center, Battelle Memorial Institute, Columbus, Ohio 43201.

This report presents information on the effects of nuclear radiation on transducers. Emphasis has been placed on transducer types frequently employed in measurement of pressure, temperature, and acceleration forces. The report provides a summarization of many laboratory experiments which have generated information pertinent to determining radiation damage thresholds for each type of transducer in accordance with the operating principles used in their design and manufacture. The technical characteristics of each type of transducer are discussed to provide some insight to the problems that exist in radiation hardening. This report deals with information from reports available at the Radiation Effects Information Center (REIC) and the Transducer Information Center (TIC). The information presented can serve a useful purpose to designers, manufacturers, and users as a guide for developing a better understanding of the problems that must be considered prior to final selection of transducers to be used in nuclear environments.

1243. R. B. Walton and R. E. Sund, Delayed Gamma Rays from Fission, AFWL-TR-66-122 (October 1966). Availability: Air Force Weapons Laboratory, Kirtland Air Force Base, New Mexico

Measurements of the energy spectrum of delayed 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 2 μsec and 80 μsec. The data showed three new prominent gamma rays with energies of 205, 390, and 1330 keV; two unresolved gamma rays having energies of approximately 110 keV; and the six gamma rays with energies of 260, 460, 720, 850, 910, and 1250 keV which were observed in previous studies for $t > 50$ μsec after fission. Absolute intensities of the new gamma rays were obtained using the normalization factor derived from the relative intensities of the last six gamma rays and their previously determined absolute intensities. The 205-, 390-, and 1330-keV gamma rays probably result from the cascade decay of a single isomer which has a half life of 3.4 ± 0.4 μsec and fission yields of 1.30 ± 0.32 and 0.63 ± 0.21 percent per fission of Pu-239 and U-235, respectively. Half lives and yields were also obtained for the two 110-keV gamma rays, but the results are tentative because of uncertainties in the data. The fission yields, cascade gamma rays, and half lives of the isomers responsible for early delayed gamma rays from U-235 (n,F) and Pu-239 (n,F) have been summarized and total intensities and energy emission rates of gamma rays with energies greater than 140 keV have been presented as a function of time after fission over the range from 2 μsec to 25 msec.

1244. Y-S Su, J-C Chou, S-C Lin, "Differential Energy and Angular Distribution of Scattered Gamma Rays Through Concrete Incident Normally," Ho Tsu K'o Hsueh 4, 3-4, 39-54 (August 1965).

The angular and energy distributions of the gamma radiation emerging from the face of the concrete barrier incident normally with parallel beams from Co-60 were first measured by the use of a 3 in. x 3 in. NaI(Tl) scintillation detector with 256 channels pulse-height analyzer. The obtained pulse-height distributions were treated by the modified inverse matrix method with the help of IBM-1620 digital computer. The experimental results were compared with theoretical predictions of Monte Carlo calculation and a fairly good agreement was revealed.

1245. F. H. Clark, N. A. Betz, and J. Brown, Monte Carlo Calculations of the Penetration of Normally Incident Neutron Beams Through Concrete, ORNL-3926 (January 1967). Availability: $1.00, CFSTI.

The 05R Monte Carlo code has been adapted to calculate fast-neutron doses within and outside of concrete slab shields. Such calculations were performed for a variety of shield thicknesses (0 to 180 cm) and for various monoenergetic sources from 0.7 to 14 MeV. The concrete chosen is ordinary concrete with approximately 6% water content. Results are plotted as tissue kerma versus penetration distance for each source energy. The source is in each case a broad normally incident beam.

1246. S. Z. Mikhail., Ed., Fallout Phenomena Symposium, Proceedings/Part I, USNRDL-RL-177 (June 9, 1966). Availability: U. S. Naval Radiological Defense Laboratory, San Francisco, California.

These proceedings incorporate a number of papers concerned with fallout problems. The general topics treated are as follows: particle formation, cloud development-particle calculation, transport and deposition, and radiation fields. (RSIC abstract.)

1247. J. W. Leake, "A Spherical Dose Equivalent Neutron Detector," Nucl. Instr. Method 45, 151-156 (AERE-R-4884) (November 1966).

A spherical neutron detector has been designed which is a development of the Andersson and Braun counter, but which weighs only 5 kg. The detector comprises a double polythene moderator system, with a LiI(Eu) crystal at the center, which is viewed by a photomultiplier. The neutron energy response of the detector matches the recommendation of the ICRP (1963) within ±50% for neutrons with energies between 200 keV and 10 MeV, the sensitivity falling above 10 MeV. Below 200 keV the sensitivity increases above the required value to a maximum at 5 keV where it overestimates by a factor of $3\frac{1}{4}$ times. At thermal energies the response is high by a factor of $3\frac{1}{4}$ times.

1248. C. M. Huddleston and W. C. Ingold, A Revised Formula for the Calculation of Gamma-Ray Shielding Properties of Shelter Entranceways, USNCEL N-843 (October 1966). Availability: U. S. Naval Civil Engineering Laboratory, Port Hueneme, California.

An improved formula has been devised for the calculation of gamma-ray dose attenuation in two-legged air ducts through concrete. Comparisons are given between the predictions of the simple empirical formula and the results of measurements as well as predictions obtained by a more complicated computational technique. The accuracy of the empirical formula is discussed. It is shown that the formula is highly accurate, having essentially zero bias and a standard deviation of less than 14 percent of the correct value.

1249. S. G. Tsipin, B. I. Sinitsin, and V. K. Duruga, "Passage of Reactor Neutron Radiation Through Nonhydrogeneous Meaid," J. Nucl. Energy, Pts. A/B, 20, 851-861 (1966).

An examination is made of the results of experimental and theoretical studies of the passage of neutrons through nonhydrogenous homogeneous materials and mixtures. The relaxation lengths obtained for neutrons of different energy groups are discussed. It is shown that, with certain assumptions, the magnitude of the attenuation of a flux of neutrons of the upper energy group ($E > 3$ MeV) in a nonhydrogenous medium can be obtained from fairly simple empirical expressions using well-known constants; the removal cross section σ_{rem}, the cross section obtained from the inverse relaxation length (σ/λ) and the asymptotic cross section. The values of these constants are given.

1250. A. Bergstom, S. Schwarz, L. G. Stromberg, and L. Wallin, Remarks on the 6-Li (n,alpha)3-H Cross Section in the Region 1 Less than e sub n Less than 600 keV, CCDN-NW 3 (October 1966). Availability: Neutron Data Compilation Centre, European Nuclear Energy Agency, France.

A comparison of existing data on the 6-Li(n,α)3-H reaction cross section is given. The renormalized data of Bame and Cubitt are included, as well as a revision of the data in the energy range $150 < E_n < 450$ keV presented earlier by Schwarz et al. The remaining discrepancies turn out to be the question of the width of the resonance peak and how far the 1/v extrapolation is valid in the keV region.

1251. M. Polan, An Analysis of the Fallout Prediction Models Presented at the USNRDL-DASA Fallout Symposium of Sept. 1962, Vol. I: Analysis, Comparison, and Classification of Models, USNRDL-TRC-68 (September 8, 1966). Availability: U. S. Naval Radiological Defense Laboratory, San Francisco, California 94135.

The nuclear fallout prediction models discussed at the USNRDL-DASA Fallout Symposium held at San Francisco in September 1962 have been analyzed in terms of the following characteristics: (1) adjustments for burst conditions; (2) the geometry and dimensions of the nuclear cloud; (3) radioactivity distribution within the cloud; (4) radioactivity distribution in relation to fallout-particle size; (5) conversion of burst yield to exposure rate; (6) division of the cloud into small elements (wafers), each containing particles of a single size; (7) particle settling rates; (8) wind treatments; (9) transport and deposition of cloud wafers; (10) accumulation and decay of fallout; and (11) the procedures for obtaining fallout contours or other outputs from the models.

It was found that fallout prediction models are used in four primary areas: (A) in operation analysis; (B) in scientific research; (C) to obtain manual predictions in the field, and (D) in special-purpose electronic computers. Two classes of models were established: one in which a single effective fallout wind is used, and the cloud is not divided into wafers; and one in which multilayer wind fields are used, and the cloud is divided into wafers.

Evaluations were made and conclusions were reached that could lead to increased consistency in fallout predictions.

1252. G. Olive, J. F. Cameron, and C. G. Clayton, A Review of High Intensity Neutron Sources and Their Application in Industry, AERE-R-3920 (June 1962). Availability: HMSO, 6 shillings, net.

This report reviews various high intensity neutron sources which are now available and discusses their present and future application in industry.

The characteristics and cost of radioisotope neutron sources are compared with accelerating machines and it is shown that for large scale applications of activation analysis and short lived isotope production where a high neutron flux is required, the neutron generator is preferred.

There are four useful reactions by which the neutrons are available from accelerating machines; these are the X-Be, d-Be, d-d, and d-t. The characteristics of each type of machine is discussed and although the characteristics and life of the tritiated titanium target is known to be restricted, it is concluded that for industrial application the (d,t) reaction is the most useful and is likely to remain so in the future. Low cost and maintenance are the chief advantages.

The most important industrial application is considered to be activation analysis by fast and thermal reactions with the possibility of extension to process control. The design of an irradiation facility suitable for use in an industrial analytical laboratory is discussed.

1253. T. C. Rozzell, J. L. Setser, and G. S. Logsdon, "Economical Massive Shield for Gamma Spectroscopy," Health Phys. 12, 5, 699-703 (May 1966).

A large shield for gamma spectroscopy has been designed and assembled. The cost was approximately one-fifth that of a commercial shield. This cubically shaped shield has 2-in. lead walls, with inside dimensions of 22 in.per side. It is lined with 1/10-in. copper to reduce the low energy X-rays resulting from photoelectric interactions within the lead. A steel frame and steel rods support the 117 various size lead bricks. It was necessary that costs and weight be kept as low as possible. The total cost was $757,000 and the weight is 3100 lb (including 48- by 36- by 1/2-in. steel plate on which the shield rests). The shield previously used was a popular 3-in. (1000 lb) lead module type. The Compton backscatter and low energy lead X-rays were significantly reduced by the new shield, and the resolution in the lower energy region was also improved. The large shield also permits the analyses of larger samples.

1254. D. R. Johnson, "Neutron Dose Conversion Factors for AmBe and AmB Sources," Health Phys. 12, 6, 856-57 (June 1966).

AmBe and AmB neutron sources are calibrated in terms of first collision tissue dose per neutrons. (RSIC abstract.)

1255. R. M. Fry, "Neutron Dose Conversion Factors for Radioactive Neutron Sources," Health Phys. 12, 6, 855-56 (June 1966).

Calculations are made for conversion of neutron fluence to kerma in soft tissue for PuBe sources.

1256. H. Hollister and J. E. Turner, "The Possible Role of Momentum in Radiation Dosimetry. III. Remarks on the 1962 ICRP/ICRU Report," Health Phys. 12, 7, 949-953 (July 1966).

Work in radiation dosimetry during the last several years has been directed in part toward formulations that bring to bear new experimental techniques and new theoretical ideas. Examples include the work of (1) Rossi and co-workers, directed toward finding a detailed description of local energy absorption in tissue, (2) Hurst and Ritchie, directed toward stating formally the problem of dosimetry in the most general way, and (3) Turner and Hollister, directed toward finding a physical equivalence of different radiations that are made biologically equivalent by producing a given end point experimentally. These investigations have been successful in clarifying some questions. In particular, the last-named work has shown that linear momentum transfer together with energy transfer describe RBE and the dependence of RBE on LET. The explicit relationship found, based originally on neutron RBE data, also bears a marked resemblance to an empirical relationship given in the 1962 report of the ICRP/ICRU RBE Committee. A discussion of these results is presented. It is found that they have a unified description in terms of incident particle velocity. The ratio of linear momentum absorbed per unit mass (P_{abs}) to energy absorbed per unit mass (E_{abs}) has, of course, the physical dimensions of reciprocal velocity.

1257. W. R. Hendee and J. L. Ellis, "Scattering of Gamma Radiation from Semi-Infinite Slabs," Health Phys. 12, 673-81 (1966).

Experimental measurements have been made of the exposure due to ^{60}Co and ^{137}Cs gamma rays scattered from thick, semi-infinite slabs of concrete, lead and water, for various separation distances between source and detector and for various heights of source and detector above the scattering slabs. The results indicate that the ratio of source and detector height above the scattering slab to the distance of separation of source and detector is the parameter most influential in determining the scattered exposure. No influence on the scattered exposure was observed by positioning the source at a height greater or lesser that that of the detector. Isotropic reflection curves closely approximating the experimental points were obtained from albedo calculations and resulted in energy albedos of 0.050, 0.0113, and 0.049 for ^{60}Co radiation on concrete, lead and water, respectively, and of 0.100, 0.013, and 0.083 for ^{137}Cs radiation on concrete, lead and water, respectively.

1258. B. T. Kenna and F. J. Conrad, "Fast Neutron Flux Pattern for a 14-MeV Neutron Generator," Health Phys. 12, 4, 564-566 (April 1966).

When the flux patterns available from a fast neutron generator are determined, two observations must be kept in mind. First, the determination of flux patterns is not intended to replace such techniques as an internal monitor for obtaining the actual flux which a sample experiences. Second, the flux patterns can change somewhat with time or with different targets. However, characteristics such as isotropic flux and an approximate $1/R^2$ relationship generally will be valid.

1259. C. W. Nestor, Jr., and H. C. Claiborne, Heating in Infinite Cylinders by Uniform Gamma Sources, ORNL TM-1260 (August 1966). Availability: Dep. mn CFSTI $2.00 cy, $0.50 mn.

As an aid to the reactor designer concerned with gamma heating in elements that may be approximated by long cylinders, we have integrated the point kernel describing the transfer of gamma energy for the case of sources distributed uniformly throughout a right circular cylinder of infinite length. Since the coefficients were available for a third-degree polynomial representation of the energy absorption buildup factor for several materials, we chose this representation and were able to reduce the solution to a single numerical integration for each term in the buildup factor expansion. In addition, we obtained an analytical expression for the contribution from the linear term in the expansion.

In this memorandum we describe the reduction of the solution to a single integration, the derivation of the analytical expression for the linear term contribution and the numerical integration procedure. The results of the calculations are given in tabular and graphical form, and listings of the FØRTRAN-63 programs are included for references.

1260. L. G. Haggmark, Monte Carlo Techniques Applied to the Calculation of Gamma-Ray Transport in Compartmented Structures, USNRDL-TR-1038 (AD-637685) (May 3, 1966). Availability: CFSTI, $2.00 (cy), $0.50 (mn).

A Monte Carlo computer code was assembled to calculate the transport of gamma rays in three-dimensional , rectangular geometry. The code uses the statistical estimation technique to compute the flux and dose at several space points. The code was applied to two compartmented iron structures that had been experimentally investigated. The Monte Carlo and experimental results compare well, i.e., within 20 percent, in most cases. In a few cases for one structure, where the radiation was incident at large angles from the normal to one wall, the comparison between the results was less than satisfactory. The experimental value was a factor of 6 greater than the Monte Carlo value in the worst case.

1261. R. M. Beckers, R. F. Denkins, H. M. Glen, T. J. Golson, J. E. Kahn, G. W. Renfro, and J. A. Steed, Engineering Design Practices at ORNL for Facilities Containing Radioactive Materials, ORNL TM-1459 (April 1966). Availability: Dep. mn. CFSTI $3.00 (cy), $0.75 (mn).

This report is a compilation of seven papers that were presented by members of the General Engineering and Construction Division Design Department of the Oak Ridge National Laboratory as part of the Applied Health Physics course for state health department personnel that was conducted by the Oak Ridge Institute of Nuclear Studies in September and October of 1965. The intent of the papers is to call attention to those problems peculiar to design for laboratories and other facilities that handle radioactive materials. The material presented here is of a very general nature but is based upon a wide range of experience gained over a number of years by personnel of the General Engineering and Construction Division. It is hoped that the information presented herein will be helpful in guiding competent engineers in designing or specifying the special tools and equipment required for handling radioactive materials. It should also prove useful in establishing criteria and evaluating designs for radioisotope facilities.

1262. F. H. Clark, Determinations of Shield Requirements for Neutron Sources, ORNL TM-1655 (October 5, 1966). Availability: CFSTI $2.00 (cy), $0.50 (mn).

A number of computation devices and charts have been assembled to assist in the rapid evaluation of common materials as neutron shields. It is intended to be included as a section in an NCRP handbook shortly to be issued as a replacement to NBS-63, Protection Against Neutron Radiation up to 30 Million Electron Volts.

1263. P. Stoches and V. Kadlets, Measuring the Attenuation of Neutron Fluxes in a Model of the Upper Biological Shield of the Reactor, ORNL-TR-1232 (Translated from UJV 1454/65) (1965). Availability: Dep. (mn) CFSTI $2.00 (cy), $0.50 (mn), JCL $3.60 (fs), $1.22 (mf).

In view of its structural complexity, the proposed design for the upper biological shield of the KS-50 reactor of the Chechoslovak atomic power plant was tested by subjecting its mockup to the neutron flux in the thermal column of the VVR-S experimental reactor at Rez. The simplified mockup consisted of a cast-iron ring and concrete. Activation measurements revealed that the horizontal flux distribution was less uniform than the vertical distribution; similar measurements indicated that a boron carbide layer placed at 10 cm from the mockup attenuated the flux by a factor of 40; it was used in all tests involving thermal and epithermal neutrons. The system was air-cooled in order to keep the temperature of the mockup at 30°C. The geometric attenuation, measured by means of phosphorus was found to lie between 3.5 and 4. A scintillation probe was also used in the measurements. Theoretical calculations of the relaxation length by means of the RASH B and RASH D codes agreed with the experimental findings. It was concluded that the thermal column is well suited for this type of measurements. For fast neutron measurements the use of activation detectors is recommended, even though they are less sensitive. Tests are in progress in the improved detection of fast neutrons by means of rhodlum and indium threshold detectors and by separating phosphorus from irradiated sulfur by combustion.

1264. A. B. Chilton, "A Review of Aperture and Ceiling-Shine Effects in Fallout Shielding Technology," Health Phys. 12, 9, 1311-1316 (September 1966).

This paper reviews the existing methods of predicting aperture and ceiling-shine effects in fallout shielding technology, indicating the extent to which the techniques used are well supported by research. The present methods of handling apertures appear quite plausible, but no confirmatory work is yet available. The method of accounting for ceiling-shine presently used is somewhat artificial and under some circumstances a more precise method is needed. Such improved methods have been suggested, but there is still room for further careful study of this problem.

1265. R. A. Atneosen, T. D. Thomas, W. M. Gibson, and M. L. Perlman, "X-Rays and Electrons Emitted in Coincidence with Fission of CF-52," Phys. Rev. 147, 3, 1206-1220 (August 1966).

We have measured the emission times of K x rays and the yield and spectra of conversion electrons produced in coincidence with the spontaneous fission of Cf-252. The measurements of x-ray yield as a function of time since fission can be analyzed into components with half-lives of 2.4, 0.4, 0.05, and < 0.01 nsec with relative numbers of nuclei 45, 25, 10, and 20, respectively. The yield of conversion electrons falls almost exponentially from 30 keV, the lowest energy measured, to about 500 keV; this falloff can be accounted for by the energy dependence of internal conversion coefficients. The yield of conversion electrons as a function of mass reaches a maximum at about mass 108, a minimum at mass 130, and a pronounced maximum at about mass 153. There is a sharp decrease in yield beyond this second maximum. The combination of x-ray half-lives and energies observed is consistent with low multipolarity transitions, mostly M1-E2 mixtures with some E2 and perhaps some E1. The gross features of the yield as a function of mass can be understood in terms of the stable nuclear deformations beyond mass 150, of postulated deformations for the neutron-rich nuclides near mass 108, and of the closed-shell properties of nuclides near mass 132. Various explanations for the sharp drop in yield beyond 153 are explored. It is suggested that the amount of spin acquired by a fragment at scission may depend critically on tha partner fragment; spherical fragments are less effective at imparting spin to their partners than are deformed ones. This suggestion provides an explanation for the drop in electron yield at mass 153 and for some of the differences between the yield and spectrum of K x rays for Cf-252 and these quantities for U-236.

1266. R. E. Maerker and F. J. Muckenthaler, "Monte Carlo Calculations, Using the Albedo Concept, of the Fast-Neutron Dose Rates Along the Center Lines of One- and Two-Legged Square Concrete Open Ducts and Comparison with Experiment," Nucl. Sci. Eng. 27, 2, 423-432 (February 1967).

Monte Carlo calculations have been carried out to determine the fast-neutron dose-rate distributions along the center lines of both a straight and a two-legged square concrete duct for a particularly demanding source geometry. The calculations incorporated doubly differential dose-albedo data for concrete which were previously reported. A comparison of the results of the calculations with those from a geometrically similar experiment shows good agreement, and places on a firm foundation the concept of treating the fast-neutron duct transmission problem as a reflection phenomenon at a point that is describable by the differential albedo properties of the walls.

1267. V. V. Verbinski, M. S. Bokhari, J. C. Courtney, and G. E. Whitesides, "Measurements and Calculations of the Spectral and Spatial Details of the Fast-Neutron Flux in Water Shields," Nucl. Sci. Eng. 27, 2, 283-298 (February 1967).

The spectral intensity of the fast-neutron flux penetrating a water medium was measured for two configurations: a large-source, poor-geometry arrangement; and a small-source, almost-good-geometry configuration.

In the large-source experiment, the spectral intensity of the angular flux was obtained at six positions in the water shield of a pool-type reactor and for as many as three angles at each position. In addition to the measurements, the spectral shape and the absolute intensity of angular flux in the shield were calculated. In conjunction with this, the absolute neutron source density was mapped throughout the reactor volume and the distribution along the reactor center line was used as inpot to two neutron-transport calculations that were carried out for a one-dimensional, spherical geometry.

In the small-source experiment, a 2-cm-thick lead target irradiated with short bursts of 33-MeV electrons provided a source of photoneutrons with approximately a fission spectrum at a distance of 40 cm from water slabs of various thicknesses. This distance, together with the large separation of slab and detector and a small-aperture collimator, approximated a good-geometry arrangement for measurements of neutrons leaking normally from the slab. Consequently, these leakage spectra were very sensitive to total neutron cross sections and a distinct peak was observed at 5 to 7.5 MeV. This peak was not at first reproduced by transport calculations that used the measured source spectrum as input; however, when the neutron total cross sections of oxygen were updated with relatively recent high-resolution data, the agreement both in spectral shape and in attenuation (the latter determined from sulfur-activation ratios) was noticeably improved.

1268. W. A. Coleman, R. E. Maerker, F. J. Muckenthaler, and P. N. Stevens, "Calculation of Doubly Differential Current Albedos for Epicadmium Neutrons Incident on Concrete and Comparison of the Reflected Subcadmium Component with Experiment," Nucl. Sci. Eng. 27, 2, 411-422 (February 1967).

Extensive Monte Carlo calculations were performed to determine the distribution in energy and angle of neutrons reflected from steel-reinforced concrete for five incident directions and ten incident energy groups extending from 0.5 eV to 200 keV. The reflected distributions were determined in terms of a doubly differential albedo for each of 54 emergent directions for each energy group lying between and including the incident group and the tenth group (0.5 to 1.8 eV). The standard deviation of the doubly differential albedo averaged around ten percent. The angular slowing down density of the incident epicadmium neutrons within the slab was computed at 0.5 eV and was used as the source distribution for a Monte Carlo single-velocity diffusion calculation using 0.025-eV cross sections. From the diffusion calculation, the differential angular albedos of the reflected subcadmium neutrons and the depth distributions of captures occurring at subcadmium energies were obtained. Measurements of the differential angular albedo of emergent subcadmium neutrons due to a measured spectrum of incident monodirectional beams of epicadmium neutrons were performed at the ORNL Tower Shielding Facility in an experiment geometrically identical to that previously reported for incident subcadmium beams. Of the 35 common points of calculation and measurement, the two largest discrepancies are 23 and 36%; the remaining 33 comparisons produced a root-mean-square deviation of 4.5%.

1269. C. E. Clifford, E. A. Straker, F. J. Muckenthaler, V. V. Verbinski, R. M. Freestone, Jr., K. M. Henry, and W. R. Burrus, "Measurements of the Spectra of Uncollided Fission Neutrons Transmitted Through Thick Samples of Nitrogen, Oxygen, Carbon, and Lead: Investigation of the Minima in Total Cross Sections," Nucl. Sci. Eng. 27, 2, 299-307 (February 1967).

Minima in the total cross sections of nitrogen and oxygen have been investigated by measuring the spectra of uncollided fission neutrons transmitted through thick samples of the elements and by comparing the results with calculated spectra based on total cross sections only. Good agreement between the measurements and calculations was obtained for oxygen when the cross sections evaluated by Slaggie and Reynolds were used and for nitrogen when the cross sections measured by Foster and Glasgow were used. Calculations with other evaluated cross sections for these elements resulted in significant differences and indicate a lack of detailed knowledge of the valleys in the cross-section data. Measurements made for carbon and lead were in good agreement with calculations, as was expected since the cross sections for these elements are well known. The technique used for the spectral measurements employed an NE-213 proton-recoil spectrometer of high sensitivity.

1270. F. H. Clark, "Variance of Certain Flux Estimators Used in Monte Carlo Calculations," Nucl. Sci. Eng. 27, 2, 235-239 (February 1967).

Calculations were set up to determine the effect of the grazing flux on the surface-crossing flux estimator and the pillbox track-length estimator used in Monte Carlo calculations. It was found that grazing particles make only higher order contributions to current or flux but significantly increase the variance. These effects are more pronounced at internal surfaces than at external boundaries. The use of nonstochastic methods to estimate contributions of grazing particles is justified.

1271. A. B. Chilton, "A Modified Formula for Differential Exposure Albedo for Gamma Rays Reflected from Concrete," Nucl. Sci. Eng. 27, 2, 481-482 (February 1967).

A modification is proposed to the Chilton-Huddleston formula for the differential albedo. (RSIC abstract.)

1272. C. E. Clifford and G. C. Wait, "Air-Ground Interfact Effects on the Exposure from Elevated 137-Cs Gamma Sources," Nucl. Sci. Eng. 27, 2, 483-485 (February 1967).

The effect of an elevated source in a contaminated field is unknown. A study was made of the air-ground correction factors for point isotropic gamma sources at heights likely to be of interest for foliar retention problems. It is appreciated that other factors, like foliage density, would also be significant. (RSIC abstract.)

1273. R. H. Karcher, "Neutron Air-Transport Data for the Design of Protective Structures," Nucl. Sci. Eng. 27, 2, 367-387 (February 1967).

The transport of neutrons from a point source of simulated weapons radiation in infinite air is calculated. Weapons neutron spectra are simulated using a mixed source composed of a chopped fission spectrum with most of the neutrons below 0.4 MeV deleted, and an equivalent number distributed uniformly in the 12- to 16-MeV range. The results obtained are generally conservative, from a shielding standpoint, for most nuclear devices. The method of track length stretching is used to improve the efficiency of the Monte Carlo analysis for deep penetration calculations. Well-converged fast-neutron flux and dose data are obtained for penetration distances of about 400 g/cm^2 (approximately 2 miles in sea-level air at 68°F). Energy spectra and angular distributions are calculated also; however, the convergence is less satisfactory in this case. It is found that the inelastic and capture gamma sources resulting from neutron interaction in air are of extremely low intensity and are probably negligible for most shielding applications. Integral and differential neutron air-transport data are tabulated as a function of penetration distance to facilitate their use in shielding calculations.

1274. E. E. Morris and L. V. Spencer, "The Small-Angle Limit for Twice-Scattered Gamma Rays from a Point-Isotropic Sources," Nucl. Sci. Eng. 27, 2, 485-488 (February 1967).

An analytic-treatment is provided for double scattering of gamma rays through small angles. Comments are provided to guide in the application of the method to neutrons. (RSIC)

1275. R. R. Coveyou, V. R. Cain, and K. J. Yost, "Adjoint and Importance in Monte Carlo Application," Nucl. Sci. Eng. 27, 2, 219-234 (February 1967).

The use of the Monte Carlo method for the study of deep penetration of radiation into and through shields entails the use of sophisticated methods of variance reduction to make such calculations economical or even feasible. This paper presents an exposition of the most useful methods of variance reduction. The exposition is unified by consistent exploitation of adjoint formulations to estimate expected values, as in previous work, and further to evaluate the variance of the resulting estimates.

The connection between adjoint formulations and the choice of biasing schemes is also investigated. In particular, it is shown that the value function (the solution of the integral equation of the adjoint formulation) is always a good choice for importance function biasing; a sharp upper bound, independent of the particular problem, is found for the resulting variance. Predicted (analytical) and experimental (Monte Carlo) results are also given for a simple one-dimensional problem.

1276. C. Eisenhauer, "A Study of the Angular and Energy Distributions of Radiation at Small Distances from a Point Source of Gamma Rays or Neutrons," Nucl. Sci. Eng. 27, 2, 240-251 (February 1967).

The theoretical expression for the singly scattered angular and energy flux of radiation from a point source is studied in the limit of very small source-detector separation distances ($\mu_o \ll 1$). It is shown that both the angular and energy distributions are related in a simple way to the scattering kernel. Examples are given for gamma and neutron point sources in air. The applicability of the distributions at separation distances of the order of a mean free path ($\mu_o r \approx 1$) is discussed.

1277. D. J. Raso and S. Woolf, "Doses Behind Vertical and Horizontal Barriers from a Plane Source of 1.25 MeV Gamma-Ray Photon," Nucl. Sci. Eng. 27, 2, 252-262 (February 1967).

Calculations were performed by the Monte Carlo method to determine the dose at various detector locations behind a vertical barrier and below an adjacent horizontal barrier. Results were obtained, using two different Monte Carlo approaches, for a 1.25-MeV simulated ground source incident on 60, 40, and 20 psf of concrete. The results of the Monte Carlo calculations were used to calculate reduction factors. The above-ground reduction factors compared with those calculated from the OCD Engineering Manual, and the agreement in this case was found to be unsatisfactory. In some instances, discrepancies were found to be as high as a factor of 3. For the case of the open basement, agreement among the three methods was found to lie within 10 to 20%. The results obtained by the two Monte Carlo methods used were found to be in excellent agreement. These results also compared to within 10 to 15% with results of experiments performed at Technical Operations Research. The results indicate that further investigation is necessary to determine the amount of scattered radiation within a basement.

1278. D. R. Mathews, K. F. Hansen, and E. A. Mason, "Deep Penetration of Radiation by the Method of Invariant Imbedding," Nucl. Sci. Eng. 27, 2, 263-270 (February 1967).

The method of invariant imbedding has been applied to neutron shielding problems in plane geometry with realistic energy and angle-dependent cross sections. The method seems to offer advantages over competing methods when the shield is heterogeneous and very thick or when the shield is composed of a very large number of different regions. Reflection and transmission equations are derived by the method of invariant imbedding and their numerical solution discussed. A simple exponential approximation is shown to work well for the solution of these equations. Results for several problems, including a thick homogeneous water shield and a thinner heterogeneous iron/polyethylene/iron shield are compared with results obtained by other methods.

1279. B. Chinaglia, D. Monti, C. Fedrighini, and A. M. Moncassoli, "Neutron Attenuation Experiments in Iron-Water Configurations," Nucl. Sci. Eng. 27, 2, 307-311 (February 1967).

Fast-, epithermal-, and thermal-neutron penetration has been measured in the ETNA facility (plane fission neutron source) for some simple shield configurations (all water or a composite of water, iron, and water slabs with iron thickness of 6.2, 10, and 19.6 cm).

Experimental results are presented as thermal fluxes or activation-detector reaction rates for the water-only configuration and as the ratios of the reaction rates observed in the other configurations to those obtained with water only. These results are compared to calculations performed with a multigroup-removal diffusion code (MACRAD) and a semi-empirical diffusion code (FOG-S) to test their ability to predict the influence of an iron slab on the neutron spectrum and the neutron attenuation.

1280. J. J. Steyn and D. G. Andrews, "Experimental Differential Number, Energy, and Exposure Albedos for Semi-Infinite Media, for Normally Incident Gamma Photons," Nucl. Sci. Eng. 27, 2, 318-327 (February 1967).

Differential angular and energy distributions of back-scattered gamma rays have been measured for photons normally incident on essentially semi-infinite scattering media of various atomic numbers. The measurements have been made by permitting a highly collimated 3- x 3-in. sodium-iodide (thallium-activated) scintillation crystal to view a sample area at the center of a scattering slab region irradiated by a broad beam of gamma radiation. The spectra have been recorded on a 400-channel pulse-height analyzer and reduced to detector incident flux by means of an IBM-7094 digital computer. The results have then been expressed graphically in terms of albedos and fitted to simple empirical expressions. In the case of high-atomic-number scatterers, K x-ray fluorescence has been separately evaluated.

1281. D. J. Dudziak, "Capture and Fission Gamma-Ray Effective Spectra for Shielding," Nucl. Sci. Eng. 27, 2, 328-337 (February 1967).

Effective two-group gamma-ray spectra have been determined for thermal-neutron capture in sodium, nickel, type 304 stainless steel, and tantalum, as well as for ^{235}U prompt-fission gamma rays. A seven-group compilation of capture gamma rays was used as the basis for this study. Absorbed dose (uncollided and builtup) in several materials was calculated for varying thicknesses of several intervening shielding materials. The resulting function for each combination was reduced to two exponential functions over a range of 0 up to 560 g/cm^2. Effective spectra were determined to be as follows:

sodium, 6.09 MeV/capture at 5.5 MeV and 5.74 MeV/capture at 2.0 MeV;
nickel, 8.33 MeV/capture at 8.0 MeV and 1.62 MeV/capture at 2.0 MeV;
type-304 SS, 5.86 MeV/capture at 8.0 MeV and 1.95 MeV/capture at 2.0 MeV;
tantalum, 3.76 MeV/capture at 4.0 MeV and 2.88 MeV/capture at 1.5 MeV;
prompt fission, 2.31 MeV/fission at 4.0 MeV and 4.92 MeV/fission at 1.25 MeV.

These effective spectra reproduce, to within an average absolute deviation of less than 7.4%, the absorbed doses (uncollided and builtup) calculated by the detailed spectra, within the ranges of areal density considered.

1282. K. Shure and C. T. Oberg, "Neutron Exposure of the PM-2A Reactor Vessel," Nucl. Sci. Eng. 27, 2, 348-359 (February 1967).

An irradiation exposure profile was calculated and subsequent experimental measurements using nondestructive dosimetry procedures were obtained for the reactor pressure vessel of the Portable Medium Power Plant (PM-2A). Near the core centerplane at the inner edge of the vessel, the maximum calculated exposure is 1.6×10^{19} n > 1 MeV/cm^2 and the maximum calculated activity from the $^{54}Fe(n,p)^{54}Mn$ reaction is 1.9×10^7 dis/(cm^3 sec) as of July 9, 1965. The average of calculated-to-measured ^{54}Mn activity for 25 azimuthal locations at the core centerplane is 1.05. This agreement is within the uncertainties of either the calculated or experimental results.

1283. R. L. French, J. H. Price, and K. W. Tompkins, "An Analysis of a Shielding Experiment Performed in a Real Fallout Field," Nucl. Sci. Eng. 27, 2, 360-366 (February 1967).

Four simple enclosure shields were exposed to fallout from Shot Small Boy of Operation Sunbeam. Gamma-ray measurements were made both inside and outside the enclosures to determine time-dependent and time-integrated dose transmission factors (dose inside divided by dose outside). The analysis of the experiment consisted of calculating similar factors and comparing with the measured data. Monte Carlo procedures were used to determine the radiation distribution incident upon the shields and to compute the radiation penetration into the shields. Fallout gamma-ray spectra resulting from several different theoretical and experimental investigations were used in calculating the dose transmission factors. The calculated dose transmission factors were found to be consistently higher than the measured factors by as much as a factor of 2, depending upon the particular source term. After investigating several possible causes, it was concluded that the discrepancy was probably the cumulative effect of a deficiency of low-energy photons in the calculated source terms, omission of support structure inside the enclosure shields in the penetration calculations, and of neglecting the effects of ground roughness in the calculations.

1284. E. T. Clarke, "Gamma-Ray Scattering Near an Air-Ground Interface," Nucl. Sci. Eng. 27, 2, 394-402 (February 1967).

This paper reviews the status of experimental and theoretical gamma-radiation intensity at or near an air-ground interface. Various measurements of radiation scattered to a detector 3 ft above an air-ground interface containing a point source of ^{60}Co or ^{137}Cs at various distances are compared with Monte Carlo data and moments method calculations for an infinite homogeneous air medium. Other measurements and calculations of skyshine radiation backscattered by the air under similar circumstances are also analyzed. It is shown that the three approaches -- experimental, Monte Carlo, and moments method -- produce consistent results for ^{60}Co. For an infinite plane source of ^{137}Cs, however, experimental measurements over ice and over concrete indicate about 20% less scattered radiation over concrete than would be received from the same source in an infinite homogeneous air medium.

1285. A. B. Chilton, "The Close-In Exposure Field from Point Isotropic Gamma-Ray Source Located at an Air-Ground Interface," Nucl. Sci. Eng. 27, 2, 403-410 (February 1967).

On the basis of a simplified theoretical model and formulas, a method is presented for calculating the exposure field in the air near an air-ground interface on (or very near) which a point isotropic source of monoenergetic gamma rays has been placed. Necessary functions have been calculated and are presented for cobalt-60 (1.25 MeV) and cesium-137 (0.662 MeV). Within the limitations of the model, calculated results have an uncertainty on the order of 1%, for source-detector distances out to about 15 m. Comparison with previous experimental and theoretical results is made. At detector locations very near the source (on the order of a meter or less), the exposure field is somewhat affected by the exact height of the source above the interface, even for variations on the order of a centimeter.

1286. E. Aalto and A. Krell, "On the Attenuation of Neutrons and Photons in a Duct Filled with a Helical Steel Plug," Nucl. Sci. Eng. 27, 2, 433-440 (February 1967).

Neutron and gamma-ray attenuation have been studied in a cylindrical duct, length-to-diameter ratio = 7, partly (36 vol%) filled by a helical (screw-like) steel plug with an adjustable number of turns. The total neutron leakage through the duct decreased by a factor of 6 when the number of turns increased from 0 to 0.5; and by an additional factor of 3 in going from 0.5 to 2 turns (saturation value). The leakage with less than 0.5 turns is governed by the fast flux and above 0.5 turns by the epithermal flux.

It is shown that the neutron attenuation (in the saturated case) can be satisfactorily predicted by homogenizing the plug and by combining the attenuation calculated in an infinite layer thus obtained with that predicted in a circular duct. The prerequisite is that the attenuation curves obtained are displaced in the streaming direction by the diameter of the duct.

The attenuation of gamma rays in the saturation case (≈ 3 turns) equals that predicted in a homogenized infinite material without the duct attenuation.

1287. Y. Higashihara, M. Nakai, and M. Okubo, "Albedo Component of Gamma Rays Passing Through a Cylindrical Duct," Nucl. Sci. Eng. 27, 2, 441-449 (February 1967).

The albedo component of gamma rays passing through a straight cylindrical lead duct has been investigated experimentally for reactor leakage gamma rays impinging on the duct mouth at various angles of 0 through 90°. The distribution of the albedo component along the duct axis may be expressed in the form $F(\alpha)\ (Z/a)^{-3}$ within the range of Z/a investigated. The contributions from multiple-reflection components to total dose have also been inferred. Also obtained were the radial distributions in both horizontal and vertical directions within and behind the duct.

For a duct with one bend of 90°, the gamma-ray dose contributions arising from scattering areas located in both the first leg and the second leg have been determined separately for the incident beam angle of 0° with respect to the axis of the first leg.

1288. M. Shindo, A. Tsuruo, S. Miyasaka, M. Kitazume, and J. Miyakoshi, "Gamma-Ray Streaming Through a Straight Cylindrical Duct," Nucl. Sci. Eng. 27, 2, 450-463 (February 1967).

The transmission of gamma rays through a straight cylindrical duct from an infinite plane source has been studied both theoretically and experimentally. The distribution of the resulting dose in the vicinity of the duct was calculated by the ray-analysis method, with the exception that the dose arising from radiation scattered within the duct was calculated by a semi-analytical Monte Carlo method. These results, together with those from experiments, lead to useful conclusions. Data applicable to the design of small ducts through shields have been developed for a range of various geometric parameters.

1289. A. F. Rohach, "Weight Optimization of Gamma-Ray Shields Using Layered Media and Spherical Geometry," Nucl. Sci. Eng. 27, 2, 464-467 (Feb. 1967).

Shielding of any gamma-ray source generally requires a material of heavy density. However, in spherical geometry, the geometry effect of the shield configuration can also be important. It is the purpose of this note to illustrate that a combination of light and heavy materials may be used to give an overall optimum spherical shield.

1290. K. Shure, "Few-Group and Multigroup Calculations of Neutron Penetration," Nucl. Sci. Eng. 27, 2, 468-472 (February 1967).

Results obtained from the P3MG1 program and a 54 energy-group structure for nonthermal neutrons have been used as the multigroup standard to which few-group results are compared. Comparison of results obtained with this multigroup program with experimental thermal-neutron flux distributions in water and in iron-water shield arrays and comparison with fast-neutron spectral results from more rigorous calculational techniques have indicated the reliability of this calculational procedure for penetrations of practical interest.

1291. R. V. Petrov, V. N. Pravetskii, Yu. S. Stepanov, and M. I. Shal'nov, Radioactive Fallout (Physics, Biological Effects and Protective Measures), AEC-tr-6634 (1966). Availability· $5.00 CFSTI.

This is a complete Russian manual of fallout, its effects, and protective devices. (RSIC abstract.)

1292. D. G. Harland, "A Radio-Active Method for Measuring Variations in Density in Concrete Cores, Cubes and Beams," Mag. Conc. Res. 18 55, 95-101 (June 1966).

The paper describes an apparatus for measuring the variation of density along cores and beams of concrete. The variation of density is determined by measuring the change in the absorption of gamma radiation. The gamma rays emanate from a one millucurie source of caesium 137 and are detected by a scintillation counter. This is connected to a rate-meter and chart-recorder which continuously display the intensity of radiation transmitted as the sample traverses along its depth dimension through the gamma beam. The variation in the intensity of the transmitted radiation is translated into density variations by a suitable calculation. Attention is drawn to the need to retain, for each type of speciment, the same source sample detector geometry, but it is shown that, with careful operation, variations in density may be measured with a standard deviation of about 1 lb/ft^3. Although designed for use with concrete specimens, the technique can be applied to any cohesive material.

1293. F. W. Kruger, "Optimization of a Shield for Mixed Radiation," ORNL-tr-1499 (Translated from Kernergie 9, 5, 165-168) (1966). Availability: Dep. mn (JCL) $1.10 fs, $0.80 mf.

A plane source of radiation emits two species of radiation which are shielded by a wall of two materials. The conditions are given under which the highest shielding effect is obtained with a given weight per unit area or given thickness of the wall. The results of the consideration are explained by some samples.

1294. A. Schaarschmidt, "Theory of Multiple Scattering of Gamma Radiation in Extended Scattering Media. Part I," ORNL-tr-1152 (Translated from Z. Ang. Phys. 19, 3, 180-187 (1965). Availability: JCL $2.60 (fs), $0.95 (mf).

In the first part of this study, a survey is presented on the principles of the theory of multiple scattering of p-radiation in extended scattering media of small atomic number for the energy region in which pair production does not yet play a role. The study is based on a three-group classification of radiation: primary radiation proper, hard straight-forward quasi-primary radiation, which determines the penetration into greater depths, and diffuse, low-energy scattered radiation of small range. The formation of scattering spectra below the primary energy is described by a straight-ahead solution, but with the use of the Wick eigenvalue; as shown by the comparison with spectra of Goldstein and Wilkins for water, E_o = 0.5 and 1 MeV, this approximation evidently is more favorable for a vertically radiating plane than for a point source. For the spectral distribution in the lower energy region, approximated as a scattering spectrum with a spatially uniformly distributed source, an equation is used which differs somewhat from that of Fano and Nelms; the various approximations are evaluated for a pure Compton scatterer, water, aluminum and iron, and are compared with numerical solutions of the corresponding integral equation. By combining the solutions for the upper and lower energy region, the total field of scattering spectra is constructed with the example of water as a scattering medium, E_o = 1 MeV.

1295. A. Schaarschmidt, "The Theory of Multiple Scattering of Gamma Radiation in Extended Scattering Media. Part II," ORNL-tr-1153 (Translated from Z. Ang. Phys. 19, 3, 187-193) (1965). Availability: JCL $1.60 (fs), $0.80 (mf).

In the second part, so-called "ideal equilibrium spectra" are introduced as a second approximation for the field of scattering spectra, with the assumption of a known course of the buildup factor as a function of the distance, and are compared with the spectra of Goldstein and Wilkins for aluminum, E_o = 1 MeV. The further propagation of scattered radiation in the low-energy region is discussed on the basis of an age-calculation and a stationary diffusion calculation for aluminum as the scattering medium, E_o = 1 MeV, with different approximations concerning the anisotropy of radiation and absorption. The Fermi age is formally expanded to gamma quanta, with consideration of the dependency of the differential cross section of the wave length before scattering. The calculation is carried out with the Flugge assumption of a spatially distributed source, which is simulated here with the use of the Taylor representation for the buildup factor by a sum or three exponential functions. The influence of the diffusion of quanta is reproduced correctly by sign, but in too large an amount, as shown by the evaluation, when compared with the spectra of Goldstein and Wilkins. The numerically calculated spectra thus are in the low-energy region between the "ideal equilibrium spectra" and the diffusion spectra. On the whole, the calculations show that a rather good description of scattering spectra can be obtained with the use of the model considerations and with a still tolerable mathematical effort.

1296. M. J. Schumchyk, M. A. Schmoke, W. O. Egerland, and E. L. Schulman, Scattered Radiation (Skyshine) Contribution to an Open Basement Located in a Simulated Fallout Field - Final Report, NDL-TR-68 (AD-643538) (December 1966). Availability: CFSTI $3.00 (cy), $0.65 (mn).

The objective of this work was to determine, experimentally the shielding afforded by an open concrete-walled basement located in a simulated fallout field and to compare these experimental results with theoretical results published in National Bureau of Standards (NBS) Monograph 42.

A cobalt-60 point-source circulation system was used to simulate a uniformly-contaminated residual gamma radiation area out to a radius of 600 feet. Experimental exposure-rate measurements were made in the free field and at various locations within the structure as a function of height above the basement floor. Ionization chamber dosimeters were used as radiation detectors. Experimental measurements were extrapolated to infinite-field conditions by use of analytical procedures and compared with other related experimental data and theoretical results.

The following conclusions were established:

(1) Analysis of the position function $f(h,\omega)$ at various detector levels above the basement floor indicates the presence of radiation backscattering contribution that is dependent on height above the floor and solid-angle fraction and is not adequately estimated by a single correction factor (1.2 suggested in NBS Monograph 42).

(2) Agreement within 21 percent was obtained between experimental reduction factors and theoretical reduction factors calculated by Equation 31.1 of NBS Monograph 42 for detector locations in the center or near the center of the basement (detector locations C and B).

(3) Theoretical reduction factors underestimate experimental reduction factors at the off-center detector locations close to the basement walls (detector locations A, D, and E) by as much as 47 percent. The differences appear to be caused by radiation scattered from the walls and floor of the basement; this scattering is not completely accounted for in the theoretical calculations.

(4) Extrapolation to the ground surface ($\omega = 1$) of the experimentally measured reduction factors at the center detector location within the open concrete basement yields a skyshine exposure rate that is 7.9 percent of the infinite free-field exposure rate at the 3-foot height. This compares favorably with 8.8 percent calculated in the NBS Monograph 42.

(5) The infinite-field exposure rate at the 3-foot height above a graded, rolled, and relatively smooth field was determined as 468 R/h at a source density of 1 ci/ft^2 of cobalt-60 radiation simulated with a circulating point source.

1297. J. D. Stevenson and J. A. Havers, Entranceways and Exits for Blast-Resistant Fully-Buried Personnel Shelters, (AD-636048) IITRI-OCD - Final Report (September 1965). Availability: CFSTI $3.00 (cy), $0.65 (mn).

The work reported in this study is a continuation of the Total Shelter Design Optimization series which, under OCD sponsorship, is currently in progress at the IIT Research Institute. These studies have developed the design requirements and identified in-place costs for single-purpose, fully-buried personnel shelters of 100, 500, and 1000-man capacities. The attack environment, for analytical purposes, is described by a megaton (W = 1 to 100 MT) nuclear explosion and by side-on surface overpressures of 10 to 200 psi. Reports issued to date in this series have included the following:

(1) "Structural Materials for Hardened Personnel Shelters," by John A. Havers, Contract No. OCD-PS-62-66, Subtask 1151-A, IIT Research Institute, December 1963.

(2) "Structural Cost Studies for Hardened Shelters," by John A. Havers and Jerry J. Lukes, Contract No. OCD-PS-64-50, Subtask 1152-E, IIT Research Institute, January 1965.

(3) "Entranceways and Exits for Blast-Resistant Fully-Buried Personnel Shelters," by John D. Stevenson, and John A. Havers, Contract No. OCD-PS-64-50, Subtask 1152-E, IIT Research Institute, May 1965 (draft).

(4) "An Investigation of Minimal Equipment Needs in Personnel Shelters," by John A. Havers, Claire B. Monk, Jr. and Erich H. Koeller, Contract No. OCD-64-50, Subtask 1216-A, May 1965 (draft).

By combining the findings of these separate studies, it is possible to perform rapid evaluations of specific attack environments, composite performance requirements, alternative analytical solutions and estimated in-place costs as a basis for the preliminary design of this particular type of shelter. It is found that structural per-occupant costs are strongly related to the design capacity of the shelter, and exhibit a decrease as the shelter capacity is increased. The structural costs are also found to increase with increasing levels of design overpressure, although this latter correlation becomes decidedly weaker as the overpressure level is increased.

1298. R. D. Schamberger, MCR Program Shield Study in the ORNL Lid Tank, NDA-2143-7 (April 21, 1961). Availability: Dep. mn CFSTI $2.00 (cy), $0.50 (mn).

A series of lid tank experiments was performed at ORNL to assist in the design of the MCR shield. Neutron and gamma ray measurements were taken behind configuration containing LiH, iron and LiH, borated iron and LiH, and lead and LiH. These experiments were primarily concerned with determining the nature of the secondary gamma production and the influence of boration, reflector thickness, and core composition on the secondary gamma production. The experimental data and descriptions of the configurations are presented.

1299. F. Kam and R. Schamberger, Military Compact Reactor Program Shield Study in the ORNL Lid Tank - Supplement, NDA Memo 2143-11 (June 6, 1961). Availability: Dep. mn CFSTI $1.00 (cy), $0.50 (mn).

Point kernel calculational methods were applied to analyze gamma and fast neutron dose rates of lid tank measurements.

The effective removal cross section for a thick piece of steel backed by LiH and H_2O was measured.

The effective energies of the secondary radiation produced in lead and steel were determined.

1300. L. R. Bunney and D. Sam, Gamma-Ray Spectra of the Products of Fast Neutron Fission of U-235 and U-238 at Selected Times after Fission, USNRDL-TR-1076 (AD-642319) (September 13, 1966). Availability: CFSTI, $3.00, $0.65 mn.

Experimental measurements of the gamma-ray pulse height distributions due to the products of fast-neutron-induced fission of U^{235} and U^{238} are presented. The measurements were made at nine selected times after fission from 15 minutes to 3 days. Irradiation times and counting intervals were chosen to minimize saturation and decay effects.

The experimental data were used to calculate 100 energy bin distributions of the absolute number of photons/fission-sec by means of absolute calibrations of the collimated TaI(Tl) detector. The number of fissions in each sample was determined radiochemically. Machine computation was used extensively.

1301. T. M. Jordan, Reactor Shielding Calculations by the Monte Carlo Method, Douglas Paper No. 3271 (AD-620149) (May 1965). Availability: CFSTI, (HC) $3.00, $0.65 (mf).

This paper outlines Monte Carlo techniques for the numerical solution of the energy-dependent Boltzmann equation for neutrons or gamma rays, which significantly reduce the computer time required for Monte Carlo shielding calculations. This reduction is a maximum for calculations involving multi-energy source and/or interaction spectra and complicated source, shield, and system geometries. The Monte Carlo techniques include the use of energy independent biasing functions, energy-dependent samples which span the entire energy range of interest, and analytic integrations of the energy dependence of the differential angular-energy interaction cross sections.

Typical numerical results obtained from the Douglas Monte Carlo computer program SOBER are presented to demonstrate the accuracy, flexibility and computational speed achieved through the use of these techniques.

Work presented herein was conducted by the Douglas Missile and Space Systems Division under the Company sponsored Research and Development Account No. SO 80331-002.

1302. R. N. Bruce, Jr., J. E. Branch, M. P. Gronbeck, J. T. Hanley, and D. F. McCahill, An Investigation of School Designs to Resist Integrated Nuclear Weapons Effects, AD-626770, Tulane University - OCD Technical Report No. 100 (October 1965). Availability: CFSTI, $3.00 (hc), $0.65 (mf).

Each school design was examined to determine the inherent protection procided against nuclear weapons effects including thermal radiation, blast, and nuclear radiation in the form of radioactive fallout. On the basis of the individual analyses, recommendations were made concerning points of weakness in the protection provided by each structure. Recommended modifications were generally minor and structural in nature. There was no intent to modify the aesthetics of the structures considered. Modifications were recommended to increase the protection to a more desireable level with little if any increase in cost.

Since the schools were designed to provide a protection factor against fallout of at least 100, no attempt was made to improve this level of fallout protection. The minor modifications necessitated by blast and thermal considerations generally served to increase the protection factor approximately 5%.

Whereas the fallout and blast analyses are quantitative in nature, the present state of knowledge of thermal effects required that the thermal evaluations and recommendations be of a qualitative nature. Those qualities considered for each school included:

1. Site layout
2. Structural considerations
3. Compartmentation and firestops
4. Grouping of hazardous areas
5. Furnishings and materials of construction
6. Firefighting from within
7. Smoke and heat venting

The blast analyses indicated that the inherent blast resistance provided by the schools as designed varied from approximately 0 psi to 2 psi overpressure resistance. With minor modifications, the level of overpressure resistance could be increased to approximately 10 psi.

The general observations stated above are based on the individual analyses of each of the school designs. The modifications recommended for increased thermal and blast protection were examined to determine any increased costs resulting from the recommended changes. In some school designs, the modifications could be accomplished at no increase in cost. In other designs, the cost was generally somewhat less than $1.00 per square foot of building area.

1303. J. B. Eggen and F. O. Leopard, Analysis of Slant-Incident Radiation Through Ducts and Voids in Shields, FZK-311 (December 20, 1966). Availability: Hdq. USATAC, Warren, Michigan, ATTN: SMOTA-RCSN.

Calculated gamma-ray and neutron transmission through ducted shields are compared to previously obtained measured data. The shield panels were at angles of 0°, 15°, and 30° from a collimated reactor radiation source. Calculational methods used were the Monte Carlo Multibend Duct Procedure (L05) and the Flexible Monte Carlo (FMC) Procedure. Comparison of results (limited with L05) from the two calculational procedures suggests that the FMC procedure is preferable for use with the plane source and plane detectors chosen for this study. Both procedures give results that are lower than the measured values. The results are presented in terms of the integrated dose rate over an area parallel to and 3 in. from the exit face of the test panel. The gamma-ray dose rates are a factor of 100 higher than the neutron dose rates; thus total dose rates and gamma-ray dose rates are approximately equal. Measured-to-calculated dose-rate ratios for slant incidence radiation vary from 1.7 to 2 for gammas and from 1.01 to 2.5 for neutrons. Results indicate a decreased dose rate as the incident radiation angle increases. At 30° incidence, the gamma-ray dose rates are from 69% to 84% of the normal-incident data; the neutron dose rates are from 34% to 58% of the normal-incident data. This reduction is sufficient to warrant consideration of angle of incidence of radiation in the design of armored vehicles.

1304. J. R. Williams and C. S. Stevens, Dose Rate from 14.7-MeV Neutrons and 1.25-MeV Gamma Rays Transmitted Through Special Optical Materials, FZK-313 (December 2, 1966). Availability: U. S. Army Frankford Arsenal, Philadelphia, Pennsylvania, ATTN: SMUFA-N5300.

Two monoenergetic radiations (14.7-MeV neutrons and 1.25-MeV gamma rays) were collimated individually onto a steel-polyethylene slab with a 6-in.-diam cylindrical duct. Transmitted dose rates were measured for the solid slab, the open duct, and the duct filled with optical materials having various thicknesses, positions, and combinations. The experiment demonstrated the effects on transmitted dose rate of varying thickness, position, and composition of the optical materials for the two incident radiations.

1305. R. J. Prevost, Reflection and Transmission of 14.6-MeV Neutrons Incident on Shield Materials, FZK-314 (December 15, 1966). Availability: Hdq. USATAC, Warren, Michican, ATTN: SMOTA-RCSN.

To provide data for preliminary evaluation of calculational procedures, an experiment was performed with a narrow-beam source of 14.6-MeV neutrons incident along the normal and at an angle of 45° to the normal of slabs of shielding materials. Reflected and transmitted fast-neutron dose rates resulting from the incident 14.6-MeV neutrons were measured. The Slab Penetration and Reflection Calculation (SPARC) procedure was used to predict those reflected and transmitted dose rates that were measured, and comparisons were made between such measurements and calculations. Thirteen slab configurations of concrete, 6% borated polyethylene, armor steel, and laminations of the last two named materials were used in the study.

The agreement between the experimental data and the SPARC calculations warrants the use of this procedure for design evaluations of current interest in the Radiological Armor Program.

The portion of the incident field of 14.6-MeV neutrons reflected from slabs of steel or from multilayer slabs for which steel was the incident layer varied from 11 to 42%. The portion of the incident field of 14.6-MeV neutrons reflected from polyethylene or from multilayer slabs for which the first layer was polyethylene varied from 2 to 17%.

For a slab of 3-in.-thick layer of steel backed by an equal thickness of polyethylene the transmitted-neutron dose rate was about 20% less than that for a slab with these materials in reverse order.

1306. G. T. Western, F. L. Gibbons, J. R. Williams, B. E. Morris, and H. G. Carter, Angular Distribution of Fast-Neutron and Nonelastically Produced Secondary-Gamma-Ray Dose Rates from 14.6-MeV Neutrons Incident Normal to Slabs, AFWL-TR-65-210 (April 1966).

To provide data for preliminary evaluation of calculational procedures, an experiment was performed with a source of narrow-beam 14.6-MeV neutrons incident normal to a slab. The procedures are used for predicting the scattered and uncollided components of fast-neutron dose rate and the secondary-gamma-ray dose rate from nonelastic reactions. Twelve slab configurations of 6% borated polyethylene, armor steel, and combinations of these materials were used in the study.

Excellent agreement was obtained between predicted and measured values of fast-neutron dose rate, and for secondary-gamma dose rate from all configurations containing borated polyethylene and borated polyethylene plus steel. Maximum disagreement between predicted and measured scattered neutron and secondary-gamma dose rates occurred for the configurations containing only armor steel; predicted values increased from 20% lower for 4-in. steel to 40% lower than experimental values for 1-in. steel.

For the pure borated polyethylene slabs, the major source of secondary-gamma dose rate was the $C^{12}(n,n'\gamma)$ reaction. For slab configurations containing steel, all other sources of secondary-gamma production were negligible compared to the gamma rays produced by fast-neutron nonelastic reactions in the iron.

The component of secondary-gamma-ray dose rate resulting from thermal-neutron capture in the slabs containing armor steel and 6% borated polyethylene is negligible for the normally incident radiation.

1307. F. O. Leopard, Monte Carlo Analysis of Fast-Neutron Spectra in an Infinite Medium of Paraffin Wax, AFWL-TR-65-218 (April 1966). Availability: U. S. Air Force Weapons Laboratory, Kirtland Air Force Base, New Mexico, ATTN: WLDN.

A Monte Carlo study has been made to investigate the fast-neutron spectrum and angular distribution in a large assembly of paraffin wax. The shield configuration and detector locations were chosen so that techniques currently used in calculations and experiments of neutron deep penetration could be evaluated. A modified version of the Flexible Monte Carlo (FMC) procedure, along with several data-preparation and analysis routines, was developed and used to analyze the results. Calculations were performed to determine the angular-flux values at penetration thicknesses ranging from 5 cm to 60 cm. A comparison of the calculated and measured angular flux data was made for several penetration thicknesses and angles of incidence for which experimental data were taken. A divergence in spectral shapes at 0° is attributed to a "streaming effect" in the experimental data for direct-beam flux. Except for this difference, the Monte Carlo results are in good agreement with a large range of comparable experimental results.

1308. M. Leimdorfer and J. Barish, A Method of Representing Two-Dimensional Distributions for Use in Monte Carlo Calculations, ORNL-4002 (Jan. 1967). Availability: CFSTI, $3.00 (cy), $0.65 (mn).

In an attempt to increase the efficiency of an existing Monte Carlo program, the Nucleon Transport Code (NTC), for treating the transport of high-energy nucleons in matter, a method has been developed which is intended to improve the procedure for selecting combinations of the energy and angle of particles released in nuclear interactions. In the present form of NTC, the process of particle emission from nucleon-nucleus collisions is simulated by a Monte Carlo treatment of the intranuclear cascade. In the proposed method, the particle emission data obtained from the intranuclear cascade calculations are represented in the form of tables which are suitable for sampling purposes. The power of the method lies in the fact that the tables are constructed directly from the intranuclear cascade (Monte Carlo) histories.

1309. D. J. Dudziak and M. E. Battat, Gamma-Ray Energy Deposition in the Molten Plutonium Burnup Experiment (MPBE) Reactor Structure, LA-3606-MS (January 12, 1967). Availability: CFSTI, $3.00 (cy).

In order to determine thermal stresses, heat removal requirements, and thermal expansions, the spatial distribution of neutron and gamma-ray energy deposition in a reactor must be known. Such distributions have been determined for the Los Alamos Molten Plutonium Burnup Experiment (MPBE), using both Monte Carlo and point kernel integration techniques to calculate gamma-ray transport. Neutron transport was calculated using both the diffusion theory and S_4 approximations in two dimensions. Preliminary calculations determined the total gamma-ray and neutron energy escape from the reactor vessel and, thus, the reactor room heat load from these sources. More detailed calculations provided point values of the absorbed dose in iron, carbon, nickel, tantalum, and sodium at 70 selected spatial points throughout the reactor system.

1310. E. R. Schleiger, M. Goldstein, and E. Tochilin, Calorimetric Measurements of Gamma Ray, Fast Neutron, and Charged Particle Absorbed Doses, USNRDL-TR-621 (January 28, 1963). Availability: CFSTI, $3.00 cy, $0.65 mn.

The energy absorbed by aluminum, carbon, or tissue-equivalent (TE) material when exposed to gamma rays, fast neutrons, or 900-MeV alpha particles was measured with a microcalorimeter. The intrument was patterned after the one described by Reid and Johns, Rad. Res. 14, 1 (1961), but has an ethylene glycol bath instead of water. Spaced mylar sheets about 6-1/2-in. in diameter were mounted on one side of the absorber inside the evacuated cylindrical chamber in order that the detector could be thermally insulated in the direction of the beam without the presence of bath solution.

Gamma-ray absorbed dose rates as low as 2 rads/min were measured with the instrument. The dose rates obtained from the exposures to Co^{60} and Cs^{137} sources at this Laboratory (NRDL) agreed with the output of the sources as measured with a National Bureau of Standards secondary standard cavity ionization chamber. Thirteen 10-min exposures of the TE absorber in the microcalorimeter to fast neutrons from the University of California 60-in. cyclotron at Crocker Laboratory resulted in absorbed dose measurements varying from 24 to 40 rads. The average absorbed dose value was 5% higher than the NRDL determination based on flux and spectrum measurements. Microcalorimeter exposures to the 900-MeV alpha particle beam from the University of California 184-in. cynchrocyclotron at Lawrence Radiation Laboratory gave absorbed doses that agreed reasonably well with ionization chamber determinations.

1311. L. E. Weaver, P. O. Strom, and P. A. Killeen, Estimated Total Chain and Independent Fission Yields for Several Neutron-Induced Fission Processes, USNRDL-TR-633 (March 5, 1963). Availability: CFSTI, $3.00 cy, $0.65 mn.

Calculated estimates are presented of unmeasured total chain yields and of independent fission yields for the fission products resulting from fission-spectrum and 14-MeV neutron bombardment of U-233, U-235, U-238, and Pu-239 and from thermal neutron bombardment of U-233, U-235, and Pu-239.

1312. T. W. Eckels and A. Smaardyk, Design, Performance, and Evaluation of Shielding Window for Faret Containment Cell, ANL-7157 (July 1966). Availability: CFSTI, $3.00 cy, $0.65 mn.

A shielding window, possessing high integrity, has been designed for use in a reactor containment structure. This has been accomplished mainly by the appropriate placement and sealing arrangement of thick glass slabs. Conservative capability to withstand, without damage, over two times the required design pressure of 3- psig has been achieved. High resistance to potential missile damage has been included, mainly by the ability of the first slab to absorb impact while protecting successive slabs. For extreme temperature conditions associated with "maximum credible accident" criteria, tests have demonstrated the window integrity in the same manner of protection. In this case the first slab, even if cracked, would serve as an insulating member protecting the next slabs. A high degree of visibility has been achieved by the selective arrangement of windows, adequate glass area, and a combination of glass slab sizes and thicknesses resulting in desirable angles of view. In addition, EBR-II design features for replacement of glass components have been incorporated in a manner such that replacement may be done while the integrity of the cell is maintained.

1313. R. C. Greenwood and J. H. Reed, Prompt Gamma Rays from Radiative Capture of Thermal Neutrons, IITRI-1193-53 (Vols. 1 and 2) (October 14, 1965). Availability: $3.00 CFSTI, $0.65 mn.

An atlas of thermal neutron capture gamma ray spectra measured under standard conditions using NaI(Tl) scintillation detectors, and a comprehensive compilation of the neutron capture gamma ray data which was available prior to May, 1965, are contained in this report. Included are prompt gamma ray spectra, and data, from 74 naturally occurring elements, together with a number of separated isotopes, unstable elements and fissionable materials.

The objective of this report is to provide sufficient data concerning neutron capture gamma ray spectra to allow evaluation of practical uses of an analytical technique based upon the detection of neutron capture gamma rays in measurement, analysis and control technology and thereby to facilitate the use of the method in these areas.

1314. D. J. Dudziak and L. B. Freeman, "Irradiated Ra:Be Neutron Sources," Nucl. Sci. Eng. 11, 3, 324-327 (1961).

An analytical and experimental investigation of irradiated Ra:Be neutron sources has been carried out at the Bettis Laboratory and at the Materials Test Reactor (MTR) in Idaho. The significant isotopes in the Ra-226, Ac-227, and Th-238 chains are described analytically and the neutron source strengths and gamma dose rates are calculated. Comparison with experiment shows a reasonable predictability in most cases.

1315. A. Shimizu and H. Mizuta, "Application of Invariant Imbedding to the Reflection and Transmission Problems of Gamma Rays (II)," J. Nucl. Sci. Technol. 3, 10, 441-447 (October 1966).

The transmission of gamma rays through slab was calculated by the method of invariant imbedding. The results are in good agreement with other reliable solutions obtained by the moments method and Monte Carlo calculations. The present method has the following advantages: (1) It permits determination of the energy and angular distributions of reflected or transmitted photons due to monoenergetic and monodirectional source photon; (2) the method can be applied to multi-layer slabs; (3) extension to thicker slabs can be easily made; (4) the execution time required for numerical solutions is short enough to permit systematic calculations for many elements.

1316. A. L. Frank, "Gamma-Radiation Characteristics--Angular Distribution Over a Desert Terrain Fallout Field," Health Phys. 12, 12, 1715-1731 (December 1966).

A gamma-radiation spectroscopy experiment was carried out at the Nevada Test Site near Mercury, Nevada, in the summer of 1962. A gamma-ray NaI scintillation spectrometer was taken onto a fallout contaminated desert terrain. Measurements were made at ten vertical angles at 9 days after shot time. All data were for a detector height of 4 ft. The pulse height data were reduced to photon spectra (photons/min-MeV-in.^{2}-steradian) by means of a matrix iteration procedure. Some comparisons were made between the vertical-angle photon spectra and data from other sources. Also a calculation of the effective ground roughness of the desert terrain was made by the absorbing overlayer model. This resulted in a calculated overlayer of approximately 24 equivalent feet of air.

1317. W. B. Terney and H. Fenech, "Shipboard Reactor Shield Optimization Using Optimum Gradient Method, Nucl. Appl. 3, 46-52 (January 1967).

The optimum gradient technique is reviewed and then used to optimize a shipboard reactor shield system consisting of a water-lead primary and a concrete-lead-polyethylene secondary shield. The shield is optimized to the point where the cost of further reductions in weight exceeds the worth (K) of these reductions to the ship, and subject to five dose point constraints. Plots of eight thicknesses as a function of K are given. For high K values (essentially weight optimization) the concrete thicknesses are zero. As K decreases and cost becomes more and more important in comparison to the weight, concrete is added, and the more expensive lead and polyethylene are subtracted.

1318. K. M. Dee, "Multi-Dimensional Spherical Harmonics Method for Energy Dependent Problems," Nuckeonic 8, 7, 383-386 (September 1966).

The explicit multi-dimensional spherical harmonics formalism is extended to energy dependent problems. Angular- and space-energy dependence are separated. The former can be treated as in the one-velocity case, the latter as in the one-dimensional case. As an illustration multi-dimensional $P_N L_Q$ approximation is developed.

1319. K. G. Darrall and G. Oldham, "The Installation of a 14-MeV Neutron Generator," Nucl. Energy, 189-94 (July 1966).

A Kaman Model 1003 14-MeV neutron generator along with its installation is described.

1320. S. M. Ermakov and E. E. Petrov, "The Problem of Gamma-Radiation Penetration Through Shields," Sov. J. Atomic Energy (English Transl.) 19, 1, 947-949 (1965).

In this paper, an effect is described which is associated with the penetration of gamma rays through plane shields consisting of two components: a primary water layer and a secondary lead layer.

The effect consists of the following: for incidence of hard gamma rays at large angles to the normal, an increase in the thickness of the water layer can lead to an increase in the intensity of the radiation penetrating the shield.

1321. E. E. Kovalev and D. P. Osanov, "On the Determination of Shielding for Radiation from Cylindrical Sources," Sov. J. Atomic Energy (English Transl.) 20, 1, 104-106 (January 1966).

Nomograms are presented to determine concrete shield thickness for cylindrical source of gamma rays. (RSIC)

1322. G. I. Lukishov and O. A. Chelyuk, "New Design for Heavy Protective Windows," Sov. J. Atomic Energy (English Transl.) 20, 1, 118-119 (January 1966).

Characteristics of some shield windows are given. (RSIC)

1323. Y. I. Bublik, S. M. Ermakov, B. A. Efimenko, V. G. Zolotukhin, and E. E. Petrov, "Dose Rate from a Unidirectional Source of Gamma Quanta Close to the Ground-Air Interface," Sov. J. Atomic Energy (English Transl) 18, 6, 791-792 (June 1965).

The dose rate from a unidirectional, monoenergetic gamma quantum source close to the earth-air surface has been calculated by the Monte Carlo method (i.e., by calculating the flux locally, separating analytically the contribution of single scattering, and treating in parallel the results of statistical tests for many points in space). The statistical accuracy of the dose rate determination is 6-8%.

A comparison is made with experimental results, obtained with a Co-60 source. The agreement is satisfactory.

1324. I. I. Bondarenko, V. G. Liforov, V. N. Morozov, M. N. Nikolaev, V. A. Parfenov, and V. A. Semenov, "Measurement of Neutron Spectra in Nickel, Iron, and Stainless Steel," Sov. J. Atomic Energy (English Transl.) 18, 6, 752-760 (June 1965).

Measurements of neutron spectra in various media, obtained by the transit time method in the fast pulse reactor IBR with a resolution of 0.04 μsec/m, are presented. The spectra of neutrons emerging from iron and nickel prisms of various thickness and also those from stainless-steel prisms are studied. The "fine structure" due to the resonance character of the cross sections of the media studied was clearly seen in the spectra measured. The experimental neutron spectra are compared with calculations made with a multigroup system of constants allowing for resonance self-screening of the cross sections. The reasons for the slight discrepancies found are analyzed.

1325. S. G. Tsypin, B. I. Sinitsyn, and V. K. Duruga, "Transmission of Neutron Radiation from a Reactor Through Hydrogen-Free Media," Sov. J. Atomic Energy (English Transl.) 18, 4, 442-50 (April 1965).

The results of experimental and theoretical investigation into the transmission of neutrons through hydrogen-free uniform materials and mixtures are discussed. The relaxation lengths obtained for neutrons of different energy groups are discussed. It is shown that under certain assumptions the magnitude of the flux attenuation for neutrons of the upper energy group ($E_n \geq 3$ MeV) in a hydrogen-free medium can be obtained from very simple empirical expressions by using well-known constants: the removal cross section σ_{rem}, the cross section obtained from the reciprocal of the relaxation length (σ_1/λ), and the asymptotic cross section σ_{as}. Values are derived for these constants.

1326. A. P. Suvorov, A. G. Guseinov, and M. N. Nikolaev, "Resonance Structure of the Cross Sections and Its Influence on the Scattering Anisotropy for Fast Neutrons and Their Transmission in Iron," Sov. J. Atomic Energy (English Transl.) 18, 3, 352-357 (March 1965).

The influence of the resonance structure of the cross sections on the propagation and moderation of neutrons in various media is discussed. This paper also presents detailed studies of the changes in neutron spectrum's "microstructure," which must be known in order to determine the mean cross sections.

1327. S. F. Degtyarev and V. I. Kukhtevich, "Spatial Distribution of the Dose Rate of Air-Scattered Neutrons from a Unidirectional Point Source," Sov. J. Atomic Energy (English Transl.) 18, 3, 315-316 (March 1965).

We measured the spatial distribution of the relative dose rate produced by unidirectional point sources of neutrons with initial energies equal to 4.2 and 1.9 MeV at a distance of 14.8 m between the source and the detector.

1328. Y. I. Kolevatov, V. I. Kukhtevich, E. S. Matusevich, and O. A. Trykpv, "Angular and Energy Distribution of Scattered Gamma-Radiation Near an Isotropic Source in an Infinite Air Medium," Sov. J. Atomic Energy (English Transl.) 18, 3, 313-14 (March 1965).

We performed an experimental investigation of the angular and energy distribution of scattered gamma-quanta from a Co-60 isotropic point source in an infinite air medium for two values of the distance R from the source (15 and 30 m) and for different angles between the source-detector line and the symmetry axis of the unidirectional detector between 12 and 180 degrees.

1329. M. V. Fedulov, "Use of the Method of Moments for Solving Equations of Neutron Thermalization in Infinite Media," Sov. J. Atomic Energy (English Transl.) 18, 3, 290-97 (March 1965).

The possibility of using the method of moments for solving the integral equation of neutron thermalization in an infinite homogeneous medium is investigated. Comparisons are made with the exact solution for certain cases of moderation in monatomic gases.

1330. S. P. Belov, V. A. Dulin, Y. A. Kazanskii, V. I. Popov, and S. G. Tsypin, "Experimental Investigations of Shields on the Riz Stand," Sov. J. Atomic Energy (English Transl.) 18, 2, 167-172 (February 1965).

The present article describes a stand with a zero-power reactor designed for investigating the processes occurring in that part of the shield which is in close contact with the reactor core. If the processes occurring in this part of the shield (formation of the neutron spectrum and generation of strong gamma radiation) are known, the dimensions and the weight of the reactor's entire shield can be properly calculated.

The results obtained in measuring neutron spectra and investigating shielding materials (iron, nickel, and borated nickel) on the stand are given.

1331. G. V. Val'skii, D. M. Kaminker, G. A. Petrov, and L. A. Popeko, "Concerning the Emission Times of Gamma-Quanta as a Result of Fission," Sov. J. Atomic Energy (English Transl.) 18, 3, 279-282 (March 1965).

The emission times of gamma quanta in the thermal neutron fission of U^{235} were measured by the method of geometrical discrimination of individual sections of the path of the fission fragments. Collisions between fission fragments and gamma quanta were recorded as a function of the distance of the collimator axis from the target. In contast to the data of certain earlier reports, the conclusion is drawn that the main bulk of fission gamma radiation is emitted after a time of less than 5×10^{-11} sec. It was found that within the interval $3 \times 10^{-10} - 2 \times 10^{-8}$ sec, not more than 5-10% of the gamma radiation accompanying fission is emitted.

1332. K. K. Popkov and S. M. Rubanov, "Dependence of the Density of Radiation Damage to the Reactor Vessel on the Composition of the Ferro-Aqueous Thermal Shield," Sov. J. Atomic Energy (English Transl.) 18, 1, 83-84 (January 1965).

The present article is concerned with an investigation of ferro-aqueous mixtures with different compositions for thermal shields from the point of view of the effect of the mixture composition on the radiation damage to the reactor vessel caused by neutrons. For this purpose, we considered a primary-shield composition consisting of 25 cm of ferro-aqueous mixture, the composition of which varied, 9 cm of iron, which simulated the reactor vessel, and 30 cm of water, which corresponded to the primary shield bath.

1333. F. A. Makhlis and I. M. Kolpakov, "Determination of Gamma-Ray and Neutron Absorbed Dose in Polymers," Sov. J. Atomic Energy (English Transl.) 18, 1, 52-56 (January 1965).

Together with results, a method is given for calculating the gamma ray and neutron absorbed dose in several polymers which is based on the elemental composition of the materials and the energy of the radiation.

1334. M. V. Blinov, N. M. Kazarinov, and A. N. Protopopov, "Angular and Energy Characteristics of the Neutrons Emitted in U-235 Fission," Sov. J. Atomic Energy (English Transl.) 18, 2, 133-139 (February 1965).

The velocities of a fission fragment and of a neutron moving in the same direction were measured simultaneously by the time of flight method for the thermal neutron fission of U^{235}. As a result, the emission spectrum of the fission neutrons was obtained. The angular distributions and energy spectra were also measured for the neutrons ejected at different angles to the direction of motion of the fission fragments. These distributions were compared with calculations in which the emission spectrum obtained was used. The comparison showed that the data agree well after excluding small deviations in the angular distribution. It follows from analysis of the results that the emission spectrum agrees with the data calculated by the statistical theory of evaporation and also that not less than 90% of the neutrons from U^{235} fission are emitted in the process of isotropic evaporation of neutrons from the completely accelerated fission fragments.

1335. A. G. Basil'ev, A. P. Veselkin, Y. A. Egorov, V. A. Kucheryaev, and Y. V. Pankarat'ev, "The Attenuation of Reactor Radiation by Means of Serpentine Concrete," Sov. J. Atomic Energy (English Transl.) 18, 2, 151-157 (February 1965).

A description is given of experiments on the study of the shielding properties of concrete that has a serpenting aggregate. It is shown that in comparison with limonite concrete the shielding properties of this type of concrete are slightly better as regards neutrons, and no worse as regards gamma radiation. At the same time, this type of concrete has a much higher capability of withstanding the effect of temperature, and its use in reactor shielding can be recommended at least up to temperatures of 450°C.

1336. J. S. Bevan, "Parameters Describing the Microscopic Distribution of Radiation Dose," Phys. Med. Biol. 11, 3, 405-409 (1966).

The parameters introduced by Rossi for the quantification of energy deposition in small spherical volumes are discussed, and more general parameters, appropriate to the distribution of radiation dose in volumes of arbitrary shape, are defined. It is shown that the Rossi parameters can be regarded as a special case of the more general parameters. The application of these new parameters to cylindrical volumes is outlined, and a resulting 'definition' of a long cylinder is suggested.

1337. S. Sasse, and C. D. Wuneke, A Method for the Derivation of Build-up Factors for Normal Concrete from Attenuation Coefficients, ORNL-tr-1510 (Transl. from ABS-THH-1002) (1966). Availability: Dep. mn, $3.00 cy, $0.65; JCL, $1.60 fs, $0.80 mf.

A scheme is presented for fitting the approximately determined buildup factors for concrete by an approximate analytic representation.

1338. W. Futtermenger, A Method of Measuring Build-Up Factors for Gamma Energies from 0.5 to 15 MeV, ORNL-tr-1514 (Transl. from ABS-THH-1006) (April 1964). Availability: CFSTI, $3.00 cy, $0.65 mn; JCL, $2.60 fs, $0.95 mn.

This report presents a measuring principle which permits a determination of the buildup factors for gamma rays in the energy region of 0.5-15 MeV as necessary for shield calculations. At the present, a few calculated dose build-up factors and energy absorption buildup factors are available. They exist only for a few elements and cannot be simply applied to composite shields. The treatment of gamma radiation from the capture of thermal neutrons, which is transmitted obliquely through the layers of the shield, is a particular problem. Suitable buildup factors are not known for this case.

1339. G. Stiller, "Neutron Flux Measurements in the Mockup of a Power Reactor Shield in the Large Irradiation Channel of the Geesthact Research Reactor," ORNL-Tr-1519 (Translated from ABS-THH-1015 (1966). Availability: Dep. mn. CFSTI, $3.00 cy, $0.65 mn, JCL, $1.60 fs, $0.80 mf.

Fast, epithermal, and thermal neutron fluxes are measured with activation devices placed around a mockup of a power reactor. (RSIC)

1340. G. Stiller, "Determination of Fast Neutron Spectra with Threshold Detectors," ORNL-tr-1520 (Translated from ABS-THH-1016) (1966). Availability: Dep. mn, CFSTI, $3.00 cy, $0.65 mn; JCL $1.60 fs,)0.80 mf.

The use of Mg-24, Si-28, Cu-63 as threshold detectors for the measurement of fast neutron spectra is described. (RSIC)

1341. G. Stiller, "Determination of the Fast Neutron Flux Spectrum with Threshold Detectors," ORNL-tr-1521 (Translated from ABS-THH-1016a) (1966). Availability: Dep. mn CFSTI, $3.00 cy, $0.65 mn; JCL, $1.60 fs, $0.80 mf.

Neutron spectra are measured with P-31, Fe-56, and Al-27 threshold detectors. (RSIC)

1342. W. Futtermenger, "Measurement of Gamma Radiation in the Mockup of a Power Reactor Shield in the Geesthact Research Reactor (FRG)," ORNL-tr-1523 (Translated from ABS-THH-1018) (1966). Availability: Dep. mn, CFSTI, $3.00 cy, $0.65 mn; JCL, $4.60 fs, $1.68 mf.

Measurements are made of radiation around a shielded mockup reactor. (RSIC)

1343. K. Heinz, Experimental Determination of Geometry Effects in the Measurement of Buildup Factors for Gamma Radiation in Concrete, ORNL-tr-1524 (Transl. from ABS-THH-1021) (1966). Availability: Dep. mn, CFSTI, $3.00 cy, $0.65 mn; JCL, $4.60 fs, $1.49 mf.

Measurements are made to determine when, in practical situations, it is proper to use a point source buildup factor and when a monodirectional plane wave buildup factor. (RSIC)

1344. H. Schubart, Calculation of Gamma Radiation Buildup Factors in Layered Shields, ORNL-tr-1527 (Transl. from ABS-THH-1025) (1966).

The problem of estimating the buildup factor in layered shields is reviewed. A modified form of the Kitazume equation is found to be relatively successful. (RSIC)

1345. S. Sasse, Tables of Shielding Constants for the Components of Typical Shielding Materials, ORNL-tr-1529 (Transl. from ABS-THH-1027) (1966). Availability: CFSTI, $3.00 cy, $0.65 mn.

The following constants are given for a large number of shielding materials: density, gamma absorption coefficients, removal cross section, scattering and total cross sections, mean cosine of scattering angle, diffusion length, diffusion constant, Fermi age, and capture gamma spectra. (RSIC)

1346. W. Marth, "Dose Constants of Gamma Rays," ORNL-tr-1542 (Transl. from Atompraxis 12, 8, 393-398, 1966). Availability: Dep. mn, CFSTI, $3.00 cy, $0.65 mn; JCL, $1.10 fs, $0.80 mf.

A tabulation of dose conversion factors for gamma rays from various isotopic sources is given. (RSIC)

1347. M. Tisljar, "Study of the Properties of Gamma Radiation in Air," ORNL-tr-1543 (Transl. from Acta Phys. Austriaca 21, 1,2, 198-203) (1966). Availability: Dep. mn, CFSTI, $3.00 cy, $0.65 mn; JCL, $1.10 fs, $0.80 mf.

For civil defense purposes it is necessary to know the influence of distant intense gamma sources such as those developing after nuclear explosions, for example. Since such cases involve distances which are a multiple of the mean free pathlengths of gamma radiation of 0.5-0.7 MeV, scattering effects in the atmosphere play a very important role.

This influence was considered for two limiting cases:

1. For a point gamma source with an energy of E_o = 550 keV. In the infinitely homogeneous medium of the atmosphere, the buildup factor at a distance of about 1.4 km -- corresponding to 15 mean free pathlengths -- amounts to $B(\mu_o r, E_o) = 155$.
2. For a semi-infinite medium of the atmosphere, filled homogeneously with a specific activity, a factor of 1.7 was calculated for the influence of scattering.

1348. A. Kania, R. Pasquet, P. Pepin, J. Rastoin, and M. Solanes, "Long Range Neutron Flux Transport in Fast Reactors," ORNL-tr-1554 (Transl. from Phys. of Fast and Intermediate Reactors, Vol. I, pp. 387-401) (1962). Availability: CFSTI, $0.65 mn; JCL, $1.60 fs, $0.80 mf.

Long-range neutron-flux propagation in fast reactors. Methods for calculation of long-range neutron-flux transmission in fast reactors are first described. Then an attempt is made to determine the accuracy of the results obtained, taking into account the fast that the cross sections are not known with certainty and that the calculation codes involve finite difference equations instead of differential analytic equations. The methods, when thus worked out in detail, are used to trace neutron flux attenuation in mockups representing the RAPSODIE reactor shielding, and the results obtained by calculation are compared with experimental data.

1349. Y. Segal and S. Zamir, "The Effect of Air Channels on the Diffusion Length of Thermal Neutrons in Light Water," Nucl. Sci. Eng. 27, 3, 564-567 (March 1967).

The effect of air channels on the diffusion length of thermal neutrons in light water was measured and calculated. The comparison between experimental results, Monte Carlo calculations, and values derived using Behrens and Benoist formulae show a good agreement for the longitudinal diffusion length. In the transverse case, Benoist's method agrees with Monte Carlo results better than Behrens' values.

1350. T. S. Bustard and J. Silverman, "Bremsstrahlung from 90-Sr 90-Y in Thin Absorbers," Nucl. Sci. Eng. 27, 3, 586-596 (March 1967).

The internal bremsstrahlung from ^{90}Sr-^{90}Y and the external bremsstrahlung yields in various target materials were measured, using a novel experimental approach. The bremsstrahlung targets employed consisted of materials from atomic numbers 13 to 73 and ranged in thickness from 4 mg/cm^2 past the range of the ^{90}Y beta particles. The experiment performed by means of pulse-height scintillator spectrometric techniques enabled the observation and semiempirical calculation of how the bremsstrahlung spectrum builds up and is simultaneously attenuated, as well as the corresponding changes in the beta spectrum transmitted by the absorbers. This approach allows determination of generated bremsstrahlung spectra without having to make large attenuation corrections to thick target data.

A brief description of internal bremsstrahlung and the theoretical aspects of external bremsstrahlung generation is given. Although there are several thick-target theories, only the Evans approximation to thick-target external bremsstrahlung is considered in detail. The Evans theory provides a good fit to the high energy portion of bremsstrahlung spectra and is therefore in prevalent use. Besides, it provides a clear example as to how yield constants or k values are determined.

The bremsstrahlung yield constants were found to differ, depending upon whether photon number or energy is considered. Further, a difference was found between the yield constants for ^{90}Sr-^{90}Y in secular equilibrium and ^{90}Y alone, indicating that an energy dependence also exists. The yield constants determined are 0.34×10^{-3} and 0.24×10^{-3} MeV^{-1} for ^{90}Sr-^{90}Y, and 0.50×10^{-3} and 0.41×10^{-3} MeV^{-1} for ^{90}Y, for bremsstrahlung energy and photons, respectively. This result indicates that the yield constant is energy dependent and also differs depending upon whether bremsstrahlung photon number or energy is being considered.

1351. R. L. Murray, W. J. Harrington, and C. Siewert, "Anisotropic Scattering Coefficients in the Constant Cross-Section Transport Equations," Nucl. Sci. Eng. 28, 1, 124-126 (April 1967).

The coefficients that enter the anisotropic scattering formulation of the constant cross-section transport equation are investigated. Exact expressions to order ten are presented as well as a useful, nested series for numerical calculations to any order.

1352. M. R. Mendelson, "The Effect of Anisotropic Scattering in Plane-Geometry Transport Theory and Monte Carlo Calculations," Nucl. Sci. Eng. 28, 1, 127-132 (April 1967).

The sensitivity of three thermal-energy model problems to anisotropic scattering was investigated by comparing double P_5 solutions with P_3 and P_1 scattering expansions. Results indicate that P_3 scattering effects can be significant in the calculation of absorption rates in certain sensitive plane-geometry configurations. Monte Carlo calculations were also performed for one of these problems, using two different anisotropic scattering representations: the transport approximation; and a "histogram" kernel, which match the first two and four Legendre moments of the scattering kernel, respectively. The transport approximation was found to give discrepancies of eight to nine percent in thermal absorption rates, and it is concluded that this scattering representation can lead to serious errors in Monte Carlo calculations.

1353. K. Tamura and A. Tsuruo, "Dose Buildup Factors of Plane Parallel Barriers for Cs-137 Plane Monodirectional Source," Nucl. Sci. Eng. 28, 1, 144-145 (April 1967).

The dose transmission and linear attenuation coefficients for eight materials and a 137-Cs source were determined experimentally. The eight materials were: graphite, water, ordinary glass, aluminum, ordinary concrete, magnetite concrete, iron, and lead. The results compared well with Monte Carlo calculations. (RSIC)

1354. J. O. Mingle, "Applications of the Invariant Imbedding Method to Monoenergetic Neutron Transport Theory in Slab Geometry," Nucl. Sci. Eng. 28, 2, 177-189 (May 1967).

The invariant imbedding theory for the monoenergetic neutron transport equation in slab geometry is determined in terms of reflection, transmission, and escape functions. The application of this theory to escape probabilities, blackness coefficients, and critical size determinations is shown along with representative numerical evaluations. The high accuracy of the calculations and the fact that the angular dependence is always obtainable make the invariant imbedding method a reasonable approach to many problems.

1355. J. C. Young, G. D. Trimble, J. M. Neill, G. K. Houghton, D. H. Houston, and J. R. Beyster, "Neutron Thermalization in Light Water," Nucl. Sci. Eng. 28, 2, 259-269 (May 1967).

Quasi-infinite medium neutron spectra have been measured as a function of moderator temperature and poison concentrations in water. Three different poisons were used: a 1/v absorber (boron) and the resonance absorbers erbium and gadolinium. These measurements were performed utilizing the General Atomic Linac and standard time-of-flight techniques. The temperature range covered was from ambient to about 300°C. The measurements were compared to the latest theoretical models using the best known calculational techniques available. The scattering kernel based on the modified Haywood frequency spectra and computed by the GASKET-FLANGE computer codes appears to give thebest agreement with the experimental data.

1356. G. I. Bell, G. E. Hansen, and H. A. Sandmeier, "Multitable Treatments of Anisotropic Scattering in SN Multigroup Transport Calculations," Nucl. Sci. Eng. 28, 3, 376-383 (June 1967).

Much theoretical work has been done in the past to represent tne 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 S_N codes. The S_N transport codes at LASL allow a representation of anisotropy in the scattering source term by means of multi-table cross-section sets and two formalisms are given here to generate these sets. Both involve the expansion of scattering cross sections in a series of Legendre polynomials, and incorporation of the expansion coefficients in the tables of transfer cross sections. One, called a consistent P approximation, involves a simple truncation of the series; while the other, called an extended transport approximation, includes an attempt to approximate the next higher term in the series. A general expression is derived for the error in the neutron flux due to either approximation. The numerical evaluation of S_N cross-section entries for these formalisms has been computerized.

Convergence with respect to Number of Tables is numerically investigated for several different neutron-transport problems: a) deep penetration of high-energy neutrons in air; b) critical size of an enriched-uranium bare sphere; c) reflector savings for an enriched-uranium sphere immersed in H_2O; and d) fast-reactor core mockup on ANL's ZPR-III.

It is concluded from these problems that both approximations converge rapidly with increasing number of tables and that the simple transport approximation gives quite accurate results for a wide range of problems.

1357. H. G. Kaper, "An Approximate Theory for the Slowing Down of Neutrons in a Nonmultiplying Medium," Nucl. Sci. Eng. 28, 3, 415-425 (June 1967).

An approximate theory for the slowing down of neutrons in a nonmultiplying medium with plane symmetry is described. The theory is based on an approximate form of the transfer function for elastic scattering under the hypothesis that the mass number of the moderator is larger than one. In the A^{-N} approximation the slowing down equation is reduced to a finite system of differential equations with respect to the lethargy variable. A detailed study has been made of the results obtained in the A^{-N} approximation with $N = 0$ and $N = 1$. Special attention has been paid to a comparison of the A^{-1} approximation with age theory and asymptotic theory.

1358. S. Kasi and H. Koskinen, "Analysis, Calculations and Measurements Concerning the Moisture Measuring by the Neutron Method," Nucl. Sci. Eng. 3, 1, 74-82 (1966).

This paper describes firstly a method of numerical analysis for the moisture measurement with neutrons, with special consideration of the effects of the dry density and the possible boron content of the material. Then a series of measurements performed by using an instrument including an additional iron reflector are rapported and the results analyzed. Finally some theoretical approaches of a more analytical nature are illustrated by applying simple physical models. Special attention is paid to the influence of a nonuniform distribution of moisture on the mathematical treatment.

Sections 2 and 3 are contributions of the first author, section 4 of the second author.

1359. W. C. Hall, "Lead Concrete - First Extra-High-Density Shielding Suitable for Installation by Mass Production Methods," Nucl. Sci. Eng. 3, 3, 476-477 (1966).

A lead additive concrete is described. A density of 10 gm/cm^3 is claimed. (RSIC).

1360. V. A. Dulin, V. G. Dvukhsherstnov, Yu. A. Kazanskii, and I. V. Shugar, "Neutron Angular and Energy Distribution at the Boundary of Two Media," Sov. J. Atomic Energy (English Transl.) 17, 1249-1255 (1964).

Measurements were made of the angular and energy distributions for fast neutrons (0.4-3.4 MeV) emerging from water, graphite, aluminum, iron, nickel, and lead as a water-medium interface after penetration of thicknesses equalling 1.5-4.6 mean free paths. The reaction $D(D,n)He^3$ was the neutron source. Measurements were made with a single-crystal fast neutron scintillation spectrometer having gamma-ray discrimination. The results of the measurements are compared with calculations.

1361. D. L. Broder, A. S. Zhilkin, and A. A. Kutuzov, "Spectra of Fast Neutrons in Heavy Media and Water," Sov. J. Atomic Energy (English Transl.) 18, 4, 519-523 (April 1965).

The behavior of the neutron spectrum near the interface between the heavy medium and water is of interest, since the presently used methods for calculating biological shields are least accurate precisely in the regions near the boundary as a result of abrupt changes in the spectrum and the angular distributions of neutrons.

In our experiments, the spectra of neutrons from monoenergetic sources with E_o = 3.4 MeV and E_o = 15 MeV which were moderated in iron, lead, or water behind iron and lead layers were measured by means of a scintillation spectrometer.

1362. A. G. Lisochkin and F. A. Predovskii, "The Weight-Effectiveness Index of Two-Component Materials Used for Shielding Against Neutrons and Gamma Rays," Sov. J. Atomic Energy (English Transl.) 18, 4, 524-525 (April 1965).

A value index is proposed for assigning figures of merit for two-component shields based upon an estimate of the shield weight. The index is based upon effects on primary radiation in a slab system. It does not take into account secondaries or system curvature.

1363. S. M. Ermakov, B. A. Efimenko, V. G. Zolotukhin, Yu. A. Kolevator, and V. I. Kukhtevich, "Spatial Energy Distribution and Dose Rate of Gamma Radiation from Unidirectional and Isotropic Co-60 Sources at the Ground-Air Interface," Sov. J. Atomic Energy (English Transl.) 18, 4, 534-537 (April 1965).

The present article provides the results obtained in measuring and calculating, by means of the Monte Carlo method, the spatial and energy distributions of scattered gamma radiation from a unidirectional Co-60 source (mean energy: 1.25 MeV) for distance R between the source and the detector equal to 15 and 30 m and the same elevation H(2, 10, 30, and 53 m) of the source and the detector above the ground. The measurements and calculations were performed for two orientation angles θ_0 (60 and 90°) of the unidirectional source.

1364. V. I. Golubev, A. V. Zvonarev, M. N. Nokolaev, M. Yu. Orlov, V. V. Peneenko, and O. P. Uznadze, "Propagation of Neutrons in Iron," Sov. J. Atomic Energy (English Transl.) 18, 5, 608-612 (May 1965).

Results of a theoretical and experimental study of the space-energy distribution of neutrons in the iron screen of the BR-I reactor are presented.

The experimental results are compared with multigroup calculations in the P_3-approximation (one-dimensional geometry) and the P_1-approximation (two-dimensional geometry). The effect of allowing for resonance blocking of the cross sections on the exactness of the description of neutron propagation in the medium is demonstrated.

1365. S. A. Kozlovskii, V. S. Kyz'yurov, K. K. Popkov, and D. N. Lebodev, "The Effect of Boron-Containing Blocking on the Yield of Capture Gamma Radiation," Sov. J. Atomic Energy (English Transl.) 18, 5, 672-675 (May 1965).

This letter investigates the effect of various boron-containing blockings on the yield of capture gamma radiation from the composition of the primary shielding of a water-moderated water-cooled reactor. The reactor vessel, thermal screens and the water layers between the screens were modeled by sheets of steel (Grade 3) and Plexiglas, respectively, of size 800 x 800 mm. The blocking layers were placed between the steel vessel and the water tank, modeling the water of the primary shielding. They were made of boron carbide (covering a density of 1.1 g/cm^3) and borated lead, i.e., a mixture of read lead and boric anhydride in the proportion 9:1 by weight. The mixture density was ~ 5.5 g/cm^3, containing about 3 wt% boron.

1366. G. V. Miroshnikov, "Attenuation by Iron and Polyethylene of Tissue Dose of Neutrons Incident Obliquely on the Shielding," Sov. J. Atomic Energy (English Transl) 18, 5, 681-684 (May 1965).

Neutron shielding is often made from polyethylene and iron, the shielding properties of which have been studied by many authors. To make accurate assessments of the values of biological shields, we need data on the spatial and energy distributions of neutrons in them, or of their attenuation of the tissue dose.

This question has been insufficiently discussed in the literature, especially for this layer (i.e., those with thickness less than three or four times the neutrons' free path). In assessing the shielding power in this case, it is essential to take account of the angle of incidence of the neutrons on the obstacle.

We investigated the shielding properties of thin layers of iron and polyethylene ($\leq$ 11 cm), using the protoneutron sources $Sb^{124}(\lambda) + Be^{9}$, $Na^{24}(\gamma) + D^{2}$, and $N^{24}(\gamma) + Be^{9}$ (with mean neutron energies 24 keV, 0.22 and 0.83 MeV), and also a cascade generator, used to obtain neutrons with E_n = 3 MeV by means of the reaction $D^{2}(d,n)He^{3}$.

1367. E. A. Kramer-Ageev, V. N. Markov, V. P. Mashkovich, V. K. Sakharov, and V. M. Sakharov, "Distribution of Neutrons in a Straight Cylindrical Channel," Sov. J. Atomic Energy (English Transl.) 19, 1, 911-913 (1965).

The passage of radiation through inhomogeneities in shielding, and in particular through slots and channels, is a quite important, though comparatively little-studied question in shielding from radiation. A number of papers have been devoted to this problem, among which the most fundamental and interested are for example.

In the present investigation we studied the energy and space distribution of neutrons in a straight cylindrical channel of diameter 14.4 cm and length 150 cm passing through a water shield. Source of neutrons was an isotropic Po-α-Be disk source which was simulated by an isotropic Po-α-Be point source emitting $2 \cdot 10^{7}$ neutrons/sec.

1368. G. A. Vasil'ev, A. P. Veselkin, Yu. A. Egorov, G. G. Moiseev, and Yu. V. Pankrat'ev, "Attenuation of Pile Radiations in Serpentinite Sand," Sov. J. Atomic Energy (English Transl.) 19, 4. 1283-1289 (April 1965).

Results of an experimental study of the attenuation of pile radiations in serpentine sand having a bulk density of 1.62 g/cm^3 are reported. The sand investigated contains about 11.5% chemically bound water liberated at temperatures upwards of 450°C.

The attenuation of fast flux and thermal flux, attenuation of neutron and gamma dose rate, and fast-neutron spectra in serpentinite sand were measured. Relaxation lengths of fast neutrons computed from experimental data are compared to relaxation lengths of fast neutrons in boron carbide, serpentinite concrete, and iron-ore concentrate.

1369. V. E. Drozdov, I. M. Zakharova, and S. P. Dobrovol'skii, "A Study of the Dose-Rate Field in an Irradiator with Gamma-Ray Source Consisting of Spent Reactor Fuel Elements," Sov. J. Atomic Energy (English Transl.) 19, 4, 1301-1306 (April 1965).

Experiments were performed to determine the distribution of activity along a fuel element from the RFT reactor. It was shown that the distribution of activity is satisfactorily described by a cosine function, and that the formulas used to compute the dose rate field of an irradiator with uniformly distributed activity along the radiation source are inapplicable to radiators consisting of fuel elements (the disagreement was 23% for the RFT and 32% for the First Atomic Power Station). Better agreement was obtained between experimental data and values computed by our formulas.

An experimental and theoretical study was made of the dose rate field from a single fuel element and from a hollow, cylindrical irradiator made up of eighteen RFT fuel elements. Working formulas were obtained for computing the dose rate field from spent fuel elements. In computing the dose rate field inside a hollow, cylindrical, irradiator made of fuel elements of finite diameter, it was shown that the activity can be considered to be distributed according to a cosine law over the surface of an infinitely thin cylinder with a radius equal to the arithmetic mean of the minimum and maximum radii of the irradiator.

1370. L. M. Shirkin, "Angular Distribution of the Intensity of Gamma Radiation Scattered by Lead and Water," Sov. J. Atomic Energy (English Transl) 19, 4. 1338-1339 (1965).

It was shown experimentally that the angular distribution curve of the intensity of scattered gamma radiation varies only weakly with the range $\mu_0 X$. The sources used by these authors had energies below 1.33 MeV, while the maximum thickness of the material investigated was eight times the free path length.

1371. I. V. Goryachev, "Angular Distribution of Neutron Dose Close to the Air-Ground Boundary," Sov. J. Atomic Energy (English Transl.) 19, 4, 1342-1343 (1965).

An empirical analytical fit is made to neutron angular distribution at an interface. (RSIC)

1372. V. A. Ionov, "Spectral Distribution in the Surface Atmosphere of Gamma Rays from a Point Source of Co-60 Shielded by Aluminum," Sov. J. Atomic Energy (English Transl.) 19, 4, 1344-1347 (April 1965).

The gamma-ray field in the surface layer of the atmosphere has been investigated by many people. The dose and angular distribution of intensity from a point source has been obtained. However, the spectral and spectral-angle functions have only been computed for extended sources; the corresponding functions for point radiation sources with an absorbing layer have not yet been obtained.

1373. G. G. Doroshenko, V. A. Federov, and E. S. Leonov, "Changes in Fast-Neutron Spectra After Penetrating Aluminum, Paraffin, and Water," Sov. J. Atomic Energy (English Transl.) 19, 5, 1434-1436 (1965).

The development of high-efficiency, fast-neutron spectrometers with time-amplitude selection of gamma-ray background, of automatic stabilization of amplification factor, as well as the development of a reliable matrix method for computing efficiencies taking into account all the factors which determine the line shape of a spectrometer, including the actual energy resolution of the detector, make it possible to carry out extended precision measurements of fast-neutron spectra at levels that are a fraction of the maximum permissible flux; in addition, it appears possible to investigate the fine structure of the spectra.

In the present work, an attempt was made to follow the changes in fast-neutron spectra with penetration through thick layers of aluminum, paraffin, and water. This is necessary if one is to explain the influence of the energy dependence of the cross sections of the basic materials under investigation on the shape of the fast-neutron spectra. A Po-Be fast-neutron source was used which was placed in a paraffin collimator with a 48° aperture angle in order to reduce the contribution from scattered neutrons. The material under study (aluminum, paraffin), in the form of sheets 70 x 70 cm in size, was located 25 cm from the source. The spectrometer detector was placed on the surface of the material directly above the source. Experimental results for aluminum (44 cm thick) and paraffin (45 cm thick) are shown in Figs. 1 and 2.

1374. A. P. Veselkin, Yu. A. Egorov, and V. A. Kycheryaev, "Shielding Properties of Iron-Serpentine Concrete," Sov. J. Atomic Energy (English Transl.) 20, 1, 101-103 (January 1966).

A study has been made of the shielding properties of concrete containing serpentinite aggregate with respect to reactor radiations. The advantage of such a concrete is that it is capable of maintaining 10-12 wt% of chemically bound water in the aggregate composition at temperatures up to $450^{\circ}C$, i.e., its shielding properties do not change even after prolonged effects of high temperature.

1375. V. B. Dubrovskii, M. Ya. Kulakovskii, M. S. Pinkahsik, N. V. Krasnoyarov, B. K. Pergamenshchik, V. I. Savitskii, J. Nucl. Energy A/B 20,1031-1038 (Sov. J. Atomic Energy (English Transl.) 19, 6 , 1498-1503, December 1965) , "The Use of Concrete for Shielding High-Temperature Reactors" Availability: Above Journal

The possibility is examined of constructing thermal shields of reactors from chromite and fireclay heat-resistant concretes. Neutron fluxes of different intensities (up to 10^{13} n/cm^2 sec, close in spectrum to the fission spectrum) absorbed by shields of these materials are examined. The passage of neutrons, γ-ray fluxes and generation of heat in the shield are taken into account. The temperatures in the shield are calculated for different neutron fluxes, thicknesses of concrete and cooling conditions. A static calculation of the temperature stresses in a shield constructed from heat-proof reinforced concrete is carried out.

It is established that heat-resistant reinforced concrete can be used for the shields of nuclear reactors for neutron fluxes to the concrete up to 10^{13} n/cm^2 sec, temperatures up to 1000-1100°C and temperature drops of up to 900° C.

1376. E. F. Plechaty, J. R. Terrall, Photon Cross Sections 1.0 keV to 15.0 MeV, UCRL-50178, Availability: Dep. mn; CFSTI

Photon cross sections from 1.0 keV to 15.0 MeV photon energy are presented for selected elements in tabular and graphical form together with a discussion of the method of evaluation.

1377. W. L. Bunch, Shielding Calculations for Irradiated FTR Fuel, BNWL-342;Availbility: Dep. mn; CFSTI

Shielding claculations have been made for irradiated FTR fuel to provide systems engineers with a basis for conceptual design of those facilities that are to contain this material. Gamma intensity as a function of shield thickness was obtained for several materials of interest, including lead, tungsten, depleted uranium, concretes of various density, water and air. Heat generation as a function of decay time was also calculated for evaluation of cooling requirements. A preliminary estimate of neutron shielding will only be required if beryllium is contained within the fuel.

1378. J. H. Thorngate, F. F. Haywood, J. A. Auzier, S. Self, Energy and Angular Distribution of Neutrons and Gamma Rays--Operation BREN, CEX-62.12 ; Availability: Dep. mn; CFSTI

In order to understand better the transmission and scattering of radiation in air, a series of measurements were made to determine the energy and angular distributions at large distances from the Oak Ridge National Laboratory Health Physics Research Reactor and an 800-curie ^{60}Co source. Both sources were positioned at various elevations on a 1527=ft tower while data were acquired by detectors located in collimators at either 750 or 1000 yd from the base of the tower. Good measurements were obtained of neutron and gamma-ray doses from the HPRR and gamma doses from the ^{60}Co source were also obtained. No useful spectrum information was obtained from the HPRR.

1379. L. G. Mooney, M. B. Wells, Effects of Neutron Energy Spectrum in Air on the Neutron-Induced Dose in Concrete Underground Structures, RRA-M65 ; Availability: Dep. ; CFSTI

A study was made to determine the effect of two different neutron energy and angular distributions in air on the neutron-induced dose in shielded underground concrete structures. The first neutron distribution used in the calculation corresponded to that expected from a fission neutron source at a distance of 700 meters in air. The second distribution was a much harder spectrum, resulting from the combination of a fission type source and a 14-MeV source.

Calculations of the capture gamma-ray and multicollision neutron doses were made at point receivers located three feet above the floor on the geometric centerline of subsurface cubical and parallelpiped concrete structures. The structures were covered with concrete slab shields varying in thickness from 4 to 60 inches.

Results of the calculation show relatively little difference in the total neutron-induced dose inside subsurface structures from the source distributions considered in this study. The higher contribution of the neutron dose from the harder source spectrum is offset by the larger capture gamma-ray dose resulting from the more abundant low energy neutrons from the fission-type source.

1380. J. J. Long, M. B. Wells, Comparison of Calculated and Measured Neutron Dose Rates Inside Steel Compartments Exposed to the Radiation Field of a Reactor, RRA-M66; Availability: Dep. CFSTI

Calculations have been performed to predict the neutron dose rate measured inside simple structures formed by 4x4 foot, three-quarter inch thick steel slabs. These structures were suspended from the Oak Ridge National Laboratory Tower Shielding Facility at a distance of one hundred feet from the Tower Shielding Reactor (TSR-II) in the COOL-I shield. The RRA-31 procedure, which has been developed for use in calculating the shielding efficiency of naval vessels against initial neutrons from weapons bursts, was used in the anlysis of neutron penetration through the steel enclosures. The RRA-31 calculations, when compared with the measured data for two different simple compartmented structures and for several single slab configurations, were found to be in good agreement with the measured data.

1381. J. D. Marshall, M. B. Wells, The Effect of Cutoff Energy on Monte Carlo Calculated Gamma-Ray Dose Rates in Air, RRA-M67; Avail.: Dep. CFSTI

Monte Carlo calculations have been made for the transport of gamma rays from a point isotropic source in an infinite air medium in which the particles are allowed to decrease in energy down to 0.01 MeV. These new calculations were prompted by discrepancies in dose build-up factors observed between earlier Monte Carlo calculations and those obtained from moments method data for infinite water. Inclusion of the low energy (0.01 - 0.10 MeV) component of the dose leads to good agreement between dose build-up factors for the two different calculational methods. The low energy comonent contributes as much as 36% of the total dose from a 0.5 MeV point isotropic gamma-ray source after an equilibrium spectrum is obtained. Good agreement was obtained not only in the dose buildup factors but also in the energy spectra. It was concluded that Monte Carlo calculations for air and moments method data for water are compatible (within ± 10 percent) when the gamma ray in the Monte Carlo calculations are allowed to scatter down to a low energy cutoff of 0.01 MeV.

1382. B. S. Sychev, M. M. Komochkov, V. V. Mal'kov, L. N. Zaitsev, High-Energy Neutron Propagation in Iron-Water Mixtures, ORNL-tr-1537, (Translated from JINR-P-2359) Availability: CFSTI

In the practice of designing the shields of high-energy accelerators a need often arises for the creation of a shield with a limited or minimum thickness. As a rule, this is achieved by using a heavy-aggregate concretes containing reinforcing fillers and scrap iron or steel. However, a considerable build-up of intermediate neutrons is possible in these materials which can be most simply calculated by the method of the age theory. To study the build-up process of intermediate neutrons, experimental investigations have been conducted of the attenuation of high-energy neutron fluxes in an iron-water medium.

1383. A. L. Pope, Neutron Cross Sections for Natural Zirconium in the Energy Range 0.0001 eV to 6 keV, AEEW-M-504 (1966) Availability: Dep. mn.

The currently available resolved resonance data for the isotopes of zirconium are evaluated, with the view to compile a file of basic tabulated neutron cross-section data. (Data file number 9 now available in UKAEA Nuclear Data Library).

These data are also compared with integrally measured data, and from the discrepancies it is concluded that:

(i) there is a need for better measurements of the radiation widths of the resonances for all zirconium isotopes, and especially for the s-wave resonances of Zr-91.

(ii) the measured values of the thermal-neutron capture cross sections of the isotopes are not consistent with the data for natural zirconium.

(iii) there are still considerable uncertainties in the resonance absorption integrals both for natural zirconium and the separated isotopes.

1384. C. K. Wiehle and W. L. Durbin, Combined Effects of Nuclear Weapons on NFSS Type Structures, URS-658-3 (September 1966). Availability: URS Corporation, 1811 Trousdale Drive, Burlingame, California.

The primary objective of this program was to examine and evaluate the protection afforded by the types of structures found in the National Fallout Shelter Survey (NFSS) against the combined effects of nuclear weapons. Since a major criterion for the designation of the NFSS structure was the radiation protection factor (PF) of the structure, the approach adopted in this investigation was to examine the sensitivity of the PF for idealized building situations to alteration by air blast and fire damage. To accomplish this, interim techniques were developed for estimation of the air blast loading and damage to selected types of structures. These techniques were developed by utilizing available experimental data together with engineering judgment to modify current generalized blast loading schemes. The procedure adopted to predict damage was to determine the blast loading on each building component and then to compare this loading with the failure loading for the component.

Similarly, by utilizing fire prediction information, a method was developed for determining the fire damage within the building. To predict damage to the various building elements by this procedure, the duration of the peak fire was calculated and then compared with the rated resistance of the components.

1385. W. E. Edwards, J. E. MacDonald, and C. S. Robertson, Jr., Nuclear Heating Calculations in a Fast-Spectrum, Refractory-Metal Reactor Assembly, GE-TM-66-6-7 (June 20, 1966). Availability: General Electric, Atomic Products Division, Cincinnati, Ohio.

A detailed nuclear heating analysis was made of a fast-spectrum, refractory-metal reactor with a moderating reflector as part of the 710 Reactor Program. Two-dimensional S_N transport theory and Monte Carlo methods were used. An unusual feature of this analysis was the breaking of the gamma source into four spatial zones and eleven energy groups. By computing the heating due to these sources separately and combining them at the end of the analysis sequence, it was possible to determine the origin of the nuclear heating in each component. The results clearly show the need for an accurate calculation of the neutron induced gamma sources in extra-core structural materials.

1386. C. McDonnell and J. Velletri, An Experimental Evaluation of Roof Reduction Factors, PSDC-TR-16 (May 1, 1966). Availability: Protective Structures Development Center, Fort Belvoir, Virginia.

A major portion of the radiation dose received within a structure is due to the fallout deposited on the roof. This report describes experiments conducted to measure the attenuation introduced by the roof to this radiation as a function of the mass thickness and the solid angle fraction subtended by the roof source to a detector in the structure.

The first series of experiments conducted on a roof of an inhomogeneous mass distribution, determined the attenuation introduced by mass thicknesses of 0, 48.6, 97.2, and 145.8 psf. Qualitative agreement between data from these experiments and calculated values made in an infinite homogeneous media is good. Measured dose rates from a rectangular source array of eccentricity 1.5 agreed within 10 percent with a circular source array subtending the same solid angle fractions.

A second series of experiments designed to closely simulate the analytical model used in the theoretical calculations was conducted for mass thicknesses of 0, 48.6, 97.2, and 145.8 psf. A maximum discrepancy of 15% is shown between theory and experiment for all mass thicknesses investigated.

1387. S. Sasse and C. D. Wuneke, Approximate Calculation of Buildup Factors of Gamma Radiation in Ordinary Concrete, ORNL-tr-1516 (transl. from ABS-THH-1008) (1966). Availability: DTIE.

Buildup factors are estimated by an interpolation scheme. (RSIC).

1388. S. Sasse and H. Schubart, Calculation of Secondary Gamma Radiation for a Mockup of a Power Reactor Shield for the Geesthact Research Reactor (FRG) with the Grace I Computer Program, ORNL-tr-1517 (transl. from ABS-THH-1010) (1966). Availability: Dep. mn, CFSTI, $3.00 cy, $0.65 mn; JCL, $3.60 fs, $1.37 mf.

The present study deals with the calculation of the secondary gamma rlux, i.e., that induced by neutron capture, in a special material configuration with the aid of the GRACE I computer program.

1389. S. Sasse, Weight Optimization for Radiation Shields of Large Nuclear Reactors, ORNL-tr-1525 (transl. from ABS-THH-1023) (1966). Availability: Dep. mn, CFSTI, $3.00 cy, $0.65 mn; JCL, $8.60 fs, $2.96 mf.

A scheme is proposed for optimal placement of bad shielding in a borated polyethylene shield. Dose from primary and secondary gamma rays and from neutrons is taken into account. A minimum weight shield is sought. (RSIC)

1390. R. H. Clark, XMA-1 Shielding Tests Using Slabs of Advanced Shielding Materials, APEX-502 (July 1958). Availability: CFSTI, $3.00 cy, $0.65 mn.

Results of a series of tests that were conducted in the General Electric Outside Tank Facility during December 1957 and January 1958 are discussed. Emphasis is placed on the studies of the production of secondary gamma rays in Hevimet while in a beryllium medium, photoneutron production in beryllium, and the effect of hydrogenous backing of nonhydrogenous materials. Some comparisons with calculating values are made.

1391. B. W. Vanzant and R. C. DeHart, Penetration of Nuclear Radiation into Shafts, Tunnels, and Entranceways, Army-SWRI Project No. 03-1238 - Final Report (June 1966). Availability: Southwest Research Institute, San Antonio, Texas.

It is important to designers of shelters to be able to determine the amount of radiation from a nuclear weapon detonation that penetrates shafts, tunnels and entranceways of the shelter. In this report, existing procedures available to designers are discussed, and a comparison is made between calculated results with those determined experimentally. This comparison clearly indicates that the agreement between experiment and calculation is not as close as is desirable. It is noted that consistent and complete laboratory experiments have not been undertaken for the purpose of determining transmission characteristics of radiation (gamma and neutron) from a nuclear weapon into tunnels with standardized cross sections, bends and intersections. Such a program would require the use of gamma and neutron radiation sources of appropriate energy levels placed at varying distances and angles from the tunnel opening. Most of the available empirical data were not directly applicable to the problem under consideration. When an appropriate source configuration was used, the tunnel configuration at the bend or intersection was not a practical one. When the tunnel configuration was appropriate, the source was clouded with background shielding, etc. In most cases, an empirical relation can be obtained for the measured data and a reasonable theoretical explanation can be given; but there is little agreement between the various sets of data. Even in full-scale tests, there was little agreement between two types of measuring devices at the same point. In short, the existing data and procedures are lacking in agreement to such an extent that it appears at this time that the shelter designer cannot with any degree of certainty calculate the amount of nuclear radiation penetrating shafts, tunnels, and entranceways.

1392. D. J. Dudziak, Capture and Fission Gamma-Ray Effective Spectra for Shielding, LA-DC-8253 (1966). Availability: Los Alamos Scientific Laboratory, University of California, Los Alamos, New Mexico.

Effective two-group gamma-ray spectra have been determined for thermal neutron capture in sodium, nickel, type 304 stainless steel, and tantalum, as well as for ^{235}U prompt fission gamma rays. A seven-group compilation of capture gamma rays was used as the basis for this study. Absorbed dose (uncollided and built-up) in several materials was calculated for varying thicknesses of several intervening shielding materials. The resulting function for each combination was reduced to two exponential functions over a range of 0 up to 560 g/cm^2. These effective spectra reproduce, to within an average absolute deviation of less than 7.4 percent, the absorbed doses (uncollided and built-up) calculated by the detailed spectra, within the ranges of areal density considered.

1393. B. W. Shumway, Experimental and Calculated Estimates of the Shielding Effectiveness of Compartmented Structures Exposed to Fallout, USNRDL-TR-1045 (AD-645552) (July 1966). Availability: CFSTI, $3.00 (HC), $0.65 (MF).

Exposure reduction factors were measured inside six compartmented steel structures having different wall thicknesses ranging from 1/4 to 1-1/2 in. These were exposed to radiation from fallout of varying age from three to nine days. Calculations based upon the Nelms-Cooper gamma-ray spectrum at H+1.12 hours were made for selected compartments in each of the structures following procedures given in the Office of Civil Defense Professional Manual, PM-100-1. Comparison of experiment and calculation reveals a sensitivity to spectral changes and shows that protection is greater during the periods D+3 to D+9 days than at H+1.12 hours. Overall agreement is generally satisfactory. The calculational methods for radiation through floors, however, appear to be inadequate.

Spectra measured on site at D+3 and D+9 days are given.

1394. M. C. Bertin, J. Celnik, H. Flesher, G. Grochowski, M. H. Kalos, J. H. Ray, and E. S. Troubetzkoy, Neutron Cross Sections of 238-U, 235-U, 237-U, 239-U, 234-U, 236-U, 239-Pu, 240-Pu, W, Pb, Ni, Cr, C, 6-Li, 7-Li, and T, UNC-5099 (AD-616629) (Dec. 31, 1964). Availability: CFSTI, $3.00 (HC), $0.65 (MF).

Neutron cross-section sets have been prepared for 16 elements or isotopes for neutron energies from 0.037 eV to 18 MeV. The cross sections tabulated include the total, elastic, inelastic, (n,2n), and fission cross sections, as well as cross sections for charged particle emission. Information is also given on the angular distribution of elastically scattered neutrons and on the energy distribution of neutrons and gamma rays following nonelastic scattering.

1395. I. C. Roberts, Measurement and Analysis of Nuclear Radiation Heating in Selected Shielding Materials, AFWL-TR-66-120 (January 1967). Availability: Air Force Weapons Laboratory, Kirtland Air Force Base, New Mexico, ATTN: WLDN.

An experiment was performed to measure the heating rates in slabs of zirconium hydride, beryllium oxide, Inconel-X, and 6 percent borated polyethylene. Measurements were also obtained with the zirconium hydride shielded by boral and by boral plus lead and with the beryllium oxide shielded by lead. The measured data are compared to calculations of the heating rates obtained by SHADRAC, a procedure that uses moments-method data. The neutron flux distributions in the test configurations were obtained by means of DTK, an S_N radiation transport code. These neutron distributions were used to hand calculations of secondary-gamma-source terms used as input in SHADRAC to compute the heating rates from the capture and scatter of neutrons. The measured and calculated heating rates differ in magnitude by less than 25 percent of all detector positions, and the curves of the heating rates as a function of material thickness show good agreement in shape. The heating from the primary gamma radiation is the most important component of the total heating rate calculated for all configurations.

1396. G. V. Miroshnikov, "Attenuation of Neutron Tissue Dose by Thin Layers of Hydrogenous Materials," Sov. J. Atomic Energy (English Transl.) 18, 5, 685-688 (May 1965).

Measurements were performed in barrier geometry. The materials investigated (slabs dimensioned 1000 by 1000 by 10 mm) were placed at distances of 2 meters from the neutron sources. Close-to-normal neutron drops took place across the barrier. The dosimeters were placed on the far side of the slabs and flush up to them. The following hydrogenous materials were investigated: water, polyethylene, plexiglas, polychlorovinyl laminates, epoxy-base glass-fiber reinforced plastic (table 2). The statistical error in the attenuation measurements remained with approximately 3%.

1397. K. D. Lathrop and N. S. DeMuth, Biorthogonal Angular Polynomial Expansions of the Boltzmann Transport Equation, LA-3617 (Oceober 1, 1966). Availability: CFSTI, $3.00 (hc), $0.65 (mf).

A new system of biorthogonal polynomials is developed for the angular expansion of the directional flux in the linear Boltzmann transport equation. Recursion relations and an addition theorem are derived for a system of biorthogonal polynomials shown by Didon (1868) to be orthogonal over the unit hemisphere. These polynomials are applied to two dimensional forms of the Boltzmann equation. Equations for the coefficients of expansion are derived when the directional flux is expanded in a series of either system of polynomials. One of these systems of equations is shown to be a linear combination (with specific coefficients) of the equations obtained when the directional flux is expanded in spherical harmonic functions, and it is shown that this same system reduces to the spherical harmonics equations in one-dimensional plane geometry.

1398. R. Vinkeloe, "Concrete as a Protection Against Radiation," ORNL-tr-1614 (Transl. from Betonstein Zt. 30, 12, 642-646, 1964) (1966). Availability: CFSTI.

Concrete is eminently suited for the shielding of radioactive rays of all kinds. It can adequately absorbe gamma rays as well as neutron rays. By the choice of suitable aggregates the bulk density of concrete can be varied within wide limits; in addition, its hydrogen content can be varied. Its absorptive capacity for thermal neutrons can be considerably increased by the admixture of small quantities of boron compounds.

Concrete is given its shape and form on the construction site. It can be economically adapted to practically any desired and required constructional shape. Besides radiation-shielding duties it can also perform load-bearing and space-enclosing functions.

The present article deals with the general conditions for radiation shielding, with special reference to considerations of concrete technology, and the efficient composition of the protective concrete.

1399. Y. Kanemori, "Measurements of Dose Buildup Factors in Slab Materials for an Isotropic Co-60 Source," ORNL-tr-1649 (Transl. from Nippin Genshiyroku Gakkaishi 8, 6, 304-08, 1966) (1966). Availability: Dep. CFSTI; JCL, $1.60 (fs), $0.80 (mf).

Gamma-ray transmission dose rates were measured for the case where the line connecting a 60-Co point isotropic source and the detector center lies obliquely against the absorber plane, to derive the relation between dose buildup factors and the oblique angles between the connecting lines and the absorber plane. The slab materials taken up were aluminum, iron, and lead.

In addition, transmission dose rates for a line isotropic source were obtained by numerical interaction of the measured dose rates for point sources, and relations similar to the case of point source were derived between dose buildup factors and oblique angles. Dose buildup factors for a plane isotropic source were also determined.

1400. T. Huodo, "Backscattering of Gamma Rays," ORNL-tr-1670 (Transl. from Nippon Genshiryoku 8, 7, 371-81, July 1966) (1966). Availability: Dep. mn, CFSTI; JCL, $2.60 fs, $0.92 mf.

A general survey of theoretical and experimental studies on the backscattering of gamma rays is presented. Typical data of albedos and the reflection buildup factor, dose of scattered gamma rays are shown in this article.

1401. A. Honig, "Equivalent Thicknesses of Materials for the Attenuation of Gamma Radiation," ORNL-tr-1666 (Transl. from Jaderna Energie 12, 296-98, 1966) (1966). Availability: Dep. CFSTI, JCL, $1.10 fs, $0.80 mf.

The article contains practical tables of thicknesses for various materials -- heavy and ordinary concrete, aluminum, water, and steel -- for the attenuation of gamma radiation equivalent to lead thicknesses, in relation to the photon energy.

1402. A. Mockel, "Invariant Imbedding and Polyenergetic Neutron Transport Theory -- Part I, Theory," Nucl. Sci. Eng. 29, 1, 43-50 (1967).

Invariant imbedding method is applied to compute the energy-dependent, time-dependent transmission and reflection of neutrons in slab geometry. A set of "characteristic functions" (which are a generalization of Chandrasekhar's X- and Y-functions) can be defined when the scattering kernel is degerate. A computational scheme is outlined which makes use of these functions to obtain the time decay constant, the time-asymptotic leakage flux, and, for a steady-state case, the thermal utilization factor for a heterogeneous system.

1403. C. D. Zerby, "Density Transformations for Time-Dependent Radiation Transport Problems," Nucl. Sci. Eng. 29, 1, 151-153 (July 1967).

The purpose of this note is to rigorously derive the required density transformations for a step function source and a delta function source. The transformations apply to an infinite homogeneous medium where the data required for solving quite general problems with time dependence can be reduced drastically by tabulating a series of selected cases and then compounding the results and transforming them to solve general problems. (RSIC)

1404. EACRP, Neutron Spectra, EACRP-L-62 (1966). Availability: European Nuclear Energy Agency, O.E.C.D. European-American Committee on Reactor Physics, Paris, France.

Graphical and tabular data giving thermal-neutron spectra for a large number of materials has been compiled by the European-American Committee on Reactor Physics. Both experimental and theoretical results are given. (RSIC)

1405. A. J. Budka, W. J. Brand, and T. J. Dolce, Secondary Gamma Ray Transmission Through Nevada Test Site Soil, BRL-R-1329 (July 1966). Availability: Ballistic Research Laboratories, Aberdeen Proving Grounds, Maryland.

Secondary gamma ray transmission data are presented for monoenergetic secondary gamma ray sources in Nevada Test Site Soil slabs of various thicknesses. The data presented have been computed using the Monte Carlo method.

These secondary gamma ray transmission data along with an appropriate weighting scheme allow one to determine the secondary gamma ray transmission information for any depth distribution of secondary gamma rays in Nevada Test Site soil.

1406. K. W. Tompkins, Analysis of Multilayer Shield Experiments III and IV, NARF-60-7T (MR-N-255) (Sept. 16, 1960). Availability: Convair, Fort Worth, Texas.

The Ground Test Reactor has been employed in the Outside Test Tank to expose slab assemblies of shielding materials to radiation. These assemblies were designed to mock up the sides, front, and rear of the reactor shield system employed by the General Electric Company in ANP research and to yield basic data.

For this series of tests, a two-inch-thick lead slab was placed between the reactor and the shield-material slabs to reduce the primary gamma rays to an extent such that small changes in the secondary gamma rays could be detected. Calculations of dose rates have been made for comparison with measurements. These calculations include both primary and secondary dose-rate components. Secondary gamma rays resulting from thermal-neutron captures and those resulting from the inelastic scattering of fast neutrons in the heavy gamma shielding materials have been found to be a significant part of the measured gamma dose rates. Photoneutrons resulting from (γ,n) reactions in beryllium have been found to contribute significantly to the measured fast-neutron dose rates for those assemblies containing large quantities of this material.

1407. E. Laumets and C. A. Young, The NRDL Manual Gamma-Ray Ship-Shielding Calculational Method. Appendix I and II, USNRDL-Technical Manual No. 67-1 (AD-813-474L) (Dec. 30, 1966). Availability: U.S. Naval Radiological Defense Laboratory, San Francisco, California 94135 ATTN: STINFO.

A graphical method for hand calculation of ship shielding factors is developed and presented. The method utilizes a piece-by-piece accounting for each slab of shielding material and a time-dependent gamma-ray pseudo-spectrum consisting of five energies. Sets of graphs for deck-deposit and transit radiation shielding calculations are presented as two appendixes printed under separate covers. Results of the calculations are usually within a factor of two of correct values when simplifying approximations are used, and within ±50% when very careful, detailed calculations are made.

1408. W. R. Kimel, R. E. Faw, J. A. Baran, J. O. Mingle, R. M. Rubin, R. C. Iotti, and W. T. Urban, Scattering of Fallout Radiation from Ceilings of Protective Structures - Final Report, Kansas State Univ. Bulletin, Spec. Report No. 72 (July 1966). Availability: Office of Civil Defense, Office of the Secretary of the Army, Washington, D.C.

A thorough study of the ceiling shine problem has been made. Ceiling shine is defined to be that contribution to the radiation dose rate in a protective structure resulting from fallout radiation entering the structure through apertures in vertical walls and scattering downward from the ceiling.

A systematic analytical analysis of this problem, verified experimentally, resulted in a set of design curves. These curves yield the ceiling shine reduction factor in any concrete structure as a function of building dimensions, size and location of apertures, and detector location.

1409. C. W. Watson, Monte Carlo Calculation of Gamma-Ray Heating Rates and Fluxes in Phoebus 1 from Core Sources, LA-3577 (Dec. 6, 1966). Availability: $0.90, USAEC TISE, P.O. Box 1001, Oak Ridge, Tennessee.

Monte Carlo estimates have been made of the gamma-ray heating rates and fluxes in and around a Phoebus 1 reactor from gamma rays originating in the reactor core. The basic input data, as well as the results, are presented on considerable detail.

1410. F. H. Welch, E. C. Phillips, J. G. Asquith, and N. F. Davies, Lithium Hydride Technology I. Properties of Lithium Hydride and Corrosion Studies, NAA-SR-9400 (May 8, 1964). Availability: $2.45 USAEC-TIE, P. O. Box 1001, Oak Ridge, Tennessee 37830

The physico-chemcal, mechanical, and nuclear properties, and corrosion effects of lithium hydride are described in a three-part report for SNAP System Applications.

The first section presents new data on the physical, mechanical, and chemical properties of various forms of lithium hydride, and also confirms previously reported data from other sources.

The second section describes the results of 1000-, 2000-, and 4000-hr exposures of selected stainless steels to both solid and liquid lithium hydride. No deleterious corrosion effects were noted on the materials examined.

The third section reviews the effects of nuclear reactor radiation exposure to lithium hydride and lithium (-7 isotope) hydride based on a new radiation effects theory. No significant radiation damage effects are expected during the current SNAP reactor shield lifetime.

1411. W. H. McMaster, N. K. DelGrande, J. H. Mallett, N. E. Scofield, R. Cahill, and J. H. Hubbell, Compilation of X-Ray Cross Sections, UCRL-50174, Section II (January 1967). Availability: Lawrence Radiation Laboratory, University of California, Livermore, Calif.

This volume contains preliminary graphs and tables of x-ray cross sections in the range of 1 keV to 1 MeV. The total cross sections presented are the sum of computed scattering cross sections and the fit to derived experimental photoelectric cross sections.

Form factors available in the literature were used to compute the scattering cross sections. Where solutions to the Hartree-Foch equations were unavailable, Bewilogua's values for the Thomas-Fermi model were used.

To derive the experimental photoelectric cross sections, the scattering cross sections were subtracted from the total attenuation data obtained by several experimenters. The derived values were then fit by an appropriate polynomial on a log-log scale with the experimental data weighted according to the accuracy of the experiment.

The fitted photoelectric cross sections were compared by hand to the theoretical photoelectric cross sections by Rakavy and Ron, and of Schmickley. The agreement was good.

L and M shell jump ratios were calculated from the theoretical values of Rakavy and Ron.

1412. J. J. Schmidt, Neutron Cross Sections for Fast Reactor Materials Part I: Evaluation, KZK-120, Part I (February 1966). Availability: Kernforschungszentrum Karlsruhe, Institute fur Neutronenphysik und Reaktortechnik, Gesellschaft fur Kernforschung M.B.H. Karlsruhe, Germany.

This volume is Part I of the tripartite report entitled

"Neutron Cross Sections for Fast Reactor Materials"

Part I. Evaluation

Part II. Tables

Part III. Graphs

published in one and the same edition as an external report of the Kernforschungszentrum Karlsruhe and as an EANDC report under the reference numbers KFK 120 (EANDC (E)-35"U"). It presents a fairly comprehensive documentation and a critical review of the available experimental and theoretical information on neutron microscopic cross section data and an extensive description of the procedures which were used to elaborate this information into the cross section sets displayed in tabular and graphical form in KFK 120/parts II and III. The present work is restricted to those materials which are of primary interest in the design of sodium, steam or He cooled, metal, oxide or carbide fuelled fast reactors with steel, Inconel or Mo canning, i.e.,

H, He, C, O, Na, Cr, Fe, Ni, Mo, U^{235}, U^{238}, and Pu^{239}

As these cross section data are primarily intended for use in reactor calculations, the neutron energies considered are restricted to the range 0.01 eV to 10 MeV, these being very nearly the lower and upper energy limits of reactor neutrons. In the small fast reactors and critical systems neutrons concentrate on keV and MeV energies. Large dilute power reactor systems contain neutrons down to the eV region. Energies down to thermal were considered to render this evaluation useful also for thermal and epithermal reactor calculations. Main emphasis, however, is given to resonance and fast neutron cross sections.

1413. V. P. Mashkovich and V. A. Klimanov, "Distribution of Gamma-Ray Intensity Inside Straight, Cylindrical Ducts," Sov. J. Atomic Energy (English Transl.) 20, 2, 152-157 (February 1966).

A study was made of gamma-ray intensity distributions from disc, isotropic, and monodirectional sources with energies of 0.412-2.75 MeV along the axis of straight, cylindrical ducts passing through shielding. Water, iron, and concrete were used as shielding materials. Experimental results and calculations agreed within 15%.

The gamma-ray intensity distribution inside and outside water shielding through which passed a straight, cylindrical duct was measured for disc, isotropic sources. The experimental data agreed with computed results from the "ray analysis" method within 20%.

1414. D. A. Kozhevnikov, "Approximate Similarity of Neutron Fields Formed by Sources with Different Spectra," Sov. J. Atomic Energy (English Transl.) 20. 2. ;08-199 (February 1966).

A transformation is described by which the neutron attenuation kernels of sources of differing energies are made to coincide. The ratio of age for two spectra, called the age modulus, is used to link the two kernels. (RSIC)

1415. E. A. Kramer-Ageev and V. S. Troshin, "The Angular Distribution of Doses of Neutrons Scattered by Shields," Sov. J. Atomic Energy (English Transl.) 20, 2, 200-201 (February 1966).

Angular distributions of doses of neutrons were measured behind shields made of water (5-15 cm thick), concrete (10-60 cm), and graphite (20 cm). The neutron source was a (γ,n) reaction on the uranium target of a 30-MeV linear electron accelerator. As is known, the spectra of neutrons from the (γ,n) reaction differ little from the fission spectrum.

1416. L. M. Shirkin, "The Angular Distribution of Energy and Dose Scattered Neutrons in Water," Sov. J. Atomic Energy (English Transl.) 20, 2, 202-204 (February 1966).

In the present study the Monte Carlo method was used for calculating the angular distribution of energy and dose of neutrons scattered in water. We investigated a plane unidirectional source with energies of 3.3 MeV and 8.0 MeV in barrier geometry. The maximum thickness of the water barrier was ten mean free paths. In order to improve the statistical accuracy of the calculation, we combined the Monte Carlo method with analytic methods.

1417. V. S. Galishev, "On the Theory of the Passage of Gamma-Quanta Through a Slab of Finite Thickness," Sov. J. Atomic Energy (English Transl.) 20, 3, 297 (March 1966).

In previous work the problem of the passage of gamma quanta through a slab of finite thickness was reduced to a set of two first-order integro-differential equations for the photon flux. In the present work we firstly converted this set of equations into a single second-order integro-differential equation, which was then replaced by a finite set of differential equations. The latter were then solved by the Green function technique.

This method was applied to the study of the passage of 10 MeV gamma quanta through an iron slab of thickness $u_o a = 0.5$ ($u_o = 0.238$ cm^{-1}, is the linear attenuation factor of iron for the given energy; a is the slab thickness in cm). The calculations were carried out on the Ural-1 digital computer.

The table shows the resulting calculated values of the spectral distribution functions $4\pi\lambda N_o(0,\lambda)$ and $2\pi\lambda N_o^+(a,\lambda)$ (multiplied by $(u_o a)$) for gamma quanta of wavelength λ (in Compton units) reflected and transmitted by the slab. The spectral distribution functions were employed to calculate the energy storage factor of the radiation transmitted through the slab. It worked out to be 1.06, which is in good agreement with the value given by Berger and Doggett.

1418. V. P. Zharkov and E. A. Panov, "Flow of Radiation for a Cylindrical Channel and a Plane Slit in a Shield," Sov. J. Atomic Energy (English Transl.) 20, 4, 393-394 (April 1966).

This paper discusses the problem of flow of radiation for a cylindrical channel and a plane slit in a shield, which is partly defined by a source located within the input cross section of the channel or slit. In deriving the formulas, the attenuation of the radiation in the shield was assumed to be exponential. It was also assumed that $L \gg r$ and $L \gg T/2$, where L is the thickness of the shield, r is the radius of the cylindrical channel, and T is the height of the slit.

1419. A. M. Panchenko, "Energy Buildup Factors for Cosine Sources in Iron and Lead," Sov. J. Atomic Energy (English Transl.) 20, 4, 405-406 (April 1966).

The energy buildup factors are given for iron and lead for point, disk and infinite plane gamma-quantum sources with angular radiation distribution following a cos theta or cos-square-theta law. The buildup factors were calculated by integrating the distribution function of the scattered-radiation intensity for a point unidirectional source, found by experiment.

1420. D. B. Pozdneev and S. A. Churin, "Backscattering of Low-Energy Gamma Rays from Shields of Finite Thickness," Sov. J. Atomic Energy (English Transl.) 20, 4, 407-409 (April 1966).

This work is concerned with a study of the dependence of differential albedo for normal incidence on the thickness of various scatterers for the case of point isotropic sources with primary gamma-ray energies in the range 0.145-0.765 MeV.

1421. Yu. A. Kazanskii, V. I. Kukhtevich, V. I. Popov, V. V. Tarasov, and B. P. Shemetenko, "Dependence of Buildup Factor on Detector Position Outside Shielding," Sov. J. Atomic Energy (English Transl.) 20, 5, 481-482 (May 1966).

The authors measured buildup factors with a radioactive source of Cs^{137} gamma rays for various detector positions and with the source behind an aluminum shield 2.8 mean free paths thick and 40 cm in diameter. The measurements were made with a scintillation detector (stilbene crystal). The distance L from the source to the surface of the shield facing the detector varied from 18 to 150 cm; for each value of L, the distance R from the surface of the shield to the detector varied from 0 to 500 cm.

1422. V. B. Dubrovskii, A. K. Shreiber, A. F. Mirenkov, and V. N. Solov-ev, "Gamma-Ray Shielding of Artificial Stone," Sov. J. Atomic Energy (English Transl.) 20, 5, 484-485 (May 1966).

Artificial stone is proposed as shielding for gamma radiation from various sources (secondary loops of atomic power stations, gamma irradiation equipment, "hot" cells, electron accelerators).

Artificial stone possesses several technical and economic advantages when compared with ordinary concrete and is finding an ever-increasing application in construction. Artificial stone is obtained by incorporating stone (ore) into a cement mixture base.

To study shielding properties, blocks were prepared which were made of concrete (densities 2250, 3300, and 4600 kg/m^3) and artificial stone containing limestone and hematite ore as the coarse aggregate. The density of the artificial stone was 2320, 3770, and 4600 kg/m^3.

The experimental studies were carried out with gamma equipment having a Co^{60} source with an activity of 500 gm-equiv Ra.

The gamma-ray dose rate behind concrete and artificial stone shields of varying thicknesses was recorded by SBM-10 gamma dosimeters (see figure). The results obtained were compared with the results of theoretical calculations for which the linear absorption coefficient μ was computed from the chemical composition of the material and the dose buildup factor B by determining the effective atomic number of the material Z_{eff} in accordance with Goldstein's method.

The excellent agreement of calculated and experimental data, and the similarity of the dose rate distribution curves behind concrete and artificial stone shields, is evidence of the high uniformity of the shielding properties of the blocks. Therefore, it is possible to consider artificial stone as a uniform, homogeneous mixture of chemical elements for which the gamma-ray shielding parameters can be computed just as for concrete (from chemical composition). The coefficients μ and B for the experimental concrete and artificial stone compositions were calculated over an energy range from 0.1 to 100 MeV, and they can be used in practical shielding calculations.

1423. V. B. Dubrovskii, Yu. S. Ryabukhin, A. F. Mirenkov, and V. N. Solov'ev, "Gamma-Ray Penetration Through Joints of Builtup Concrete Shields," Sov. J. Atomic Energy (English Transl.) 20. 5. 496-487 (May 1966).

Experimental studies of builtup and monolithic shields were carried out with a cobalt source having an activity of 500 gm-equ. Ra in two geometries (collimated and isotropic source). The small SBM-10 counter with improved characteristics, which has practically no energy dependence in the 0.1 to 1.25 MeV energy range, was used as a radiation detector. Joint thickness (2-3 cm) was considerably greater than the counter diameter (0.6 cm); thus the detector was considered as point isotropic.

As a result of the work that was performed, it was established that the use of builtup shielding does not lead to significant local increase in dose rate behind shielding for practically important situations. The penetration of gamma rays through joints is described by the formula given with satisfactory accuracy. In cases where the value of the radiation penetration coefficient turns out to be fairly large, one can use the proposed methods.

1424. I. V. Goryachev, V. A. Dulin, S. M. Ermakov, V. V. Koluzhenkov, A. P. Suvorov, and L. A. Trykov, "Angular Distribution of Fast Neutrons Outside Iron Shielding," Sov. J. Atomic Energy (English Transl.) 20, 6, 541-545 (June 1966).

Angular energy distributions of fast neutrons outside iron shielding were measured. Experimental results were compared with Monte Carlo calculations and with multigroup calculation by the "transmission matrix" method in the $2P_7$ approximation. The satisfactory agreement of experiment with calculation, both in absolute magnitude and in the shape of the angular spectra, is noted.

1425. O. A. Barsukov, V. S. Avzyanov, and V. N. Ivanov, "Study of the Spectra and Doses Created in the Iron-Water Shielding of a Monoenergetic Neutron Source," Sov. J. Atomic Energy (English Transl.) 21, 1, 643-651 (July 1966).

Many-group calculations were made for the penetration of neutrons, emitted from monoenergetic sources, through water, iron, and water-iron systems of finite dimensions; the results of these calculations are presented. The neutron spectra resulting from the passage of such neutrons through water and iron shielding layers were calculated on the twenty-group diffusion-transport approximation. Detailed attention was paid to the high-energy part of the spectrum, certain peculiarities in neutron migration and moderation processes in shielding of the type in question were elucidated. Dose curves D(t) were plotted for neutrons of various energies.

By using the superposition principle, the results enable the neutron spectrum to be determined for sources having any arbitrary spectrum.

1426. V. B. Dubrovskii, Sh. Sh. Ibragimov, A. Ya. Ladygin, and B. K. Pergamenshchik, "Effect of Neutron Irradiation on Some Properties of Heat-Resistant Concretes," Sov. J. Atomic Energy (English Transl.) 21, 2, 740-744 (August 1966).

In contrast to the structural materials of nuclear reactors, the radiation resistances of concretes used in biollogical shielding have not been sufficiently studied. A tendency has recently arisen for the preferential use of heat-resistant concretes in biological shielding instead of materials such as steel, cast iron, graphite, boron, etc., which are costly and in relatively short supply. In this paper we shall indicate the effect of reactor neutron irradiation on certain properties of Portland-cement and liquid-glass heat-resistant chromite concretes. The integral neutron flux used in this investigation was $(2\text{-}2.4) \times 10^{21}$ neutrons/cm^2 and the irradiation temperature up to 550°C.

It was found experimentally that these concretes retain quite high strength and elastic properties. The thermal conductivity and thermal expansion coefficient change very little. It is concluded that such concretes may be recommended for use in the biological shielding of nuclear reactors.

1427. D. C. Irving, R. G. Alsmiller, Jr., H. S. Moran,"Tissue Current-to-Dose Conversion Factors for Neutrons with Energies from 0.5 to 60 MeV", Nucl. Instrum. Methods 51, 1, 129-135 (ORNL-4032 and ORNL-TM-1522)

To assist in the evaluation of the hazard associated with exposure to high-energy neutrons, a Monte Carlo computer program was used to calculate the energy disposition as a function of depth in a 30-cm thick infinite slab of tissue resulting from neutrons incident on the slab at energies up to 60 MeV. The program treated nonelastic and elastic interactions, including evaporation processes and nuclear recoils. Cases of both normal and isotropic incidence were calculated for neutrons of 0.5,2,10,18, 30 and 60 MeV. From these data, current-to-dose conversion factors were extracted for the average whole-body dose, the dose at a 5-cm depth, and the maximum dose. A set of quality factors (QF's) was adopted for transforming rad dose to rem dose.

1428. M. B. Vasil'ev, "Scattering of Gamma Radiation from Rs-226 and Cs-137 by a Surface of Earth or Water", Sov. J. Atomic Energy (English Transl.) 21, 2 777-779 (August 1966).

In this letter we find the buildup factor B for reflection of gamma rays emitted by radium and cesium sources at various heights above earth and water surfaces. The radium source contained 0.1 mg-equiv radium. Measurement was by an SRP-2 scintillation gamma radiometer.

1429. F. A. Makhlis, L. A. Sugak, E. A. Plandin, and I. K. Shmyrev, "Absorption of the Energy of Gamma Radiation from a Unidirectional Point Source of Gamma-Quantum in Plane Geometry," Sov. J. Atomic Energy (English Transl.)21, 3, 855-857 (September 1966).

By using the Monte Carlo method, we have studied the absorption of gamma radiation from a unidirectional point Co^{60}source located on the surface of a plane-parallel water-equivalent plate of infinite extent and finite thickness. From the results we estimated the accuracy of the calculation, using the absorbed-energy buildup factors and the so-called gamma method. The calculation was performed with "Minsk-1" computer by a previously-devised method. We calculated the histroies of over 5,000 gamma-quanta. We summed the energy absorbed in ring-shaped coaxial layers with radial step equal to 3 or 5 cm. The plate was divided in thickness into plane layers with thickness also equal to 3 or 5 cm.

1430. L. A. Trykov, I. V. Goryachev, and V. I. Kukhtevich, "Measurement of the Fast Neutron Albedo Dose for Different Shields," Sov. J. Atomic Energy (English Transl.), 21, 4, 912-918 (October 1966)

The energy distribution of neutrons reflected from iron, soil, water, and polyethylene is measured with a monocrystal scintillation spectrometer for various angles of incidence of a beam of reactor neutrons at the surface of the medium being studied. The relationship is developed between the magnitude of the neutron albedo dose and the layer thickness for iron. The experimental data from this paper are compared with the results obtained by other authors.

1431. P. A. Yampol'skii, V. F. Kokovikhin, A. I. Golubkov, N. A. Kondurushkin, and A. V. Bolyatko, "Neutron Penetration in Air," Sov. J. Atomic Energy (English Transl.), 21, 4, 926-930, (October 1966).

A Monte Carlo calculation of the neutron distribution arising from monoenergetic isotropic point sources in an infinite homogeneous medium of normal density is presented. Initial neutron energies of 0.001; 0.025; 0.2;0.8;2;5;10; and 14 MeV are considered. The space-energy and time neutron distribution at distances of 10 to 1300 m from the source are obtained. The results are compared with certain of the author's experimental data.

1432. M. A. Kartovitskaya, S. M. Rubanov, and L. S. Shskorbatova, "Efficacy of Boron in Metal-Water Shields," Sov. J. Atomic Energy (English Transl.) 21, 4, 957-958 (October 1966).

The dependence of weight and size of iron-water and lead-water shields on the boron content and the place where it is introduced into the shield is investigated.

Boration of lead-water shields is effective only up to A = 0.5 wt% of boron; for B equal to 0.2 to 0.3, where omega is the volume fraction of the heavy component in a homogeneous metal-water shield, the decrease in shield thickness amounts to 1% for for B = 0.7 the decrease is 3%. The saving in shield weight in the case of an axially symmetric arrangement amounts to 1.5 to 2% for B = 0.2 to 0.3%.

Boration is more effective in iron-water shield since the yield of capture gamma radiation from iron is greater than that from lead. For shields with B = 0.5 the relative decrease in weight is from 5 to 8% and with B = 0.9 it is 9%. For a lower iron concentration (B = 0.2) the introduction of 5 wt% boron into the shield leads to a decrease in shield weight of 6%.

1433. D. L. Broder, V. B. Dubrovskii, P. A. Lavdanskii, V. P. Pospelov, and V. N. Solov'ev, "Shielding Properties of Fireproof Chromite and Magnesite Concretes," Sov. J. Atomic Energy (English Transl.) 21, 4, 959-60 (October 1966).

An experimental study has been made of the shielding properties of fireproof chromite and magnesite concretes, which can be used as thermal shielding in reactors. Fireproof concretes become dehydrated at high temperatures, which changes their shielding properties. In order to make the experimental conditions under which fireproof concretes operate in actual shield designs, slabs of the materials were given a heat treatment.

The conclusion is drawn from the data obtained that even in the dehydrated state, fireproof chromite and magnesite concretes make good shielding materials, and are to be recommended for use in building shields to operate under high temperature conditions (up to 800-1700°C).

1434. S. M. Rubanov and L. S. Shkorbatova, "Optimum Composition of Homogeneous Shields," Sov. J. Atomic Energy (English Transl.) 21, 4, 956-957 (October 1966).

This article is devoted to the investigation of the dependence of the thickness and weight of a two-component shield for a nuclear power reactor on the composition of the shielding material.

The calculations are performed by a numerical method which permits taking into account all the components of the total dose outside the shield, including the effect of slow neutron buildup and the production of capture gamma rays.

The contributions made by intermediate neutrons and by capture gamma sources are computed by a seven group age-diffusion calculation of the space and energy dependence of the neutron distribution in the reactor and shield. The method presumes that the spatial distribution of neutrons with energies above 1.5 MeV is known experimentally. The spatial distribution of the total dose is obtained for shields of various compositions.

In the table are listed minimum values of the thickness x_D^{min} and the weight $G^{min} = \gamma x_D$ (where γ is the density of the shielding material) of a slab shield necessary to reduce the dose rate to the maximum permissible level of 0.6 μ rem/sec, and, in the case of homogeneous two-component shields, the volume fraction of the heavy component insuring minimum thickness ω_x^{min} and minimum weight α_G^{min}. The radiation source is a hypothetical water-cooled water-moderated 50 MW (thermal) reactor.

On the basis of the calculations performed and by comparison with the other data, it may be stated that for certain optimum weight-size characteristics it is not permissible to neglect the buildup factors for low-energy neutrons and capture gamma radiation from the shield. Discrepancies are observed in the majority of the shielding compositions considered.

1435. O. S. Marenkov, "Some Laws of the Distribution of the Gamma-Field of a Soft Emitter," Sov. J. Atomic Energy (English Transl.) 21, 4, 966-967 (October 1966).

Monte Carlo calculations were performed for initial source energies of 0.279 MeV (Hg^{203}) and 0.145 MeV (Ce^{141}). The scatterers used were water, quartz sand, aluminum, and iron. The minimum source-detector distance was 3 cm, the maximum 25 cm. The maximum number of quantum "histories" followed was of order 16,000, giving a statistical error of less than 2-3% in the integral flux. Analysis of the theoretical results shows that the integral gamma radiation intensity is given by a simple exponential function. This simple law can be used in applied methodological investigations with soft emitters.

1436. N. F. Andryushin, B. P. Bulatov, and G. M. Fradkin, "Some Characteristics of the Field of Backscattered Gamma Radiation in Working Premises," Sov. J. Atomic Energy (English Transl.) 21, 4, 968-9 (October 1966).

As a rule, work with radioactive preparations is performed in small rooms. The intensity of radiation at various points in the space is thus determined not only by the primary radiation, but also by the radiation scattered from the walls of the room. In some cases, scattered radiation may appreciably effect the results of measurements. However, until recently there was practically no work in the literature on the quantitative characteristics of radiation inside a closed cavity.

In this article the authors verify results from model cavities (with sizes less than 50 x 50 x 50 cm) by experiments with an actual room designed for work with powerful gamma emitters (the room was 440 x 320 x 260 cm, with concrete walls 100-cm thick).

1437. R. V. Stavitskii, "Energy Characteristics of X-Rays with Maximum Voltages of 40-120 kV," Sov. J. Atomic Energy (English Transl.) 21, 4, 980-82 (October 1966).

In designing and testing shielding against direct and scattered x-radiation with maximum tube voltages of 40-120 kV, account must be taken of the energy characteristics of both forms of radiation. We have used an indirect method of determining these characteristics, based on single and double measurements of half-attenuation layers, followed by determinations of the effective energy.

The half-attenuation layer was measured for three media -- (1) water, which is a tissue-equivalent material; (2) brick (density 1.6 g/cm^3); (3) concrete (2.3 g/cm^3).

1438. B. P. Bulatov, "Method of Calculating the Intensity of Backscattered Gamma Radiation," Sov. J. Atomic Energy (English Transl.) 2., 5, (November 1966).

Formulas and graphs enabling the energy flux density of backscattered gamma radiation from certain widely-used isotopes (Co^{60}, Cs^{137}, Au^{198}, Cr^{51}) to be calculated are presented for several typical cases, for example, narrow and wide beams of gamma quanta falling at different angles on to a scattering surface, and an isotropic source in contact with the surface of a reflector. The accuracy of calculations based on these formulas is $\pm 20\%$. The method here presented constitutes the first attempt to produce engineering formulas and nomograms for determining quantities of scattered gamma radiation.

1439. B. K. Fedyushin, "Propagation of Capture Gamma Radiation in a Uniform Spherical Shield," Sov. J. Atomic Energy (English Transl.) 21, 5, 1062-1063 (November 1966).

If thermal neutrons emitted isotropically from a spherical source of radius R_0 enter a uniform concentric spherical shell shield of thickness $h = R - R_0$, where they give rise to ν γ's each of energy ϵ (MeV) per thermal neutron capture, and if it is assumed that the gamma rays are emitted isotropically and cannot penetrate the spherical neutron source, the capture gamma dose rate at the outside of the shield is given by an integral which is evaluated by using diffusion theory and buildup factors. An integral which is evaluated by using diffusion theory and buildup factors.

1440. O. S. Marenkov, "Calculating Photoelectric Attenuation Coefficients for Gamma Radiation," Sov. J. Atomic Energy (English Transl.) 21, 5, 1071-2 (November 1966).

An analytical approximation is given of the mass photoelectric attenuation coefficients for gamma rays.

1441. S. F. Degtyarev, V. I. Kukhtevich, A. P. Suvorov, V. V. Tarasov, V. K. Tikhonov, and S. G. Tsypin, "Angular Distribution of Fast Neutrons Emerging from a Medium Which Contains Hydrogen," Sov. J. Atomic Energy (English Transl.) 21, 5, 1076-1078 (November 1966).

The present article gives the results of experiments on the angular distribution of the flux (dose rate) of fast neutrons, with energies $E \geq 0.7$ MeV, emerging from a plate of lithium hydride with a density of 0.5 g/cm^3 and a thickness of 15, 30, 45, or 60 cm, or from a layer of water of thickness 15 or 45 cm. The radiation source was a collimated beam of neutrons (plane unidirectional source).

1442. G. M. Bozin, S. F. Degtyarev, V. I. Kukhtevich, B. I. Sinitsyn, V. B. Staroverov, V. K. Tikhonov, and S. G. Tsypin, "Passage of Fast Neutrons Through Thick Layers of Lithium Hydride," Sov. J. Atomic Energy (English Transl.) 21, 5, 1079-1080 (November 1966).

The attenuation of the flux (dose rate) of fast neutrons in lithium hydride of density 0.5 g/cm^3 was studied using a plane unidirectional neutron source.

The fast neutron flux was measured by a scintillation counter with a tablet of ZnS(Ag) mixed with Plexiglas (tablet diameter 4 cm, thickness 0.4 cm).

1443. S. F. Degtyarev, V. I. Kukhtevich, V. K. Tikhonov, S. G. Tsypin, "Build-up Factor of Fast Neutrons Versus the Relative Positions of Shielding and Detector", Sov. J. Atomic Energy (English Transl.) 21,5,1081-83.

The present article gives experimental results and calculations on the relation between the flux of fast neutrons (with energies E greater than 0.7 MeV) and the shield-detector distance for constant source-shielding distance.

1444. A. V. Larichev, O. F. Partolin, E. D. Chistov, and O. M. Zaraev, "Passage of Gamma Rays Through Spherical and Cylindrical Shielding Barriers," Sov. J. Atomic Energy (English Transl.) 21,5,1084-1086 (November 1966).

The present letter gives the results of experiments on the attenuation of gamma rays from Co^{60}, Cs^{134}, and Tm^{170} in cylindrical and spherical shielding barriers.

The source-detector distance was 50 or 100 cm. At these distances the gamma rays could be taken as coming from point sources. The spherical and cylindrical barriers were sets of hollow hemispheres and hemicylinders of various thicknesses which were fitted into one another. The shielding thickness increased towards the detector. The spherical screens of lead, iron and aluminum had maximum thicknesses of 75 mm, while those of VNM-2-3 tungsten alloy had maximum thicknesses of 41 mm (converted to density 19.3 g/cm^3). The cylindrical barriers of lead, iron and aluminum had maximum thicknesses of 100 mm.

Similar measurements were made in the same conditions for flat barriers shields which consisted of lead, iron, and aluminum sheets up to 100 mm thick and blocks of tungsten alloy.

1445. V. E. Drozdov, I. M. Zakharov, and S. P. Dobrovol, "Dose-Rate Field of an Irrariator with Used Fuel Elements as Source of Gamma Radiation," J. Nucl. Energy, Pts. A/B 20, 1011-1018 (1966).

Experiments were carried out on the RFT reactor to determine the activity distribution along a fuel element. It was shown that the distribution of activity is satisfactorily described by a cosine function, and that the equations used to calculate the dose-rate field for irradiators with an even distribution of activity along the radiation source are inapplicable when such sources are fuel elements [the discrepancy is 23% for RFT and 32% for the First Atomic Power Station (AES)]. Good agreement was obtained between our calculations and the experimental data.

1446. J. H. McCrary, E. H. Plassman, J. M. Puckett, A. L. Conner, and G. W. Zimmerman, "X-Ray Attenuation-Coefficient Measurements," Phys. Rev. 153, 2, 307-312 (1967).

X-ray attenuation coefficients were measured in the energy range of 25 keV to 130 keV for the following elements: Be, C, Mg, Al, S, Ti, Fe, Ni, Cu, Zn, Zr, Nb, Mo, Ag, Sn, La, Gd, Hf, W, Au, Pb, Th, U, and Pu. Narrow beam collimation was used with a Bragg diffraction monochromator. Experimental errors were less than 2% for most of the measurements.

1447. B. P. Hughes, I. R. G. Lowe, and J. Walker, "The Diffusion of Water in Concrete at Temperatures Between 50 and 95°C," Brit. J. Appl. Phys. 17, 1545-1552 (December 1966).

The water content of concrete used in nuclear reactor shields plays an important part in settling the shielding characteristics, and the movement of water is largely responsible for the long-term behaviour of the shields. The prediction of the long-term behaviour is clearly desirable. The paper describes drying tests which have been made on concrete and neat cement spheres of various diameters and indicates that elementary diffusion theory is useful for the temperatures investigated (between 50 and 95°C). The diffusion coefficient is, however, strongly dependent on temperature. Surface drying mechanisms are considered and it is shown that they can retard the moisture diffusion process in normal applications.

1448. V. V. Verbinski and M. S. Bokhari, "A Fast-Neutron Spectrometer for Reactor Environments," Nucl. Instr. Methods 46, 2, 309-316 (1967).

A fast-neutron spectrometer which can operate in a reactor environment has been tested in the water shield of the Bulk Shielding Reactor I in gamma-ray fluxes as high as 10^6 R/h. Basically, the spectrometer consists of a very thin layer of ^{6}LiF between, but not in contact with, two silicon-gold diodes. Neutrons traversing the spectrometer collimator strike the ^{6}Li foil, and the alpha particle and triton emitted in the resulting ^{6}Li(n,α)T reaction (Q = 4.78 MeV) are detected in coincidence in the two diodes. Shadow shielding in the collimator protects the diodes from direct reactor radiations. Background measurements are made by substituting a ^{7}LiF foil for the sensitive ^{6}LiF element. The measured spectra are in good agreement with calculated spectra, the measurements and calculations having been normalized to the same absolute neutron source intensity.

1449. A. Rubbino, D. Zubke, and C. Meixner, "Neutrons from (Alpha,n) Sources," Nuovo Cimento 44B, 178-181 (July 11, 1966).

The neutron energy spectrum of a Pu-Be source has been measured using a particular fast-neutron spectrometer. Good energy resolution was achieved and the detailed structure of the spectrum was evidenced. Results are discussed and compared with other measurements obtained with Pu-Be, Po-Be, and Am-Be sources.

1450. V. P. Mashkovich, A. N. Nikolaev, V. K. Sakharov, B. I. Sinitsin, and S. G. Tsipin, "Axial Distribution of Fast Fission Neutrons in Straight Cylindrical Channels in Water, J. Nucl. Energy 21, 435-439 (May 1967).

An investigation was carried out into the penetration of neutrons from the BR-5 reactor in straight cylindrical channels 144 and 90 mm in diameter. Sulphur and aluminum threshold indicators were used as detectors.

Data were obtained on the axial attenuation in straight cylindrical channels of fast neutron fluxes from a disk and finite plane monodirectional fission-neutron source, and the effect on the attenuation of the neutron radiation of displacement of the channels was studied. The results can be used to calculate the axial penetration of neutron radiation in stepped channels.

1451. A. B. Chilton, "Buildup Factors for Point Isotropic Gamma Ray Sources in Infinite Medium of Ordinary Concrete," Nucl. Eng. Design, 6,3,205-212.

In order to provide more definitive values for buildup factors for point, isotropic gamma ray sources in an infinite medium of ordinary concrete, a Monte Carlo program has been undertaken to make such calculations. Previous approaches to this problem had been through the assumption of a Z_{eff} for concrete and an interpolation in the existing Goldstein-Wilkins results for single elements. In addition to the approximate character of the old procedure, such calculations have been variable because different values of Z_{eff} have been used by different workers.

The specific method of Monte Carlo calculation is briefly described, with emphasis on the assumptions, approximations, and sources of data.

The resulting data are estimated to have an uncertainty less than 5%. They are compared with results using the Z_{eff} approach, and the asymptotic trends at very small and very large penetration distances are discussed.

1452. J. A. Davis, L. A. Hageman, and R. B. Kellogg, "Singular Difference Approximations for the Discrete Ordinate Equations in x-y Geometry," Nucl. Sci. Eng. 29, 2, 237-243 (1967).

Two well-known finite difference approximations to the discrete ordinate equations in x-y geometry are shown to lead to a singular system of equations for the case of reflecting boundary conditions. These difference schemes are the diamond approximation of Carlson, and the central difference approximation. Despite this singularity, it is shown for the diamond scheme that a solution always exists and is, in some sense, unique. For the central difference scheme, however, it is shown that a solution need not, and in most cases will not exist.

1453. L. R. Bunney and D. Sam, "Gamma-Ray Spectra of the Products of Fast-Neutron Fission of 235-U and 238-U at Selected Times After Fission," Nucl. Sci. Eng. 29, 3, 432-443 (September 1967).

Experimental measurements of the gamma-ray pulse-height distributions due to the products of fast-neutron-induced fission of 235-U and 238-U are presented. The measurements were made at nine selected times, $\frac{1}{4}$, $\frac{1}{2}$, 1, 2, 5, 10, 24, 48, and 72 h after fission. Irradiation times and counting intervals were chosen to minimize saturation and decay effects.

The experimental data were used to calculate 100 energy bin distributions of the absolute number of photons/(fission sec) by means of absolute calibrations of the collimated NaI(Tl) detector. The number of fissions in each sample was determined radiochemically. Machine computation was used extensively.

1454. R. E. Maerker and F. J. Muckenthaler, "Neutron Fluxes in Concrete Ducts Arising from Incident Thermal Neutrons: Calculation and Experiment," Nucl. Sci. Eng. 29, 3, 444-453 (ORNL TM-1758) (September, 1967).

Monte Carlo calculations, employing the albedo concept, have been carried out to determine the thermal-neutron flux distributions along the centerlines of a straight, a two-legged, and a three-legged square open concrete duct for a particularly demanding source geometry. The neutrons entering the duct were of thermal energy. The calculations used differential angular albedo data for concrete which have been reported previously. A comparison of the results of the calculations with those from a geometrically similar experiment shows good agreement and places on a firm foundation the concept of treating the thermal-neutron duct transmission problem as a reflection phenomenon at a point that is describable by the differential albedo properties of the walls.

1455. H. F. Priest, F. C. Burns, and G. L. Priest, "Neutron Flux Distribution from a 14-MeV Neutron Generator," Nucl. Instr. Methods, 50, 141-146 (1967).

In order to carry out accurate activation analyses using a neutron generator employing the $^{3}H(d,n)^{4}He$ reaction, it is important to know the neutron flux distribution relative to the tritium target. Methods have been developed for determining this flux distribution and typical examples are presented.

1456. R. E. Bentley, J. C. Jones, and S. C. Lillicra, "X-Ray Spectra from Accelerators in Range 2 to 6 MeV," Phys. Mod. Biol. 12, 3, 301-314 (1967).

The photon flux spectra from two 6 MeV Linear Accelerators and a 2 MeV Van de Graaff Generator which are in normal use for radiotherapy, have been measured using a 7.5 x 7.5 cm NaI(Tl) crystal spectrometer. The x-ray beam was scattered by aluminum sheets to reduce the intensity and energy range, and the primary spectra calculated from the measured scattered spectra using the Klein-Nishina formula. The crystal spectrometer was calibrated from theoretical Monte Carlo calculations which were converted to photon flux spectra by a digital computer. Photon flux spectra were obtained for different field sizes and under different operating conditions, and the variation in the spectra over the x-ray field was investigated. The results were compared with theoretically calculated thick target spectra using published formulae for the bremsstrahlung cross sections.

1457. M. Kitazume, A. Tsuruo, and M. Shindo, "Gamma Rays and Neutrons Streaming About a Cylindrical Duct by the Ray Analysis Method," J. Nucl. Sci. Technol., 4, 1, 11-20 (1967).

Studies were made on the fluxes and doses produced by an infinite plane source at the further side of duct exit surface of a shield barrier. The plane of infinite source is divided into four regions, and the contribution of each region is calculated separately by the ray analysis method. The total dose at the observation point and the contribution to it from each region of the plane source are calculated, using as parameters such factors as position of the observation point, duct radius, duct length, energy of radiation and shield material. The radius of influence created around the duct by its presence in respect of dose at the observation point, and the radius of contribution of the plane source in respect of flux at the duct centerline have been determined. The characteristics of the integrand that are significant for the results obtained should provide useful data for practical design work involving duct streaming problems. The method of calculation presented should not only be applicable to gamma rays but also to fast neutrons.

1458. F. B. K. Kam and F. H. S. Clark, "Fission Neutron Attenuation and Gamma-Ray Buildup Factors for Lithium Hydride," Nucl. Appl, 3, 433-435 (July 1967).

A fission-spectrum neutron dose-rate attenuation curve has been computed by the Monte Carlo method for lithium hydride penetrations up to 67.5 g/cm^2. Gamma-ray exposure dose buildup factors have been computed for lithium hydride using the moments method. Discrete gamma-ray source energies in the range from 0.5 to 10 MeV were used, and calculations were made out to 20 mean free paths (at source energy).

1459. P. E. Wilkniss and G. J. Wynne, "Flux Density Close to the Target of a Neutron Generator," Int. J. Appl. Radiat. Isotop. 18, 77-81 (January 1967).

An expression is developed for calculating the neutron flux distribution in the neighborhood of a disk-shaped target, taking account of the orientation of the detector relative to the source.

Calculated values are compared with data obtained empirically, using a commercial 14 MeV neutron generator.

This paper points out some of the problems involved in ignoring detector orientation, outlines a development of an expression for flux which considers detector orientation, and compares calculated values with data obtained empirically.

1460. E. Storm and H. I. Israel, Photon Cross Sections from 0.001 to 100 MeV for Elements 1 Through 100, LA-3753 (Nov. 15, 1967). Availability: CFSTI, $3.00, $0.65.

The report contains tables of photon cross sections in the range 0.001 to 100 MeV for Z = 1 to 100. Supplementary tables of mass attenuation coefficients for air, water, concrete, and sodium iodide; x-ray atomic energy levels; line energies; average energies, relative intensities; and fluorescence yields are also presented.

1461. M. Alberg, H. Beck, K. O'Brien, and J. E. McLaughlin, "An Investigation of Differential Energy and Angle Spectra for 137-Cs Gamma Rays in Water and Air," Nucl. Sci. Eng. 30, 1, 65-74 (1967).

Differential energy and angle spectra from a point isotropic ^{137}Cs source in an effectively infinite medium of water have been determined for gamma-ray penetrations of 1, 2, 3, and 4 mean free paths at 15° intervals. The spectra were unfolded from scintillation spectrometer measurements by an analytic method based on the Scofield iteration scheme. An integration of the results over all angles yielded differential energy spectra which were consistent with multigroup transport calculations.

The measurements were also carried out in a condensed, air-like medium. A comparison of the results with those obtained in water showed that the differences in attenuation coefficients between the two materials caused spectral differences only at very low energies for small separations between source and detector, which were consistent with theoretical calculations.

1462. R. B. DeBar, "Truncation of the Spherical Harmonic Expansion of the Transport Equation," Nucl. Sci. Eng. 30, 2, 159-165 (1967).

The most general linear truncation relation for the spherical harmonic representation of the transport equation in three dimensions is shown in any order to be a partial differential equation. This equation is uniquely determined up to two independent scalar parameters in the time-dependent case and one scalar parameter in the time-independent case. In the time-dependent situation, one of the parameters may be related to the other parameter, which is pertinent to the time-independent limit, in such a way as to give correct retardation in all orders.

1463. R. W. Stooksberry and E. E. Carroll, Jr., "Spatial Distributions in Water of Neutrons from a 2.84-MeV $T(p,n)^3He$ Source," Nucl. Sci. Eng. 30, 2, 213-221 (1967).

The distributions in water of indium resonance and $^{115}In(n,n')^{115m}In$ neutrons from a 2.85-MeV $T(p,n)^3He$ source have been measured. The values of the integrated plane activities perpendicular to the proton beam direction (including the extrapolated region beyond the range of measurement of each plane) are presented. The experimental results are in agreement with Monte Carlo calculations utilizing two oxygen elastic-scattering descriptions and agree within quoted standard deviations in both cases.

1464. F. R. Channon and R. L. Seale, "The Study of Thermal Neutrons in Shields Containing Ducts by Source-Separation Techniques," Nucl. Sci. Eng. 30, 2, 242-260 (1967).

The behavior of thermal neutrons in a water shield containing ducts was studied experimentally and compared with theoretical models. In addition to measuring the total flux, a source separation technique was used to isolate the various flux components, which are:

1) direct flux, uncollided neutrons which enter the duct mouth

2) scattered flux, collided neutrons which enter the duct mouth

3) penetration flux, collided neutrons which originally enter the fhield.

Duct diameters from 1 1/8 to 4 in. were considered. The shield absorption properties were altered by dissolving various amounts of boric acid in the water. The duct cross-sectional shape was changed by partially flooding the interior of the duct.

The experimental results indicated that the direct flux is inversely proportional to distance squared. For sufficiently long ducts, the direct flux is nearly the total flux. For shorter ducts, either the scattered flux or penetration flux may produce the largest contribution to the total flux. Each of these components peak near the duct mouth and then attenuate more rapidly than the direct flux.

Successful calculational models were developed for each of the flux components. These permitted determination of the total flux to within a factor of 1.3 at distances greater than two or three duct diameters from the mouth.

1465. Y. Furuta and Y. Kanemori, "Experimental Studies on Gamma-Ray Dose Rates from a 60-Co Cylindrical Source," Nucl. Sci. Eng. 30, 2, 261-267 (1967).

Gamma-ray dose rates from a ^{60}Co cylindrical source were obtained experimentally in the radial direction at the half-height of the source.

The concept of the dose buildup factor was introduced for a volume source. The dose buildup factor for a cylindrical source, which is represented as a function of the distance between source and detection point, has a value of about five at the point nearest to the source surface. The factor then decreases rapidly, passes through a minimum value, and approaches a constant value. These features were analyzed experimentally with a line and a disk source.

An empirical formula for the dose buildup factor is proposed which agrees with the experimental values to within about $\pm 15\%$.

1466. J. J. Wagschal, "An Improvement of the S_N Calculation Method for Slab Geometry," Nucl. Sci. Eng. 30, 2, 305-307 (1967).

Although the DSN method is capable of giving good results with particular choices of angles and weights one is quite limited in the choice of the angles. The modified SNG method gives accurate results leaving considerable freedom in the choice of the angular interval. This freedom is very important in problems having a directional preference such as anisotropic sources.

1467. T. E. Dudley, M. R. Mendelson, and N. E. Holden, "The Use of Infinite Medium Spectra in Gamma-Ray Heating Calculations," Nucl. Sci. Eng. 30, 3, 328-339 (1967).

A reasonable physical model for the slowing down of gamma rays in infinite media is presented, and a method of numerical solution is described. Equilibrium energy spectra due to a fission source of gamma rays are shown for water, aluminum, iron, zirconium, and lead. In addition, energy spectra in aluminum, iron, and lead, due to the corresponding (n,γ) source in each metal, are presented.

The use of infinite medium calculations to obtain a lower energy cutoff for a gamma heating problem is suggested. It is shown that for the case of a fission source, essentially all of the source energy is absorbed above 0.05 MeV in the materials studied, except in the case of water where approximately three percent of the energy is absorbed below 0.05 MeV.

The infinite medium spectra are used to average absorption and slowing-down cross sections for fuel materials and metals, and the resulting group constants are compared with similar calculations using a fission-source spectrum as a weighting function. Large differences are noted in many instances. Calculations of spatial energy deposition in simple model problems indicate that such differences in group constants can lead to local errors of significant magnitude.

1468. R. E. Maerker and F. J. Muckenthaler, "Neutron Fluxes in Concrete Ducts Arising from Incident Epicadmium Neutrons: Calculations and Experiments," Nucl. Sci. Eng. 30, 3, 340-354 (1967).

Monte Carlo calculations, using the albedo concept, have been carried out to determine subcadmium and epicadmium neutron flux distributions along the centerline of a straight, a two-legged, and a three-legged square concrete duct arising from the slowing down of incident epicadmium neutrons for a particularly demanding source geometry and spectrum. The calculations used albedo data differential both in the reflected angles and reflected energy which have been reported previously for concrete. A comparison of the results of these calculations with those from a geometrically similar experiment shows good agreement and places on a firm foundation the concept of treating neutron slowing down in a concrete duct as a reflection phenomenon at a point which is describable by the differential albedo properties of the walls. The conclusion is also reached that the dose rates arising from the subcadmium neutrons (whether due to an epicadmium source or a subcadmium source) and associated secondary wall-capture gamma rays can comprise a very important part of the total absorbed dose rate in tissue deep inside a multilegged duct.

1469. N. E. Holden, M. R. Mendelson, and T. E. Dudley, "Total Gamma Energy Release Due to Thermal-Neutron Fission in 235-U," Nucl. Sci. Eng. 30, 3, 461-463 (1967).

The time and energy dependence of the gamma-ray spectrum following a thermal-neutron fission of ^{235}U has been investigated by several authors. A compilation and evaluation of much of this data has been made and an estimate of the total gamma-ray energy release has been obtained. Following the convention of most authors, the data are broken down into four time ranges.

1470. H. E. Hungerford and R. F. Mantey, Fast Reactor Technology -- Plant Design. Chapter 8. Shielding, pp. 425-578, The MIT Press, Cambridge, Massachusetts, 1966.

This chapter on shielding of fast reactors was prepared as a guide in the design, fabrication, construction, testing, and operation of fast reactor power plants. It presents little theory, leaving this to the quoted references. The chapter should give the user a broad knowledge of the many facets of shielding for fast reactors. The chapter encompasses material available in the literature as well as material hitherto unpublished.

In general, the chapter's contents fall into five categories: (1) basic shielding considerations and principles, material requirements, environmental effects, design criteria, and effect of coolant, (2) shield calculations, (3) test programs, (4) nuclear data and mathematical aids, and (5) a description of fast reactor power-plant shields.

1471. R. Oliver, Radiation Physics in Radiology, Blackwell Scientific Publications, Oxford, 1966.

This is a brief introduction to the principles of radiation physics with an emphasis on x-rays. The topics treated are: atomic structure, production of x-rays, absorption of x-ray energy, mechanisms of absorption, measurement of x-radiation, radioactivity, personnel protection, visible light and optics. (RSIC).

1472. H. Schultz, S. Sassa, Analytical Estimation of the Optimal Selection and Arrangement of Shielding Materials in Reactor Shields, ORNL-TR-1526 (Transl. from ABS-THH-1024) (September 1965). Availability: CFSTI, $3.00, $0.65; JCL, $0.60, $2.06.

On the basis of simple attenuation models, an analytical optimization model is constructed for one energy group and several energy groups. Coupled energy groups are also discussed. Previous work in optimization is briefly reviewed. (RSIC).

1473. R. L. Heath, Scintillation Spectrometry, Gamma-Ray Spectrum Catalogue (Vol. I and II), IDO-16880 (August 1964). Availability: Dep. CFSTI.

A new edition of the Scintillation Spectrometry Gamma-Ray Spectrum Catalogue has been issued. This edition is a complete revision of the original data compilation, which was issued as an AEC R and D report (KDO-16408) in 1958. As in the original catalogue, this edition contains a collection of spectra representing the response of a scintillation spectrometer to individual radioactive nuclides. In addition to the graphs representing the response of a 3" x 3" NaI detector in a standard geometrical arrangement, the data are presented in digital form for the preparation of punched-card, perforated tape, or magnetic tape libraries for data analysis. An important edition to the catalogue is data for neutron-deficient isotopes.

The new edition is prepared in two loose-leaf volumes and contains data for almost 300 isotopes. All spectra are normalized to a standard set of gain scales and a text is presented which describes the fundamentals of gamma-ray spectrometry. This includes a discussion of spectrometer design, electronics, instrumental calibration, and data processing.

To facilitate the use of these data, tables of detector efficiency, photopeak efficiency, and other information useful for quantitative data analysis have been included. An extensive index has also been added with separate tables of data listed according to gamma-ray energy, half-life, method of source production, and other specialized categories.

1474. A. Honig and V. Zapletal, Ultrasonic and Radiometric Measurement of Homogeneity and Density of Concrete in Biological Shield of Nuclear Reactor, VUT 65-D-3 (1965). Availability: Technical University BRNO, Hlinky 48, Czechoslavkia.

Methods for using radiometry and ultrasonics are presented which can be used for checking the compacting of concrete in biological shields of nuclear reactors. Methods are given for measuring the transmission time and velocities of ultrasonic waves and concrete wall thickness. The statistical evaluation methods and mathematical processing of measured values are described. A determination was made of the relation between velocity of ultrasonic wave propagation and density for standard (2.2 g/cm^3) and heavy (3.6 g/cm^3) concretes. The relations between gamma-ray attenuation, ultrasonic impulse velocity and concrete densities were studied. (RSIC)

1475. Subcommittee ANS-6, Shielding, "Proposed Standard Program for Testing Biological Shielding in Nuclear Reactor Plants," Nucl. Eng. Bull. 5, 1, 1-8 (ANS 6.1) (July 1967).

This standard describes an operational shield testing program that is to be used in evaluating the installed biological shielding in nuclear plants. A general testing procedure is outlined and the necessary radiation measurements and types of radiation-measuring instruments are prescribed.

1476. C. P. Ross, C. L. Angerman, and F. D. R. King, Development of 60-Co Capsules for Heat Sources, DP-1096 (June 1967).

Radioactive cobalt containing 100 curies of ^{60}Co per gram of metal has been encapsulated in "Inconel" 600 and "Hastelloy" C containers. Such capsules, producing up to 250 watts, have been tested successfully at 850°C for 100 hours in air. Longer tests are under way.

Preliminary tests indicate that TD Nickel (2% thoria-dispersed) and TD Nickel Chromium (78 nickel - 20 chrome) are also promising materials for encapsulation.

About 8 megacuries of ^{60}Co at activities up to 400 Ci/g will be available for heat source development and demonstrations by May 1967.

Progress is reported in the areas of ^{60}Co production, fuel form properties, capsule design, capsule material properties, and capsule fabrication and testing.

1477. M. P. Ruffle, Computation of Certain Solid Angles and Some Useful Associated Functions, AERE-R-5419 (March 1967). Availability: Dep., CFSTI; JMSO 3 s. 0d.

Formulae are presented for evaluating the solid angle subtended by a plane polygon and a circular disc at a general point in space. The averages of the latter function over a coaxial cylindrical surface and a coaxial parallel disc are expressed in terms of standard elliptic integrals.

1478. H. S. Bloomfield, Calculation of Fast-Neutron Flux Emerging from a Reactor Beamhold and Comparison with Experiment, NASA TM X-1418 (August 1967). Availability: CFSTI, $3.00.

A comparison of calculated and measured fast-neutron fluxes emerging from the HB-6 beamhole of the NASA Plum Brook reactor is presented. Integrated dose rates were calculated with the QAD line-of-sight computer program using Albert-Welton removal theory to account for neutron attenuation. The program output was converted to fast flux with the Snyder-Neufeld conversion factors and then modified by the use of collimated removal cross sections to account for the reduction of the scattered flux component emerging from the duct. In addition, the use of neutron removal theory in nonhydrogenous media was assumed valid because of recent measurements.

Measured flux values were calculated from sulfur foil activation data and then extrpolated to a lower energy of 0.3 MeV (0.08 picojoule) to correspond to the conventional energy range of the calculational method. The extrapolation was carried out by using both a fission spectrum and a beryllium altered fission spectrum. These spectra were chosen to represent the possible extremes of the true duct spectrum, which was estimated to be a hardened fission spectrum.

Results of the comparison indicate that the calculational method can predict fast-neutron fluxes emerging from the HB-6 beamhole to within a factor of about 1.5 when suitable corrections to the method are applied.

1479. C. B. Clifford, "Design and Fabrication of a Prototype Laminated Uranium Metal Shipping Cask for Large Shipments of Cobalt-60," KY-521.

This report presents the design standards, engineering drawings and the manufacturing techniques used in fabricating a prototype laminated depleted uranium metal shipping cask for transporting 150,000-curie lots of cobalt-60.

The cask design was based on criteria furnished by the Oak Ridge National Laboratory and the latest drafts of the proposed 10 CFR 71 regulations and Chpater AEC-0529. Emphasis was placed on reducing the weight and volume of the casek in order to minimize the problems associated with transporting and handling the casek in the cells located at ORNL and their customer's facilities.

Heat transfer test data are presented to show the temperatures obtained by dissipating up to 1.8 times the design heat load of electrical energy inside the cask.

1480. P. J. Brockwell, Stochastic Problems in Transport Theory, ANL-7131 (August 1966). Availability: CFSTI, $3.00,$0.65.

The application of the theory of discontinuous Markov processes and stochastic population processes to problems in the field of transport theory is investigated.

In Chapter I it is shown how a scattering process (in which there may, at each collision, be a constant probability that the scattered particle is absorbed and lost to the system) can be characterized by transition probabilities $P_k(S, \theta_1, \ldots, \theta_k | \omega; s)$ of a transition $\omega \to S$ in distance s with exactly k scatterings, the angles of scattering $\theta_1, \ldots, \theta_k$ (in order of occurance) being such that $\theta_1 \varepsilon \theta_k$. Such a characterization is of importance when there is a change of state of the particle at each collision and the corresponding transition probabilities depend on the angle of deflection.

For one-dimensional processes the problems simplify considerably owing to the fact that the angular deflections θ_i can take only two values, 0 or π. A detailed investigation of one-dimensional problems in which the probability of a collision in a small element of path length δs is $\lambda\delta s + 0(\delta s)$ is made in Chpater II. In order to apply the theory to scattering in three dimensions, an approximation is made in which scattered particles are assumed to move in one of a set of 30 directions in space. This model is considered in detail in Chapters VI and VII.

Chapter III deals with a particular one-dimensional problem which arises in the anlysis of bubble chamber tracks and is concerned with the stochastic population process generated by the distances travelled between successive collisions in a fixed length of track. In Chapter V we consider one-dimensional process in which the distances travelled by a particle between successive collisions have either a gamma distribution or a distribution which is a linear combination of negative exponential distributions. Processes of the latter kind can be used to improve the approximation (used by Fermi and others) to the projection on a fixed axis of the three-dimensional motion of a particle which is scattered isotropically.

Chapter IV is concerned with multiplicative scattering processes in one dimension. Several mathematical models for one-dimensional neutron multiplication in a rod are considered, with particular reference to the concept of criticality and the distribution of the number of emergent neutrons produced by a single trigger neutron.

1481. R. B. V. Simmons, Design for Radiation Protection at Douglas Point, AECL-2877 (May 1967). Availability: Scientific Document Distribution Office, Atomic Energy of Canada Limited, Chalk River, Ontario, Canada, $1.50.

The paper summarizes the considerations in the Douglas Point design which provide radiation protection for the plant staff during normal operations. There are two basic aspects considered; the shielding and exclusion of personnel from high radiation areas and the limitation of their contact with contamination.

1482. J. W. Lewellen, P. S. Lacy, and T. M. Raby, Pathfinder Atomic Power Plant. Shielding and Radiation Survey Measurements, ACNP-67520 (June 1967). Availability: Dep. CFSTI.

The results of the shielding measurements and radiation surveys performed at the Pathfinder Power Plant during the reactor startup and power escalation programs are presented. Calibrated portable radiation survey meters including both neutron and gamma monitors and the installed nuclear and temperature instrumentation in the biological shield were utilized for these measurements. Radiation surveys were originally made at a nominal 6.5 Mw (thermal), and then repeated up to, and including, 90% full power.

1483. M. Heusinkveld, Transmission of 14-MeV Neutrons in Liquid Nitrogen, UCRL-50259 (June 9, 1967). Availability: CFSTI, $3.00, $0.65.

The range of neutrons from a 14-MeV neutron source in liquid nitrogen has been measured with various neutron detectors. The distance over which the flux of high-energy neutrons is attenuated by a factor of e was measured to be 36 ± 2 cm. Some results are also presented for lower-energy neutrons.

1484. M. Alberg, H. Beck, K. O'Brien, and J. E. McLaughlin, Tabulated Values of Experimentally Determined Differential Energy and Angle Spectra for 137-Cs Gamma Rays in Water and Air, HASL-187 (December 1966). Availability: CFSTI, $3.00.

In the 1967 Nuclear Science and Engineering article, An Investigation of Differential Energy and Angle Spectra for ^{137}Cs Gamma Rays in Water and Air, we described the measurements and analyses leading to the gamma-ray energy and angle spectra in water and a condensed air-like medium. This report contains tabulated values of these spectra at the angles and penetration distances that were studied. It also includes the differential energy spectra that were obtained by an integration of the results over all angles, and interpolations of the water spectra that were used for comparison with the spectra obtained in the air-like medium, a 6.15% solution of NaOH.

1485. M. E. Battat, D. J. Dudziak, and R. L. LaBauve, 6-Li and 7-Li Data in the ENDF/B Format, LA-3685-MS (June 26, 1967). Availability: CFSTI, $3.00, $0.65.

At the Cross Section Evaluation Working Group (CSEWG) Meeting on June 9-10, 1966, at Brookhaven National Laboratory, the Los Alamos Scientific Laboratory was assigned the responsibility of preparing the data for the isotopes ^{6}Li and ^{7}Li for the first version of the Evaluated Nuclear Data File/B (ENDF/B) tape. These data have been assembled in the ENDF/B format and sent to the Cross Section Evaluation Center (CSEC) at Brookhaven National Laboratory. Most of the data are from the AWRE data file originated by K. Parker of Aldermaston. Values of $\bar{\mu}_{LAB}$ and ξ, along with Legendre coefficients for the elastic scattering angular distributions, were received from H. Alter of Atomics International.

Appendix A of this report presents plots of the original AWRE cross-section data converted to the ENDF/B format. The ENDF/B listings for the ^{6}Li and ^{7}Li data, as they appeared on the first version (approximately February 1967) of the ENDF/B tape, are shown in Appendix B.

1486. K. Lathrop and B. Carlson, "Numerical Solution of the Boltzmann Transport Equation," J. Comp. Phys. k, 2, 173-197 (November 1966).

A numerical model for solution of the linear Boltzmann Transport Equation is formulated. By applying the same techniques used in the derivation of the analytic equation, a discrete analog of the Boltzmann equation is derived for a finite cell in phase space. Initially undetermined coefficients in the analog are determined by requiring the numerical formulation to include properties (e.g., particle conservation) of the analytic equation. Terms occurring in the finite-cell analog are defined, and two treatments of the angular dependence are illustrated. A discrete ordinates representation is derived based on a connected straight-ling-angular representation. This formulation maintains optical reciprocity and may be generalized.

The second portion of the paper describes the systematic derivation of difference relations necessary to complete solution of the numerical formulation. Both representation schemes, based on assumed forms of particle fluxes in the cell, and characteristic schemes are examined. Difficulties encountered in extrapolation made by the use of representation schemes are clarified, and insight is gained for methods that can be used to prevent flux oscillations in numerical calculations.

1487. I. Kataoka, "A Study of the Methods for Analyzing Multilayered Gamma Ray Shields," ORNL-TR-1671 (Transl. from Rept. of Ship Research Institute 3, 4, 161-343 ((July 1966). Availability: Dep. CFSTI, JCL, $13.05 fs, $6.11 mf.

The paper presents two methods for analyzing gamma rays in stratified slab shields. In Part I, a numerical integration method of the Boltzmann transport equation is described, which solves the equation at the discrete points of ordinates of spatial, angular and energetic. Under the same interval of spatial mesh points, results of test calculations reveal that the method presented gives more correct attenuation of gamma fluxes than, for instance, the discrete S_n method does. A supplemental procedure is provided for a monoenergetic and monodirectional incident source besides the treatments for continuously distributed sources.

Part II presents the Response Matrix method as well as a table of the response matrices in the Appendix. The transmission and reflection responses of the gamma rays for an elemental slab of each material have been prepared with the Monte Carlo method for the gamma-ray injection of the unit intensity to the slab surface under a certain set of incident angle and energy. The transmitted or reflected gamma rays from a given stratified shield of different materials as well as single material are obtained by operating the response matrices of the elemental slabs composing the shield of. The procedures for synthesizing and interpolating the response matrices are given. And the estimated errors accompanied with the procedures have been evaluated.

In Part III, various results of sample calculations are illustrated. The results computed with the present methods show good agreement with those from the Moments method, the Monte Carlo calculations and the experimental measurements.

1488. W. Hart, Fission Cross Section Data Files for 232-Th, 233-U, 234-U 235-U, 236-U, 238-U, 237-Np, 239-Pu, and 242-Pu in the Energy Range 1 keV to 14 MeV, AHSB(S) R 124 (1967). Availability: Authority Health and Safety Branch U.K.A.E.A., 11 Charles II Street, London, S.W. 1.

The fission cross section evaluation by W. G. Davey for the isotopes of thorium, uranium, neptunium, and plutonium has been improved, in the light of later experimental data, and the energy range extended to 14 MeV. The tabulated cross sections have been put on to punched cards in the UK Nuclear Data Library format for use in foil activation work, Data File Numbers (DFN) 332 through 342 being adopted.

1489. W. Hart, Neutron Cross Sections of Pu-239 in the Energy Range 1.0E-10 MeV to 15.0 MeV, AHSB(S) R 125 (1967). Availability: Authority Health and Safety Branch, U.K.A.E.A., 11 Charles 11 Street, London, S.W. 1.

A survey has been made of recent, accurate experimental determinations of neutron cross sections and other relevant data for Pu-239. From these, $\bar{\nu}$ and total, fission and radiative capture cross sections have been evaluated over the energy range 1.0E-6 MeV to 15.0 MeV. In an effort to complete this work in a relatively short time, the other relevant nuclear parameters (which are less significant for most neutronics calculations) have been taken from the latest UK Nuclear Data Library evaluation - Data File Number (DEF) 184('). It is thought that this omission will not seriously devalue the improvements which have been effected.

These values have been grafted onto the revised data of Story, covering the energy range 1.0E-10 MeV to 1.0E-6 MeV, and written on punched cards as part of the IKAEA Nuclear Data Library. It has been assigned the Data File Number 329 and Mark Number 5/Pu-239.

Some test calculations have been performed using this data set with the Monte Carlo Code GEM and the results are presented at the end of the report.

1490. M. L. Eaton and C. M. Huddleston, A Critical Evaluation of the Markov Matrix Treatment of Neutron Diffusion in Slabs, USNCEL-TR-529 (AD-653444, NP-16791) (May 1967). Availability: CFSTI, $3.00.

It is known that a method based on the concept of a Markov chain can be used to treat the problem of absorption, transmission, and backscatter when single-velocity neutrons are incident on a plane slab of finite thickness, under the assumption that scattering is isotropic in the laboratory system. Such conditions can exist when neutrons impinge on a slab of material, such as a shield or an inside wall of a shelter entranceway. Since the Markov matrix method is a new approach to the problem, the practical limits in applying the method have not previously been explored. This report outlines the region of usefulness of the matrix method. The solutions to some sample problems are given. Comparisons are made with other theoretical treatments. It is concluded that the Markov matrix method is useful over a wide region of interest. Within its limits of practical applicability, the method gives highly exact answers without requiring exorbitant computing time.

1491. P. V. R. Rao, J. R. Rao, and V. Lakshminarayana, "Influence of Electron Binding on the Incoherent Scattering of Low Energy Gamma Rays," Indian J. Pure Appl. Phys. 4, 2, 56-59 (February 1966).

The incoherent scattering cross sections (σ_b) in elements carbon, aluminum, copper, and cadmium have been determined at photon energies 280, 145, 129 and 100 keV and are compared with the corresponding integral free electron incoherent scattering cross sections (σ_f) calculated using the Klein-Nishina formula to assess the influence of electron binding. The diminution of σ_b relative to σ_f is found to increase progressively with decreasing energy and increasing atomic number, thereby indicating the growing severity of binding effects in these two directions. The experimental values of the ratio (σ_b/σ_f) are compared with theoretical estimates of the same computed on the basis of the Thomas-Fermi atomic charge distribution. Fair agreement is noticed between the two ratios whenever the diminution in the value of the cross section due to binding effects does not exceed about 10 percent. In other cases, progressively increasing deviations between the two values of the ratio (σ_b/σ_f) are found towards decreasing energy and increasing atomic number. The deviations are attributed to systematic errors inherent in the theoretical estimates made on the basis of the Thomas-Fermi model.

1492. B. G. Bennett and H. L. Beck, Legendre, TSCHEBYSCHEFF, and Half-Range Legendre Polynomial Solutions of the Gamma Ray Transport Equation in Infinite Homogeneous and Two Media Plane Geometry, HASL-185 (August 1967). Availability: CFSTI, $3.00, $0.65.

The polynomial expansion matrix equation method for obtaining semi-analytical solutions of the one-dimensional Boltzmann equation has been extended for use with full range Legendre polynomials, Tschebyscheff polynomials, and half-range Legendre polynomials. Each of the polynomial approximations has been used to calculate the differential energy spectra and angular distributions for various gamma-ray source distributions in several scattering media.

Comparisons of our infinite medium results were made with the exact calculations at the source energy, a careful experimental measurement of the energy spectrum from a 0.662 MeV point isotropic source in an effectively infinite water medium, the differential energy spectra and buildup factors from Goldstein and Wilkens' moments method calculations, and with the angular distributions from a moments method calculation for a 1 MeV plane isotropic source. The comparisons of these results show good agreement. In general, we find that at distances close to the source plane, the half-range Legendre approximation provides the most accurate solutions. At greater distances, use of full range Legendre approximations is preferable. The Tschebyscheff solutions give slightly improved angular flux values in the extreme forward and backward directions but provide somewhat less accurate differential energy flux spectra than the corresponding order Legendre approximations. We concluded that the low order P3 and/or DP1 polynomial expansion methods can be used to obtain the differential energy flux to better than 5% accuracy on the average for distances of from .1 to 10 mean free paths. The angular distributions are somewhat less accurate.

We have extended this method to two media boundary problems. Several examples of plane isotropic sources on the soil-air interface were considered. The differential energy spectra in air above the interface show slight reductions in scattered energy flux at the low energy part of the curve as compared to infinite air medium results. This is in accord with qualitative expectations, since the interaction cross sections for air and soil when normalized to the same electron density differ appreciably only below about 200 keV. For a .662 MeV plane isotropic source on the interface, the actual differences in scattered flux exposure rates for the soil-air medium as compared to infinite homogeneous air medium calculations range from a 5% reduction at 1 meter above the interface to 2% at 1 mean free path. Reasonable

1492. (Continued)

results were also obtained for sources distributed exponentially with depth in the soil. In this case the use of infinite medium results gives even more significant differences in exposure rates compared to the more accurate two media treatment.

The use of this efficient and practical analytical method has allowed a wide range of comparisons of polynomial approximations and of the physical parameters of the gamma-ray propagation in the analysis of several problems with infinite medium and two media configuration. These investigations suggest that this method can be applied advantageously to similar two or even three region slab geometry problems in weapons effects, shielding, and environmental radiation for various types of materials and plane source distributions.

1493. S. Pearlstein, An Extended Table of Calculated (n,2n) Cross Sections, Nuclear Data, Section A, 3, 3, 327-341 (October 1967).

The table presents, for some 500 stable and unstable targets, computed values of the (n,2n) cross section for three neutron energies in the neighborhood of 14 MeV and for a fission neutron spectrum. Cross sections for (n,2n) for fission neutrons are also given. The calculations are based on the statistical model and on empirical expressions for inelastic cross sections and level density.

1494. L. R. Bunney and D. Sam, "Exposure Rates from the Products of Fast Neutron Fission of 235-U and 238-U at Selected Times After Fission," Health Phys. 13, 1033-1037 (1967).

The exposure rates per fission per square centimeter at 3 ft above a uniformly contaminated plane were calculated for nine selected times ($\frac{1}{4}$, $\frac{1}{2}$, 1, 2, 5, 10, 24, 48, and 72 hr) after fast neutron fission of ^{235}U and ^{238}U. The calculations were made from recent experimentally determined gamma-ray spectra of the fission products. The results of the calculation of exposure rates from these experimental gamma-ray spectra of unfractionated fission products of ^{235}U and ^{238}U are compared graphically with those based on calculated spectra and with those computed from the few earlier experimental spectra taken at comparable times.

1495. N. Oi, I. Tanabe, S. Takayanagi, and Y. Matsushima, "Gamma-Ray Spectra of Fission Products Observed with Li-Drifted Germanium Detectors (II)," J. Nucl. Sci. Technol. 4, 7, 372-377 (July 1967).

The gamma-ray spectra of short-lived fission products from thermal neutron irradiation of highly enriched U were observed with an encapsulated Li-drifted Ge gamma-ray spectrometer. The spectra at various periods -- 10 min, 30 min, 1, 2, 5, 10, and 20 hr -- after irradiation were measured up to about 1 MeV. The relative activities of fission products at various periods after irradiation (10 min-20 hr) were calculated and used for assigning photopeaks.

1496. H. Schultz and C. D. Wueneke, "Radiation Field of a Disk-Shaped Source in an Absorbing Half-Space," ORNL-TR-1637 (Transl. from Atomkernenergie 11, 247-50, May-June 1966) (1966). Availability: Scripta Technica, Inc. 275 Madison Avenue, New York, New York 10016.

Using a calculational model, the penetration into a homogeneous medium of monoenergetic gamma rays or fast neutrons from a disk-shaped source was studied. Detailed tables of the flux density were calculated with an electronic computer. Approximate procedures and nomograms associated with them were developed to aid in the practical application of model experiments and to facilitate rapid surveys of the radiation field.

1497. S. A. W. Gerstl, "Double P-1 Approximation for Gamma Rays," ORNL-TR-1539 (Transl. from Atomkernenergie, 10, 432-5, Nov.-Dec, 1965) (1967). Availability: Scripta Technica, Inc., 375 Madison Avenue, New York, New York 10016.

This paper deals with the solution of the space-, energy-, and angle-dependent transport equation for the differential energy flux of gamma rays incident on a slab of finite thickness from a plane, monoenergetic source with an arbitrary angular distribution. The double P_1 approximation, well known from the theory of neutron shielding and often called "Yvon's method," is applied to the problem of gamma-ray transport. It has two basic advantages over methods heretofore applied to calculating the penetration of gamma rays through slabs of finite thickness, namely:

1. An improvement of Yvon's method allows the exact treatment of strongly anisotropic scattering cross sections.

2. The boundary conditions in slab problems can be treated exactly.

The differences in principle between the ordinary P_L approximation and this modified method of calculation are discussed.

Numerical calculations have been carried out using the double P_1 method; their accuracy corresponds to the ordinary P_5 approximation. In addition to the energy spectrum in the interior of the slab, the spectra of the reflected and transmitted energy fluxes are given and compared with Monte Carlo calculations.

1498. D. P. Osanov, "Temperature Distribution in Gamma-Ray Shielding," J. Eng. Phys. 8, 6, 563-67 (June 1965).

Analytical expressions are obtained for the temperature distributions in gamma-ray shielding for isotropic and unidirectional beams with boundary conditions of the third kind. These are solved numerically for concrete shielding, and the effects of shield thickness. Gamma-ray scattering, boundary conditions and beam geometry on the temperature distribution profile are examined.

1499. R. C. Lawson and D. E. Watt, Thermal Neutron Depth Dose Dependence on Beam and Phantom Size for Incident Collimated Beams of Fast Neutrons, PG-774(CC) (1967). Availability: HMSO, 2s. 6d NET.

Thermal neutron flux distributions produced at depth in tissue-equivalent liquid were measured for several collimated beams of incident fast neutrons with the object of obtaining data for calculating the $^{1}H(n,\gamma)^{2}D$ and $^{14}N(n,p)^{14}C$ doses delivered under conditions experienced frequently in radiobiological experiments.

It is demonstrated that two group theory can be used to predict accurately the distribution of thermal neutron flux with depth and also the position of the peak value.

A simple formula is given which permits the magnitude of the peak to be determined for any beam size between 5 cm and 17 cm diameter relative to that measured for one beam size. By a combination of these formulae a complete description of the thermal neutron flux at depth in tissue can be obtained.

Studies on the magnitude of the flux distribution caused by changes in the phantom dimensions lead to the conclusion that the naturally occurring statistical variations about the size of the standard man phantom do not produce an important change in dose level.

1500. W. R. Kimel, L. V. Spencer, J. C. LeDoux, A. B. Chilton, and C. Eisenhauer, Radiation Shielding, Analysis and Design Principles as Applied to Nuclear Defense Planning, Office of Civil Defense, Kansas State University TR-40 (November 1966). Availability: Superintendent of Documents, GPO, $6.00.

This publication is a compilation of lectures and papers presented at the Kansas State University Summer Institute on Fundamental Radiation Shielding Problems as Applied to Nuclear Defense Planning during the summers of 1962, 1963, and 1965. The topics covered are: gamma ray shielding theory, Engineering analysis (basis and methods), experimental program of Summer Institute, and the development of the Engineering Method and some simplified methods of structure shielding analysis. (RSIC)

1501. M. D. Goldberg, S. F. Mughabghab, S. N. Purohit, B. A. Magurno, and V. M. May, Neutron Cross Sections, Vol. IIC, Z = 61 to 87, BNL-325, Sec. ed., Sup. 2, Vol. 2C (August 1966). Availability: CFSTI, $3.00, $0.65.

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 elements whose atomic numbers are in the range 61 to 87.

The format differs from preceding versions in several ways. The more normal size, arrangement of all data by nuclide, the fluid format is likely to make this supplement more convenient for the user. (RSIC).

1502. R. M. Freestone, Jr., Monte Carlo Calculations of the Neutron Flux Within a $^{10}B_4C$ Shell, ORNL TM-1816 (June 9, 1967). Availability: Dep. CFSTI.

A Monte Carlo code, O5R, has been employed to study the transport of neutrons through a nearly spherical $^{10}B_4C$ shell. Such shells are often used to shield neutron detectors against the intense background of thermal neutrons associated with reactor or reactor-like neutron fluxes. An aluminum disk simulated a solid-state detector within the shell.

Calculations were made for isotropic monoenergetic neutrons having energies of 0.5, 1, 2, 4, 6, 8, 10, 12, and 14 MeV, for an isotropic fission spectrum source, and, for comparison purposes, for a 4-MeV normal beam source.

Results indicate that only a small distortion of the incident spectral shape is created by passage through the shell. The effect seems to be confined to incident neutron energies below 2 MeV. The degree of isotropy of the incident flux, however, importantly affects the absolute fraction of neutrons reaching the detector.

1503. M. G. Silk, The Determination of the Fast-Neutron Spectrum in Positions in the Daphne Reflector, AERE-R-5347 (1967). Availability: HMSO, 4s. 6d. NET.

The difficulties inherent in spectrum determinations in reactors at positions remote from the core centre are considered, and the suitability of lithium-6 and helium-3 semiconductor sandwich spectrometers for this work is discussed.

Spectra obtained in many positions in the DAPHNE reflector are compared with calculated spectra.

1504. W. L. Bunch and R. E. Shaver, Shielding Calculations for Conceptual Design of FFTF Sodium Systems, BNWL-388 (April 1967). Availability: CFSTI, $3.00, $0.65.

Calculations indicate that the sodium coolent in the Fast Test Reactor will attain saturated specific activities of the order of 0.1 Ci/cm^3 of ^{24}Na and 30 $\mu Ci/cm^3$ of ^{22}Na. About 400 g/cm^2 of shield material will be required to protect operating personnel from the gamma associated with the ^{24}Na activity. Following decay of this isotope (15 hour half life), about 100 g/cm^2 will provide adequate shielding. Fission products entering the sodium as a result of fuel cladding failures are not expected to increase basic shield requirements for the sodium systems, but will probably influence conceptual design of maintenance and handling techniques for the systems.

1505. O. K. Harling, Compilation of Doubly Differential Cross Sections and the Scattering Law for H_2O and D_2O at 299 degrees K and for H_2O at 268 degrees K, BNWL-436 (June 1967). Availability: CFSTI, $3.00, $0.65.

This report contains a compilation of slow neutron inelastic scattering cross sections for room temperature H_2O and D_2O and for H_2O at five degrees below its freezing point. The double differential cross section and, from these, the Egelstaff scattering law have been obtained from measurements using the Battelle Rotating Crystal time-of-flight spectrometer. Experimental conditions were designed to provide cross sections over a wide range of energy and momentum transfer space. Maximum energy transfers, by downscattering of 0.6, 0.21, and 0.6 eV with wave vector ($\bar{k} = \bar{P}/h$) transfers up to 32, 9.5 and 32$Å^{-1}$ were used, respectively, with H_2O, D_2O, and light ice. Such an extensive grid is required to accurately represent the scattering kernel for reactor thermalization calculations and for the derivation of a generalized frequency distribution, which is useful in understanding the dynamics and structure of the scatterer.

1506. J. F. Batter and A. W. Starbird, The Preparation of Simplified Manuals for Shielding Analysis, AD-652873 (March 1967). Availability: Dep. CFSTI.

Calculational methods currently used to predict shelter from radiological fallout are examined with respect to both experimental and analytical data such that simplified methods could be developed. Emphasis is placed upon "In and Down Scattering," the effects of finite fields of contamination and how interior partitions affect the dose rate in above and below ground areas.

The discrepancy found between calculation and experiment for in and down scattering is attributed primarily to the attenuation afforded by the basement ceiling. A preliminary analytical estimate of this attenuation, based upon Compton single scattering, is presented and a modified calculational technique suggested for use in shelter calculations.

Little full-scale theoretical and experimental data on the dose contributed by limited strips of contamination exists of the current time. Fair agreement exists between values computed by the engineering method of shelter calculation and experiments on scale models. The simplified calculational technique at times creates values that differ from the more complex techniques. A modified simplified technique is suggested that provides significantly better agreement.

The existing methods of computing the effects of interior partitions in above-ground areas shows good agreement with what few experiments that are currently available. There is a strong indication, though that the dose attributed to the radiation scattered in the walls of the structure is underestimated.

1507. W. A. Coleman, R. E. Maerker and F. J. Muckenthaler, Differential Neutron Current Albedos for Concrete in the Incident Energy Range 0.5 eV to 200 keV. I. Descriptions of Monte Carlo Calculation and TSF Experiment and Comparison, ORNL-3967, Vol. I (March 1967). Availability: CFSTI, $3.00 cy, $0.65 mn.

Extensive Monte Carlo calculations were performed to determine the distribution in energy and angle of neutrons reflected from steel-reinforced concrete for five incident directions and ten incident energy groups extending from 0.5 eV to 200 keV. The reflected distributions were determined in terms of a doubly differential albedo for each of 54 emergent directions for each energy group lying between and including the incident group and the tenth group (0.5 to 1.8 eV). The standard deviation of the doubly differential albedo averaged around 10%. The angular slowing-down density of the incident epicadmium neutrons within the slab was computed at 0.5 eV and used as the source distribution for a Monte Carlo single-velocity diffusion calculation using 0.025-eV cross sections. From the diffusion calculation, the differential angular albedos of the reflected subcadmium neutrons and the depth distributions of captures occurring at subcadmium energies were obtained. Measurements of the differential angular albedo of emergent subcadmium neutrons due to a measured spectrum of incident monodirectional beams of epicadmium neutrons were performed at the ORNL Tower Shielding Facility in an experiment geometrically identical to that previously reported for incident subcadmium beams. Of the 35 common points of calculation and measurement, the two largest discrepancies were 23% and 36%; the remaining 33 comparisons produced a root mean square deviation of 4.5%.

1508. W. A. Coleman and R. E. Maerker, Differential Neutron Current Albedos for Concrete in the Incident Energy Range 0.5 eV to 200 keV. Results of Monte Carlo Calculations -- 0, 45, 60, 75, and 85 deg., ORNL-3967, Vols. 2-6 (November 1966). Availability: CFSTI, $3.00, $0.65 mf for each volume.

This report is one of a series of six reports presenting calculated differential albedos for monodirectional beams of 0.5 eV to 200 keV neutrons incident on concrete and comparing calculated and reflected and subcadmium neutron currents with experimental results obtained at the ORNL Tower Shielding Facility. The bulk of the calculations were performed with the 05R Monte Carlo computer code and covered ten different source energy bands for each of five angles of incidence of a "gun barrel" source of monodirectional neutrons. Volume I describes the experimental and calculational procedure, compares experimental and calculated results of the reflected subcadmium component, 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.

1509. A. Shimizu, Tabulation of Dose Transmission Factors for Homogeneous Slabs, NBS Report 9617 (September 21, 1967). Availability: Office of the Director, National Bureau of Standards, Washington, D.C.

A series of calculations on the penetration of gamma rays through homogeneous slabs has been made by the method of invariant imbedding. The dose transmission factors are tabulated for fifteen source energies in the range from 0.66 to 10.0 MeV, for seven source obliquities, for slab thickness up to fifteen mean free paths, and for seven materials.

1510. A. Shimizu, Tabulation of Dose Transmission Factors for Two-Layer Slabs, NBS Report 9618 (September 21, 1967). Availability: Office of the Director, National Bureau of Standards, Washington, D.C.

The method of invariant imbedding has been applied to the penetration problem of gamma rays through two-layer slabs, and turned out to be accurate and less time-consuming than the Monte Carlo method. A series of calculations were made for two-layer slabs of water, iron, and lead for plane oblique sources. An approximate formula for synthesizing the buildup factor of a composite slab from those of elementary layers is derived from numerical solutions by modifying the formula originally proposed by Kalos.

1511. F. A. Bryan, M. D. Weight, A. B. Nicholls, A. J. Benjamin, and J. C. Wright, Initial Radiation Spectra Definition. Final Report, DASA-1934 (RTI Report No. R-OU-246) (May 1967). Availability: Research Triangle Institute, Research Triangle Park, North Carolina.

Calculations have been performed to determine the initial gamma radiation spectra from nuclear weapons as a function of weapon-shelter distance. The purpose of these calculations was to determine standard gamma spectra which could be used in initial weapon radiation shielding calculations.

The report describes the calculations by which the initial radiation spectra were determined. The sources of photons that were considered in the calculations included prompt fission gammas, early fission product gammas, and gammas due to radiative neutron capture by air nitrogen. It is shown that the differential energy spectra associated with each of the photons sources obtain a pseudo-equilibrium in their scattered components. These pseudo-equilibria obtained after photon penetrations in air of 1.0 to 1.5 miles. It is recommended that the one-mile distributions, as defined by the calculations, be used as standards for initial radiation shielding calculations.

1512. L. R. Bunney and D. Sam, Exposure Rates from the Products of Fast Neutron Fission of U-235 and U-238 at Selected Times after Fission, USNRDL-TR-1106 (October 28, 1966). Availability: U. S. Naval Radiological Defense Laboratory, San Francisco, California.

The exposure rates per firssion per square centimeter at three feet above a uniformly contaminated plane were calculated for nine selected times (1/4 to 72 hours) after fast neutron fission of U-235 and U-238. The calculations were made from experimentally-determined gamma-ray spectra of the fission products.

1513. P. Kirkegaard, Comparison Between Gamma-Shielding Calculations, Using Buildup Methods and Monte Carlo Methods, Respectively, RISO-M-528 (February 1967). Availability: Dep. mn.

This paper describes a comparison between calculations carried out by two different types of γ-shielding computer programmes developed at Riso. One type uses the build-up method (P-18 PRIGAM, P-19 SEGAM I, P-20 SEGAM II), and the other the Monte-Carlo method (P-220 MC 4).

The comparisons are confined to iron-water and lead-water shields, and it is shown that the simple and quick build-up programmes are satisfactory in most of the practical cases.

Further, the report contains a description of a change in the formalism for constructing build-up factors for laminated media.

1514. M. Schumchyk, M. Schmoke, E. Schulman, and E. Pollock, Scattered Radiation (Skyshine) Contribution to a Concrete-Covered Basement Located in a Simulated Fallout Field, NDL-TR-69 (AD-654141) (July 1967). Availability: Nuclear Defense Laboratory, Edgewood Arsenal, Maryland.

This report determines, experimentally, the effect of overhead mass thickness on the shielding characteristics of a concrete-walled basement located in a simulated fallout field and compares the results with theoretical calculations set forth in NBS Monograph 42.

A uniformly contaminated residual gamma radiation area was simulated to a radius of 600 feet with a cobalt 60 point-source circulation system. Experimental exposure-rate measurements were obtained in the free field and also within a concrete-covered basement as a function of height above the basement floor. Ionization chamber dosimeters were used as radiation detectors. Experimental measurements were extrapolated by analytical procedures to infinite field conditions and divided by the infinite free-field exposure rate measured 3 feet above ground to find the reduction factors. These reduction factors were compared with the theoretical results.

1515. Lockheed Georgia Nuclear Laboratory, Evaluation of Methods for Computing Nuclear Rocket Radiation Fields, ER-8236 (NP-16954) (January 1966).

The report describes critical evaluations of a series of computer programs suitable for theoretical predictions of radiation fields in RIFT vehicles and other nuclear rocket systems. Ten computer programs covering point-kernel, discrete ordinates and Monte Carlo methods have been examined as a part of this study.

1516. D. C. Irving, Evaluation of Neutron Cross Sections for Boron-10, ORNL-TM-1872 (ENDF-109) (October 9, 1967). Availability: Dep. mn.

The neutron cross sections for ^{10}B have been evaluated from 10^{-4} eV to 15 MeV. The existing experimental data are reviewed, and theoretical calculations and other reasoning are used to fill in the gaps. A complete and consistent set of cross sections is presented and an explanation is given for the choices made in developing this cross section set.

1517. R. Beaudry, R. Mantey, H. Hungerford, and W. Chaltron, Design of the Radiation Shield System for the Enrico Fermi Atomic Power Plant, APDA-203 (May, 1967). Availability: CFSTI 3.00 HC, 0.65 nef.

The raidation shielding of fast reactor systems such as the Enrico Fermi fast breeder reactor poses many special shielding problems which are not encountered with thermal reactors.

The primary shield which surrounds the Fermi reactor vessel is quite complex, consisting of a stainless steel thermal shield within the reactor vessel, and a steel, graphite, and boronated graphite shield outside of the reactor vessel. The rotating plug (part of the primary shield system), which allows access to the reacotr for fuel handling and control, is a complicated structure whose shield is composed of stainless steel, plain steel, boron steel, and plain and boronated graphite. Special methods and materials were required to minimize radiation streaming through the rotating plug and around the fuel-handling port.

In addition to the shielding for the Reactor Building, shielding was designed for: (1) the primary sodium service system; (2) the primary inert gas (argon) system; (3) the waste-gas system; (4) the secondary sodium coolant system (leakage radiation); (5) the Steam Generator and the Control Buildings; and (6) the Fuel and Repair Building. The main Material used in the shields was ordinary concrete, supplemented with steel, steel shot, sand, gravel, and magnetite, as necessary and desirable.

Each shield area listed above is described in detail in this report, and important engineering features are amplified and illustrated with charts, diagrams, and photographs.

1518. T. Jordan, Energy-Independent Biasing Functions for Monte Carlo Transport Calculations, Douglas Paper No. 3636 (November, 1965). Availability: Douglas Aircraft Co., Inc., Missile and Space System, Div. Santa Monica, California.

This paper describes analytical and numerical techniques developed at Douglas for Monte Carlo solution of the Boltzmann equation. These techniques permit the simultaneous treatment of a multitude of statistical samples of different energies, and a stratified sampling of every possible scattering event. This is accomplished through the consistent use of energy-dependent samples which span the entire energy range.

The application of these techniques eliminates many similar calculations and can result in order-of-magnitude reductions in the computer time required for shileding problems. This is

particularly true for problems involving complicated geometries, energy-dependent sources, multiple scattering events, and point detectors. Typical numerical results for several idealized shielding problems, obtained using these techniques, are presented to demonstrate the accuracy which can be obtained with moderate computer times. Comparisons are also made with correlated biased analog calculations to indicate the overall Superiority of energy-independent biasing.

1519. T. Auerbach, J. Mennig, P_N-Theory in r-θ-Geometry, EIR-Bericht Nr. 115 (April, 1967). Availability: Eidg. Institut für Reactor forschung Würenlingen, Switzerland (NSA: 21: 29825).

Kofink's method of solving the transport equation in P_N-approximation (1) has been extended to r-θ-geometry. It is shown how the treatment may be applied to multigroup theory. Two types of interface boundary conditions are discussed. Since the method is primarily intended for use with heterogeneous theory, no cell boundary is defined. Instead, there is a condition on the asymptotic behavior of the scalar flux. Numerical results are given for monopole and dipole extrapolation lengths at the surface of a natural uranium rod immersed in D_2O.

The present report wes first published as TM-PH-180 (2). A few minor modifications were made in the text.

1520. M. Mendelson, S. Congdon, Particular Solutions of the One-Speed Transport Equation, KAPL-P3335 (1967). Availability: Knolls Atomic Power Lab., Schenectady, N. Y.

In Cases' formalism, the general solution of the non-homogeneous equation is constructed from a particular solution and the solution of the homogeneous equation. The purpose of this note is to present a methos for obtaining particular solutions of the one-speed transport equation for a general source, $S(x,\mu)$. (a) Explicit examples are given for sources of the form $x^n P_m(\mu)$, which by superposition can be used to expand an arbitrary spatial and angular source diatribution.

1521. J. Gylfe, Reactor-Shield Subsystems for Manned Orbiting Research Laboratories (MORL), NAA-SR-MEMO-12373, Vol.1 (June 15, 1967). Availability: Atomics International, H. Div. of North American Aviation, Inc., Canoga Park, California.

This report presents the results of the work performed by Atomics International under contract to the AEC for the joint NASA/AEC study to determine the design requirements of reactor power systems for large manned orbiting laboratories, as typified by the MORL. The scope of AI's work included design and analysis of the reactor and its associated control system, radiation shielding, and the primary coolant system for nuclear power plants utilizing mercury Rankine, Brayton, and thermoelectric power conversion systems.

1522. H. Blinn, R. Lane, M. Mayer, C. Thompson, J. Hanson, R. Markley, P. Tauson, Space Power Plant Study, X-65-18835 (NASA-CR-64507) (WANL-PR(B)-009) pp 4-23 to 4-47 (December 31, 1963). Availability: Westinghouse Astronuclear Laboratory, P. O. Box 10864, Pittsburgh, Pennsylvania 15236.

This report presents the final results of a portion of the space power plant study conducted under contract NAS 5-250.

The study evalustes cesium and potassius as working fluids in an advanced high-temperature Rankine cycle power conversion system. The evaluation was accomplished by designing major components of cycles using each working fluid and comparing the results.

It is necessary to have as high a maximum cycle temperature as possible in order to minimize power plant weight. The creep of the first turbine rotor is probably the most serious upper temperature limitation on Rankine cycle power plants for space applications. Consequently, this was adopted as a design criterion. With this design basis, the maximum cycle temperature for the system using cesium was higher than for the system using potassium.

The study was conducted for 1 Mw shaft output power but it is believed that the conclusions drawn from comparing the two systems on a plant weight basis are generally valid. The total weights of the major components of the power plants compared in this study are 3894 lbs for cesium and 4883 lbs for potassium. This is a consequence of the difference in radiator size.

Because of the high expense and long development time of high temperature alkali metal Tankine Cycle technology, it is necessary to select the working fluid with the greatest potential as early as possible. For this reason, it is recommended that cesium should be seriously considered for such applications.

In addition to the work described above, some reactor and shielding studies were performed. A reactor thermal hydraulic code and a buckling iteration neutron diffusion theory code were written.

1523. S. Mathur, I. Morgan, P. Buchanan, A Compilation of Gamma Ray Angular Distribution Coefficients, ORO-2791-21 (August, 1966). Availability: Texas Nuclear Corp., Austin, Texas. CFSTI.

The theoretical formulation for the calculation of gamma ray distributions in (η, Gγ) processes, where η represents the ingoing nucleon and G the outgoing nucleon, has been developed by Satchler.

The theoretical formulation of the (η, Gγ) processes follows the familiar compound nuclear statistical continum approach. It is assumed that the predominant reaction mechanism is the formation of a compound nucleus and subsequent decay into a residual nucleus, which de-excites with the emission of gamma rays.

The gamma ray angular distribution function consists of a Legendre polynomial expansion weighted by suitable transition parameters and by Hauser-Feshbach penetrability terms. The transition paramenters depend on the initial and final nuclear spins involved and the angular momenta of the nucleons involved in the transition.

This report presents a compilation of angular distribution coefficients computed on the basis of Satchler theory. These coefficients can be used for hand calculations, or in conjunction with a computer program, for the calculation of gamma ray angular distributions in (η,Gγ) reactions. The compilation presents the angular distribution coefficients for approximately 20,000 spin sequences with ground state spins of 0, 1/2, 3/2, 5/2, 7/2, 9/2, 11/2, 13/2, and 15/2.

1524. Department of the Army, X-Ray Shielding, TM-5-805-12 (NP-16296) (August, 1966). Availability: Army, Headquarters, Washington, D. C.

This manual provides guidance for design of medical and dental X-ray shielding in floors, walls, and ceilings. The requirements included are intended to fully satisfy the criteria which a qualified expert would apply under the most rigid conditions. This manual is limited to design of shielding for typical dental, radiographic, fluoroscopic and therapeutic X-ray installations up to 300,000 volts.

1525. H. Dunster, The Application and Interpretation of ICRP Recommendations in the United Kingdom Atomic Energy Authority, AHSB(Rp) R-78 (June, 1967). Availability: H. M. Stationary Office. Price 3s. 6d. NET.

The dose limits in this report are with one exception the same as those recommended by ICRP. The intention to adopt these figures in the Authority has already been notified to the Health Committee of the Authority and endorsed by the Authority Committee on Health and Safety, which has withdrawn the two Health and Safety Codes. This report summarises the dose limits and makes recommendations on some aspects of their practical application. Recommendations not originating from ICRP, or implying extensions or modifications of ICRP recommendations, are clearly identified in the text, usually by context.

1526. G. Bartholomew, K. Eastwood, L. Groshev, V. Pelekhov, A. Doveika, S. Monaro, A. Demidov, L. Sokolovskii, Compendium of Thermal-Neutron-Capture γ-Ray Measurements, Nucl. Data, 3, 4-6, 367-645 (1967).

A compilation is presented of thermal-neutron-capture γ-ray energies and intensities with decay schemes, examples of original data, and fully corrected neutron radiative-capture cross sections, spins, parities, and other information of value to neutron-capture γ-ray spectroscopy is included.

1527. F. Clark, D. Trubey, Energy and Dose Buildup Factors for Various Concretes, Nuclear Appl. 4,1, 37-41 (September 20,1967).

Buildup factors have been determined by moments method calculations for ordinary, magnetite, and barytes concretes. They agree well with approximate evaluations in the range 3 to 6 MeV. Outside that range, there is noticeable disagreement.

1528. E. Fleming, The Fission Product Decay Chains (239-Pu with Fission Spectrum Neutrons), UCRL-50243, Vol. 1, Vol.2, Vol.3 (March 31, 1967). Availability: Lawrence Radiation Laboratory, Livermore, California.

Three parameters for each radionuclide in eadh fission product mass chain have been computed and plotted vs. time from 0.6 seconds to about two years. The parameters are: (1) the (atom) fraction of the total chain yield, i. e. the

ratio of the number of atoms of a given nuclide in a mass chain to the sum of the atoms of all nuclides in the mass chain; (2) the number of kilocuries per kiloton of fission; (3) the number of roentgens per hour per kiloton per square mile. Each decay chain diagram is presented together with the folowing values for each nuclide in the chain - the half-life, the independent yield, the photon dose-rate multiplier, the zero time atom fraction, the kilocuries per kiloton per fraction and the roentgens per hour per kiloton per square mile per fraction. Total fission product sums of KCi/kt and of $R \cdot Mi^2/h \cdot kt$, each vs. time from 0.6 seconds to about two years, have been computed and plotted.

1529. C. Huddleston, N. Shoemaker, Monte Carlo Calculations of Gamma-Ray Albedo, AD 621 441 (USNCEL TN-N-764) (September 7, 1967). Availability: CFSTI.

A Monte Carlo Computer program has been used to generate values for the differential dose albedo of gamma rays on concrete. The values have been fit to a semiempirical formula containing two adjustable parameters. Values for the parameters are reported as a function of gamma-ray energy.

1530. C. Huddleston, N. Shoemaker, Calculations of the Energy-Angular Relationship for Neutrons from a Neutron Generator, AD 621 473 (USNCEL TN-N-763) (September 2, 1968). Availability: CFSTI.

A relativistic calculation is made to determine the energy of neutrons emitted from the NCEL neutron generator as a function of the angle of emission. Both the tritium reaction, $T(d,n)He^4$, with a neutron energy of approximately 14 MeV, and the deuterium reaction, $D(d,n)He^3$, with a neutron energy of approximately 2.5 MeV, are considered. Values of neutron energy as a function of angle are tabulated and plotted.

1531. L. Faust, Measured and Calculated Surface Dose Rates of Plutonium and Plutonium Oxides, BNWL-SA-22 (June 7, 1965). Availability: CFSTI.

An extrapolation chamber was used to measure the X-ray and gamma ray surface dose rates of metallic plutonium, PuO_2 and PuO_2-UO_2 mixtures. Data are presented showing the changes in the plutonium isotopic composition. The surface dose rate of plutonium metal was compared to the dose rate after the metal was oxidized. Surface dose rate values are presented for different weight per cents of PuO_2-UO_2 mixtures. Table I

summarizes the measured values for the different materials.

Measured and calculated surface dose rates for plutonium metal were found to be in excellent agreement; they are 1.69 and 1.62 rad/hr respectively. An equation to calculate the surface dose rate of plutonium due to neutrons is presented but no direct comparisons between the equation and measured surface dose rate values are made. Attenuation factors are presented for plastic "transfer bagging" material and hood gloves currently in use at Pacific Northwest Laboratory - Battelle Memorial Institute.

1532. Second Status Report Shielding Standard Design Method, WANL-TME-1466 (March 1, 1968). Availability: CFSTI.

This report is the second in a series of reports describing the Shielding Standard Design Method employed by WANL to perform NERVA radiation and shield analyses and design. This method, designated Shielding Standard Design Method II (SSDM-II) is the second generation of methods which has evolved from evaluation comparisons of calculated versus experimental radiation environment data. SSDM-II employs three widely-used analytical techniques; Monte Carlo, S_n transport code, and point kernel. Monte Carlo Program 18-0 is specifically used to calculate gamma ray heating within the pressure vessel. The two dimensional S_n transport code, TDC, generate neutron fluxes by group in an r z geometry. Point kernel Program is used to compute external radiation environment. Many subsidiary linked programs are also employed to generate input data and process output data.

Extensive somparisons of SSDM II calculated data with NRX experimental data and other applicable experimental data are presented. The experimental data include internal and external reactor gamma ray dose rates external fast neutron ($E > 2.9$ MeV) and thermal neutron environment, and liquid hydrogen propellant heating data.

A continuous effort is in progress to improve and to more fully automate the SSDM II system. These improvements, Which are described in detail in the report, include : (1) replacing the TDC transport code with the DDK two dimensional transport code for more accurate calculation of neutron fluxes, and (2) replacinf the 14-0 point kernel code with the WANL KAP point kernel code. Two unique neutron-photon S_n transport code system (for one and two dimensional analysis) are being developed. Application of the S_n transport to photon transport analysis is expected to facilitate radiation analysis of reactor-shield systems.

Although the WANL SSDM II and its planned improvements were developed for analysis of the NERVA system, almost all computer codes can be employed in analysis of other types of reactor systems and isotope heat source systems. It is felt that the methods and their applications described here will benefit others associated with radiation analysis and shield design.

1533. P. Schreiber, F. Kodras, Data Report of Radiation Level Measurements within and Behind Beryllium, DC-61-369 (February 14, 1961). Availability: CFSTI.

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 measurements and thermal flux measurements 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.

1534. J. Moteff, Proposed Two-Component Method of Nuclear Shield Analysis, DC-60-4-87 (April 13, 1960). Availability: CFSTI.

When broad beam removal cross sections are used in the conventional Albert-Welton IBM shielding programs a certain fast neutron dose rate will be obtained for the 50-foot position. Assuming a dose rate uncertainty factor of 2 the side shield material weight for the ACS P140 power plant can be given as 10,620 13.6% pounds.

If the Albert-Welton IBM shielding program is slightly modified to use collimated removal cross sections; and the analysis also includes the dose contribution from the surface of the shield a lower dose rate is obtained for the 50-foot position. This new dose rate indicated that the side shield material weight can be given at 7850 9.5% pounds if the original neutron dose rates are desired. Although the neutron dose rates are changed by large factors for the LiH thicknesses considered, the gamma dose rates are only changed by about 30%. This side shield gamma increase would not change the crew shield dose rate by note than a few per cent.

An alternate performance gain would be to reduce the crew

shield dose rate by more than a few per cent.

An alternate performance gain would be to reduce the crew shield weight, which for a nearly optimum design would result in an equivalent weight savings.

The two component shield analysis method that is proposed in this report is compared to existing experimental data. The significance of the neutron spectrum is also discussed.

1535. W. Selph, H. Claiborne, Methods for Calculating Effects of Ducts, Access Ways and Holes in Radiation Shields, ORNL-RSIC 20 (DASA-1892-1) (January, 1968). Availability: CFSTI.

A review of methods and data is presented for calculating radiation transmission through shields containing ducts or voids. Analytic, ray analysis, Monte Carlo, albedo, and other methods are described. RSIC.

1536. W. Selph, Neutron and Gamma-Ray Albedos, ORNL-RSIC 21 (DASA 1892-2) (January, 1968). Availability: CFSTI.

A review of the determination and application of neutrons and gamma-ray albedos in radiation transport calculations is presented. Many types of albedos are treated including differential and total dose albedos and variation with neutron energy. Capture, activation and inelastic-scattering gamma-ray, albedos are also considered. RSIC.

1537. J. Hopkins, H. Conde, D. Drake, Elastic and Inelastic Scattering of Fast Neutrons from ^{6}Li and ^{7}Li, LA-3765 (November 21, 1967). Availability: CFSTI.

The differential elastic and inelastic neutron scattering cross sections of ^{6}Li and ^{7}Li have been measured at incident neutron energies of 4.83, 5.74, and 7.5 MeV for ^{6}Li and at energies of 3.35, 4.83, 5.74, and 7.5 MeV for ^{7}Li. Scattered neutrons and gamma rays were observed independently. The cross sections were measured with a neutron time-of-flight spectrometer relative to the well-known cross section for neutron scattering from hydrogen. The gamma-ray spectra were measured with a NaI(Tl) spectrometer using time-of-flight techniques to eliminate the neutron background. The 2.184-MeV state in ^{6}Li was was excited by neutron inelastic scattering. The 3.56-MeV

state in ^{6}Li was not observed in either the neutron or gamma-ray studies. Scattered neutrons from the 0.478-MeV state in ^{7}Li were resolved at both 3.35 and 4.83 MeV. Scattered neutrons from the 4.63-MeV state in ^{7}Li were observed at incident neutron energies of 5.74 and 7.5 MeV. A continuum of neutrons due to three-particle breakup was observed for both ^{6}Li and ^{7}Li at 4.83, 5.74, and 7.5 MeV.

1538. G. Hansen, H. Sandmeier, The Effect of Basic Neutron Reaction Cross Sections of Nitrogen (n,n′), (n,2n), (n,γ), (n,p) and (n,α) on High Energy Neutron Penetration in Air, LA-3810 (December 7, 1967). Availability: CFSTI.

In an evaluation of an integral quantity one is interested in the relative sensitivity of this quantity to the basic input data.

This report assesses numerically the influence of the basic reaction cross section of nitrogen, the major constituent of air, on the penetration of high energy neutrons in air. The reaction cross sections investigated are (n,n′), (n,2n), (n,γ), (n,p), and (n,α). The collision cross section is kept constant in all perturbations and the above individual reaction cross sections are assumed to become elastic scattering collisions, i. e., (n,n′) → (n,n) . . . . . (n,p) → (n,n). For a 10 m spherical source of (12-14) MeV neutron we evaluate unperturbed and perturbed neutron fluxes as a function of energy and distance. Numerical values for both flux and energy spectra are given at a distance of 825 m. As a representative example of integral data we evaluate the first collision dose to soft tissue for the perturbed and unperturbed neutron flux. It is concluded that the (n,n′) → (n,n) and the (n,p) → (n,n) perturbations result in an increase of the dose by a factor of 2 whereas the (n,α) → (n,n) perturbation increases the dose by a factor of 1.8. The effect of the (n,2n) and (n,γ) perturbation couses insignificant perturbations in the first collision dose. One of us (H. A. S.) in a classified report has also investigated the ingluence of the above perturbations in the basic nitrogen cross section data to the neutron vulnerability of nuclear weapons.

The present report follows a previous publication including the same authors where the effect of claculational methods of the deep penetration of high energy neutron in air has been investigated.

1539. A. Frank, Spectral Measurements of Scattered Photons from a Point Source Near the Air-Ground Interface, USNRDL-TR-67-124 (October 10, 1967). Availability: U. S. Naval Radiological Defense Laboratory, San Francisco, California 94135.

A gamma-ray spectroscopy experiment was performed to measure the scattered photons from a point source near the air-ground interface. Source-to-detector radial distances of 9.7 and 20.6 ft. were used for a Cs^{137} source and 9.7 ft. for a Co^{60} source. The source height was 8 in. and the detector height wes about 3 ft. The measured flux was separated into 10 degree vertical angular bins and the data were reduced to photon spectra by a matrix iteration procedure.

Comparisons were made between the measurements and the corresponding Monte Carlo calculated spectra. The angular exposures were also compares and the buildup factors were presented along with other buildup data from the literature.

1540. P. Klann, W. Paulson, Comparison of Absoluted Calculated and Measured Gamma and Neutron Doses in Tungsten - Water-Moderated Critical Assembly, NASA TN D-4223 (November,1967). Availability: CFSTI.

The mixed gamma and neutron dose was measured at 60 locations within the Tungsten - Water-Moderated Reactor critical Assembly. Graphite wall thimble ionization chambers filled with carbon dioxide and polyethlene wall chambers filled with ethylene were used. The chambers were absolutely calibrated in a bremsstrahlung beam against a secondary calibration standard and in a reactor against a water filled calorimeter. These calibrations were used to obtain gamma and neutron response coefficients for the chambers which permitted partitioning of the measured mixed radiation dose into a gamma dose and a neutron dose. The measured gamma doses were compared with a ATHENA Monte Carlo calculation. Good agreement was found for the 16 locations compared with a "first collision" calculation of the dose. The calculated neutron doses were uniformly low with an average deviation from the measurement of 18 per cent.

1541. M. Anderson, Neutron Energy Spectra of ^{239}Pu-Be, ^{238}Pu-F, and ^{238}Pu-$^{18}O(\alpha,n)$ Sources, MLM-1422 (October 31, 1967). Availability: CFSTI.

The energy spectra of isotopic neutron sources are being determined with a single stilbene crystal, fast neutron spectrometer. Pulse shape discrimination is used to separate

pulses originating from neutron interaction and those originating from gamma interactions. Spectra were obtained for (α,n) sources containing plutonium-239 with beryllium, plutonium-238 with fluorine, and plutonium-238 with oxygen-18.

1542. R. Maerker, F. Muckenthaler, Calculations, Using the Albedo Concept, of Thermal-Neutron Fluxes, Epicadmium-Neutron Fluxes, and Fast-Neutron Dose Rates Along the Center Lines of Square Concrete Ducts: Comparison with Experiment, ORNL-4147 (October, 1967). Availability: CFSTI.

Monte Carlo calculations that use the albedo concept and a particularly demanding source geometry and spectrum have been carried out to determine subcadmium- and epicadmium-neutron fluxes and fast-neutron dose rates along the center lines of one-, two-, and three-legged square concrete ducts open on the source end. Secondary gamma-ray dose rates arising from the capture of neutrons in the duct walls were also calculated. The calculations used previously reported albedo data that are differential in both the reflected angles and the reflected energy. A comparison of the calculated neutron fluxes and dose rates with those obtained in geometrically identical experiments conducted at the Tower Shielding Facility shows agreement generally within 25 percent through five orders of duct attenuation. The results thus place on a firm foundation the method of treating neutron transmission through large-sized ducts as a multiple-reflection phenomenon describable by the differential albedo properties of the wall. The conclusion is also reached that wall-capture gamma-ray dose rates arising from capture of thermal-neutrons and the dose from the thermal neutrons themselves (whether due to an epicadmium source or a subcadmium source) can comprise a very important part of the total absorbed dose rate in tissue at locations deep inside a multilegged duct.

1543. H. Bloomfield, Shielding Requirements for the NASA Plum Brook HB-6 Beamhole Radiation Effects Facility, NASA TM X-1461 (November, 1967). Availability: CFSTI.

The calculational methods and results used to design a shield assembly for the NASA Plum Brook Reactor HB-6 beamhole facility are described herein. The function of this shield assembly is to provide adequate biological protection during reactor operation and to permit personnel access to a test chamber after shutdown.

Radiation sources examined during reactor operation

were neutrons and gamma rays scattered by an experimental package in the test chamber, direct core radiations, and radiation through rectangular slits which may be formed by misalinement of shield sections. Fission product decay gammas and neutron-induced gamma activity in the test chamber were the important radiation sources investigated after shutdown.

The bulk of the calculations was performed by specialized digital computer shielding programs which accounted for considerable detail of the core-beamhole geometry. Calculational results compare favorably with experimental measurements on on the final shield assembly.

1544. C. Smith, N. Scofield, The Moments Method Used to Determine the Energy Albedo of Gamma Rays from Cesium-137 Impinging on Aluminum and Iron Barriers, USNRDL-TR-67-120 (September 6, 1967). Availability: U. S. Naval Radiological Defense Lab., San Francisco, California 94135.

The moments method, as outlined by Spencer and Fano, has been used to calculate the differential angular energy distributions of gamma ray fluxes backscattered from semi-infinite media. A plane normal source is simulated in the calculation. The data in this report are specifically for the angular energy distributions of .662 MeV gamma rays from Cs^{137} backscattered from semi-infinite media of aluminum and iron. The angles of detection are 110, 120, 130, 140, 150, and 160 degrees. The data are presented in tabular form. A graphical comparison is also shown between results from an albedo experiment and the moments method. All of the moments method data were obtained from a computer program developed at NRDL using moments generated at the National Bureau of Standards.

1545. M. Kangilaski, The Effects of Neutron Radiation on Structural Materials, REIC Report No. 45 (June 30, 1967). Availability: Defense Documentation Center, Cameron Station, 5010 Duke St., Alexandria, Virginia 22314.

A compilation of the data available on the effects of radiation on tensile, creep, fatigue, impact, and hardness properties of various structural materials is presented. These properties are given as a function of test temperature, irradiation temperature, and radiation fluence. Specifically the following reactor materials are covered: (1) aluminum alloys, (2) magnesium alloys, (3) beryllium, (4) zirconium alloys, (5) mild steels, (6) stainless steel, (7) nickel

alloys, and (8) refractory metals. Data on the effects of radiation on the mechanical properties of selected materials at cryogenic temperatures also are included.

1546. H. Feldman, Effects of Faulty Random Number Generators on Monte Carlo Calculations, ORNL-TM-1625 (July 12, 1967). Availability: CFSTI.

Following Coveyou and MacPherson's estimations of serial correlation in certain uniform random-number generators, we attempt to find and measure the influence of this correlation on various random sampling algorithms, used in Monte Carlo calculations, which draw on a uniform generator. Theoretical results concerning some of these algorithms are presented, along with sketchy data on their response to faulty generators.

1547. J. Ritts, P. Stevens, E. Solomito, Calculations of Neutron Fluence-To-Kerma Factors for the Human Body, ORNL-TM-2079 (January, 1968). Availability: CFSTI.

Fluence-to-kerma factors(where fluence is the time-integrated neutron flux and kerma is equal to the total kinetic energy released in materials by charged particles resulting from direct neutron interaction per unit mass of the irraidated medium) were claculated at discrete neutron energies from 0.045 eV to 15 MeV for various compositions in the human body - tissue, muscle, bone, lung, brain, red marrow, and the "standard man" composition. The calculation applied the latest cross sections for all the 11 most common elements in man. A number of reactions were included which were not considered in previously reported work, namely, an anisotropic elastic-scattering correction, inelastic scattering, (n,2n) reactions, (n,charged particle) reactions, and beta positron emissions from these reactions. These calculations are improvements in the 10-15 MeV and 2-100 eV ranges over the neutron first-collision flux-to-dose conversion factors previously reported by Snyder.

1548. Dose-Effect Modifying Factors in Radiation Protection, Report of Subcommittee M-4 (Relative Biological Effectiveness) of the Nat. Commission on Radiation Protection, BNL, 50073 (T-471) (August, 1967). Availability: CFSTI.

In reviewing the experimental values available for the relative biological effectiveness (RBE) and attempting to

specify quality correction factors (QF) for application in radiation protection, the Subcommittee found that these values would be of limited usefulness if presented independently of the manner in which dose-effect modification factors were to be used in practical radiation protection. It was deemed desirable to characterize an individual's entire cumulative exposure status by a single number. To do this, it is necessary to extend the present methods for summing exposures so that the incremented dose-equivalent values can be summed for all types of external and internal irradiations. To achieve this goal, a new system was devised for evaluating, summing, and recording occupational exposures.

1549. R. Maerker, F. Muckenthaler, Measurements and Single-Velocity Calculations of Differential Angular Thermal-Neutron Albedos for Concrete, ORNL-4090 (july, 1967). Availability: CFSTI.

Measurements and single-velocity Monte Carlo calculations have been performed to determine the differential angular thermal-neutron albedos for a reinforced concrete form nomodirectional beams of incident thermal neutrons. Preliminary calculations using a statistical estimation technique indicated up to 50 scatterings should be followed for each neutron in order to produce good estimates of the capture gamma-ray differential dose albedos. Deviation between experiment and calculation can be reduced to an average of 5.1% for 72 points of comparison if an anisotropic scattering law for water deduced from earlier Argonne National Laboratory measurements is assumed.

1550. M. Burrell, J. Watts, Plane Isotropic Buildup Factors for Bremsstrahlung Calculations, NASA TN D-4096 (August, 1967). Availability: CFSTI.

Gamma-ray isotropic dose, energy flux, and energy current buildup factors for finite slabs were calculated by Monte Carlo techniques for initial energies ranging from 0.05 MeV to 8.0 MeV with source backings from 0. mfp to 3.0 mfp and source-exit plane distances from 0.5 mfp to 7.0 mfp. Special emphasis was given to thin-source backings and to the energy range from 0.05 MeV to 0.5 MeV for these thin-source backings. Low energies were needed for accurate bremsstrahlung dose calculations and other applications for which they had not been previously available. In comparisons of the calculated finite slab buildup factors to buildup factors for infinite nedia generated by the moments method, differences of up to 50 per cent were observed. These differences were thought to be due mainly to

geometric factors.

1551. R. Robinson, S. Ewing, J. Nichols, T. Chapman, A. Miller, Brayton-Cycle Radioisotope Heat-Source Design Study Phase II (Preliminary Design) Report, ORNL-TM-1829 (August, 1967). Availability: CFSTI.

A preliminary design was prepared for a 25-kw(th) $^{238}PuO_2$ heat source that can be coupled with a Brayton-cycle power-conversion system for use in space. Primary containment is provided by multilayered fuel capsules, with tantalum-based alloy T-111 as the principle stress-bearing layer and platinum as the outer corrosion-resistant layer. The multi-purpose mission module specified by NASA is used as the vehicle configuration for integration studies. The assembly of fuel capsules is contained within a single reentry body for protection against reentry heating and subsequent impact in the event of a mission abort. The fuel capsule complies with the design criterion of no more that 1% creep in five years of service. The safety guideline that the radioisotope fuel can be contained within the fuel capsules for a period of ten half-lives of ^{238}Pu or longer is a goal toward which the capsule design is directed. The weight of the heat-source system, including the reentry body, fuel, heat exchanger, shield, etc., is 2670 lb. Preliminary designs are described of facilities for fueling and assembling capsules, shipment of capsules to launch site, and handling at the launch site. Since the heat-source design is based partly on near-future materials technology and on extrapolation from present aerodynamic data, identification is made of areas where further investigations will have to be carried out to test the validity of the design.

1552. R. Robinson, S. Ewing, A. Miller, J. Nichols, R. Stephenson, R. Beaver, D. Burton, T. Chapman, C. Craven, Jr., Brayton-Cycle Radioisotope Heat Source Design Study Phase I (Conceptual Design) Report, ORNL-TM-1691-Del. (August, 1967). Availability: CFSTI.

Conceptual designs were prepared for $^{238}PuO_2$, $^{244}Cm_2O_3$ and $^{147}Pm_2O_3$ radioisotope heat source systems that will provide 25 kw of thermal power to a Brayton-cycle power conversion system for space applications. The conceptual designs for all three fuels show primary containment provided by multilayered fuel capsules, with tantalum-based alloy T-222 as the principal stress-bearing layer and platinum as the outer corrosion-resistant layer. All the designs show the assembly of fuel capsules contained within a single reentry body for protection against reentry heating and subsequent impact in the event of a mission abort. The designs

are based on containing the radioisotope fuel within the fuel capsules for a period of time equal to ten half-lives or longer of the radioisotope. Estimated system weights based on the conceptual designs range from approximately 3000 to 25,000 lb, depending on the fuel design concept.

1553. R. Smith, Principles of Lead Shielding Design, Nuclear Engineering and Design 4 (1966) 459-479 (November 21, 1966).

The correct choice of the composition and form of lead to give the required physical properties is the first essential in the production of an efficient shield. The application of the correct principles by the designer will then ensure that the manufacturer is able to employ the most suitable techniques in the production of the shield, whether it is a massive thick shield for a container, or thin sheet lead to be applied to the walls to form a room shield.

Where local shielding of a more temporary nature is required, it is essential to know the different systems of lead bricks which are available, as it is with the special products such as lead plastic composites which have been produced for specific requirements.

1554. R. Coveyou, R. MacPherson, Fourier Analysis of Uniform Random Number Generators, Journal of the Association for Computing Machinery, Vol. 14, No. 1, pp. 100-119 (January, 1967).

A method of analysis of uniform random number generators is developed, applicable to almost all practical methods of generation. The method is that of Fourier analysis of the output sequences of such generators. With this tool it is possible to understand and predict relevant statistical properties of such generators and compare and evaluate such methods. Many such analyses and comparisons have been carried out. The performance of these methods as implemented on differing computers is also studied. The main practical conclusions of the study are: (a) Such a priori analysis and and prediction of statistical behavior of uniform random number generators is feasible. (b) The commonly used multiplier for computers with an adequate ($\geq \sim 35$-bit) word length. (c) Further work may be necessary on generators to be used on machines of shorter word length.

1555. G. Plummer, Geometry and Barrier Attenuation of Cobalt-60 Gamma Rays by a Vertical Steel Slab, Nuclear Science and Engineering: Vol. 31, No. 2, pp 183-190 (February, 1968).

This experiment was designed to test the barrier and geometry attenuation factors for ^{60}Co gamma rays developed from moments-method calculations by Spencer as given in National Bureau of Standards Monograph 42. A set of vertical, plane steel barriers was employed. Selection of the detector distance from a given barrier and the degree of collimation permitted exposures to be measured as a function of the solid angle subtended by a constant circular area on the barrier. The effective mass thickness of the barriers ranged from 0 to 73 lb/ft^2 and the solid angle subtended ar the detector varied from 0.2 to 5.5 sr.

A uniform plane radiation field was simulated by a traveling ^{60}Co source that was pumped through plastic tubing that covered a 100-ft semicircular area. Extrapolation of the experimental data gave estimates of the exposures to be expected from an infinitely extended field.

The final results for a collimated detector, located behind a steel barrier, were normalized to the free-field exposure received by a detector located 3 ft above the extended field. The experimental values were compared to a family of curves based on calculated results. For all cases except those for relatively small solid angles, the agreement was within 20%.

1556. J. Grundl, A Study of Fission-Neutron Spectra with High-Energy Activation Detectors. PartII: Fission Spectra, Nuclear Science and Engineering: 31, No.2, pp 191-206 (February, 1968)

The energy spectra of neutrons from the thermal-neutron-induced fission of ^{235}U, ^{233}U, and ^{239}Pu have been compared by means of eight activation detectors that cover the energy range 0.8 to 16 MeV. The detectors are exposed to fission neutrons produced at the center of a 10-cm-diam spherical cavity within a heavy-water moderator. Comparison of detector responses for the three spectra yield average energy ratios, ^{235}U: ^{233}U: ^{239}Pu = (1): (1.021 ± 0.005): (1.039 ● 0.002). Differences between the normalized spectra are most pronounced at high energies as exemplified by the relative ^{239}Pu: ^{235}U flux ratios 1.17 for $6 < E < 11$ MeV and 1.35 for $E > 11$ MeV. Special indexes for the ^{235}U fission spectrum, based on measurements with monoenergetic neutrons, show progressively fewer neutrons above 6 MeV than given by the usual Maxwellian description of the fission spectrum,

χ_{235} (E) = (0.770)$E^{1/3}$ exp (-0.775E). At lower energies, the U observed spectral indexes involving the ^{235}U, Np, and ^{238}U fission detectors are significantly discrepant with those predicted.

1557. J. Beyster, Neutron Scattering from Light Water, Nuclear Science and Engineering: Vol. 31, No. 2, 254-271 (February, 1968).

The single differential cross section for neutron scattering from light water has been measured over the energy range 0.006 to 10 eV. The experimental techniques for making the measurement and correcting the data to obtain an absolute cross section are discussed. It is found that the multiple scattering of neutrons in the sample constitutes a large effect and procedures are utilized and tested for making this correction. The resulting cross sections are compared with predictions of theoretical models describing the molecular motion in water. These models include various versions of the free gas model, the Nelkin model, variations of the Haywood model, the McMurry model, and Radkowski prescription. Completely satisfactory agreement with the available neutron scattering data does not appear possible for any of the above models. The Haywood model seems to provide good agreement, however, for the widest range of data.

1558. P. Jauho, H. Kalli, Accuracy of Transport Theory Calculations of an Anisotropic Neutron Flux, Nuclear Science and Engineering: Vol. 31, No. 2, pp 318-324 (February, 1968).

In this accuracy study, we have solved with the discrete S_2, S_4, and S_8 approximations a monoenergetic half-space problem with a delta-function pencil of neutrons as a source and compared the values of four suitable integral quantities with the results of Monte Carlo calculations. The errors in the space-dependent integrated neutron flux are shown in several figures. Also, tests with different mesh resolutions of the iteration net and with different standard quadrature sets have been made.

1559. V. Klimanov, Yu. Podsevalov, V. Mashkovich, The Gamma Ray Buildup Factor for Finite Media, Soviet J. At. Energy, English Transl., 22,3, 273-274 (1967).

The solutions of many problems of shielding physics and radiation technology requires knowledge of the buildup

factor for finite barriers, that is, for media in which at least one of the transverse dimensions may not be considered infinite.

This paper reports the results of an experimental study of energy buildup factors for γ-rays in the energy range 0.412 - 2.75 MeV from point isotropic source in finite homogeneous cylinders of water and aluminum. In the experiment a point isotropic source of γ-rays and a detector were placed on the axis of a cylinder shield close to its end faces, and the dependence of the γ-intensity on the length l and radius r of the cylinder were measured. A minimum SBM - 10 counter with special filters without energy dependence over the range 0.06 - 3.0 MeV was used as a detector.

Analysis of the results of the experiment showed that for a given shielding material and a fixed source energy the ratio of the intensity of the scattered radiation in a finite medium to the intensity of the scattered radiation for an infinite slab of the same thickness depends only on r/l.

1560. D. Trubey, B. Maskewitz, A Review of the Discrete Ordinates S(sub n) Method for Radiation Transport Calculations, ORNL-RSIC-19 (March, 1968). Availability: CFSTI.

Resumés are presented of papers given at a seminar-workshop on "The Discrete Ordinates S_n Method for Radiation Transport Calculations" held at Oak Ridge, Tennessee, August 14 - 16, 1967. The papers, in their review of the state-of-the-art, establish the fact that this method of choice for one-dimensional and sometimes two-dimensional deep penetration calculations.

A large number of computer codes which solve one-dimensional problems were reported. These include DTF-IV, DDK, GAPLSN, GGSN, 1DF, ODD-K, DTA, and ANISN. In addition, there were some two-dimensional codes reported including 2DF, TDSN, and DOT.

The recently developed ANISN computer code, described in detail in the workshop, has been placed with the Radiation Shielding Information Center for routine dissemination.

1561. Y. Song, Fast-Neutron Streaming Through Concrete Ducts, USNCEL-TR-R565 (February, 1968). Availability: CFSTI.

As a part of the Naval Civil Engineering Laboratory's studies of shielding for personal shelters, fast-neutron streaming through two-legged rectangular air ducts in concrete was investigated theoretically and experimentally. This is the final report on that investigation.

In the theoretical study of the problem, the albedo concept was employed, and numerical values of neutron dose in the second leg of ducts of various sizes were $T(d,n)He^4$ reaction (14 Mev), $D(d,n)He^3$ reaction (2.5 Mev), and Plutonium-Beryllium. Three sizes of concrete ducts were used as models: (1) 3 x 3 feet with a first-leg length of 15 feet; (2) 2 x 3 feet with a 12-foot first leg; and (3) 2 x 2 feet with a 12-foot first leg.

In experiments, counting rates of a spherical dosimeter were measured in both the first and second legs of the ducts described for the neutron sources named.

Comparisons show very good agreement between theoretical and experimental results.

1562. W. Cross, Tables of Beta Dose Distributions, AECL-2793 (November, 1967). Availability: CFSTI, Scientific Document Distribution Office, AECL $1.00.

Tables are given of dose distributions near isotropic point and plane beta sources in air and water for 37 nuclides of importance in medicine and health physics. These were calculated from Spencer's does distribution tables for monoenergetic electrons. A simple method is given for deriving dose distributions in other low-Z media.

1563. G. Lahti, J. Miller, E. Lantz, Preliminary Considerations for Fast-Spectrum, Liquid-Metal Cooled Nuclear Reactor Program for Space-Power Applications, NASA TN D-4315 (March, 1968). Availability: CFSTI.

A nuclear reactor suitable for space-power applications should ideally be compact, light weight, easily controlled, and be capable of operation for a substantial period of time with little or no maintenance. In order to develop a reliable nuclear reactor for space power, however, it may be necessary to compromise some of these requirements to some degree due to other considerations such as shielding

requirements which may be necessary to protect personnel and equipment.

To identify and evaluate the relative importance of some of these parameters, a preliminary study was made in which reactor size and shield weight requirements were calculated for several compositions of uranium nitride and uranium dioxide (UN and UO_2) fuel materials under various operating conditions. A fast-spectrum, lithium-cooled reactor was selected for the study since it potentially will result in the smallest reactor that is capable of operating at the required temperature levels. In order to determine man-rated shielding requirements, a 2-millirem-per-hour radiation dose at 66 feet (20 m) was selected as a reasonable value, being of the same order of magnitude as that which would be received from galactic cosmic rays in interplanetary space.

For the relatively small reactors considered in this study, nuclear criticality and the allowable fuel burnup limits essentially determine the required reactor size and resulting weight of the shielding. The importance of the other variables, such as allowable steady-state heat-transfer limits, coolant temperatures and flow rates, coolant hole sizes, etc., is only secondary in determining the weight of the reactor plus shield. Choice of the proper value for each of these parameters can be based on overall system performance, availability of design data, material limitations, and manufacturing feasibility.

From the results of this preliminary study, it appears that nuclear stability and control, fuel material development, and shielding optimization are areas in which addition work must be performed to develop a reliable nuclear reactor system for space-power application.

1564. R. Karcher, O. Baldonado, R. Erdmann, The Application of Track-Length Distribution Biasing in Monte Carlo Deep-Penetration Calculations, Nucl. Sci. Eng., 31, 3, 492-499 (March, 1968).

The feasibility of track-length distribution biasing for the calculation of spatial and angular particle flux far from the source in an infinite medium is investigated. Calculations are performed for an idealized particle transport model having an exact analytical solution, and results for scalar flux are given to a penetration distance of approximately 60 mean-free-paths. Results are also given for angular distributions. It is found that the "optimum" biasing parameter "a" for scalar flux prediction can be approximated by a $\exp(-a) = 0.368c^{n/40}$ (a revised form of

an empirical expression from an earlier study), where c is the probability of scatter and *n* is the desired penetration distance. If appropriate corrections are made for the effects of inelastic and hydrogen scatter, the present results can provide useful guidance in the application of the track-length distribution biasing technique to more realistic systems.

1565. L. Levitt, The Use of Self-Optimized Exponential Biasing in Obtaining Monte Carlo Estimates of Transmission Probabilities, Nucl. Sci. Eng., 31, 3, 500-504.

A method of increasing the sampling efficiency in Monte Carlo calculations of thick shield penetration has been developed. The procedure alters the effective mean-free-path in such a way as to maximize the rate of convergence of the transmission probability. The approach is semiempirical in nature and has been shown to be remarkably insensitive to geometry. The primary dependence appears to be on the nonabsorption probability at each collision, with secondary dependence on the distance to escape. The procedure is simple enough to permit its incorporation into existing Monte Carlo codes with a minimum of programming effort.

1566. H. Murphy, Summary of Neutron and Gamma Dosimetry Techniques, AFWL-TR-66-111 (September,1967). Availability: CFSTI.

Neutron and gamma dosimetry techniques currently in routine use in the Biophysics Branch of the Air Force Weapons Laboratory are described. Each technique is described in sufficient detail to permit a person who has some training in nuclear physics to use the report as a guide to these techniques of radiation dosimetry. The report covers the following topics: (1) gold and gold-cadmium thermal neutron detectors; (2) fission foil and sulfur pellet threshold detectors; (3) magnesium and aluminum threshold detectors; (4) gamma thermoluminescent dosimetry through the use of a commercially available calcium fluoride dosimeter; and (5) computation of absorbed tissue dose from threshold detector data. Five appendixes cover (1) decimal time notation; (2) tables of the second and third exponential integral function for small arguments for computation of self-absorption and transmission correction factors; (3) tables of normalized decay values for plutonium, neptunium, and uranium fiols; (4) propagation of errors in count rate measurements; and (5) computation of absorbed dose from a bare critical assembly.

1567. K. Yost, A Method for the Calculation of Neutron-Capture Gamma-Ray Spectra, Nucl. Sci. Eng., 32, 1, 62-75 (April, 1968).

A model suitable for the prediction of gamma-ray spectra subsequent to the capture of neutrons into nuclear states of identifiable spin and parity has been developed. The dependence of radiative transition probabilities on the nuclear selection rules is explicitly accounted for. Means are provided for allowing dipole and quadrupole transitions in conjunction with variations in the magnitudes of corresponding transition "matrix elements." Comparisons are given between experimental capture spectra and corresponding spectra calculated with varying assumptions with respect to pertinent nuclear parameters for two capture states of ^{28}Al and one of ^{25}Mg. A comparison of calculated spectra using fitted and crudely approximated cascade parameters for the ^{28}Al capture state indicates few, if any, differences that would significantly change resultant neutron-capture gamma-ray-production cross sections.

1568. W. Coleman, Mathematical Verification of a Certain Monte Carlo Sampling Technique and Application of the Technique to Radiation Transport Problems, Nucl. Sci. Eng. 32, 1, 76-81 (April, 1968).

The first section of this paper is a mathematical construction of a certain Monte Carlo procedure for sampling from the distribution

$$F(X) = \int_0^X \Sigma(x) \exp\left[- \int_0^X \Sigma[v]\, dv \right] dx, \quad 0 \leq X.$$

The construction begins by defining a particular random variable λ. The distribution function of λ is developed and found to be identical to $F(X)$. The definition of λ describes the sampling procedure. Depending on the behavior of $\Sigma(x)$, it may be more efficient to sample from $F(X)$ by obtaining realizations of λ than by the more conventional procedure described in the paper.

Section II is a discussion of applications of the technique to problems in radiation transport where $F(X)$ is frequently encountered as the distribution function for nuclear collisions. The first application is in charged particle transport where $\Sigma(x)$ is essentially a continuous function of x. An application in complex geometries where $\Sigma(x)$ is a step function, and changes values numerous times over a mean path, is also cited. Finally, it is pointed out that the technique has been used to improve the efficiency of estimating certain quantities, such as the number of absorptions in a material.

1569. K. Lathrop, N. Demuth, Biorthogonal Angular Polynomial Expansion of the Two-Dimensional Transport Equation, Nucl. Sci. Eng. 32, 1, 120-130 (April, 1968).

A new system of biorthogonal polynomials is developed for the angular expansion of the directional flux in the linear Boltzmann transport equation. It is shown in systems infinite in one space dimension that the angular integral in the Boltzmann equation can be reduced to a weighted integral over the unit circle. The corresponding system of orthogonal functions is found to be a system of two sets of polynomials. The angular dependence of the particle flux is derived for these polynomials. Systems of partial differential equations are derived for the expansion coefficients, that is, for angular moments of the particle flux. One of these systems is shown to be a specific linear combination of the equations obtained when the directional flux is expanded in spherical harmonics functions specialized for the geometry considered. It is shown that this same system, in (x,y) geometry, reduces simply to the spherical harmonics equations in one-dimensional plane geometry.

1570. L. Gardner, B. Burdick, A. Mettler, Experimental Determination of Neutron Energy Spectra in Concrete Ducts, USNCEL-TR-523 (DASA 11.058) (May, 1967). Availability: CFSTI.

Neutron Energy spectra were experimentally determined at various positions in air ducts through concrete. The ducted concrete structures studied were built as model entranceways for protective shelter. The spectra were determined for the case of 14-Mev neutrons impinging on the duct entrance, and were based on activation analysis of threshold foil detectors, foil sandwiches, and bare and cadmium-covered indium foils. The experimentally determined spectra were compared to available spectra calculated by Monte Carlo techniques utilizing the ADONIS computer code. From the spectra the dose was calculated and the results were normalized to the dose in free air at one centimeter from the source. The normalized dose, as a function of distance in the duct from the source, is graphically compared for the severalanalysis techniques. The data presented demonstrates that it is possible to determine the neutron energy spectrum in ducts by the technique of foil activation. Such determinations are useful during the experimental optimization of the protection factor for the duct in which the inclusion of doors, traps, liners, and special wall materials are considered. The data presented herein form a basis for understanding the effect of duct geometry on the

protection factor, that is, the inverse of the dose rate at various positions in the duct normalized to the dose rate at the entrance of the duct.

1571. D. Johnson, J. Poston, Radiation Dosimetry Studies at the Health Physics Research Reactor, ORNL-4113 (June, 1967). Availability: CFSTI.

The dosimetric aspects of the Health Physics Research Reactor (HPRR) have been carefully investigated and are reported herein. Studies of parameters such as quantity and symmetry of neutron leakage from the reactor core were included in the work which was done primarily to obtain neutron and gamma-ray air-dose curves as a function of distance. These curves are used both for selecting reactor power levels and exposure times to obtain predesignated radiation exposures and, together with a dependable and reproducible normalization channel, to provide quantitative dose information with an estimated accuracy of ±10%.

The fast neutron leakage from the HPRR is 1.3 neutrons per fission. The first collision neutron dose rate at a distance of 2 m with reactor and detector heights of 2 m, is 3.3 ± 0.13 mrad/unit activation of the normalization sulfur pellet. For this reasonably good experimental geometry (H/D = 1), the scattered-to-total neutron dose ratio is 21%, and the neutron-to-gamma-ray dose ratio is approximately 7:1.

1572. K. Parthasaradhi, V. Rao, Incoherent Scattering of Gamma-Rays in Low- and Medium-Z Elements, Nuovo Cimento B, 49, 218-19 (June 11, 1967).

The ratio of the integral bound-electron incoherent scattering cross-section to the integral free electron scattering cross-section is compared for C, Al, Cu, and Sn at 320, 411, and 662 keV. It can be seen from the results that the agreement between the present experimental and theoretical values is satisfactory. (RSIC)

1573. Y. Fukai, Calculation of Monoenergetic Neutron Leakage from Two-Dimensional XY Small System by Collision Probability Method, J. Nucl. Energy, 21, 447-468 (June, 1967).

New collision probabilities developed here for a two-dimensional XY system are used in the calculation of mono-energic neutron leakage from the small systems consisting

of black absorbers and a homogeneous medium with an infinite height and isotropic scattering for neutrons. First the derivation of the two-dimensional collision probabilities is presented and, using the probabilities, neutron fluxes at each mesh point in a square homogeneous medium or systems with cruciform or box-type control rods are accurately calculated and exact flow-out neutron currents from these systems are obtained from the resultant neutron fluxes and the escape probabilities from each mesh point.

An equivalent radius for the square homogeneous medium is obtained by comparing the leakage rates from it with those from the infinite cylinder medium which have been already calculated. Comparisons of the exact neutron leakage with the calculated one by the diffusion theory for each system are discussed, and it is also elucidated that calculation for such a small system shows a significant error.

1574. S. Izumi, C. Watarumi, Interaction of Adjacent Thermal Neutron Detectors in Water, J. Nucl. Sci. Technol., 4, 3, 115-120 (March, 1967),

Thermal neutron flux perturbation due to two or three adjacent detectors in water has been experimentally investigated. The detectors employed were gold foils and pins. The results have been compared with theoretical values calculated on the basis of Dalton's theory. The comparison gives fairly good agreement.

1575. R. Carter, W. Carlton, R. Rohrer, G. Dyer, Low Energy X-Ray Mass Absorption Coefficients from 1.49 to 15.77 keV for Scandium, Titanium, Vanadium, Iron, Cobalt, Nickel, and Zinc, Health Phys., 13, 593-99 (1967).

Mass absorption coefficients have been measured for scandium, titanium, vanadium, iron, cobalt, nickel, and zinc for photon energies from 1.49 to 15.77 keV with an average standard deviation of the mean of 1.2 per cent. A high intensity X-ray beam was provided by a partially evacuated, Bragg-crystal spectrometer. The absorbers were cold-rolled from spectroscopic grade material with thickness determined by the weight-area measurement method. An empirical equation relating photon energy and mass absorption coefficient is proposed for photon energies between the L- and K-absorption energies of the absorber. This equation is used to predict values of mass absorption coefficients for lead, tin, copper, and calcium and the results compared with the values found in the Norelco Reporter.

1576. B. Lawson, C. Cook, Solution of the (n, n′, γ) Transport Problem in Carbon--Monte Carlo, J. Nucl. Energy, 21, 653-669 (1967).

The Monte Carlo technique has been applied to the determination of the steady-state and time-distribution of the unscattered 4.43 MeV γ-flux arising from multiple inelastic scattering of neutrons with carbon nuclei in an infinite graphite medium. The neutrons emanate from a 14 MeV isotropic point source. The computer program requires 0.85 minutes per thousand histories with 5000 histories more than sufficient to obtain significant answers. The steady-state single and multiple inelastic neutron produced unscattered γ-flux can be approximated by a simple formula of the form

$$\Phi est = \frac{Ae^{-\mu r_m}}{4\pi r_m^2}$$

where μ is the γ-absorption coefficient of 4.43 MeV γ-rays in carbon, r_m is the source-detector separation distance, and A is a pseudo source strength.

Time-distribution studies reveal that the peak intensity of the 4.43 MeV γ-flux following a neutron pulse of zero time width monotonically increases from 2 to 5 nsec as the source-detector separation increases from 10 to 80 cm. This unscattered γ-flux decreases a factor of 100 in the next 10 nsec with an exponential character.

1577. A. Profio, T. Williamson, H. Antunez, Measurements of the Neutron Angular Flux Spectrum in Graphite, DASA-1933 (GA-7470 and GA-7409) (October 30, 1966). Availability: Director Defense Atomic Support Agency, Washington, D. C. 20301.

Measurements of the neutron angular flux spectrum in graphite have been made to provide a standard of comparison for neutron penetration calculations. The configuration approximated a point, fission-like source in an infinite medium. The spectrum from 0.002 eV to 15 MeV was measured with an accuracy of ±10% to ±20%, and resolution of ≈ 10%, by the pulsed-source, time-of-flight method. The source spectrum and other characteristics were also measured. Spectra were obtained at angles from 0° to 60° and penetration to 66 cm. In addition, dieaway measurements, indium subcadmium and epicadmium flux traverses, sulfur and aluminum threshold foil traverses, and a threshold foil spectrum measurement were made. Procedures, apparatus, and results are described in detail so that others can evaluate our results and use them to test transport codes and input approximations. Comparison of the time-of-flight data with

a preliminary descrete-ordinates calculation indicates fairly good agreement except for detail in the carbon resonance region, where the flux oscillates strongly. Both calculations and experiment indicate the intermediate spectrum for $r > 20$ cm from the source is not 1/E as usually assumed. Low energy foil activation and time-of-flight results are consistent, but threshold foils give poor results.

An appendix is included (GA-7409), titled "Angular Flux Spectrum and Other Characteristics of the 7.62 cm Uranium Target."

1578. Lead Development Assoc., Lead Shield Applications, Lead Shielding Bulletin No. 1 (1967). Availability: Lead Development Association, 34 Berkeley Square, London W1.

The modern use of nuclear power and the resulting application of radioactive isotopes requires stringent safety precautions to ensure that no one is exposed to the dangerous radiation involved.

The design of efficient shielding systems is therefore of paramount importance and the LDA booklet, "Lead for Radiation Sheilding", describes the principles of design and techniques of manufacture which have been devised to ensure the necessary efficiency of lead shielding systems.

"Lead Shielding Applications" gives illustrated examples of good practice in the shielding industry as a whole. Subsequent booklets will include more detailed examples of the use of lead in each of the specific fields covered here.

1579. R. Loevinger, M. Berman, Pamphlet No. 1 - A Schema for Absorbed-Dose Calculations for Biologically-Distributed Radionuclides, Journal of Nuclear Medicine, MIRD, Sup. No. 1, p. 7-14 (February, 1968). Availability: Dr. Edward M. Smith, Division of Nuclear Medicine, University of Miami School of Medicine, P. O. Box 875, Biscayne Annex, Miami, Florida 33152.

Methods for the calculation of absorbed dose for biologically-distributed radionuclides have evolved over the years to a point where a number of seemingly different procedures are in use. Some of the procedures contain simplifying assumptions that were adequate for the radionuclides and the biological models of interest at the time, but which may now be inadequate. The ever-increasing number and types of radionuclides in medical application and the growing interest in detailed biological models (sub-organ, cellular, molecular,

etc.) prompted the re-evaluation of the various computational procedures and also the physical and biological data required for the calculation.

This pamphlet is based on a paper by Loevinger and Berman (1), which makes use of the quantity absorbed fraction first introcuded by Ellett, Callahan and Brownwll (2,3), and also on an extension of this concept, the specific absorbed fraction. The absorbed-dose calculations for both homogeneous and inhomogeneous media and for various types of radiations are developed around these concepts.

To use the present formalism, certain input data must be assumed as "known." Among these are the fractions and energies in the radionuclide disintegration schemes, the model (which includes the kinetics of the radionuclide in the organs of the body), buildup factors and absorbed fractions and specific absorbed fractions between sources and targets. Some of this information is available in the accompanying MIRD pamphlets, and additional data will be given in future MIRD publications. This report is intended only for calculations connected with biologically-distributed radionuclides.

1580. M. Berger, Pamphler No. 2 - Energy Deposition in Water by Photons from Point Isotropic Sources, Journal of Nuclear Medicine, MIRD, Sup. No. 1, p. 15-25 (February, 1968). Availability: Dr. Edward M. Smith, Division of Nuclear Medicine, University of Miami School of Medicine, P. O. Box 875, Biscayne Annex, Miami, Florida 33152.

This pamphlet contains a tabulation of buildup factors and related data which describe the deposition of energy by photons in a water medium. Water has been chosen as the reference medium because it is clearly defined and quite representative of other biological media (such as tissue or muscle) in regard to photon scattering and absorption. The radiation sources considered are monoenergetic and point-isotropic, i.e., of negligible dimensions and emitting photons of fixed energies uniformly in all directions. Nineteen source energies between 15 keV and 3 MeV are included. The energy grid is fine enough so that one can interpolate to other source energies. The results for point sources are of direct interest themselves and can also be used to obtain, by superposition, corresponding results for extended sources. To facilitate such superpositions and other manipulations, polynomial representations of the buildup factors are given in addition to numerical values.

The results in this pamphlet apply only to homoge-

neous media and do not cover situations in which there are significant variations in composition or density (such as those due to the presence of bone, ling tissue or voids in a human body). The applicability of the results is further restricted to media which are in effect unbounder, i.e., to configurations of medium, source and detector such that leakage of radiation asross the boundaries of the medium has no influence on the absorbed dose received by the detector. Data on energy deposition in bounded media of various shapes will be presented by Brownell, Ellett, and Reddy in Pamphlet 3(1) in this series.

1581. G. Brownell, A. Reddy, W. Ellett, Pamphlet No. 3 - Absorbed Fractions for Photon Dosimetry, Journal of Nuclear Medicine, MIRD, Sup. No. 1, p. 27-39 (February, 1968). Availability: Dr. Edward M. Smith, Division of Nuclear Medicine, University of Miami School of Medicine, P. O. Box 875, Biscayne Annex, Miami, Florida 33152.

Traditionally the calculation of photon exposure has been performed by the integration of a point-source kernel consisting of simple inverse-square and exponential attenuation factors and the results expressed in roentgen. The accuracy of this type of approximation is limited because such simple kernels cannot adequately consider the dose from scattered radiation or the effects of boundaries. For internal emitters, the calculation of absorbed dose in rads is preferred to dose calculation expressed as a roentgen exposure in air. Recent calculations of buildup factors and new techniques for treating boundary problems by means of Monte Carlo sampling have improved the methods of calculating the photon dose directly in rads. These methods rigorously consider the contribution to the absorbed dose from both direct and scattered radiation and give results in a form directly applicable to absorbed-dose calculation.

While the results obtained with these newer methods do not differ greatly from those obtained with simple exponential kernels for initial photon energies greater than a few tenths of an MeV, estimates of the absorbed dose are significantly different for photons of lower energy. It is perhaps the increased use of low-energy photon emitters in clinical practice that has provided the chief impetus for the development of exact photon-scattering calculations for dosimetry problems.

Monte Carlo calculation of photon diffusion provides a direct and accurate means of calculating absorbed dose from internal photon emitters in situations

where the source and absorber geometries are defined. Calculations have been carried out for point and uniform-source distributions within bounded, tissue-equivalent material. The photon absorbed fraction ϕ is simply the fraction of the emitted photon energy that is absorbed in the region of interest (1-3). It is clear that the absorbed fraction can be determined only after specifying the source and absorber geometry, the photon energy and the intervening material. Results of such calculations are presented in this pamphlet for a sufficient number of geometries and energies to permit accurate interpolation. More recently the term specific absorbed fraction Φ has been proposed to represent the absorbed fraction per gram of absorbing medium (4).

1582. B. O'Reilly, Sr., R. Lewis, Discrete Ordinates Representations of the Scattering Kernel, NAA-SR-12493 (November 30, 1967). Availability: CFSTI.

In this report it is demonstrated that acceptably accurate DSN estimates of the slowing-down source can be obtained using a low-order truncation of the poorly convergent Legendre series expansion of the angle-energy transfer probability function, while very poor estimates can be expected when its angle-to-angle energy-to-energy representation is used.

1583. F. Binford, F. Kam, J. Barish, Estimation of Radiation Doses Following a Reactor Accident, ORNL-4086 (February, 1968). Availability: CFSTI.

Based upon the "Gaussian Plume" formula, a model has been developed which permits calculation of the downwind doses to be expected following a reactor accident. Although the calculation presented here is concerned only with the thyroid dose due to iodines and the whole body dose due to iodines and noble gases, it can easily be adapted to handle other radioisotopes. The basic assumptions include instantaneous release of a given quantity of radioactive material into a reactor building followed by an emission at a constant (volume) rate from a point source. Radioactive decay both within the building and during passage downwind are accounted for as are the growth and decay of daughter products and the effects of physical separation such as filtration. A Fortran IV computer program "PLUME" has been written for the purpose of solving the equations developed.

1584. A. Kaplan, N. Koepp-Baker, Final Report - The Use of Scale Models in Structure Shielding Experiments, TO-B67-25 (April 21, 1967). Availability: CFSTI.

Various aspects of using scale-model shielding data to predict the shielding effectiveness of full-scale structures have been studied. Among the important problems considered were the estimate of the far-field contribution and the application of model data to the corresponding full-scale structure. Our work has shown that for these two operations, separate treatment of the skyline and non-skyline components of the dose rates in the model structure produces good agreement with experimental and theoretical results for full-scale structures. This agreement is significantly better than that obtained by using the total dose rates to perform these operations, as was done in the earlier methods of handling model data. In considering the materials used in model structures, we have found that the use of steel in model experiments on a psf basis to simulate concrete in the full-scale structure produces no difference in barrier reduction, within experimental accuracy, up to a barrier thickness of 150 psf. The results of the evaluation of our methods for designing and carrying out scale-model shielding studies and applying the results of these studies to full-scale structures indicate that scale-model experiments can be used to predict the shielding effectiveness of a full-scale structure to within ±10%.

1585. J. Singh, C. Gross, R. Adams, X-Ray Spectra Produced When Thick Silicon Targets Are Bombarded with 1.05 and 1.25 Million Volts Electrons, NASA TN D-2773 (December, 1966). Availability: CFSTI.

Measurements have been made on the X-rays produced when thick silicon targets are bombarded with 1.05 MeV and 1.25 MeV electrons. (These energies are based on range-energy relationship for electrons in aluminum.) The various cross sections, after correction for efficiency and finite resolution of the sodium iodide crystal for X-rays, have been compared with the predictions of the Bethe-Heitler theory. The predicted pulse height and angular distributions of the X-rays are in qualitative agreement with the measured values. However, the absolute magnitudes of the measured X-ray intensities, at both energies of the electrons, are smaller than the theoretical values (relativistic, nonscreened, and no backscattering considered).

1586. K. Mizukami, T. Hyodo, T. Matsumoto, Backscattering of Gamma Rays from Polyethylene, Aluminum, and Lead Slabs, J. Nucl. Sci. Tech. 4, 12, p. 607-613 (December, 1967).

Spectra of backscattered raidation from finite thick slabs of polyethylene, aluminum and lead were obtained by means of a scintillation spectrometer. A point ^{60}Co γ-source was placed in close contact with a scattering slab. Variations of number and energy albedo values with increasing slab thickness are given as experimental formulas and graphical representations.

1587. R. French, C. Garrett, K. Thompkins, Calculated Gamma-Ray Dose Distributions in a Phantom Exposed to Fallout and Simulated Fallout, AFRPI CR67-1, (RRA-T71, AD-658043) (February, 1967). Availability: CFSTI.

Gamma-ray depth-dose distribution in a phantom exposed to fallout and to simulated fallout were calculated by the Monte Carlo method. The phantom consisted of a tissue equivalent vertical right cylinder of 60 cm height and 30 cm in diameter. The center of the phantom was 3 ft 8 in (111.8 cm) above a smooth ground surface uniformly contaminated with ^{235}U fission products. The energy and angle distribution of the gamma rays incident upon the phantom were taken from previous Monte Carlo calculations.

The depth-dose distributions were found to be relatively insensitive to fallout age over the period investigated (1 hour to 9 days). The dose rate at the center of the phantom is approximately 65 per cent of the free-field dose rate, while that at the lateral surface is approximately 80 per cent. Except near the extremities, the dose rate along the vertical axis of the phantom varies at approximately the same rate with height above ground as does the free-field dose rate. Approximately one-half of the dose rate at the center of the phantom is from photons which have suffered previous collisions in the phantom.

The depth-dose distributions were calculated for two arrangements of artificial sources which, although not duplicating the fallout energy spectra, were intended to produce similar depth-dose distributions. The patterns produced by revolving the phantom on its vertical axis while exposed to a point ^{60}Co source at a horizontal distance of 200 ft (61 m) are similar to those from the fallout, except for internal positions near the bottom of the phantom. A special arrangement of ^{60}Co, ^{137}Cs and ^{144}Ce sources produced substantially the same depth-dose patterns throughout the phantom as did the fallout.

An additional calculation was performed of the depth-dose distribution within a phantom positioned at the center of a 4 ft diameter by 5 ft deep foxhole located in a 1.12-hr fallout field. The distribution in the horizontal midplane of the phantom was found to be quite similar in shape to the above ground case except for a sharper dropoff of the dose away from the lateral surface. As would be expected, however, the axial distribution is quite different from that found above ground.

1588. H. Zimmermann, Radiation Dose Rate of Reprocessed Na 1 Breeder Fuel, ORNL-tr-1840 (Translated from KFK 603) (July, 1967). Availability: CFSTI.

The activity of reprocessed Na 1-breeder fuel is calculated in this study. The fuel from the more highly enriched core zone 2 is considered as the more unfavorable case. The activity is composed of the γ-activity of the fission product and the plutonium isotopes as well as their daughter products and the neutron activity of plutonium. It is dependent on the reprocessing technique. A common reprocessing of core and blanket material is more advantageous, for it leads to smaller source strengths than a separate core processing. For example, the fission product activity is reduced to about 30% by common reprocessing. The decontamination factor should be greater that 10^6, but does not need to be greater than 10^7.

The shield consists of iron and paraffin. The necessary material thicknesses depend on the reprocessing technique, and on the quantity and density of the fuel.

1589. L. Gardner, Neutron and Gamma-Ray Streaming Through Thin-Steel Shelter Entranceways, USNCEL Tech. Report R569 (March, 1968). Availability: CFSTI.

Neutron and gamma-ray streaming through a thin-steel, two-legged ducted entranceway was studied, and the dose rate at different positions in the duct was measured and calculated by several different techniques. The results were used to determine the protection afforded by a duct of this type, and they were compared to similar results obtained from ducts having thick steel or concrete walls. The albedo calculations were found to be in good agreement with the Monte Carlo calculations and the experimental measurements. It was also found that concrete ducts provide slightly greater attenuation of radiation than do steel ducts. Simplified formulas were developed to provide crude

approximations to the solution of an optimum duct design.

1590. Y. Song, Calculating the Dose Variation of Fast-Neutrons Streaming Through an Iron Duct, USNCEL Tech. Report R570 (March, 1968). Availability: CFSTI.

The dose variation of fast-neutrons streaming through an iron duct is computed by using the concept of the differential dose albedo. The basic concept of the differential albedo is itself studied and is used to formulate the neutron transport equation.

The albedo method developed in this report and the Monte Carlo method are used to obtain numerical values of the multicollision absorbed dose in the first and second legs of an 11 x 11-inch iron duct along the centerline. Comparisons are then made between the values obtained by both methods. Results show good agreement.

1591. A. Komarovskii, Design of Nuclear Plants, AEC-TR-6722 (1965). Availability: CFSTI.

This book is a revised and up-to-date edition of Stroitel'stvo yadernykh ustanovok (Construction of Nuclear Plants) (Moscow, Gosenergoizdat, 1961). It deals mainly with problems of selection of materials, design of the structural elements of stationary nuclear plants, and with the particular features of structural and installation work involved.

The first and second sections of the present book as well as Chapter 28 of the fifth section (which is about the construction of heavy-concrete shielding) deal with both reactor design and construction of biological shielding. Chapters 22 and 30 deal with special coatings of structures of radioactive premises and handle coating problems for nuclear plants, the radiochemical industry, and laboratories.

In addition to some general problems of nuclear radiation and protection, the first two parts of the book deal with the physical and engineering characteristics of structural materials used for nuclear plants, and present recommendations as to the selection of materials for the various types of plants in their particular locations. The first two parts of the present edition were completely revised and supplemented with new data from scientific work carried out in the USSR during the years 1961-1963. For the convenience of the designer, static and heat-engineering characteristics of the

various brands of concrete are presented in special sections in the second part of the book.

1592. D. Broder, Yu. Yegorov, S. Tsypin, A. Veselkin, V. Orlov, Eds., Problems in Reactor Shielding Physics, NASA-TT-F-411 (1963). Availability: CFSTI.

This collection of articles includes works which are concerned with the experimental and theoretical investigation of the biological shielding of nuclear reactors as well as works on the methodology and instruments for studying the shielding.

The works consider different analytical methods which make it possible to determine the three-dimensional energy and angular distribution of radiations in the shielding, the results of many experimants with radiations penetrating through homogeneous and multilayer shieldings through shieldings with heterogeneities and others.

Some of the works describe various devices to determine the fluxes and the spectral radiation distribution beyond the shielding and inside the shielding. These works also consider the method of analyzing the characteristics of devices and the process of measurement results.

This collection will be useful to a wide circle of readers concerned with the analysis and design of biological shielding of reactors and to specialists interested in the transmission of radiations through a substance, as well as to those working in the field of experimental nuclear physics.

1593. P. White, G. Warner, The Fission Cross Sections of ^{233}U, ^{234}U, ^{236}U, ^{238}U, ^{237}Np, ^{239}Pu, ^{240}Pu, and ^{241}Pu Relative to That of ^{235}U for Neutrons in the Energy Range 1--14 MeV, J. Nucl. Energy, 21, 8, p. 671-679 (August, 1967).

Fission cross sections of a number of fissile isotopes have been measured relative to the fission cross section of ^{235}U to an accuracy of approximately $\pm$ 2% at neutron energies of 1.0, 2.25, 5.4, and 14.1 MeV. Combining these ratios with the known values of the fission cross section of ^{235}U leads to fission cross sections having an estimated uncertainty of $\pm 3.5\%$ and which are mostly in agreement with other recent measurements.

1594. Y. Kanemori, Semi-Analytical Estimation of Gamma-Ray Intensity from a Volume Source, J. Nucl. Sci. Technol., 4, 2, p. 98-99 (February, 1967).

Integrations of experimental line and disk source kernels yielded a result in good agreement with a measured dose rate attenuation function for a cylindrical source. (RSIC)

1595. S. Ray, G. Muchow, D. Kaup, C. Dodd, G. Neilson, Derivation and Experimental Verification of an Equation for Exposure-Dose Rate Due to X-Ray Continua, J. Appl. Phys., 38, 8, p. 3122-6 (July, 1967).

An equation has been derived for the exposure-dose rate due to radiation emitted by x-ray tubes and transmitted through media which absorb all characteristic radiation. The parameters involved in this expression are the operating voltage and current of the tube and the linear-absorption coefficient and thickness of the absorbing media. Experimental verification of the equation shows that exposure-dose rates can be calculated within ±6% on a relative basis.

1596. D. Pozdneev, Back Scattering of Low-Energy Gamma Radiation, Soviet J. of At. Energy, English Transl., 20, 4, p. 354-360 (April, 1966).

Experimental data regarding the spectral and angular distribution of γ quanta back-scattered from various scattering substances are presented for the geometry of an isotropic point source; the scattering substances include paraffin, graphite, aluminum, iron, zinc, cadmium, and lead (semiinfinite media). The energy of the primary γ radiation was 0.145; 0.190; 0.279; 0.320; 0.396; 0.661 and 0.765 MeV. The numerical and energetic albedos (A and A_E) determined experimentally agree closely with the values calculated by the Monte Carlo method. The results are presented in the form of tables and graphical relationships for A, A_E, and A/A_E as functions of A and $E\gamma$.

1597. V. Mashovich, V. Sakharov, S. Tsypin, A. Nikolaev, B. Sinkitsyn, Distribution of Fast Fission Neutrons Along Straight, Cylindrical Ducts in Water, Soviet J. of At. Energy, English Transl., 20, 4, p. 469-472 (May, 1966).

A study was made of the transmission of neutrons from the BR-5 reactor inside straight, cylindrical ducts, 144 and 90 mm in diameter, located in water. Sulfur and aluminum threshold detectors were used to detect the neutrons.

Data was obtained for the attenuation along the axis of straight, cylindrical ducts of the fast neutron flux from disc, and plane, infinite monodirectional sources of fission neutrons; the effect of displacement on the attenuation of neutron radiation in the ducts was studied. The information obtained can be used to calculate the transmission of neutron radiation along stepped ducts.

1598. L. Shirkin, Calculation of the Passage of Fast Neutrons Through Graphite, Soviet J. of At. Energy, English Transl., 22, 2, p. 101-103 (February, 1967).

Using the Monte Carlo method, the author calculates the results of passing neutrons from a plane unidirectional source, with $E_0 = 3.3$ or $E_0 = 8.0$ MeV, through graphite. The angle of incidence is taken as 0°. The graphite layer thicknesses considered are from 0.9 to 6 times the free-path length. The author calculates the dose, energy and numerical albedos of graphite and also the angular and energetic distributions of the reflected neutrons and the angular distribution of the energy of scattered neutrons. He plots the mean cosine of the angle of scattering vs. the albedo, and also vs. the value of the energy angular-distribution constant. The data obtained may be found useful in the design of shadow and labyrinth shielding.

1599. A. Chilton, Buildup Factors for Point Isotropic Gamma Ray Sources in Infinite Medium of Ordinary Concrete, Nucl. Eng. Design, 6, 3, p. 205-212 (1967).

In order to provide more definite values for buildup factors for point, isotropic gamma ray sources in an infinite medium of ordinary concrete, a Monte Carlo program has been undertaken to make such calculations. Previous approaches to this problem had been through the assumption of a Z_{eff} for concrete and in interpolations in the existing Goldstein-Wilkins results for single elements. In addition to the approximate character of the old procedures, such calculations have been variable because different values of Z_{eff} have been used by different workers.

The specific method of Monte Carlo calculation is briefly described, with emphasis on the assumptions,

approximations, and sources of data.

The resulting data are estimated to have an uncertainty less than 5%. They are compared with results using the Z_{eff} approach, and the asymptotic trends at very small and very large penetration distances are discussed.

1600. C. Eisenhauer, Gamma Radiation Fluxes Near a Ground-Air Interface Using an Image Source Technique, Nucl. Sci. Eng., 32, 2, 166-177 (May, 1968).

Calculations are made of the radiation flux of gamma rays that have originated from a point isotropic source and have been singly scattered in the air lying beyond a plane interface. Calculations are made in the limit that the source-detector separation distance is small compared to a mean-free-path in air. These results are interpreted in terms of an image source. The results, combined with earlier calculations of the radiation flux reflected from a condensed medium, such as ground, predict the effect of the ground-air interface on radiation fluxes in air near the interface. The results are extrapolated to source-detector separation of the order of a mean-free-path by using infinite-medium buildup factors. Comparisons with experiment show that the model produces results that are in qualitative agreement with experiment.

1601. R. Howerton. E. Plechaty, A Formalism for Calculation of Neutron Induced Gamma-Ray Production Cross Sections and Spectra, Nucl. Sci. Eng., 32,2, p. 178-183 (May, 1968).

A formalism is developed from observable systematics and elementary energetics of nuclear reactions for the calculation of neutron induced gamma-ray production cross sections and spectra. The calculations are compared with experiments performed at several laboratories. The comparison shows a mean diviation of 18%.

1602. A. Shimizu, Calculation of the Penetration of Gamma Rays Through Slabs by the Method of Invariant Imbedding, Nucl. Sci. Eng., 32, 2, p. 184-194 (May, 1968).

The method of invariant imbedding has been applied to the problem of penetration of gamma rays through slabs. The accuracy of the method was examined by comparison with other reliable calculations and proved to be competitive

with other well-developed methods. The method has the advantages that it is much more efficient than the Monte Carlo method and that it is readily applicable to the multilayer problems. An extensive series of calculations on the transmission of gamma rays through homogeneous slabs was carried out. The energy and angular distributions of transmitted photons from monoenergetic and oblique sources were obtained for slab thickness up to 15 mean-free-paths and for 7 materials. The results of calculations were compared with experiments and found to be in good agreement.

1603. P. Benoist, A. Kavenoky, A New Method of Approximation of the Boltzmann Equation, Nucl. Sci. Eng., 32, 2, p.225-232 (May, 1968).

In a new method of approximation of the Boltzmann equation, one starts from a particular form of the equation that involves only the angular flux at the boundary of the considered medium and where the space variable does not appear explicitly. Expansion of the angular flux of neutrons leaking from the medium in spherical harmonics with no assumption about the angular flux within the medium, gives a very good approximation of several classical plane geometry problems. These problems include the albedo of slabs and the transmission by slabs, the extrapolation length of the Milne problme, and the spectrum of neutrons reflected by a semi-infinite slowing down medium. The method may be extended to other geometries.

1604. D. Hoffman, Neutron Energy Spectrum - A Classical Derivation, AD-655625 (USNRDL-TR-67-57) (May 23, 1967). Avaliability: CFSTI.

The epithermal neutron energy spectrum is derived as a function of radial distance from a point isotropic monoenergetic neutron emitter in an infinite homogeneous scattering and absorbing medium. The derivation uses Condon and Breit's neutron energy spectrum as a function of number of collisions and Chandrasekhar's general solution to the problem of random flights.

1605. T. Armstrong, A V^{o} Importance Function for the Monte Carlo Calculation of the Deep Penetration of Gamma Rays, University Microfilm, Inc, 68-9787 (December, 1967) Availability: Univ. Microfilm, Inc., Michigan.

The approach used in the present investigation is to

formulate an importance function which is based on the first term of the series sloution of the value function equation. The importance function so obtained provides both spatial and direction biasing. This function will be referred to as the V^o importance function. The merits of the V^o importance function are exemplified in terms of a one-dimensional gamma-ray transmission problem.

Results from application of the V^o importance sampling to a one-dimensional gamma-ray transmission problem are presented and discussed. The improvement due to biasing each of the variables (intercollision distance, polar scattering angle, and azimuthal scattering angle) is determined in a systematic fashion. The V^o importance sampling is compared in efficiency with a form of the exponential transform method.

The proposed method was found to be considerably more efficient than the exponential transform technique for the particular source-geometry configuration investigated.

1606. M. Drake, M. Allen, J. Garrison, Neutron and Gamma Ray Production Cross Sections for Sodium, Magnesium, Chlorine, Potassium, and Calcium. Part I through VI, NDL-TR-89-I through Vi (AD-6653591-1A-665364) (November, 1967). Availability: CFSTI (Part I-VI sold individually).

A study has been made of the neutron interaction probabilities for sodium, magnesium, chlorine, potassium, and calcium. Sets of neutron and gamma ray production cross sections have been prepared for these elements. This report, which is divided into six parts, describes the methods used to prepare the recommended set of data. Part I contains general information and the data format used. Parts II through VI describe the neutron cross sections for each of the five elements. The titles for Parts I through VI, respectively, are: General Information and Data Format, Sodium, Magnesium, Chlorine, Potassium, and Calcium.

1607. S. Gerstl, The Analytic Treatment of Arbitrary Anisotropic Scattering in the Energy Dependent Transport Theory, Nukleonik, 10, p. 227-236 (November, 1967).

A generalized spherical harmonics method is outlined to solve the one-dimensional energy and angular dependent transport equation in slab geometry, whereby any anisotropy of the scattering cross section can be treated analytically exact. The angular dependency of the scattering cross

section is described in a very general form by means of so-called "Anisotropy Functions" for which analytic expressions and recurrence formulas are stated. The applicability of this method os demonstrated by shielding calculations for gamma rays. Here, the spectra of the transmitted and reflected energy flux show new discontinuities which represent a typical boundary effect but were not observed before.

1608. C. Karzmark, T. Capone, Measurements of 6 MV X-Rays 1. Primary Radiation Absorption in Lead, Steel, and Concrete, Brit. J. Radiol., 41, 481, p. 33-39 (January, 1968).

Narrow and broad-beam absorption coefficients were evaluated for lead, steel and concrete using the primary X-ray beam from a 6 MeV linear accelerator. For narrow beams the absorption coefficients in cm^{-1} are 0.50, 0.31, and 0.084 for lead, steel and concrete respectively, while the corresponding tenth-value thicknesses are 4.6, 7.5, and 27.4 cm respectively. For broad beams the absorption coefficients in cm^{-1} are 0.41, 0.23, and 0.066 for lead, steel and concrete respectively, while the corresponding tehth-value thicknesses are 5.7, 10.0, and 34.8 cm respectively. Measurements were made over six tenth-value thicknesses of lead and four to five for steel and concrete. Frequently, leakage and unwanted scatter are larger than the transmitted primary intensity by factors of 5 or greater. The methods of preferentially shielding and discriminating against these spurious components and the corrections for them are described. The attenuation curves, the experimental arrangements and the problems of measurement are discussed.

1609. H. Vogt, Determination of the Dose Buildup Factors for Gamma-Rays in the Energy Region of 0.5-15 MeV in Laminated Shields of Iron and Water, ORNL-TR-1856 (ABS-THH-1029 and Nucl. Eng. Design, 6, p. 325-341 in German) (1967). Availability: Dep., CFSTI, JCL $3.60 fs, $1.28 mf.

The dose buildup factors were determined for a shield with two layers of iron and water which are to permit a comparison with the results of these calculations. The measuring method is based on differential scattered radiation measurements. The buildup factors here are valid for the case of an infinitely extended, parallel radiation field with a constant flux density which is incident perpendicular to the plane shields. The measurements were made behind 20 shield combinations with different iron and water thicknesses. (NSA)

1610. R. Erdmann, Neutron Transport from a Point Source in Two Adjacent, Dissimilar, Semi-Infinite Media, J. Math. Phys., 8, 5, p. 1040-4 (May, 1967).

A solution to the steady-state, two-dimensional Boltzmann equation is obtained for the flux due to a point source of neutrons located in a system of two adjacent half-spaces having the same mean-free path. Isopropic scattering and monoenergetic neutrons are assumed. The solution for an arbitrary source location is related to that for an interface source which is obtained by the Wiener-Hopf technique. Asymptotic expansions for the flux far from the source, both along and away from the interface, are derived and compared with approximate theory.

1611. F. Kruger, The Energy Flux of Gamma Radiation for Large Depths of Penetration, ORNL-TR-1793 (Translated from Kernenergie 10, 5, p. 147-149 (1967). Availability: CFSTI, JCL $1.10 fs., $0.80 mf.

For great distances from a monoenergetic gamma source the energy flux will be proportional to $e^{-\mu_o z} z_o^{-p}$ where μ_o is the absorption coefficient for the source photons and z the distance from the source. The parameter p is determined by the source geometry. The exponent η_o depends on the energy of the source photons and on the material which the radiation penetrates. We have calculated η_o for the elements H, Be, Na, Mg, Al, Si, Fe, Cu, Sn, W, Pb, U as well as for water. The results are given in a table.

1612. H. Yamakoshi, M. Nakata, A Method to Determine the Compton Scattering Angle from Its Cumulative Probability in the Monte Carlo Calculation, ORNL-TR-1857 (Senpaku Gijutsu Kenkyujo 2, 6, p. 311-16 in Japanese) (1965). Availability: Dep., CFSTI, JCL $1.60 fs, $0.80 mf.

A new way to decide the Compton scattering angle from a random number given as its cumulative probability is presented. This way is useful to trace rather higher-energy γ rays in the Monte Carlo method calculation. As an example, four tables necessary for the γ ray energy range from 5 to 10 MeV are shown. The present example gives the scattering angle with accuracy less than one degree through a one-table lookup step and, at most, five-multiplying and two dividing steps.

1613. J. Turner, The Possible Role of Momentum in Radiation Dosimetry--II. Extension to Charged or Uncharged Particles and Implications, Health Phys., 11, p. 1163-1175 (1965).

An earlier investigation of the possible role of momentum in radiation dosimetry has been extended to include charged particles. The earlier work considered neutrons and gamma radiation and the present treatment shows that the results found there can be applied in a consistent manner to the description of the dosimetry of charged particle radiation. The earlier calculations are reviewed and results are worked out here for protons, electrons and alpha particles. The relation of this work to a recent paper by Ivanov is shown. It appears that new methods of radiation dosimetry, whether in radiobiology or in radiation protection work, can be developed as a consequence of including absorbed momentum-or, more generally, other physical quantities-in addition to absorbed energy in the basic definition of radiation "dose."

1614. K. Morgan, J. Turner, Principles of Radiation Protection, John Wiley and Sons, Inc. (1967).

Sixteen chapters are presented on the following topics: the history of damage and protection from man-made and natural sources of ionizing radiation; the mechanisms and effects of the passage of heavy charged particles, γ rays, and x rays through matter; quantities and units of radiation dose; the physical basis of radiation dosimetry; detection and measurement of ionization, ionization methods of mixed radiation dosimetry, and special methods in dosimetry; theory of electron and β-ray spectroscopy; calculation of and standards for external and internal doses of radiation; biophysics of direct and indirect effects of radiation on biological materials; radiation biology at the molecular, cellular, and organismic levels; evaluation of human exposure data; characteristics, calibration, and maintenance of radiation detectors and component instrumentation used in health physics; and theory and criteria useful in the prevention of criticality accidents. Reference citations and problems are presented at the end of each chapter, and author and subject indexes are included at the end of the text. (NSA)

1615. H. Hilsdorf, A Method to Estimate the Water Content of Concrete Shields, Nucl. Eng. Design, 6, 251-263 (Condensed version of

NRSS 4, DASA-1875) (1967).

The water content of concrete is of decisive importance for the effectiveness of a concrete shield against neutron radiation. In addition, it influences the thermal conductivity of concrete and thus the temperature distribution in a concrete shield.

In the following, the modes by which water may be held within the concrete are described. Then, the major parameters affecting the water content of concrete are discussed and a method is given to estimate the water content of concrete both for the case of a drying concrete and for a concrete which is in moisture equilibrium with the surrounding atmosphere. The method is applicable for structural members exposed to temperatures up to 40°C. Some data are also given for the water content of concrete at elevated temperatures. In the final sections means of maintaining a high water content and experimental procedures to determine the water content of concrete are described.

1616. H. Hilsdorf, The Water Content of Hardened Concrete, DASA-1875 (NRSS 4), (Expanded version of Nucl. Eng. Design, 6, 251-263) (February, 1967). Availability: CFSTI.

Parameters affecting hydrogen content of concrete are discussed. Method is described to compute water content of concrete depending on mix proportions, age, size and exposure conditions of concrete. Sample calculations are given, and values of the water content of concrete members under various climatic conditions are tabulated.

1617. J. Diaz, Modeling Relationships in Neutron Transport Through Ducts and Cavities, AD-663570 (NRSS 5) (September, 1967). Availability: CFSTI.

If radiation reflection within a duct or cavity is treated by the usual albedo approximation, that is, on the assumption that radiation emerges from the wall at the same point at which it is incident, it is possible to establish modeling relationships between similar configurations differing only by a scale factor. The foregoing assumption is investigated, and it is found that the modeling relationships are approximately valid for fission neutrons as long as the linear dimensions are on the order of two and a half feet or more. This lower

limit may be decreased if the "effective reflection plane," which is always a few centimeters beneath the actual cavity interface, is utilized as the bounding surface of the duct or cavity.

1618. H. Beuchat, On the Analytical Solution of the Energy-Dependent Neutron Transport Equation in S_N-Formulation, ORNL-TR-1813 (EIR-98 in German) (November, 1966). Availability: Dep., CFSTI, JCL $8.10 fs., $2.81 mf.

An attempt was made to find an analytical solution of the $S_N$differential equation for the case of a cylindrical symmetrical arrangement (the case of a plain arrangement is contained in this arrangement as a special case). The following problems were treated extensively: explicit solution of the S_N differential equation in vacuum, solution of the S_N differential equation by Lie series at a development stage $r_o \neq 0$, solution of the S_4 differential equations at the singular development place $r_o = 0$ with special series, solution of the S_4 differential equations at the site $r_o \neq 0$ using the Lie series, convergent detection of the solution of the latter tow, solution of the S_N differential equation at the singular development site $r_o = 0$ using a special series recursion, and detailed statement of the preceding in the solution of a concrete boundary value problem. The basic studies were carried through to permit the complete analytical solution of the S_N differential equation. These analytical solutions have with respect to the usual S_N solutions the advantage of accuracy which can be assumed independent of the calculation method. The analytical solutions are so developed that in the case of long convergents or in the case of numerical instability the possibility of analytical continuation is given. A generalization of the analytical method to multidimensional problems appears possible. (tr-auth)

1619. R. Campbell, R. Berland, H. Alter, V. Luco, Legendre Expansion Coefficients for Theoretical Angular Distributions of Elastically Scattered Neutrons, NAA-SR-12533 (March 1, 1968). Availability: CFSTI.

Theoretically calculated angular distributions for elastically scattered neutrons were analyzed in terms of Legendre polynomial expansions. The theoretical calculations were performed by Agee and Rosen (LA-3538-MS, Vol. I and II0 and were based on combined optical model and Hauser-Feshback theory. From their results, thirty-six isotopes, ranging

in mass from sodium to plutonium-240, were analyzed. The resulting Legendre expansion coefficients have been forwarded to Brookhaven National Laboratory (BNL) in an ENDF/A compatible format, and are available from the National Neutron Data Center (NNDC) at BNL on request.

1620. B. O'Reilly, R. Lewis, Grams: A Fortran Program to Solve Gamma Ray Penetration Problems in Multislab Shields. Volume I: Numerical Methods, NAA-SR-12337, Vol. I (June 22, 1967). Availability: CFSTI.

The numerical analytic methods used to solve the one-dimensional energy-dependent transport equation for gamma rays migrating through a multislab shield are described. These methods are organized into a systematic procedure for solving the entire program. This procedure has been used to develop the Fortran code GRAMS, which implements the numerical solution of the problem.

1621. G. Matlack, C. Metz, Radiation Characteristics of Plutonium-238, LA-3696 (October 11, 1967). Availability: CFSTI.

The primary hazard in the use of ^{238}Pu as a biomedical power source results from the neutron flux caused by spontaneous fission and (α,n) reactions with light elements. Spontaneous fission will cause an activity of approximately 2200 neutrons per gram-second for material containing 81% ^{238}Pu. Neutrons from (α,n) reactions can increase this value to 8,000 to 10,000 unless every effort is made to produce ^{238}Pu which is extremely free of light element impurities. Most of the gamma radiation is of low energy and easily shielded. A significant fraction of the gamma radiation will, however, be caused by the growth of ^{241}Am, and by the 4n heavy element decay chain resulting from the decay of small amounts of ^{236}Pu. Gamma radiation from spontaneous fission and fission-product will be negligible.

1622. A. Babin, Modular Neutron Shield, UCRL-18093 (February, 1968). Availability: CFSTI.

A flexible, low-cost shielding system, consisting of water-filled steel cans and lead bricks, is described.

The shield is designed to protect against radiation, primarily from a spontaneous fission source of 50 to 100 μg ^{252}Cf and from a 1-curie gamma source.

Because of the modular construction used, shielding wall thickness and overall geometry can be readily modified, in order to attenuate different levels of radioactivity or to accommodate different enclosure sizes. Portable components include plain tank sections, zinc bromide windows, turntable unit, master-slave manipulators, interlocking 2-in. lead bricks, and structural support for assembled shield.

1623. L. Soffer, Activation of Sodium, Lithium, and Potassium in Compact Fast Reactors and Its Effect on Shielding, NASA TM X-1512 (April, 1968). Availability: CFSTI.

Neutron activation of sodium (Na), lithium (Li), and potassium (K) used as possible coolants in a "typical" compact fast reactor were analytically investigated. The natural elements, all individual isotopes, and one common eutectic (NaK 78) were examined.

Suitable energy dependent neutron fluxes were used in conjunction with activation cross sections to calculate saturated specific activities. Activation cross section data for the elements studied were compiled and tabulated, and these data are judged to be adequate for engineering calculations.

It was also assumed that 1 per cent of all the bromine (Br), krypton (Kr), xenon (Xe), and iodine (I) fission products were released into the coolant. The unshielded dose rates from both the fission products and the activated coolant were then calculated.

Results show that when zero holdup time is assumed, the best coolants, on the basis of unshielded activation dose rate, are K^{40}, K^{39}, Li^{7}, and natural Li. The worst are K^{41} and Na. Because of low natural abundance, K^{40} is impractical as a reactor coolant. Lithium dose rates are very sensitive to holdup time because of the short (0.8 sec) half-lives of the Li products. Holdup of even a few seconds greatly reduces the Li dose rates but not those for any other element. Lithium 7 held up for 8 seconds and gave the lowest activation dose rate of any coolant considered. The unshielded dose rate from the assumed fission product release was the highest of all. It was about two orders of magnitude greater than the zero-holdup Li^{7} rate, and about five orders of magnitude greater than the dose rate for Li^{7} held up for 8 seconds.

Gas evolution resulting from coolant irradiation was also investigated and found to be negligible except for Li^{6}.

1624. R. Bacigalupi, B. Thinger, Gamma Flux and Heat Deposition in Plum Brook - Comparison of Calculation and Experiment, NASA TM X-1539 (April, 1968). Availability: CFSTI.

A computerized point-kernel technique (QAD-P5) was used to calculate a gamma-heating map of a thermionic diode experiment proposed to be irradiated in the Plum Brook reactor core. A replica of the diode was used in the low-power mockup reactor at Plum Brook to generate corresponding gamma-heating maps experimentally using lithium fluoride dosimetry. A detailed comparison of experimental and calculated values generally showed agreement to within ±25%. Proper interpretation of the measured values is discussed herein.

1625. K. Lathrop, Ray Effects in Discrete Ordinates Equations, Nucl. Sci. Eng., 32, 3, p. 357-369 (June, 1968).

The nature of anomalous computational effects due to the discretization of the angular variable in transport theory discrete ordinates approximations is described and analyzed. The origin of these effects within the derivation of the S_n discrete ordinates equations is shown, and the effects are related to the nonequivalence of the general geometry discrete ordinates equations and the corresponding spherical harmonics equations. Procedures are given for the definition of two-dimensional discrete ordinates equations that are equivalent to the spherical harmonics equations. Elimination of ray effects from the two-dimensional S_2 equations by reduction to the diffusion theory equations is verified in a numerical example. Recipes for the elimination of ray effects are analyzed in the analytic solution of the infinite medium, isotropic line-source problem in the rectangular geometry, S_2 approximation. Optimum magnitudes for corrective source terms are indicated by the analysis. It is concluded that ray effects may be eliminated by midification of the discrete ordinated formulation, but that the extra computational effort may be more expensive than the alternative of increasing the order of angular quadrature and that the presence of discretization effects may serve as an indicator of the adequacy of the angular quadrature used.

1626. A. Shimizu, Calculation of the Penetration of Gamma Rays Through Two-Layer Slabs, Nucl. Sci. Eng., 32, 3, p. 385-391 (June, 1968).

The method of invariant imbedding has been applied to

the problem of penetration of gamma rays through two-layer slabs and turned out to be accurate and less time-consuming than the Monte Carlo method. A series of calculations wea made for two-layer slabs of water, iron, and lead for plane sources. An approximate formula for synthesizing the build-up factor of a composite slab from those of elementary layers is rerived from numerical solutions by modifying the formula originally proposed by Kalos.

1627. E. Straker, F. Mynatt, Calculations of the Effect of the Air-Ground Interface on the Transport of Fission Neutrons Through the Atmosphere, ORNL-TM-1819 (November 1, 1967). Availability: Dep., CFSTI.

Two-dimensional discrete ordinates calculations of neutron transport in an air-over-ground geometry have been performed for three different ground compositions. Results for a neutron fission source at 92 meters above the interface are compared with infinite air results. The calculated values of dose and thermal flux for both a fission and a Godiva leakage spectrum are compared with results from Operation BREN. Although the ground interface has a significant effect on the transport, the hydrogen content of the ground also has a large effect. Agreement between calculations and the BREN results is satisfactory when the Godiva leakage spectrum is used.

1628. J. Barthe, Tables of Neutron and Gamma Constants Straight-Line Attenuation Between Source and Point of Measurement with Accumulation Factor, ORNL-TR-1504 (CEA-N 571) (December, 1965). Availability: Dep. mn., CFSTI, JCL $1.60 fs., $0.80 mf.

The constants are collected in tables so that they might serve directly a number of Fortran programs that pertain to attenuation of gamma rays and of fast neutrons (Mercure, Orphee, Carmen and other programs presently being elaborated).

All of these constants are punched on IBM cards.

The radiation considered are subdivided into 12 groups of energy. The constants are tabulated for 12 elements, 8 compounds, 7 different types of concrete and air. There still remains 13 available spaces for other materials (new concretes, sands, etc.).

1629. M. Brainina, V. Kuznetsov, V. Generozov, V. Sakavich, Calculation of Dose Derivatives by Monte Carlo for Optimization of Shape and Composition of Shields, ORNL-TR-1884 (Zh. Vychisl. Mat. Fiz., 7, 4, p. 953-7 (July, 1967). Availability: Dep., CFSTI, JCL $1.10 fs., $0.80 mf.

Optimum shape and composition for a minimum weight shield may be determined using gradient programming; at each step of the iteration process, it is necessary to know only the first derivatives of dose beyond the shield. The derivitives may be calculated simultaneously with the dose calculations by the Monte Carlo method. Two cases of shield optimization are considered, with examples: optimum volume densities for given shield shape, and optimum shield shape for uniform composition. (NSA)

1630. B. Skvortsov, L. Sheiman, V. Chernyakhovskii, D. Pozdneev, The Scattering of 0.511 MeV Gamma Rays by Sheilding Barriers. (Summary of a stored article.), Soviet J. At. Energy, English Transl., 22, 5, p., 505 (May, 1967).

Results are presented by experiments with narrow beams of 0.511 MeV gamma rays incident on graphite and cadmium slabs. The energy and angular distributions of transmitted rays were measured as were the distributions of backscattered rays. Two scintillation spectrometers were used and the instrument spectra were unfolded by means of inverse matrices of the spectrometer response on a computer. An expression for the albedo if derived from the results and a discussion of the role of various interaction processes is given.

1631. N. Andryushin, B. Bulatov, The Spatial Distribution of Backscattered Gamma Radiation from Sources on the Surface of a Reflector. (Summary of a stored article.), Soviet J. At. Energy, English Transl., 22, 5, p. 507-8 (May, 1967).

Results are presented of measurements of backscattered gamma ray flux at various distances along the reflecting surface from isotropic Co-60 and Cs-137 sources. The reflectors investigated were ground and concrete. Curves of intensity versus distance are given, and an expression that approximates the data is derived. (RSIC)

1632. W. Chwetzoff, P. Poitevin, F. Dubois, Development of the Shielding for the Rapsodie Reactor, BNWL-TR-22 (Bull. Inform. Sci. Tech. No. 110, p. 3-29 in French) (December, 1966). Availability: Dep., CFSTI, JCL $4.60 fs., $1.67 mf.

The reactor Rapsodie 4 is shielded laterally by concrete which must be able to support working temperatures of the order of 200°C. It was developed on the basis of tests carries out on samples at room temperature, at high temperature, and finally under irradiation. Tests were performed on a mock-up representing a lateral shielding element of Rapsodie, to determine the possibilities of using fresh concrete and to verify that the concrete behaves satisfactorily after hardening. Special precautions were taken in the manufacture, the placing in position and the hardening of this concrete in order to keep its temperature at 60°C during the time necessary for the cement to set.

1633. R. Fiebig, Calcualtion of Position-Dependent Neutron Spectra in H_2O-Fe-H_2O Shield Configurations and Relations of Diffusion Approximations to Exact Transport Theory in Regions of Steep Flux Gradients, ORNL-TR-1849 (EUR-3472.d in German), (Expanded version of ORNL-TR-1859) (1967). Availability: Dep., CFSTI, JCL $8.10 fs., $2.90 mf.

The course of neutron spectra in a plane water-iron-water shield configuration was calculated with a neutron transport program BINE and group constants for a multi-group diffusion calculation were derived from the resuls. The position-dependence of the group constants was studied on the basis of the neutron fluxes and currents and considerable differences were found compared to the group constants obtained with the P_1 approximation. The transport results were recalculated by means of group constants determined by region and the results are discussed.

1634. Lead Development Association, Lead for Radiation Shielding, Lead Development Association, (1965).

This publication comprises in one volume the four booklets on radiation shielding first issued by the Lead Development Association during 1963 and 1964. The text of the original series has now been expanded and brought up to date. A new section describes the Regulations for the Safe Transport of Radioactive Materials issued by the International Atomic Energy Agency and their effect on the design of lead shielded containers. Also included are details of practical examples of lead shields which meet the requirements of good

practice described in this publication.

1635. W. Kreger, R. Mather, Scintillation Spectroscopy of Gamma Radiation, Volume 1 - Chapter III, Background, and Collimation, Gordon and Breach Science Publishers (1967).

Many of the circumstances of use of a gamma ray scintillation spectrometer require that the source or detector be enclosed in a partial shield which serves to reduce background, define or limit the direction of incident radiation, etc. This chapter treats sources of radiation background, shielding to reduce background, or provide proper geometry, and collimation.

1636. IAEA, Safety Series No. 6 - Regulations for the Safe Transport of Radioactive Materials, International Atomic Energy Agency, Vienna (April, 1965).

In 1961 the International Atomic Energy Agency, within the framework of its statutory functions and in accordance with recommendations made by its Preparatory Commission and by the Economic and Social Council of the United Nations, published safety regulations which could be applied to the national and international transport of radioactive materials by all means of transport.

At the same time, the Director General of the Agency indicated that these regulations would be revised at appropriate intervals in consultation with Member States and the organizations concerned and invited suggestions for their improvement in the light of experience and increased knowledge.

In preparing the revised regulations presented in this document, the Agency has received considerable support from its Member States and the organizations concerned, which have made extensive studies and suggestions in order to assist in its work. The Agency also convened several meetings of experts from its Member States and of representatives of a number of international organizations, and has been represented in several meetings convened by those organizations.

In publishing the revised transport regulations which result from that co-ordinated effort, the Agency aims at proposing a lasting framework of principles and rules, complemented by appropriate technical data, acceptable

for the safe transport of radioactive materials by air, land, and water. In particular, the developments which have been introduced concerning the packaging requirements, the methods for testing packages should facilitate the international acceptance of packages by the authorities concerned.

1637. NCRP, Medical X-Ray and Gamma-Ray Protection for Energies Up to 10 MeV - Equipment Design and Use, NCRP Report No. 33 (February 1, 1968). Availability: NCRP Publications, P. O. Box 4867, Washington, D. C. 20008 $0.75.

This report, which supersedes parts of NCRP Reports 24 and 26 (NBS Handbooks 73 and 76), is concerned with protection against radiation emitted by medical x-ray equipment operating at energies up to 10 MeV and medical gamma-beam therapy equipment. It presents recommendations pertaining to equipment design, use, and operating conditions, and to radiation protection surveys and personnel monitoring. To be covered in separate reports are recommendations concerning structural shielding design and recommendations concerning dental x-ray and veterinary x-ray protection and brachytherapy sources.

This report includes sections for the specific guidance of (1) the physicial and his associates, (2) the equipment designer and manufacturer, (3) the radiological physicist concerned with calibration procedures, equipment performance, and protection surveys and, (4) the radiological health inspector concerned with equipment inspection and survey measurements.

1638. D. Nachtigall, M. Heinzelmann, Measurement of the Flux Density and Effective Removal Cross Sections of Various Neutron Shielding Materials, UCRL-Trans-10128 (Jul-435-ST in German) (December, 1966). Availability: Dep., CFSTI, JCL $2.60 fs., $0.86 mf.

The attenuation of neutrons by different materials has been investigated. Flux densities and effective removal cross sections for neutron energies between 0.03 and 4.5 MeV and the corresponding buildup factors were determined. The flux dendity and the dose-rate were measured independently of energy for values ranging from thermal energies up to the maximum energy. Measurements were performed with radioactive neutron sources and with the nuclear reactor and those obtained with the long counter and the Anderson and Braun dosimeter were compared with those found with the five-sphere method.

1639. P. Kirkegaard, Nuclear Particle Transport with Emphasis on Monte Carlo and Shielding Calculations, RISO Report No. 136 (June, 1966). Availability: Dep. mn.

A description is given of a number of methods of calculating the transport of neutrons and γ-rays, mainly in reactor shields. Emphasis has been laid on the Monte Carlo technique. All the methods described have been programmed for the electronic computer GIER at Riso, and some of them also for the IBM 7090 computer at the Danish Technical University, Lyngby.

A simple bulk shield programme system has been developed which calculates the penetration of neutrons as well as γ-rays throughout a reactor shield. The neutrons are treated according to the removal-diffusion theory with one removal and one diffusion group. The γ-ray calculation relies on the build-up concept.

An improved removal-diffusion neutron shielding programme has been worked out, permitting an arbitrary number of removal and diffusion groups. A cross-section library with the elements most frequently encountered in shielding has been prepared.

A general outline of the Monte Carlo calculation method for problems in nuclear particle transport is given. Some minor reactor problems concerning self-absorption of γ-rays in fuel rods and distribution of the energy deposition on the various components of a fuel lattice cell are solved by the Monte Carlo method.

A Monte Carlo γ-ray shielding programme has been written, especially suited for laminated shields, in which case it is superior to build-up methods. The programme calculates the energy deposition rate throughout the shield as well as the dose rate at the external surface.

Finally, a Monte Carlo method for computing the axial neutron flux distribution in a short, absorbing rod is presented.

Whenever possible, the calculations described in this report have been supported by experimental results.

1640. J. Hacke, Dosimetry and Shielding in a 14-MeV Neutron Generator, ORNL-TR-1757 (Intern. J. of Appl. Radiation Isotopes, 18, 33-44 in German) (1967). Availability: Dep., CFSTI, JCL $2.60 fs., $0.86 mf.

With a 14-MeV neutron generator, which is used mainly for fast neutron experiments, the distribution of flux density and dose equivlent as well as the effectiveness of shielding have been investigated. Dose equivalent measurements were performed with an Andersson and Braun type dosimeter, which fits the whole energy range $0.025\ eV \leq E \leq 14\ MeV$. Dose equivalent transmission factors of normal concrete were determined in the range of 130-160 cm thickness, resulting a relaxation length of 15 cm. The results of shielding measurements are compared with results which follow from different methods of shielding calculations. To compare different neutrons dosimeters, some measurements have been taken in regions with different neutron spectra.

1641. T. Tagami, M. Kitazume, Analysis of Photoneutron Flux Distribution in Water-Shielded Reactors, NSJ-TR-93 (Nippon Genshiryoku Gakkaishi, 8, 12-15 in Japanese) (1966). Availability: CFSTI, JCL $1.60 fs., $o.80 mf.

In a water-shielded reactor, the photonuclear reaction $\gamma(d,p)n$ takes place between γ-rays emitted from reactor core and deuterons contained in the shielding water surrounding the core. Photoneutrons produced by the reaction are distributed in the water layer in addition to the neutrons originated from the core. The distribution of the thermal photoneutron flux in water not close to the core surface is approximated. Comparison of the calculated results to the experiments in BSR-1 shows that the approximation reproduces the photoneutron distribution reasonably well in a region where the neutron flux originated from the reactor core is negligibly small. (NSA)

1642. D. Pozdneev, L. Gokhshtein, S. Churin, Backscattering of Gamma Rays from Aluminum Slabs of Finite Thickness, Soviet J. At. Energy, English Transl., 22, 4, 358-363 (April, 1967).

Albedo for Al barriers of thicknesses up to 2 photon path lengths were calculated by the Monte Carlo method. Energies of 0.3, 0.5, and 1.0 MeV and incident angles of the monodirectional gamma force (relative to the normal to the barrier) of 0, 30, 45, 60, 75, and 85 deg. were investigated. From analysis of the data the empirical formula for calculation of the albedo from the barrier of finite thickness $A_x = A_\infty (1 - e^{-\alpha x})$ is proposed. Data on the dependence of A_∞ (albedo semi-infinite media) and A (empirical magnitude) on the energy of the primary

photon E_o and the incident angle Θ_o were given. The results of the claculation agree well with the data obtained by the member of statistical analysis and experiments.

1643. D. Pozdneev, Scattering of Gamma Rays by Iron Slabs, Soviet J. At. Energy, English Transl., 22, 4, 385 (April, 1967).

Results of Monte Carlo calculations of backscattering and transmission of gamma rays through iron slabs are presented. The incident radiation is 0.5 MeV and 1.0 MeV and the angle of incidence varies from normal (0°) through 30°, 45°, 60° and 75°. The thickness of slab varies from 0.1 mfp to 1 mfp. Empirical equations for calculating the flux albedo and the energy albedo are proposed as representative of the calculation results. Examples are given of calculations by the proposed equations and Monte Carlo method, and good agreement is obtained. A discussion is given of the contributions to the albedo of single scattering events, of the reflected energy spectrum and angle of incidence for various slab thicknesses. The transmitted gamma ray energy and angular distributions are also discussed and results for various angles of incidence and slab thicknesses are presented.

1644. B. Bulatov, N. Andryushin, Energy and Flux Albedo of Gamma Radiation (Summary of stored article), Soviet J. At. Energy, English Transl., 22, 4, 383-384 (April, 1967).

Relatively simple relations were derived for conversion from integral flux to integral energy albedos for gamma rays. It is shown that in the energy interval of primary gamma quanta of 0.2 MeV to 2.0 MeV, and for incidence angles of less than 60°, the ratio of flux albedo to energy albedo is a linear function of energy. Values of the proportionality factor are given for 0, 30, and 60° for water, concrete, iron and lead. A comparison is made with experiment, and curves of the albedo ratio as a function of energy below 2 MeV are presented.

1645. J. Brunner, W. Edwards, U^{235} Fission Product Decay Energy Release Rates, GEMP-579 (December, 1967). Availability: Dep.

Recent calculated and measured data were used to

prepare revised graphs of fission product decay gamma energy release rate as a function of time between 1 and 10^8 seconds following instantaneous thermal U^{235} fission and reactor operation from 1 minute to 1000 hours. Empirical expressions for fission product decay beta energy release rate were confirmed by recent measured data and were evaluated for the same operating and shutdown histories considered in the gamma analysis. Beta energy release rates, gamma energy release rates for seven energy groups, total gamma energy release rates, and the sum of the beta and gamma energy release rates are all presented graphically as a function of time after reactor shutdown.

These data were prepared under Contract No. AF04 (611) 10384. The total fission product energy release rates were reported previously in AFRPL-TR-65-196. The individual group data following reactor shutdown are included in this report for use in detailed shutdown reactor-shield analyses. The release rates following instantaneous fission are included so that data may be generated for other reactor operating and shutdown histories as needed.

1646. M. Nishi, T. Hyodo, T. Nakamura, Backscattering of Gamma Rays from Tin Slabs, Memoirs of the Faculty of Engineering, Kyoto Univ., Vol. XXIX, Part 4, p. 474-482 (October, 1967).

The backscattering of gamma rays from tin slabs was investigated as a function of slab thickness. The point isotropic ^{60}Co and ^{137}Cs sources were placed on the front face of the scatterer. The obtained data was converted to the response-corrected spectra with the aid of an inverse response matrix, and the angular and energy distributions of backscattered photons were obtained and compared with the other experimental and calculated results.

1647. J. Renken, Legendre Polynomial Expansion for the Klein-Nishina Formula, J. Appl. Phys., 38, 4925-6 (November, 1967).

When the photon transport equation is solved by a technique such as the method of discrete ordinates, the differential scattering cross section is represented by a sum of increasingly anisotropic components. In order to assess the relative importance of a given anisotropic component, it is useful to have an expansion of the Klein-Nishina formula in terms of Legendre

polynomials.

In this paper we consider the derivation of the required energy-dependent expansion coefficients. The behavior of the formulas is illustrated by numerical results.

1648. H. Davis, Aggregates for Radiation Shielding Concrete, Mater. Res. Std., 7, 11, 494-501 (1967).

Special and heavy aggregates used in the construction of concrete shielding structures are described, including iron ores, barite, serpentine, metallic products, and boron additives. Data on composition, physical properties, and aggregate costs are presented. A typical specification which covers practical aspects of aggregate procurement is included.

1649. J. Lott, C. Kesler, Service Behavior of Concrete for Radiation Shielding, Mater. Res. Std., 7, 9, 375-382 (September, 1967).

The behavior of portland cement concrete as a polyphase material is reviewed. Concrete for radiation shielding is subjected to stresses caused by structural loads and volume changes caused by moisture exchange and thermal gradients and cycling. Distress of concrete is associated with the incompatibility of the various phases of hardened concrete. Microcracks develop at the highly stressed paste-aggregate interface and propagate through the paste.

1650. S. Arnell, A. Hasselgren, O. Skeppstedt, R. Hardell, L. Jonsson, Thermal Neutron Capture Gammas Measured with Ge(Li) - Spectrometers and Interval Reactor Targets, Nucl. Instr. Methods, 54, 2, 165-180 (October, 1967).

An experimental arrangement for the study of thermal neutron capture gammas from reactions with low cross sections or where use is made of small amounts of separated isotopes is described. In this arrangement a through evacuated reactor channel and Ge(Li)-spectrometers for gamma detection are used. For the measurement of capture gammas above the pair production threshold a three crystal pair spectrometer employing a Ge(Li)-crystal surrounded by a bisected NaI-annulus is used. For the measurement of the low energy part of (n,γ)-spectra the same annulus is used in the anticoincidence mode.

The efficiency of the anticoincidence mantle for detecting backscattered quanta from the Ge(Li)-crystal is increased by adding a 5" dia. x 4" long NaI-crystal with a through axial hole to the anticoincidence assembly. The functioning of the spectrometers is illustrated with capture gammas from the A(n,γ)- and ^{54}Fe(n,γ)-reactions. The efficiency of the spectrometers and the background conditions are discussed and related to the minimum target quantities required.

1651. Y. Kanemori, Spatial Distribution of Gamma-Rays from a Cylindrical ^{60}Co Source, J. Nucl. Sci. Technol., 4(1), 32-36 (January, 1967).

The spatial distribution of γ-dose rates from three sizes of cylindrical ^{60}Co sources were experimentally obtained. A cylindrical source of long, slender form was found to produce dose rates in the radial direction higher than from a point source with the same total source activity. The spatial distribution of γ-dose rates thus measured agreed fairly well with calculated values estimated for cylindrical surface sources emitting γ-rays with a cosine angular distribution around the direction normal to the surface of the cylinder.

1652. J. Broerse, Penetration Parameters for 15-MeV Neutron Beams, Kerntechnik, 9, 10, 446-451 (1967).

The shielding characteristics of the following materials were examined: concrete, paraffin, laminated wood, water and steel. In these experiments we used flat slabs of concrete (thickness 32.6 mm, dendity 2.3 g/cc), paraffin (thickness 31.4 mm, density 0.9 g/cc), Jabroc densified wood laminate (thickness 62.5 mm, density 1.15 g/cc) and steel (thickness 3.1 mm, density 7.8 g/cc). All the slabs have an area of 60 cm x 60 cm. For the measurements of the attenuation characteristics of water an aluminum vessel was used for which the thickness could not be varied. Consequently the shielding effectiveness of water was only determined for a slab thickness of 38.5 cm.

The neutrons were produced by the D-T reaction; the total neutron yield was 3 x 10^{10} per sec, in the forward direction the emitted neutrons have an energy of 15.6 MeV. The radiation levels without any shielding at occupied areas amounted to several rem/h. The required shielding should decrease these values by a factor of about 2500 in order to reach permissible radiation levels for occupational exposure.

1653. M. Leimdorfer, R. Boughner, Monte Carlo Study of a Method for Lunar (Surface) Analysis Based on Spectroscopy of Gamma Rays Following 14-MeV Pulsed-Neutron Bombardment, Nucl. Instr. Methods, 50, 2, 302-308 (May, 1967).

In an investigation of the feasibility of analyzing the surface of the moon by gamma-ray spectroscopy techniques, Monte Carlo calculations were performed to estimate the spectral intendities of neutron-induced photons leaking from the moon surface during various time intervals following a burst of 14-MeV neutrons onto the surface. Four sets of calculations were done, all based on the lunar surface composition corresponding to an average composition of chondritic meteorites given by Urey and Craig, with varying amounts of carbon and hydrogen (or deuterium) added. The four spectra for the time interval $0-10^{-6}$s following the burst are all essentially the same, clearly showing the inelastic de-exitation gamma rays from O, Si, Mg, and Fe and also from C when that element was present. The spectra for later time intervals (10^{-6}-10^{-5}s: 10^{-5}-10^{-4}s, and 10^{-4}-10^{-3}s) are predominantly capture spectra dominated by iron capture gamma lines at 6.0 and 7.63 MeV and by the 2.23 MeV hydrogen capture gamma line when hydrogen was present. An examination of all the spectra indicates that, providing proper calibrations are made, the abundances of the various isotopes present on the lunar surface can be determined by gamma-ray spectroscopy techniques.

1654. I. Umeda, Monte Carlo Calculation of the Transmission of Gamma Rays from a Line Isotropic Source Through Steel Pipes, Nippon Kokan Technical Report, No. 4, 97-104 (February, 1965).

On a nuclear powered ship, radiation protection for passengers and crews is of paramount importance. From an economic point, radiation shielding, while designed to efficiently absorb radiation emitting from the reactor should be as small and light as possible. The steel piping of a ship utilizing nuclear energy constitutes an assembly of many line radioactive sources. The steel pipes act as an effective shield against these line sources. In order to quantitatively determine such shielding properites, the author calculated the transmission of gamma rays from the line sources through steel pipes by the Monte Carlo method. The present paper describes the procedure of calculation and cites examples of calculation of the buildup factors, the energy spectra and the collision numbers of transmitted gamma rays through the pipe, for the cases where ^{60}Co and ^{137}Cs line sources are shielded with steel pipes having a wall thickness of 10.0≈51.5mm.

The case histories of 5,000 photons for ^{60}Co and 10,000 photons for ^{137}Cs were followed on the IBM 7090 digital computer.

1655. J. Miyakoshi, A. Tsuruo, S. Miyasaka, Albedo Component of Gamma-Ray Streaming Through Straight Cylindrical Ducts, J. Nucl. Sci. Technol., 4, 8, 423-426 (August, 1967).

An experimental study was performed to investigate the effect of γ-ray streaming through cylindrical steel and lead ducts having various lengths and radii. In the experiment, ^{60}Co and ^{137}Cs point sources, and small sized detector (pulse dosimeter with plastic scintillator) was adopted, source and detector being located at the centers of opposite duct mouths. The albedo-to-direct component ratio is about 30% for steel ducts and 10% for lead at γ-ray energies of 1.25 and 0.661 MeV, with $L/R_0 \simeq 10$.

The measured results were compared with those calculated by the Monte Carlo method; they showed a good agreement. Consequently, the prediction of doses for straight cylindrical ducts having various length-to-radius ratios can be made by the procedures based on the Monte Carlo calculations.

1656. H. Greiss, Neutron Spectrum of Am-Be Source, Nucleonik (In German), 10, 283-287 (January, 1968).

The spectrum of a 10 Curie Am-Be-source was measured by means of the nuclear emuldion technique. The obtained spectrum chiefly confirms the results of other authors but shows a higher degree of resolution, having peaks of smaller halfwidth, a much more distinct shape of the minimum at 6 MeV and an additional peak at 9 MeV.

1657. D. Nachtigall, Average and Effective Energies, Fluence-Dose Conversion Factors and Quality Factors of the Neutron Spectra of Some (α,n) Sources, Health Phys., 13, 2, 213-219 (February, 1967).

Radioactive neutron sources are frequently used in dosimetry research and in Health Physics. In particular, (α,n) sources are extensively utilized. Radium, polonium, plutonium, and americium are used as α-emitters and beryllium is the chief target material. Detailed descriptions of the characteristics of (α,n) sources and methods of producing

them can be found in the literature. From this literature, tables of average energies, fluence-to-dose conversion factors, effective energies, and quality factors have been compiled.

1658. M. Shindo, JRR-4, JAERI-1120 (In Japanese) (August, 1966). Availability: Dep. mn.

Characteristic tests of experimental facilities of JRR-4 were done Data of these tests for Dry Shielding Test Facility, No. 1 Pool Facility and Lid Tank Shielding Facility are presented. Although all characteristics of the facilities were not tested, the most urgent data for the future use of these facilities were obtained and reported.

1659. Y. Kamemoto, Gamma-Ray Spectra of Fission Products, JAERI-1124 (In Japanese) (September, 1966). Availability: Dep.

A collection of the gamma-ray spectrograms for 76 different fission products with half lives longer than a few minutes, for uranium-239, and for neptunium-239 is presented. Half life, absorber, gamma-ray detector, geometry, decay scheme, etc., are indicated on each spectrogram.

1660. V. Zharkov, T. Zorina, V. Terentev, A Correction to the Spherical-Harmonics Method for Solving the Transport Equation, Soviet J. At. Energy, English Transl., 22, 1, 45-46 (January, 1967).

A solution of the one-velocity transport equation in spherical geometry by a modification of the spherical-harmonics method is discussed. Even in the lower P_N^{mod} approximations the accuracy is appreciably greater than in the ordinary P_N approximation. The results may be used to solve problems related to the behavior of localized absorbers in neutron fields.

The essence of the modification consists in taking into account the effects of the higher harmonics in the expansion of the solution in Legendre polynomials - the harmonics which are discarded in a given P_N approximation. A correction factor is introduced based on the use of a fictitious source.

1661. D. Broder, M. Kulakovski, V. Savitski, A, Mirenkov, Dubrovski, P. Lavdanski, V. Solovev, Sheilding Properties of Borated Heat-Resistant Chromite Concretes, Soviet J. At. Energy, English Transl., 22, 2, 126-127 (February, 1967).

Results of the experimental investigation of the shielding properties of borated chromite heat-resistant concrete are presented and compared with the results of theoretical calculations.

The experimental study of shielding properites was made in the horizontal beam of the VVR-TS reactor at the L. Ya. Karpov Physics and Chemistry Institute using boron trifluoride counters, phosphorus threshold detectors, and indium resonance detectors.

A method for calculating the spatial distribution of neutrons from a monodirectional source is described. The calculated neutron spectrum at the outer edge of the VVR-TS core is given as are the results of the calculation of the neutron penetration through the concretes performed by this method, and also the results of calculating the readings of the detectors used in the experimental part of the work. The data obtained are listed in the table.

It was extablished that the introduction of more than 30 kg/m^3 of boron into the chromite concrete is useless since it does not lead to any significant improvement in shielding properties. The good agreement between the calculated and experimental results shows that the claculational method described in the report is accurate enough and gives a correct description of the spatial distribution of neutrons from monodirectional sources in shields.

1662. V. Mashkovich, B. Sinitsyn, S. Tsypin, A. Nikolaev, V. Sakharov, Thermal Column Converter for Shielding Studies, Soviet J. At. Energy, English Transl., 22, 2, 132 (February, 1967).

The article deals with the design and radiation characteristics of the thermal column converter in the Br-5 reactor facility. This converter was designed to study radiation transmission through non-uniformities in the shielding.

The converter assembly consists of a tank of stainless 2 x 2 x 2.5 m, the front face 2 m^2 and adjacent to the reactor thermal column. The converter, extracted from

its shielded container (a source-holder combination, U^{235}-enriched uranium dioxide), is set into operating position by remote control. A 2 cm thick boron carbide filter is placed on the outer surface of the front face of the tank. The filter has a hole 250 mm in diameter and a cadmium sheet 1 mm thick on the inner surface for absorbing thermal neutrons. The radiation detectors can be arranged at any point in the experimental tank by means of a special remote positioning device.

The measurements showed that the converter functions as a disk type isotropic source 250 mm in diameter with a capacity of $1.57 \cdot 10^8 \pm 0.07 \cdot 10^8$ neutron fission events per cm^2 sec at a reactor power rating of 5 MW.

1663. V. Dubrovski, P. Lavdanski, V. Solovev, M. Kulakovski, V. Savitskii, Heat Release in Borated Concrete Shields, Soviet J. At. Energy English Transl., 22,2, 127-128 (February, 1967).

Results of an investigation of the heat release due to the absorption of the capture γ- and α-particles from the $B^{10}(n,\alpha)\ Li^7$ reaction in five concretes having different boron concentrations are described. The average energy liberated in the capture of a neutron by boron is less than half that liberated in the capture of a neutron by most of the other constituents of concretes. Therefore the addition of boron to concrete leads to a decrease in heat release and to a redistribution of it over the thickness of the shield (cf. table).

The study included ordinary concrete, concretes with iron ore and metallic scrap added, and also heat-resistant chromite and magnesite concretes which may be used in shields up to temperatures of 1770°C.

Graphs showing the distribution of heat release allow the shield designer to determine the efficacy of such concretes in various structural elements of the shield for hard spectra.

On the basis of an analysis of the data recommendations are made for the boron concentration in shielding concretes. (15 kg/m^3 of natural boron).

1664. N. Andryushin, B. Bulatov, The Spatial Distribution of Scattered Gamma Radiation in a Cavity for a Collimated Primary Source, Soviet J. At. Energy, English Transl., 22, 5, 522-523 (May, 1967).

The spatial distribution of scattered gamma-radiation in a spherical cavity from a collimated primary source has been investigated. The collimated flux of primary gamma radiation is incident on the wall of the cavity. The radiation source is at the center of the cavity. All calculations were permormed for a Co^{60} source and a cavity with "infinitely" thick iron walls. Since the scattering properties of iron are nearly the same as those for concrete, the calculations apply to cavities in concrete - the material most frequently used in structures housing intense gamma sources.

The calculations used data on the angular distribution of scattered gamma-radiation for normal incidence of a narrow beam on an iron structure.

1665. S. Sugiyama, T. Tomimasu, High Energy X-Ray Albedo for Pb, Cu, and Duralumin, Nucl. Instrm. Methods, 53, 346-348 (1967).

Angular distribution of the energy albedo for Pb, Cu, and duralumin irradiated by X-rays with maximum energy from 11.3 MeV to 23.3 MeV have been carried out by a plastic scintillation detector. Experimental results have been used to determine the total energy albedo with an accuracy of the order of 12%. The energy albedo for Pb, Cu, and dyrakynub are 2.4%, 2.0%, and 1.5% respectively at 23.2 MeV, and 1.7%, 2.0%, and 2.2% respectively at 15.5 MeV.

1666. J. Price, R. French, Monte Carlo Study of Directional Response Functions for Fallout Radiation Shielding Calculations, RRA-M73 (June 9, 1967). Availability: DTIE.

Directional response functions of the type used in reduction factor calculations employing the engineering method outlined in NBS-76 were calculated from Monte Carlo data. Response functions were determined for both "thin" and "thick" barriers and for radiation reaching the receiver from both above and below the plane of the detector. The data used in the calculations were obtained using the COHORT Monte Carlo procedure. The functions for "thin" barriers indicate slight differences in magnitude from the directional response functions based on moments method data. The response function for "thick" barriers was divided into two separate functions: one for radiation entering above and one for radiation entering below the detector plane. These two functions differed in both shape and magnitude from the single

function given in NBS-76 and lead, in the cases examined, to large changes in the reduction factor. However, it is not presently clear that use of the "improved" directional response functions leads to any overall improvement in the engineering method.

1667. S. Belov, Yu. Kazanskii, V. Popov, V. Demin, A. Lobakov, Coefficients of Secondary Gamma-Radiation for Aluminun, Copper, and Tungsten, Soviet J. At. Energy, English Transl., 19, 5, 1419-20 (1965).

Earlier papers provided the experimentally determined coefficients of secondary gamma-radiation for iron, nickel, and nickel with boron on the basis of the neutron spectra of a Po-Be source and of the reactor of the RIZ stand.

In our experiments, we determined the coefficient of secondary gamma-radiation for aluminum, copper, and tungsten. The measurements were performed on the RIZ stand. A single-crystal scintillation spectromenter with sodium iodide, whose diameter and length were equal to 40 mm, served as the gamma-radiation detector.

The values of the β coefficients and of the intermediate quantities necessare for calculating β are given in the table.

1668. C. Maeder, A Three-Dimensional P_N Spherical Harmonics Theory in Cylindrical Geometry, Nucl. Sci. Eng., 33, 1, 128-138 (July, 1968).

A cylindrical system of finite height is being considered, consisting of a series of homogeneous, concentric annular zones surrounded by moderator in the radial direction. For this system, a complete solution of the monoenergetic, stationary neutron transport equation is found in the P_N spherical harmonics approximation. In the moderator, the scalar flux converges asymptotically towards the solution of the diffusion equation. When coupled with the heter ogeneous (source-sink) theory, the method yields the scalar flux in a reactor and, after minor modifications, also applies to the exponential assembly.

1669. S. Gerstl, W. Kofink, Description of Anisotropic Scattering in the Double P_1 Method by Means of Anisotropy Functions, Nucl. Sci. Eng., 33, 2, 249-251.

With the summation of the anisotropy function series, we have shown the feasibility of the continuation of analytic calculations in the energy-dependent transport theory. The aim of this theory is to carry out the limit process as L appreaches infinity from a double P_L approximation towards the exact solution achieved by Kofink in the case of monoenergetic neutron transport in the formulation of the ordinary spherical harmonics method. As we can treat any arbitrary anisotropy of the scattering cross section analytically exact by means of the anisotropy functions, we applied this theory to shielding calculations for gamma rays. The energy spectra in the interior and at the surface of a slab, calculated from a double P_1 approximation, are found to be in good agreement with other available calculations.

1670. P. Juaho, H. Kalli, A Variational Approach to the Selection of the Direction Sets in the Discrete S_n Approximation to Neutron Transport Theory, Nucl. Sci. Eng., 33, 2, 251-254 (August, 1968).

In this paper, we study an approach to the selection of the directions and the weights. We start from the analytic S_n solution for the scalar flux and treat the discrete directions and the respective weights as free parameters. By variationally choosing the optimal form of the S_n scalar flux, we develop conditions for the selections of the free parameters. We give a detailed introduction to the method by selecting the S_2 directions for a bare homogeneous slab reactor. Two variational formulations are used: 1) the well-known variational method vased on the integral transport equation, and 2) the self-adjoint variational formulation for monoenergetic Boltzmann equation derived by Toivanen.

1671. R. Ackroyd, The Accurate Solution of Transport Theory Problems by the Use of a New Version of Diffusion Theory, TRG-1568(R) (1968). Availability: Dep., CFSTI, UK 7s.

Each group of a transport theory model is treated so that each group equation and its associated boundary conditions are replaced by spherical harmonic equations, of an odd order for the moments of the angular flux, and the associated boundary conditions for the moments. The moment equations and boundary conditions are then cast into a form with the appearance of a generalised version of multi-group diffusion theory. The even moments correspond to the net group currents of diffusion theory. At an interface the moments, i.e. the analogous group fluxes,

are continuous, and if the scattering is isotropic, the product of the gradient of an analogous group flux and the diffusion coefficient of diffusion theory is continuous.

The principal application foreseen for the analogy is in thr transport theory calculations. This version of transport theory (in the mould of multi-group diffusion theory) can be used in areas where classical diffusion theory fails and it can be grafted onto classical diffusion theory in areas where the latter is adequate. This method for transport theory calculations is illustrated by an analytical treatment of boundary transients at the interface of a core and reflector. The method is probably of greatest interest for numerical work. For example, a 16-energy-group transport calculation by the method of the effect of a grey control rod requires, in effect, a 16-group classical diffusion theory calculation away from the rod and $16[(M+1)/2]$-group diffusion theory calculation for the rod and its neighbourhood, when M spherical harmonics are used for the angylar flux in the rod. The method os of some interest because it suggests a way in ehich transport theory calculations on a reactor might be accomplished with very little more expenditure of machine time than is required for diffusion theory calculations.

1672. W. Reinig, Advantages and Applications of ^{252}Cf as a Neutron Source, Nucl. Appl., 5, 1, 24-25 (July, 1968).

Californium-252 is an intense neutron emitter. It can replace other neutron sourcea in many applications. Its unique characteristics may make new uses for neutrons practical.

1673. W. Edwards, Shield Kernel-Diffusion Analysis, GEMP-599 (March, 1968). Availability: Dep., CFSTI.

A kernel-diffusion analysis sequence, and associated nuclear data, were developed for shield nuclear analysis. The methods, analysis sequence, and programs are described briefly in this report; and the nuclear data are tabulated.

1674. G. Spangler, Fast Neutron Weighting Factors for Manrated Snap Reactor Applications, NAA-SR-MEMO-11901 (April 20, 1966). Availability: Dep., CFSTI.

A review of the RBE problem is presented and its relation to fast neutron weighting factors for SNAP manrated reactor systems is discussed. Reasonably justified fast neutron weighting factors can be specified for particular SNAP reactor applications; however, there is insufficient experimental justification for a firm specification of general weighting factors.

1675. E. Slaggie, J. Reynolds, C-12 Fast Neutron Cross Sections and Legendre Moments Below 15.0 MeV, KAPL-3099 (June 30, 1966). Availability: Dep., CFSTI.

An evaluated library of cross sections and elastic scattering Legendre Moments is given for C^{12} between 0.01 and 15 MeV. The cross sections below 4 MeV and the moments below 7 MeV were calculated using a fitting formula based on a combined resonance and potential well theory. Above 8.4 MeV the moments were obrained from a non-spherical optical potential calculation. These calculations are described and an extensive list of references to available data is given.

1676. S. Abdullah, M. Hussain, A. Hossain, A. Siddigi, Radiation Levels Around the 3-MeV Dacca Van de Graaff Generator, AECD-HP-4 (February, 1966). Availability: Dep.

An estimate of the Gamma radiation level at different places within the Van de Graff, generator room was made with the help of gamma monitors for the purpose of safety of personnel working around the machine. The measurements were conducted with the machine operating at 2 MeV with different beam currents. From the readings, isodose curves have been drawn. It was interesting to find that the dose rate was high only at the back end of the machine as expected, but also at one side of the Van de Graaff.

1677. R. French, On the Buried-Source Model for Computing Fallout Ground Roughness Effects, Health Phys., 14, 4, pp. 331-338, (RRA-M72) (April, 1968).

Most calculations of the gamma radiation environment produced by fallout assume that the radioactive material is uniformly deposited on a smooth surface. A recent study of possible mathematical models for including the effects of ground roughness on the gamma-ray energy and angle distributions in air above fallout placed emphasis on the "buried-source model." This model is based on the assumption that the fallout is covered by a uniform layer of soil whose thickness corresponds to the degree of ground roughness. Uncollided and singly-scattered photon energy and angle distributions 3 ft above several different ages of fallout were computed as a function of source depth using the buried-source model. After adjusting the results to account for the multiply-scattered component (approximately ten percent of the total), comparisons were made with gamma-ray energy and angle distributions measured over fallout deposited on a dry lake bed and on rough desert terrain. The comparisons indicate that the buried-source model adequately predicts the gross effects of ground roughness. In addition, the calculations performed employing the model provide some interesting insight on the detailed effects of ground roughness.

1678. H. Whittum, J. Witthaus, ML-1 Shield Study, AGN-TP-149 (CONF-651101-50) (1965). Availability: Dep., CFSTI.

Operational and twenty-four hour shutdown computed dose rates for the ML-1 reactor were compared to measured dose rates and dose rates computed during ML-1 design. The computational methods used and the results from these calculations are discussed. (NSA)

1679. F. Biggs, R. Lighthill, Analytical Approximations for X-Ray Cross Sections, SC-RR-66-452 (February, 1967). Availability: Dep. mn., CFSTI.

Analytical approximations efficient for use in computer programs are given for the photoelectric, incoherent-scattering, and energy-transfer cross sections for the range of atomic numbers $1 \leq Z \leq 100$ and in the photon energy interval $E \geq 0.1$ keV. The photoelectric approximations are based largely on experimental data at low energies, whereas at high energies they are constrained to approach the theoretical high-energy limit. The incoherent

scattering cross sections are approximated by rational functions fitted to the Klein-Nishina relations. The effect of round-off error in the usual form of the Klein-Nishina relations is also shown. Approximation parameters are given in tables.

1680. E. AAlto, A. Krell, Attenuation of Neutrons and Photons in A Duct Filled with a Helical Plug, AE-261 (December, 1966). Availability: Dep. mn.

Neutron and gamma attenuation was studied in a cylindrical duct having a length-to-diameter ratio of 7 partly 36 vol %) filled by a helical (screw-like) plug with an adjustable number of turns. The total neutron leakage through the duct decreased by a factor of six from zero to 0.5 turn, and by an additional factor of three from 0.5 to 2 turns (saturation value). The leakage with less than 0.5 turn is governed by fast flux and above 0.5 turn by epithermal flux. It is ovserved that the attenuation calculation in a homogenized, infinite material combined with the predicted attenuation in a duct gives the measured saturation attenuation. The length coordinates used must be slightly modified in this case. The attenuation of gamma radiation in the saturation case (about 3 turns) equals that predicted without the duct attenuation in a homogenized infinite material.

1681. C. Joenemalm, Nomogram for Determining Shield Thickness for Point and Line Sources of Gamma Rays, AE-252 (October, 1966). Availability: Dep. mn.

A set of nomograms is given for the determination of the required shield thickness against gamma radiation. The sources handles are point and infinite line sources with shields of Pb, Fe, magnetic concrete (= 3.6), ordinary concrete (= 2.3) or water. The gamma energy range covered is 0.5 to 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. Also included are data for calculation of the source strength for the most common materials and for fission-product sources.

1682. F. Attix, Ed., W. Roesch, Ed., Radiation Dosimetry, Second Edition, Volume I - Fundamentals, Academic Press, New York and London (1968).

The topics treated are (1) Basic concepts of dosimetry, (2) Microscopic energy distribution in irradiated matter, (3) X-ray and gamma-ray interactions, (4) Charged particle (5) Mathematical theory of radiation fields, (6) Neutron interactions and penetration in tissue, (7) Ionization, and (8) Cavity-chamber theory. (RSIC)

1683. J. Fishwick, Radiation-Absorbing Glass, AD-624874 (October 27, 1965). Availability: CFSTI.

The regions of glass formation in the system Li_2O-B_2O_3-PbO-SiO_2 were determined at the levels of 10, 15, and 20 wt % Li_2O. Three compositions were selected at the levels of 10, 30, and 50 wt % PbO as potential glasses for the use in the optical system of a radiation-protected pod. The manufacture of the three glasses is described, and their properties are reported.

1684. A. Szilvasi, Main Shield for Whiteshell Reactor No. 1, AECL-2544 (R-64CAP59) (November, 1964). Availability: Dep. mn.

Shield thicknesses for the WR-1 main shield system were calculated. The detailed results of these calculations are presented. (NSA)

A 7ft. thick biological shield made of ilemenite concrete (220 lbs/cu.ft.) is required on the side in addition to an 11 in. thick thermal shield made of equal volume iron and water which protects the concrete from overheating.

A 38 in. thick iron-water shield (equal volume iron and water) is required above and below the reactor after shut down. This shield if 44 in. thick in order to form an operational shield together with the 18 in. thick steel deck plate above the reactor. No additional shield is required downward.

1685. N. Francis, Variational Methods in the Calculation of Spatially Dependent Spectra, CONF-660204-4 (1965). Availability: Dep. mn., CFSTI.

A discussion is presented of variational methods for

cases in which the quantities studied are stationary with respect to the variable of interest. Resonance capture is not studied. After an introduction of the spatially independent problem, a specific thermal-energy-range spatially dependent neutron problem is discussed. The method is then applied to calculations of group constants for a multigroup Monte Catlo method, multigroup space-energy problems in slab geometry, and the use of overlapping groups in diffusion theory. (NSA)

1686. K. Lathrop, N. DeMuth, Biorthogonal Angular Polynomial Expansions of the Two Dimensional Transport Equation, LA-DC-8478 (1966). Availability: Dep., CFSTI.

In forming approximations to the Boltzmann transport equation, a common approach involves the expansion of the angular dependence of the directional flux in spherical harmonic functions. A method is described in which the directional flux is expanded in either of two systems of polynomials in two variables. (NSA)

1687. N. Steen, A Simple Method to Improve the Efficiency of the Sigma sub t/Estimator in Certain Monte Carlo Programs, WAPD-TM-609 (October, 1966). Availability: Dep. mn., CFSTI.

The $\Sigma_a/\Sigma t$ estimator (often used in Monte Carlo programs to estimate neutron capture rates in assemblies or reactor materials) is inefficient in regions having a high probability of neutron entry but a low collision probability. A simple method is described to enhance the efficiency of this estimator in such regions by fictiously increasing the optical thickness. Proof is given that this method will not bias the capture estimates or the flux, and examples are provided to show the improvement that may be achieved.

1688. J. Rappeneau, J. Gilbert, M. Lagorio, P. Piron, Irradiation Tests of Concretes, BNWL-tr-21 (Bull. Inform. Sci. Tech. (Paris), No. 110, 31-48, in French) (December, 1966). Availability: Dep., CFSTI, JCL $2.60 fs, $1.04 mf.

The tendency in nuclear reactor development at present is to bring the concrete shielding closer to the reactor core, thus subjecting it to increasingly higher irradiation

doses and working temperatures. It is therefore necessary to know how concrete behaves under the effects of neutron radiations and heat. A review is presented of the important properties required in special concretes and the methods used to study them and the hot cell in which irradiated samples are measured is described. The irradiation devices are also described along with an irradiation experiment. The results of the first experiments are given; these concern the variations in physical and mechanical characteristics of the types of concrete studied.

1689. D. R. Rogers, ML-15 Percent Boric Acid Shield Solution Experiment, TID-23940 (AN-AGCR-936; EAD 314) (October 5, 1965). Availability: Dep., CFSTI.

The experimental results of the addition of 5 wt % boric acid to the water shielding of the ML-1 reactor are presented. Results show that dose rates with 5 wt % solution were about 20% higher than dose rates reported for 10 wt % solution, that no correlation between fast neutron dose rate and boric acid concentration exists, and that previous predictions of the effect of boric acid concentration on both the total shutdown gamma dose rate and that portion of the total dose rate due to stainless steel activation have been overestimated. (NSA)

1690. M. S. Coates, P. D. Goode, D. B. Gayther, Measurements of Fast Neutron Spectra in Reactor Materials, AERE-R 5364 (February 1968). Availability: HMSO 8s

A series of neutron energy spectra emerging from spherical shells of natural uranium, polyethylene, graphite and sodium have been measured in the energy region 300 eV to 6.5 MeV. The time of flight method was used with a 60 m path and a pulsed source of fast neutrons provided by the 45 MeV linear accelerator. The source was located at the centre of the shell and spectra leaving the surface at 0° and 45° to a radius vector were determined. In addition a few measurements were made of spectra in the 0° direction from different penetration depths into the shell wall. The source was designed to emit neutrons isotropically and the aim of the measurements was to provide spectra in simple one dimensional systems in order to test the nuclear data sets used in reactor calculations. The experimental method is described in detail and some comparisons based on discrete ordinate solutions of the Boltzmann equation are presented.

1691. H. M. Kelley, Graph of Curies Versus Half-Life for All Radionuclides, Health Phys., 14, 271-272 (March 1968).

In its usual form, the equation for the specific activity of a radionuclide is an inverse function of both the half-life and the atomic weight, since specific activity requires a different number of atoms for every nuclide. By changing the dependent variable from specific activity to activity (disintegration rate) per gram atomic weight the equation specifies the same number of decaying atoms for any nuclide, and the disintegration rate is inversely proportional to half-life alone. A single linear log-log plot relates decay rate and time, and calculation of specific activity for all radionuclides is facilitated. A graph is presented showing this relation with three overlapping scales from 10^{-3} to 10^{6} yr and from 10^{8} to 10^{-1} Ci/g atomic weight. In order to obtain specific activity the disintegration rate of a nuclide, obtained from the graph, is merely divided by the atomic weight. (NSA)

1692. L. V. Spencer, P. Flusser, Properties of a Useful Biorthogonal System, J. Res. Nat. Bur. Std. B Math. Phys., 71B, 4, 197-211, (December 1967).

In radiation penetration theory, infinite medium flux distributions have for some years been calculated using biorthogonal functions called $U_n^k(z)$. The spaces spanned by these functions, transformation kernels, generating functions, recursion relations, asymptotic trends for large n, many expansions, and relations with well-known orthogonal polynomials are worked out.

1693. D. J. Jones, P. Munnery, Production of Tungsten Alloy Penetration Radiation Shields, Powder Met., 10, 20, 156-73, (1967).

Manufacturing procedures are given for a series of high-density sintered tungsten-nickel-copper and tungsten-nickel-iron alloys covering a density range of ~ 17 g/cm^3 to > 18.5 g/cm^3.

Physical and mechanical property data for these alloys are given and the effect of sintering variables on these properties is discussed.

Reference is made to equipment and techniques employed in the manufacture of radiation shields to meet the most stringent demands brought about over the years by the increased strength of radiation sources and the widening scope for the application of artificial radioactive elements.

1694. H. Schultz, S. Sasse, Analytical Evaluations for the Optimum Selection and Arrangement of Shielding Materials in Reactor Shields, Nucl. Eng. Design, 6, 447-464 (1967).

The designer of radiation shields is interested in an optimum selection and arrangement of shielding materials with respect to some optimization parameters as expenses, weight, thickness, etc. This becomes very difficult because of the complicated nature of the attenuation process and the complex linkage between the shielding characteristics in the various materials. The problems of such calculations are so complicated that generally a complete solution is not possible. Since the beginning of shielding theory, many efforts have been made to derive simplified calculations for this purpose with which at least a first approximation of the optimum conditions can be achieved. All previous studies concerning this matter are collected and discussed with respect to their applicability.

1695. C. M. Davisson, Interaction of Gamma-Radiation with Matter. Alpha-, Beta- and Gamma-Ray Spectroscopy, Vol. 1, 37-78, North-Holland Publishing Company, Amsterdam (1965).

The following topics are treated:- photoelectric effect, scattering by electrons, pair production, nuclear interactions and Delbruck scattering, total attenuation coefficients, and absorption methods.

1696. L. L. Meisberger, R. J. Shalek, R. J. Keller, Effective Attenuation in Water of Gamma Rays of Gold 198, Iridium 192, Cesium 137, Radium 226, and Cobalt 60, Radiology, 90, 5, 953-957 (1968).

The attenuation in water of the gamma rays of gold 198, iridium 192, cesium 137, radium 226 (in equilibrium with daughter products), and cobalt 60 has been measured experimentally and compared with calculations of the transport of gamma rays in water and air. From the experiments and calculations, which were in reasonable but not perfect agreement, the ratio of absorbed dose in water compared to that in air was derived for each type of source for distances of 1 to 10 cm from point sources. A third order polynomial was fitted in each case for this ratio related to distance.

1697. J. A. Lonergan, D. F. Herring, Gamma-Ray Shielding Studies with Monoenergetic Gamma Rays from Positron Annihilation, Nucl. Appl., 5, 2, 79-84 (August 1968).

A positron beam is produced and accelerated by an electron linear accelerator. The positrons annihilate in-flight when passed through a beryllium foil to produce nearly monoenergetic gamma rays with energies comparable

with the positron energy and an energy spread essentially equal to the energy spread of the positron beam, i.e., 2%. Such a 10-MeV gamma-ray beam was collimated and directed at a slab of aluminum that had a thickness corresponding to one mean-free-path for 10-MeV gamma rays. The energy spectra at 0, 15, and 30° to the incident beam were measured with a NaI crystal. The results were integrated and compared with Monte Carlo calculations. The experiments and calculations agreed within the experimental uncertainty.

1698. E. M. Gelbard, L. A. Hageman, J. A. Davis, Solution of the Discrete Ordinate Equations in One and Two Dimensions, WAPD-T-2028 (CONF-670410-2) (July 1967). Availability: Dep., CFSTI.

Some methods of solving discrete-ordinate equations are reviewed. All of the methods considered are applicable to multigroup transport equations, but only the monoenergetic neutron transport equation is discussed. Numerical techniques for treating one-dimensional problems are considered first; two-dimensional configurations are then discussed. (NSA)

1699. M. Ladu, P. Picchi, M. Pelliccioni, G. Verri, A Contribution to the Skyshine Study, Nucl. Instr. Methods, 62, 51-56 (1968).

The results obtained by application of the Monte Carlo method to the skyshine study are described. We found that the neutron fluence Φ per primary neutron decreases with distance r according to the laws $\Phi(r) = Ar^{-\alpha}$ for distances between 20 m and 300 m, and $\Phi(r) = B\exp(-r\lambda)$ for greater distances. The dependence of the coefficients A, B, α and λ, on a number of different parameters is studied. The results are compared with the Lindenbaum theory, and a good agreement is found. The effects of shields over the intensity and energy distribution of the diffused neutrons are also studied. Finally, the values of the back-scattering coefficient of the earth's surface is obtained.

1700. M. Kitazume, Gamma-Ray Scattering from Point Sources by Infinite Plane Surfaces, J. Nucl. Sci. Technol., 5, 3, 98-103 (March 1968).

A method of evaluating the γ dose-rate resulting from a point source and scattered by a semi-infinite plane surface has been developed in the form of a single scattering formula. In this formula, the effect of multiple reflection within the medium is incorporated as buildup factor for the reflected γ rays. Results of calculation with this formula are compared with values based on a semi-empirical expression using the differential dose albedo of a semi-infinite concrete expanse, and with experimental values obtained on dose rates of γ rays reflected by concrete and lead surfaces from point sources of ^{60}Co (1.25 Me), ^{137}Cs (0.66 MeV) and ^{192}Ir (0.4MeV). It is seen that the results of the calculation agree fairly well with the experimental results.

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