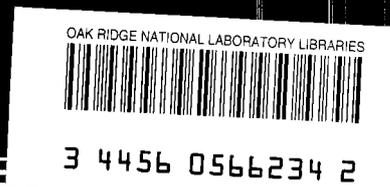




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Vol. III

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ABSTRACTS of DIGITAL COMPUTER CODE PACKAGES

Assembled by the Radiation Shielding Information Center

Betty F. Maskewitz

AUGUST 1972

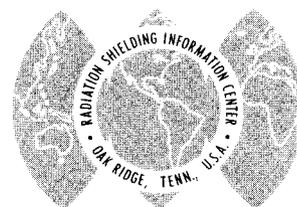
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RADIATION SHIELDING INFORMATION CENTER



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PREFACE

The collection and dissemination of the information contained in this book of abstracts represent the work of many people. We are grateful to the code contributors, without whom there would be no computer code collection; to the USAEC, to DNA, and to NASA, for their encouragement and their financial support of the RSIC programs.

The increasing efforts of developers and report authors to adhere to guidelines to aid in the interchangeability of computer codes is much appreciated. We are pleased to cite the work of Subcommittee 10, American Nuclear Society Standards Committee, as being very helpful in this respect. ANS Std. 2 and ANS Std. 3 on recommended practices (documentation and programming, respectively) to facilitate the interchange of digital computer programs are available from the Society.

ORNL-RSIC-13, Volume I contains abstracts of each code package in the RSIC Computer Code Collection CCC-1 through CCC-58, published 1966-67. Volume II continues with CCC-59 through CCC-115, 1968-1969. Volume III currently contains CCC-116 through CCC-168, packaged 1970-1972. Additional abstracts may be added until the capacity of the loose-leaf binder has been filled.

The main purpose of the abstracts is to give to a potential code user several criteria for deciding whether or not he wishes to request the code package. It is suggested that the reading of the references cited in each abstract would be the logical next step before requesting the complete package.

The letter of request for code packages should always be accompanied by the required number of reels of magnetic tape as indicated in each abstract. A statement as to how they should be written should accompany the tapes. Information on the requester's local computer environment is helpful.

Code contributors are increasingly helpful in the writing of abstracts. A copy of the RSIC format is included in the introduction to Volume I as a guide for this purpose.

RSIC cooperates with the Argonne Code Center and the OECD Nuclear Energy Agency's Computer Programme Library (NEA CPL), located in Ispra, Italy. RSIC code packages are placed in NEA CPL in order to expedite service to the European shielding scientist.

It is a pleasure to acknowledge the contributions of other members of the RSIC staff. The check-out and packaging of computer codes and the writing of abstracts are a continuing process. Hemma E. Comolander, Henrietta R. Hendrickson, and Juanita B. Wright of the Codes Section staff verify the accuracy and completeness of that part of the abstract which describes the hardware, the software, and the contents of the code package. Frances S. Alsmiller, F. H. Clark, R. W. Roussin, and D. K. Trubey assist in writing and in editing the abstracts. We are grateful to Virginia Glidewell for the care with which she prepared the material on the MT/ST typewriter and to Vivian Z. Jacobs, who coordinated the work of preparing the abstracts for publication in ORNL-RSIC-13, Volume III.

Betty F. Maskewitz
RSIC Coordinator

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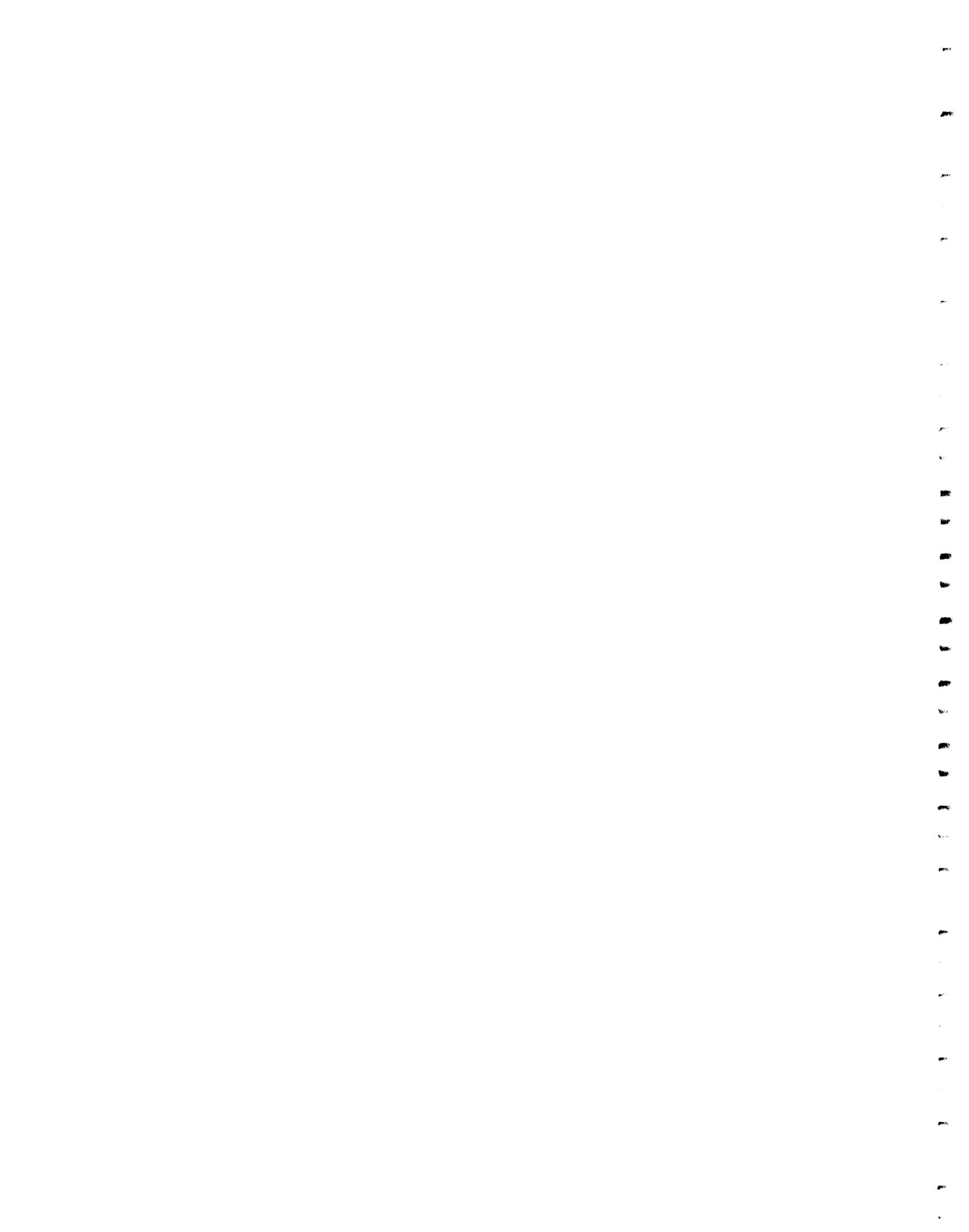
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 CYLINDRICAL COORDINATE SYSTEM, contributed by Nuclear Materials
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 CYLINDERS, contributed by USAF Nuclear Aerospace Research
 Facility, General Dynamics, Fort Worth, Texas.
 FAP, FORTRAN; IBM 704, 709, 7090, and 7094
 (References: NARF 61-39T, FZK-9-170, General Dynamics N-S Memo
 1/348)
- CCC-6: L-63 and Auxiliary Routine
 KERNEL INTEGRATION CODE - CYLINDERS, SPHERES, AND COMPLEX
 GEOMETRY, contributed by USAF Nuclear Aerospace Research
 Facility, General Dynamics, Fort Worth, Texas.
 FAP, FORTRAN; IBM 704, 7090, and 7094
 (References: NARF 61-39T, FZK-9-170, General Dynamics N-S
 Memo 1/348)
- CCC-7: NTC and Auxiliary Routines
 MONTE CARLO HIGH ENERGY NUCLEON TRANSPORT CODE INCORPORATING
 CASCADE AND EVAPORATIVE PROCESSES, contributed by Neutron
 Physics Division, Oak Ridge National Laboratory, Oak Ridge,
 Tennessee.
 FORTRAN, FAP; IBM 7090 and 7094
 (References: ORNL-3610, ORNL-TM-196, Phys. Rev. 131, 1801
 [1961])

- CCC-8: K-74
A MONTE CARLO CALCULATION OF NEUTRON FLUX IN INFINITE MEDIUM FOR POINT ISOTROPIC SOURCES, contributed by USAF Nuclear Aerospace Research Facility, General Dynamics, Fort Worth, Texas. FAP; IBM 704, 7090, and 7094
(References: NARF 60-8T, FZK-9-14 Vol. 1 and II, FZM-1267)
- CCC-9: L-05
MONTE CARLO MULTIBEND-DUCT SHIELDING CODE, contributed by USAF Nuclear Aerospace Research Facility, General Dynamics, Fort Worth, Texas. FAP; IBM 7090
(References: NARF 61-33T, MR-N-286 and NARF 62-13T, MR-N-297, RRA-T44)
- CCC-10: C-18 and Auxiliary Routines
MONTE CARLO MULTILAYER SLAB GEOMETRY CODE, contributed by USAF Nuclear Aerospace and Research Facility, General Dynamics, Fort Worth, Texas; and U. S. Army Tank Automotive Center, Warren, Michigan. FAP, STRAP FORTRAN, FORTRAN II, FORTRAN IV; IBM 7090 and 7094
(References: FZK-134-3 and RRA-N413)
- CCC-11: SANE and Auxiliary Routines
MONTE CARLO SPHERICAL MULTILAYER GEOMETRY NEUTRON TRANSPORT SHIELDING CODE, contributed by United Nuclear Corporation, Development Division - NDA, White Plains, New York. FORTRAN, FAP; IBM 7090 and 7094
(References: UNUCOR-633 and UNUCOR-634)
- CCC-12: SAGE and Auxiliary Routines
MONTE CARLO SPHERICAL AND MULTILAYER GEOMETRY GAMMA TRANSPORT SHIELDING CODE, contributed by United Nuclear Corporation, Development Division - NDA, White Plains, New York. FORTRAN, FAP; IBM 7090 and 7094
(References: UNUCOR-633 and UNUCOR-634)
- CCC-13: ADONIS and Auxiliary Routines
A and B MONTE CARLO THREE-DIMENSIONAL RECTANGULAR GEOMETRY SHIELDING CODE, contributed by United Nuclear Corporation, Development Division - NDA, White Plains, New York. FORTRAN, FAP; CODAP; IBM 7090 and 7094, CDC 1604
(References: UNUCOR-635, USNCEL-tr-R379)
- CCC-14: FMC-G and Auxiliary Routines
MONTE CARLO SIMULATION OF GAMMA-RAY LIFE HISTORIES IN A SOURCE-SHIELD CONFIGURATION, contributed by Nuclear Materials and Propulsion Operation, General Electric, Cincinnati, Ohio. FAP, FORTRAN II; IBM 7090 and 7094
(References: APEX-706; GEMP-113, 115; XDC 61-4-52)

- CCC-15: FMC-N and Auxiliary Routines
 MONTE CARLO SIMULATION OF NEUTRON LIFE HISTORIES IN A SOURCE-SHIELD CONFIGURATION, contributed by Nuclear Materials and Propulsion Operation, General Electric, Cincinnati, Ohio.
 FAP, FORTRAN II; IBM 7090 and 7094
 (References: APEX-706; GEMP-113, 115, 116, 117; XDC 61-4-52; GEMP-384)
- CCC-16: 18-0, 18-1 and Auxiliary Routines
 MONTE CARLO SIMULATION OF NEUTRON AND GAMMA-RAY LIFE HISTORIES IN REACTOR-SHIELD ASSEMBLIES, contributed by Nuclear Materials and Propulsion Operation, General Electric, Cincinnati, Ohio.
 FORTRAN II, FAP; IBM 7090 and 7094
 (References: XDC 61-1-91; GEMP-102, 113, 114, 115, 116, 117, 123, 272; APEX 605, 610)
- CCC-17: 05R and Auxiliary Routines
 A GENERAL PURPOSE MONTE CARLO NEUTRON TRANSPORT CODE SYSTEM, contributed by Neutron Physics Division, Oak Ridge National Laboratory, Oak Ridge, Tennessee.
 FORTRAN, FAP/CODAP; IBM 7090 and CDC 1604
 References: ORNL-3622, ORNL-3715, ORNL-TM-1192, ORNL-TM-1245)
- CCC-18: 05-0
 ANISOTROPIC POINT SOURCE CODE - SINGLE SCATTERED GAMMA RAYS IN AN INFINITE HOMOGENEOUS MEDIUM, contributed by Nuclear Materials and Propulsion Operation, General Electric, Cincinnati, Ohio.
 (Reference: XDC 59-8-218)
- CCC-19: 09-0
 ANISOTROPIC POINT SOURCE CODE - SINGLE SCATTERED NEUTRONS IN AN INFINITE HOMOGENEOUS MEDIUM, contributed by Nuclear Materials and Propulsion Operation, General Electric, Cincinnati, Ohio.
 FAP; IBM 7090 and 7094
 (Reference: APEX-533)
- CCC-20: TRIGR-P
 MONTE CARLO GAMMA-RAY PENETRATION CODE - PLANE GEOMETRY, contributed by TRG, Incorporated, Melville, New York; and USAF Aeronautical Research Laboratories, Wright-Patterson Air Force Base, Ohio.
 SAP; IBM 704
 (Reference: WADC TR-59-771)
- CCC-21: MORTIMER
 KERNEL INTEGRATION CODE - TWO-COMPONENT ANALYSIS FOR SNAP SHIELD GEOMETRIES, contributed by Atomics International, Canoga Park, California.
 FORTRAN, FAP; IBM 7090 and 7094
 (References: NAA-SR-9327, NAA-SR-MEMO-8968, TIM No. 798, NAA-TDR-5772)

- CCC-22: MAC (A)
NEUTRON AND GAMMA-RAY ATTENUATION CODE - SPINNEY (REMOVAL-DIFFUSION) CALCULATION IN PLANE GEOMETRY, contributed by Hanford Atomic Products Operation, General Electric Company, Richland, Washington.
FORTRAN II; IBM 7090
(References: HW-73381, HW-73381 SUP I, ORNL-TR-610, EUR 2152.e)
- MAC-RAD (B)
NEUTRON AND GAMMA-RAY ATTENUATION CODE - SPINNEY (REMOVAL-DIFFUSION) CALCULATION IN PLANE GEOMETRY, contributed by Allegemeine Elektricitats-Gesellschaft, Kernenergieanlagen (AEG-KEA), Frankfurt (Main), Germany.
FORTRAN II; IBM 7090
(References: HW-73381, HW-73381 SUP I, AEG-KEA-116, EUR 2152.e)
- CCC-23: MAVRAC
MODEL ASTRONAUT AND VEHICLE RADIATION ANALYSIS CODE, contributed by Northrop Space Laboratories, Hawthorne, California; and Aerospace Medical Research Laboratory, Wright-Patterson Air Force Base, Ohio.
FORTRAN; IBM 7090
(Reference: NSL 63-159)
- CCC-24: CARSTEP
TRAJECTORY AND ENVIRONMENT CODE - ELECTRON AND PROTON FLUXES IMPINGING ON SPACECRAFT IN ORBIT, contributed by Northrop Space Laboratories, Hawthorne, California; and Aerospace Medical Research Laboratory, Wright-Patterson Air Force Base, Ohio.
FORTRAN II; IBM 7090
(Reference: NSL 63-63R-1)
- CCC-25: TRG-SGD
CALCULATION OF SECONDARY GAMMA-RAY DOSE FROM A NUCLEAR WEAPON DETONATION - MONTE CARLO METHOD, contributed by Biophysics Branch, Air Force Weapons Laboratory, Kirtland Air Force Base, New Mexico; and TRG, Incorporated, Melville, New York.
FORTRAN; CDC 1604
(Reference: WL-TDR-64-46)
- CCC-26: GRACE-II
GAMMA-RAY KERNEL INTEGRATION DOSE RATE AND HEATING CODE - CYLINDERS AND SPHERES, contributed by Atomics International, Canoga Park, California.
FORTRAN; IBM 7090-4, CDC 1604
(Reference: NAA-SR-MEMO 4649)

- CCC-27: ACT II
ACTIVATION GAMMA-RAY SOURCE STRENGTH CODE - SIMPLE GEOMETRY.
FINITE DILUTION, contributed by Astronuclear Laboratory,
Westinghouse Electric Corporation, Pittsburgh, Pennsylvania.
FORTRAN; IBM 7090-4
(Reference: WANL-TNR-063, Addendum 1)
- CCC-28: FPIC
FISSION PRODUCT INVENTORY CODE, contributed by Nuclear
Analysis Department, Lockheed-Georgia Company, Marietta,
Georgia.
FORTRAN; IBM 7090-4
(Reference: ER-6906)
- CCC-29: MARTY-G
MONTE CARLO GAMMA-RAY RADIATION TRANSPORT AND HEAT DEPOSITION
RATES IN LIQUID HYDROGEN - SLABS AND CYLINDERS, contributed
by Research Projects Laboratory, NASA George C. Marshall
Space Flight Center, Huntsville, Alabama.
FORTRAN, FAP; IBM 7090 and 7094
(Reference: NASA TN D-1115)
- CCC-30: MARTY-N
MONTE CARLO NEUTRON RADIATION TRANSPORT AND HEAT DEPOSITION
RATES IN LIQUID HYDROGEN - SLABS AND CYLINDERS, contributed
by NASA, George C. Marshall Space Flight Center, Huntsville,
Alabama.
FORTRAN, FAP; IBM 7090 and 7094.
(Reference: NASA TN D-1115)
- CCC-31: BREMRAD
EXTERNAL AND INTERNAL BREMSSTRAHLUNG CALCULATION CODE, con-
tributed by Chemical Laboratory, Battelle-Northwest
Laboratories, Richland, Washington.
FORTRAN; IBM 7090 and 7094
(Reference: HW-83784)
- CCC-32: CLOUD
GAMMA-RAY DOSE RATE FROM A RADIOACTIVE CLOUD - KERNEL
INTEGRATION CODE, contributed by Atomics International,
Canoga Park, California.
FORTRAN; IBM 709, 7090, and 7094
(Reference: NAA-SR-MEMO-4822)
- CCC-33: SALOMON
MONTE CARLO GAMMA TRANSPORT CODE (LAMINATED SLABS), con-
tributed by Research Institute of National Defense, Stockholm,
Sweden.
FORTRAN; IBM 7090-4
(Reference: FOA 4A 4403-411)

- CCC-34: TOPIC
Sn NEUTRON TRANSPORT CODE - CYLINDRICAL GEOMETRY, contributed by Phillips Petroleum Company, Atomic Energy Division, Idaho Falls, Idaho.
FORTRAN; IBM 7040, 7090, 7094
(Reference: IDO-16968)
- CCC-35: DIPSEA
MONTE CARLO DOSE CALCULATION - ISOTROPIC POINT SOURCE IN AN EXPONENTIAL ATMOSPHERE, contributed by Technical Operations Research, Burlington, Massachusetts; and MIT Lincoln Laboratory, Lexington, Massachusetts.
FORTRAN, FAP; IBM 704, 709, 7090, 7094
(Reference: TO-B 64-12)
- CCC-36: EMPIRE-2
MULTI-GROUP DISCRETE ORDINATE TRANSPORT CODE - SLAB GEOMETRY, contributed by Bettis Atomic Power Laboratory, Westinghouse Electric Corporation, Pittsburgh, Pennsylvania.
FORTRAN, TAC; PHILCO 2000 and IBM 7090
(Reference: WAPD-TM-436)
- CCC-37: LIPRECAN I
MONTE CARLO TWO-DIMENSIONAL NEUTRON PENETRATION AND ENERGY DEPOSITION IN LIQUID HYDROGEN, contributed by Advance Space Technology, Missile and Space Systems Division, Douglas Aircraft Company, Inc., Santa Monica, California.
FORTRAN, FAP; IBM 7090 and 7094
(Reference: SM-43594)
- CCC-38: TAEC
SPACECRAFT TRAJECTORY AND ENVIRONMENT CODE, contributed by Research and Technology Division, Air Force Weapons Laboratory, Kirtland Air Force Base, New Mexico; and Aerospace Division The Boeing Company, Seattle, Washington.
FORTRAN; CDC 1604 and IBM 7044, 7094
(References: WL-TDR-64-71, Vol. I and II; D2-90684-1)
- CCC-39: PROP
PRIMARY PROTON PENETRATION CODE, contributed by Research and Technology Division, Air Force Weapons Laboratory, Kirtland Air Force Base, New Mexico; and Aerospace Division, The Boeing Company, Seattle, Washington.
FORTRAN; CDC 1604 and IBM 7094
(References: WL-TDR-64-71, Vol. I and II; D290684-1)
- CCC-40: NIOBE
NEUTRON DISCRETE ORDINATE CODE - MULTILAYER SPHERICAL GEOMETRY, contributed by United Nuclear Corporation, Development Division-NDA, White Plains, New York.
FAP; IBM 7090
(References: UNUCOR-631, UNUCOR-632)

- CCC-41: RENUPAK
NEUTRON MOMENT CALCULATIONS IN INFINITE HOMOGENEOUS MEDIA,
contributed by United Nuclear Corporation, Development
Division - NDA, White, Plains, New York.
FAP, binary; IBM 7090
(Reference: NDA 2120-3)
- CCC-42: DTF-IV and Auxiliary Routine
MULTIGROUP NEUTRON TRANSPORT DISCRETE ORDINATES CODE, ONE-
DIMENSIONAL, ANISOTROPIC SCATTERING, contributed by Los Alamos
Scientific Laboratory, T Division, Los Alamos, New Mexico.
FORTRAN IV; IBM 7090 and 7030
(References: LA-3373, LA-3267, UNC Phys/Math-3321, Vol. I and II)
- CCC-43: PROTOS
MONTE CARLO PROTON TRANSPORT CODE, contributed by Research
Institute of National Defense (Forsvarets Forskningsanstalt)
Stockholm, Sweden.
FORTRAN II and FAP; IBM 7090
(References: FOA 4 A 4411-411, FOA 4 A 4436-411)
- CCC-44: TORN and Auxiliary Routines
MONTE CARLO NEUTRON TRANSPORT CODE SYSTEM, contributed by
Technical Operations Research, Burlington, Massachusetts;
U. S. Army Nuclear Defense Laboratory, Edgewood Arsenal,
Maryland; Radiation Research Associates, Inc., Fort Worth,
Texas.
FORTRAN II and FAP; IBM 7090
(References: TO-B 63-82, RRA 29, RRA-N59, RRA-N57, RRA-N520,
RRA-N519, RRA-N522)
- CCC-45: TORG and Auxiliary Routine
MONTE CARLO HIGH ENERGY GAMMA-RAY TRANSPORT CODE FOR PLANE
GEOMETRY, contributed by Technical Operations Research, Burling-
ton, Massachusetts; U.S. Army Nuclear Defense Laboratory, Edge-
wood Arsenal, Maryland; Radiation Research Associates, Inc., Fort
Worth, Texas.
FORTRAN II; IBM 7090
(References: TO-B 63-73, RRA-N521, RRA-N522)
- CCC-46: OGRE and Auxiliary Routine
A GENERAL-PURPOSE MONTE CARLO GAMMA-RAY TRANSPORT CODE SYSTEM,
contributed by Neutron Physics Division, Oak Ridge National
Laboratory, Oak Ridge, Tennessee.
CCC-46A: FORTRANII and FAP; IBM 7090
CCC-46B: FORTRAN 63 and CODAP; CDC 1604
CCC-46C: FORTRAN IV; IBM 360/50 and 360/75
CCC-46D: FORTRAN IV; IBM 7090
(References: ORNL-3805, ORNL-TM-1212)

- CCC-47: LEP and Auxiliary Routines
LOW-ENERGY INTRANUCLEAR CASCADE CODES, contributed by Neutron Physics Division, Oak Ridge National Laboratory, Oak Ridge, Tennessee.
FORTRAN and FAP; IBM 7090
(References: ORNL-3844, ORNL-TM-1033, Phys. Rev. 131(4), 1801, ORNL-3433, ORNL-TM-1225, ORNL-TM-196)
- CCC-48: QAD and Auxiliary Routines
KERNEL INTEGRATION CODE SYSTEM, contributed by Los Alamos Scientific Laboratory, N Division, Los Alamos, New Mexico.
QAD-HD and QAD-P5A were contributed by NASA Lewis Research Center, Shielding Analysis Section, Cleveland, Ohio.
FORTRAN II (QAD-P5), FORTRAN IV (all versions), IBM 7090 and 7094; FORTRAN IV (QAD-P5), IBM 360
(References: LA-3573, NASA TM-X-1397)
- CCC-49: TAPER II
A FISSION PRODUCT CONCENTRATION CODE, contributed by Reactor Computing Systems Unit, Atomic International, Canoga Park, California.
FORTRAN II; IBM 7090 and 7094
(Reference: AI-64-MEMO-67)
- CCC-50: LRSPC
RANGE AND STOPPING POWER CALCULATOR, contributed by Lockheed-Georgia Company, Nuclear Analysis Department, Marietta, Georgia.
FORTRAN II and FAP; IBM 7090 and 7094
(Reference: ER-7777 Vol. I)
- CCC-51: LPPC and Auxiliary Routines
PROTON PENETRATION CODE, contributed by Lockheed-Georgia Company, Nuclear Analysis Department, Marietta, Georgia.
FAP (LPPC), FORTRAN II and FAP (LMFC, LSSC, LIGHT, MSGAM), FORTRAN II (NCON), FORTRAN IV and MAP (FLARE); IBM 7090 and 7094
(References: ER-6643, ER-7777 Vol. II and III)
- CCC-52: LEBC and Auxiliary Routine
ELECTRON BREMSSTRAHLUNG CODE, contributed by Lockheed-Georgia Company, Nuclear Analysis Department, Marietta, Georgia.
FORTRAN II and FAP; IBM 7090 and 7094
(References: ER-6643, ER-7777 Vol. II)
- CCC-53: LSVDC and Auxiliary Routines
SPACE VEHICLE DOSE CALCULATION, contributed by Lockheed-Georgia Company, Nuclear Analysis Department, Marietta, Georgia.
FORTRAN II and FAP (DOSE, GEOM, GEOTST, LMFC, LSSC), FAP (DIP), FORTRAN IV and MAP (FLARE); IBM 7090 and 7094
(References: ER-7777 Vol. II and III, ER-6643)

- CCC-54: NRN and Auxiliary Routines
MULTIGROUP REMOVAL-DIFFUSION CODE SYSTEM FOR PLANES, CYLINDERS,
AND SPHERES, contributed by Aktiebolaget (AB) Atomenergi.
Stockholm, Sweden.
FORTRAN IV and MAP; IBM 7090 and 7044
(References: AE-FFA-673, AE-RFN-213; Nucl. Sci. Eng., 24-2,
165-174 and 22-4, 443-450)
- CCC-55: ISOGEN
RADIOISOTOPE GENERATOR CODE, contributed by Battelle Northwest
Laboratories, Chemical Laboratory, Richland, Washington.
FORTRAN; IBM 7090
(References: HW-83785, HW-SA-3769)
- CCC-56: MYRA
CALCULATION OF SHIPPING COSTS AND CASK DESIGNS FOR IRRADIATED
FUEL ELEMENTS, contributed by Oak Ridge National Laboratory,
Chemical Technology Division, Oak Ridge, Tennessee.
FORTRAN; CDC 1604 and IBM 7090
(References: ORNL-3648, ORNL-3931)
- CCC-57: STERNO and Auxiliary Routine
TWO-DIMENSIONAL GAMMA-HEATING KERNEL INTEGRATION CODE, con-
tributed by Pratt and Whitney Aircraft, CANEL, Middletown,
Connecticut.
FORTRAN 63; CDC 1604
(Reference: TIM No. 829)
- CCC-58: SPARC
MONTE CARLO SLAB PENETRATION AND REFLECTION CODE, contributed
by USAF Nuclear Aerospace Research Facility, General Dynamics,
Fort Worth, Texas; U. S. Army Tank Automotive Center, Warren,
Michigan; U. S. Army Ballistic Research Laboratories, Aberdeen
Proving Ground, Maryland.
FORTRAN II (E59) and FAP (P09); IBM 7090 and 7094
(References: FZK-200-5A, FZK-200-5A SUPP, FZK-234, FZK-234
SUPP)
- CCC-59: COMBINE and Auxiliary Routine
KERNEL INTEGRATION OF TRANSMISSION AND REFLECTION PROBABIL-
ITIES FOR SHIELDED COMPARTMENTS, contributed by USAF Nuclear
Aerospace Research Facility, General Dynamics, Fort Worth,
Texas; U. S. Army Tank Automotive Center, Warren, Michigan;
U. S. Army Ballistics Research Laboratories, Aberdeen Proving
Ground, Maryland.
FORTRAN II and FORTRAN IV, IBM 7090 and 7094
(References: FZK-200-5B and SUPP; FZK-200 5C and SUPP;
FZK-235 and SUPP)

- CCC-60: SDC and Auxiliary Routines
KERNEL INTEGRATION SHIELD DESIGN CODE FOR RADIOACTIVE FUEL-
HANDLING FACILITIES, contributed by Chemical Technology
Division, Oak Ridge National Laboratory, Oak Ridge, Tennessee.
FORTRAN II, IV; IBM 7090. FORTRAN IV; IBM 360, 7090.
FORTRAN 63; CDC 1604.
(References: ORNL-3041, 3931)
- CCC-61: CEP
MONTE CARLO CALCULATION OF NEUTRON FIRST-FLIGHT ESCAPE
PROBABILITIES FOR FINITE CYLINDERS, contributed by Phillips
Petroleum Company, Atomic Energy Division, Idaho Falls, Idaho.
MAP; IBM 7040
(Reference: IDO-17075)
- CCC-62: K009
SOLID ANGLE INTEGRATION CHARGED PARTICLE PENETRATION CODE, con-
tributed by NASA Manned Spacecraft Center, Houston, Texas.
FORTRAN II; IBM 7090
(Reference: NASA Project 3208P)
- CCC-63: OPEX
SHIELD WEIGHT OPTIMIZATION CODE, contributed by Atomic
International, Canoga Park, California.
FORTRAN II; IBM 7090
(Reference: NAA-SR-TDR-11516)
- CCC-64: LPSC and Auxiliary Routine
PROTON PENETRATION CODE - MULTILAYER SLAB GEOMETRY, contributed
by NASA Lewis Research Center, Cleveland, Ohio.
FORTRAN IV; IBM 7090 and 7094
(References: NASA TM X-52166 and Phys. Rev. 131, 1801)
- CCC-65: TDSN
TWO-DIMENSIONAL MULTIGROUP DISCRETE ORDINATES NEUTRON TRANSPORT
CODE, contributed by NASA Lewis Research Center, Cleveland, Ohio.
FORTRAN IV and MAP; IBM 7090 and 7094
(Reference: NASA TN D-3573)
- CCC-66: BIGGI-3P
NUMERICAL GAMMA-RAY TRANSPORT CODE FOR PLANE MULTILAYER
GEOMETRY, contributed by EURATOM, Ispra (Varese), Italy.
FORTRAN; IBM 7090
(References: EUR-2488.e; ANL 7050, p. 113; EUR 1643.e., p. 92;
EUR 3555.e; Provisory Description, BIGGI 3P).

- CCC-67: STORM and Auxiliary Routines
SOLAR FLARE RADIATION HAZARD TO EARTH ORBITING VEHICLES,
contributed by NASA Langley Research Center, Hampton, Virginia;
Aerospace Systems and Services, Republic Aviation Corporation,
Farmingdale, Long Island, New York.
FORTRAN IV; IBM 7090 and 7094
(References: RAC-1395-1; RAC-1395-2; FHR 1395-3)
- CCC-68: TYCHE III
MONTE CARLO CODE - NEUTRON SLOWING DOWN MOMENTS, contributed
by Atomics International, Canoga Park, California.
FORTRAN, FAP; IBM 7090 and 7094
(References: NAA-SR-7357; NAA-SR-MEMO-9802; NAA-SR-MEMO-9721;
NAA-SR-MEMO-9069)
- CCC-69: CURIE-DOSE-THUNDERHEAD
CALCULATION OF EXTERNAL AND INTERNAL DOSE FROM A RADIOACTIVE
CLOUD, contributed by Atomics International, Canoga Park,
California.
FORTRAN II, FAP; IBM 7090 and 7094
(Reference: NAA-SR-8884)
- CCC-70: CHARGE
SPACE RADIATION SHIELDING CODE - PROTON AND ELECTRON PENETRA-
TION OF MULTILAYERED SLABS AND SPHERES, contributed by MSSD,
Douglas Aircraft Company, Inc., Santa Monica, California.
FORTRAN IV; IBM 7090 and 7094
(Reference: SM-46335)
- CCC-71: MIST
MULTIGROUP DISCRETE ORDINATES TRANSPORT CODE FOR SLAB GEOMETRY,
contributed by Phillips Petroleum Company, Atomic Energy Division,
Idaho Falls, Idaho.
FORTRAN II, IV; IBM 7090
(Reference: IDO-16856)
- CCC-72: COMPRASH
SPINNEY REMOVAL-DIFFUSION SHIELDING CODE, contributed by UKAEA-
AEEW Shielding Group, Harwell, England.
FORTRAN II and FAP; IBM 7090
(References: AEEW-R361 and AEEW-M648)
- CCC-73: ASTROS and Auxiliary Routine
CALCULATION OF PRIMARY AND SECONDARY PROTON DOSE RATES IN
SPHERES AND SLABS OF TISSUE, contributed by Lawrence Radiation
Laboratory, Berkeley, California.
FORTRAN IV and MAP; IBM 7090 and 7094
(References: UCRL-10980, UCRL-16154)

- CCC-74: CAPS-2 and Auxiliary Routine
ANALYSIS OF STRUCTURES FOR FALLOUT RADIATION SHIELDING,
contributed by Office of Civil Defense, Office of the
Secretary of the Army, Department of the Army, Washington, D. C.
FORTRAN 63; CDC 1604
(References: NBS Monograph 42, PM 100-1 and Suppl., TR 20
(Vol. 1), OCD-JD1)
- CCC-75: G-33
KERNEL INTEGRATION CODE - MULTIGROUP GAMMA RAY SCATTERING,
contributed by Los Alamos Scientific Laboratory, Los Alamos,
New Mexico; NASA Lewis Research Center, Cleveland, Ohio;
Aerojet-General Nucleonics, San Ramon, California.
G-33A - FORTRAN II and FAP; IBM 7090
G-33B - FORTRAN IV; IBM 7090, 7094, and 7094 II
(References: EAD-119, AN-COMP-196, Informal Notes [G. P. Lahti])
- CCC-76: BPPC and Auxiliary Routines
PROTON PENETRATION CODE FOR SPACE VEHICLES, contributed by
NASA Langley Research Center, Hampton, Virginia; Aerospace
Division, The Boeing Company, Seattle, Washington; Air Force
Weapons Laboratory, Kirtland Air Force Base, New Mexico.
FORTRAN II, IV; IBM 7090 and 7094
(References: WL-TDR-64-71, DC-90684-1)
- CCC-77: BEBC
ELECTRON BREMSSTRAHLUNG PENETRATION CODE FOR SPACE VEHICLES,
contributed by NASA Langley Research Center, Hampton, Virginia;
Aerospace Division, The Boeing Company, Seattle, Washington;
Air Force Weapons Laboratory, Kirtland Air Force Base, New Mexico.
FORTRAN IV; IBM 7090 and 7094
(References: WL-TDR-64-71, DC-90684-1)
- CCC-78: BED
ELECTRON PENETRATION CODE FOR SPACE VEHICLES, contributed by
NASA Langley Research Center, Hampton, Virginia; Aerospace
Division, The Boeing Company, Seattle, Washington; Air Force
Weapons Laboratory, Kirtland Air Force Base, New Mexico.
FORTRAN IV; IBM 7090 and 7094
(References: WL-TDR-64-71, DC-90684-1)
- CCC-79: ISOSHL
KERNEL INTEGRATION CODE - GENERAL PURPOSE ISOTOPE SHIELDING
ANALYSIS, contributed by Battelle Memorial Institute, Pacific
Northwest Laboratories, Richland, Washington.
FORTRAN IV; IBM 360 and 7090
(References: BNWL-236; Unpublished Data, Douglas-United
Nuclear, Hanford Atomic Products Operation, Richland, Wash.;
HW-83784; BNWL-236 SUP1)

- CCC-80: GASS
MONTE CARLO CALCULATION OF SELF-SHIELDING BY ENCAPSULATED
GAMMA-RAY SOURCES, contributed by University of Illinois,
Civil Engineering and Nuclear Engineering Program for the
Office of Civil Defense.
FORTRAN IV and MAP; IBM 7090 and 7094
(Reference: UI-NRSS-3)
- CCC-81A: UNC-SAM and Auxiliary Routines
MONTE CARLO THREE-DIMENSIONAL COMPLEX GEOMETRY SHIELDING
CODE SYSTEM, contributed by United Nuclear Corporation,
White Plains, New York; U.S. Army Ballistics Research
Laboratory, Aberdeen Proving Ground, Maryland.
FORTRAN and CODAP-2; CDC 1604
(Reference: UNC-5093)
- CCC-81B: UNC-SAM-2 and Auxiliary Routines
MONTE CARLO TIME DEPENDENT THREE-DIMENSIONAL COMPLEX
GEOMETRY SHIELDING CODE SYSTEM, contributed by United
Nuclear Corporation, White Plains, New York; U. S. Army
Ballistics Research Laboratory, Aberdeen Proving Ground,
Maryland.
FORTRAN and CODAP-2; CDC 1604
(References: UNC-5157 and 5093)
- CCC-82: ANISN and Auxiliary Routine
MULTIGROUP ONE DIMENSIONAL DISCRETE ORDINATES TRANSPORT CODE
WITH ANISOTROPIC SCATTERING, contributed by Computing Tech-
nology Center and Oak Ridge National Laboratory, Union Carbide
Corporation Nuclear Division, Oak Ridge, Tennessee.
FORTRAN IV, IBM 7090 and 7094; FORTRAN IV (H), IBM 360
(References: K-1693, NAA-SR-10951)
- CCC-83: RAID
MONTE CARLO MULTIBEND DUCT SHIELDING CODE, contributed by
USAF Nuclear Aerospace Research Facility (NARF), General
Dynamics, Fort Worth, Texas.
FORTRAN IV; IBM 7090 and 7094
(Reference: NARF-DC-Memo 1.115)
- CCC-84: SHADRAC
KERNEL INTEGRATION CODE - SHIELD HEATING AND DOSE RATE CAL-
CULATION IN COMPLEX GEOMETRY, contributed by USAF Nuclear
Aerospace Research Facility, General Dynamics, Fort Worth,
Texas.
FORTRAN IV; IBM 7090 and 7094
(Reference: NARF-DC-Memo 1.097)

- CCC-85: MOMGEN and MOMDIS and Auxiliary Routines
MOMENTS METHOD RECONSTRUCTION OF SCATTERED GAMMA-RAY MOMENTS
METHOD RECONSTRUCTION OF SCATTERED GAMMA-RAY DISTRIBUTIONS,
contributed by U.S. Naval Radiological Defense Laboratory, San
Francisco, California; National Bureau of Standards provided the
total scattering moments which are used as input to MOMGEN.
FORTRAN II, IV; IBM 704 and 7090
(References: NRDL-TR-67-9, NYO 3075)
- CCC-86: HANGER and Auxiliary Routines
MONTE CARLO CYLINDRICALLY SYMMETRIC SHIELD, NEUTRON TRANSPORT
AND HEATING CODE, contributed by United Nuclear Corporation,
Development Division-NDA; and NASA Lewis Research Center, Cleve-
land, Ohio.
FORTRAN IV; IBM 7090
(Reference: UNC-5043)
- CCC-87: LG-H
RAY ANALYSIS CYLINDRICAL DUCT KERNEL CODE FOR NEUTRON AND GAMMA
RAYS, contributed by Shielding Codes Group, Tokai Establishment,
JAERI, Ibaraki, Japan; and Hitachi Central Research Laboratory,
Hitachi, Ltd.
FORTRAN IV; IBM 7090, 7040, and 7044
(Reference: Informal Memo 1966))
- CCC-88: RADOS
GAMMA-RAY DOSE ESTIMATION FROM CLOUD OF RADIOACTIVE GASES BY
KERNEL INTEGRATION, contributed by Savannah River Laboratory,
E. I. du Pont de Nemours and Company, Aiken, South Carolina.
FORTRAN IV; IBM 360/65 and 360/75
(Reference: DP-1098)
- CCC-89: DOT
MULTIGROUP TWO-DIMENSIONAL DISCRETE ORDINATES TRANSPORT CODE WITH
ANISOTROPIC SCATTERING, contributed by Computing Technology Center
and the Oak Ridge National Laboratory Neutron Physics Division,
Union Carbide Corporation, Nuclear Division, Oak Ridge, Tennessee;
Atomics International, Canoga Park, California; U. S. Air Force
Weapons Laboratory, Kirtland Air Force Base, New Mexico.
FORTRAN IV; IBM 7090, IBM 360/75, and CDC 6600
(References: K-1694; Nucl. Sci. Eng. (1969), to be published)
- CCC-90: AMC
MONTE CARLO ALBEDO CODE FOR NEUTRON AND CAPTURE GAMMA-RAY DISTRI-
BUTIONS IN RECTANGULAR CONCRETE DUCTS, contributed by Neutron Physics
Division, Oak Ridge National Laboratory, Oak Ridge, Tennessee.
FORTRAN IV and MAP; IBM 360/75
(Reference: ORNL-3964)

- CCC-91: NEFIRS
MULTIGROUP SPINNEY METHOD REMOVAL-DIFFUSION CODE FOR NEUTRONS,
contributed by Gulf General Atomic, San Diego, California.
FORTRAN IV and MAP; IBM 360/75
(Reference: GA-8069)
- CCC-92: SAP - N and G
NEUTRON AND GAMMA-RAY ALBEDO MODEL SCATTER SHIELD ANALYSIS CODE,
contributed by Westinghouse Astronuclear Laboratory, Pittsburgh,
Pennsylvania.
FORTRAN IV; IBM 7090 and 7094
(References: WANL-TME-1273, WANL-TME-1273 Revision A)
- CCC-93: MCFLARE
MONTE CARLO CODE TO SIMULATE SOLAR FLARE EVENTS AND ESTIMATE
PROBABLE DOSES ENCOUNTERED ON INTERPLANETARY MISSIONS, contribu-
ted by NASA Lewis Research Center, Shielding Analysis Section,
Nuclear Systems Division, Cleveland, Ohio.
FORTRAN IV; IBM 7090 and 7094
(Reference: NASA TN D-4311)
- CCC-94: KAP-V
KERNEL INTEGRATION CODE IN COMPLEX GEOMETRY, contributed by
Westinghouse Astronuclear Laboratory, Pittsburgh, Pennsylvania;
NASA George C. Marshall Space Flight Center, Huntsville, Alabama.
FORTRAN IV; IBM 7090 and 7094
(References: WANL-PR-(LL)-010, Vol. 4; WANL-PR-(LL)-010, Vol. 1;
WANL-PR-(LL)-014, Vol. 7)
- CCC-95: TAPAT
MULTIGROUP ONE-DIMENSIONAL DISCRETE ORDINATES CODE, contributed
by Westinghouse Astronuclear Laboratory, Pittsburgh, Pennsylvania;
NASA George C. Marshall Space Flight Center, Huntsville, Alabama.
FORTRAN IV; IBM 7090 and 7094
(References: WANL-PR-(LL)-010, Vol. 2; WANL-PR-(LL)-010, Vol. 3;
WANL-PR-(LL)-014, Vol. 1; WANL-PR-(LL)-014, Vol. 2; WANL-PR-(LL)-
014, Vol. 3)
- CCC-96: TIC-TOC-TOE
ON AXIS LIQUID HYDROGEN PROPELLANT TANK HEATING CODE, contributed
by Westinghouse Astronuclear Laboratory, Pittsburgh, Pennsylvania.
FORTRAN IV; IBM 7090 and 7094
(References: WANL-PR-(LL)-010, Vol. 5; WANL-PR-(LL)-014, Vol. 1;
WANL-PR-(LL)-014, Vol. 9)

- CCC-97: ODD-K
MULTIGROUP TWO-DIMENSIONAL DISCRETE ORDINATES CODE, contributed by Westinghouse Astronuclear Laboratory, Pittsburgh, Pennsylvania; NASA George C. Marshall Space Flight Center, Huntsville, Alabama. FLOCO (ODD-K), FORTRAN IV (NAGS), FORTRAN IV and MAP (DAFT); IBM 7090 and 7094
(References: WANL-PR-(LL)-010, Vol. 2; WANL-PRR-(LL)-010, Vol. 7; WANL-PR-(LL)-010, Vol. 8; WANL-PR-(LL)-010, Vol. 6; WANL-PR-(LL)-014, Vol. 1; WANL-PR-(LL)-014, Vol. 2; WANL-PR-(LL)-014, Vol. 4; WANL-PR-(LL)-014, Vol. 5; WANL-PR-(LL)-014, Vol. 6)
- CCC-98: FASTER
MONTE CARLO TRANSPORT CODE IN COMPLEX GEOMETRY, contributed by Westinghouse Astronuclear Laboratory, Pittsburgh, Pennsylvania; NASA George C. Marshall Space Flight Center, Huntsville, Alabama; ART Research Corporation, Los Angeles, California; Aerojet General Corporation, Sacramento, California; Boeing Company, Huntsville, Alabama.
FORTRAN IV; CCC-98A: IBM 7090 and 7094, CCC-98B and CCC-98C: 360/75, CCC-98D: UNIVAC 1108
(References: WANL-PR-(LL)-010, Vol. 9; WANL-PR-(LL)-014, Vol. 1; WANL-PR-(LL)-014, Vol. 8)
- CCC-99: PLUME
GAMMA-RAY DOSE RATE FROM A RADIOACTIVE CLOUD - KERNEL INTEGRATION CODE, contributed by Operations Division, Oak Ridge National Laboratory, Oak Ridge, Tennessee.
FORTRAN IV; IBM 360/75
(Reference: ORNL-4086)
- CCC-100: K019
SHIELD THICKNESS CALCULATION PROGRAM FOR SPACE VEHICLES, contributed by NASA Manned Spacecraft Center, Houston, Texas.
FORTRAN IV; IBM 360/75
(Reference: NASA-MS-3066)
- CCC-101: NAP
MULTIGROUP TIME-DEPENDENT NEUTRON ACTIVATION PREDICTION CODE, contributed by IIT Research Institute, Chicago, Illinois; NASA George C. Marshall Space Flight Center, Huntsville, Alabama.
FORTRAN IV; IBM 7090 and 7094
(References: IITRI-A 6088-21, IITRI-A6088-22)
- CCC-102: SURF
CONICAL AND PLANE SURFACE SINGLE SCATTERING CODE, contributed by Nuclear Materials and Propulsion Operation, General Electric Company, Cincinnati, Ohio.
FORTRAN IV; GE-625 and IBM 360/75
(Reference: GEMP 582)

- CCC-103: OPEX-II
RADIATION SHIELD OPTIMIZATION CODE, contributed by NASA Lewis
Research Center, Cleveland, Ohio.
FORTRAN IV; IBM 7090/7094
(Reference: NASA TM X-1769)
- CCC-104: EDNA
ELECTRON DOSE AND NUMBER ANALYSIS CODE BY KERNEL INTEGRATION,
contributed by Space Sciences Laboratory, NASA George C. Marshall
Space Flight Center, Huntsville, Alabama.
FORTRAN IV; IBM 7090 and 7094
(References: IN-SSL-N-68-13, NASA SP-169)
- CCC-105: RDMM
NEUTRON SPECTRA DETERMINATION BY ACTIVATION DETECTOR ANALYSIS,
contributed by EURATOM Joint Nuclear Research Center, Ispra
Establishment, Italy.
FORTRAN IV; CCC-105A: IBM 7090 and 7094, CCC-105B: IBM 360/75
(References: EUR 2985.e, Nuc. Sci. Eng. 23, 344-353)
- CCC-106: PF-COMP
BUILDING FALLOUT RADIATION PROTECTION FACTOR ANALYSIS, con-
tributed by Research Triangle Institute, Raleigh, North Carolina;
U. S. Army Office of Civil Defense, Washington, D. C.
FORTRAN 64; CDC 3600
(References: NBS Monograph 42; Research Triangle Institute
RM-205-1, Part II; OCD Working Paper (March 1965); OCD Working
Paper (Jan. 1966); OCD Publication PM-100-1; OCD Publications
prepared by RTI (Oct. 1965); PM-100-1 Supplement)
- CCC-107: ETRAN
MONTE CARLO CODE SYSTEM FOR ELECTRON AND PHOTON TRANSPORT THROUGH
SLABS, contributed by Center for Radiation Research, National
Bureau of Standards, Washington, D. C.
FORTRAN IV & V; CCC-107A/ETRA 15: UNIVAC 1108
FORTRAN IV; CCC-107B/ETRA 16 and CCC-107C/ETRA 16B: IBM 360/75
(References: NBS-9836, NBS-9837)
- CCC-108: SPECTRA
DETERMINATION OF NEUTRON SPECTRA FROM ACTIVATION MEASUREMENTS,
contributed by Sandia Laboratories, Albuquerque, New Mexico.
FORTRAN; CCC-108A: CDC 1604, CCC-108B: CDC 3600
(Reference: SC-RR-67-746)
- CCC-109: SOSUM
MULTIGROUP BETA AND GAMMA-RAY ENERGY SOURCES FROM RADIOISOTOPE
ACTIVITIES, contributed by Atomics International, Canoga Park,
California.
FORTRAN IV; IBM 360/75
(Reference: AI-AEC-MEMO-12693)

- CCC-110: AIRTRANS
MONTE CARLO TIME AND ENERGY-DEPENDENT THREE-DIMENSIONAL RADIATION
TRANSPORT CODE, contributed by United Nuclear Corporation, Elms-
ford, New York; Lockheed Missile and Space Company, Sunnyvale,
California
FORTRAN; CCC-110A: CDC 1604, CCC-110B: UNIVAC 1108
(References: LMSC-5234, UNC-5179)
- CCC-111: FLORA
CALCULATION OF THE CONTRIBUTION OF FLUORESCENCE RADIATION, con-
tributed by Douglas Missile and Space Systems Division, Santa
Monica, California.
FORTRAN IV; UNIVAC 1108, IBM 7090
(Reference: DAC-60654)
- CCC-112: SAND
NEUTRON FLUX SPECTRA DETERMINATION BY MULTIPLE FOIL ACTIVATION -
ITERATIVE METHOD, contributed by Air Force Weapons Laboratory
(RTD), Kirtland Air Force Base, New Mexico; Battelle Memorial
Institute Pacific Northwest Laboratory, Richland, Washington;
Atomics International, Canoga Park, California; TRW Systems Group,
TRW, Inc., Redondo Beach, California.
FORTRAN IV; CCC-112A: UNIVAC 1108, CCC-112B: IBM 360/75,
CCC-112C: CDC 6600
(References: AFWL-TR-67-41, Vol. I; AFWL-TR-67-41, Vol. II;
BNWL-855)
- CCC-113: ATHENA
MONTE CARLO RADIATION TRANSPORT AND GAMMA-RAY HEATING CODE SYSTEM
IN COMPLEX THREE-DIMENSIONAL GEOMETRIES, contributed by United
Nuclear Corporation, Elmsford, New York, NASA Lewis Research
Center, Cleveland, Ohio.
FORTRAN IV; IBM 7090 and 7094
(Reference: UNC-5148 (NASA CR-54905))
- CCC-114: SAM-C
MONTE CARLO TIME-DEPENDENT THREE-DIMENSIONAL COMPLEX GEOMETRY
(COMBINATORIAL) SHIELDING CODE SYSTEM, contributed by Mathemati-
cal Applications Group, Inc., White Plains, New York; U. S. Army
Ballistic Research Laboratory, Aberdeen Proving Ground, Maryland,
and U. S. Army Nuclear Defense Laboratory, Edgewood Arsenal,
Maryland.
FORTRAN IV; CDC-6600
(References: UNC-5157, MAGI-6701)
- CCC-115: GADJET
MONTE CARLO GAMMA-RAY ADJOINT ENERGY TRANSPORT CODE IN COMPLEX
THREE-DIMENSIONAL GEOMETRY, contributed by Radioptics, Inc.,
Plainview, New York; U. S. Naval Radiological Defense Laboratory,
San Francisco, California; Office of Civil Defense, Washington,
D. C.
FORTRAN IV; CDC 6600
(References: UNC-5093, NRDL-TRC-68-27, NRDL-TRC-68-25)

- CCC-116: TRECO
AN ORBITAL INTEGRATION ESTIMATION OF TRAPPED RADIATION, contributed by National Space Science Data Center, NASA Goddard Space Flight Center, Greenbelt, Maryland.
FORTRAN IV; CCC-116A: IBM 360; CCC-116B: CDC 6600.
(References: NASA SP-3024, Vol. I-VI, NSSDC68-02)
- CCC-117: BETA
MONTE CARLO BREMSSTRAHLUNG AND ELECTRON TRANSPORT ANALYSIS CODE IN COMPLEX GEOMETRY, contributed by A.R.T. Research Corporation, Los Angeles, California, and Air Force Weapons Laboratory, Kirtland Air Force Base, New Mexico
FORTRAN IV; IBM 7090
(Reference: AFWL-TR-68-111)
- CCC-118: SIGMA
SPACE RADIATION DOSE ANALYSIS WITHIN COMPLEX CONFIGURATIONS, contributed by McDonnell Douglas Astronautics Company, Western Division, Huntington Beach, California.
FORTRAN IV; IBM 7090 and 7094
(Reference: DAC-60878)
- CCC-119: ELBA
ELECTRON AND BREMSSTRAHLUNG DOSE RATE CODE, contributed by Space Sciences Laboratory, NASA George C. Marshall Space Flight Center, Huntsville, Alabama.
FORTRAN IV; IBM 7090 and 7094
(References: NASA-SP-169 and informal notes)
- CCC-120: SPACETRAN
DOSE CALCULATIONS AT DETECTORS AT VARIOUS DISTANCES FROM THE SURFACE OF A CYLINDER, contributed by the Neutron Physics Division, ORNL, Oak Ridge, Tennessee.
FORTRAN IV; IBM 360
(Reference: ORNL-TM-2592)
- CCC-121: SABINE
SPINNEY (REMOVAL-DIFFUSION) SHIELDING CODE IN COMPLEX GEOMETRY, contributed by EURATOM, Ispra (Varese), Italy, through the ENEA Computer Programme Library.
FORTRAN IV; IBM 360
(References: EUR 3635.e and Addendum)
- CCC-122: RAD 2
FISSION PRODUCT RADIOACTIVITIES CALCULATION, contributed by Gulf General Atomic, San Diego, California.
FORTRAN IV; IBM 7090
(Reference: GAMD-6519)

- CCC-123: XSDRN
MULTIGROUP ONE-DIMENSIONAL DISCRETE ORDINATES SPECTRAL AVERAGING
NEUTRON TRANSPORT CODE, contributed by Mathematics, Neutron Physics,
and Reactor Divisions, ORNL, Oak Ridge, Tennessee.
FORTRAN IV; IBM 360
(Reference: ORNL-TM-2500)
- CCC-124: KDLIBE
KERNEL-DIFFUSION SHIELD ANALYSIS SYSTEM, contributed by Nuclear
Systems Programs, Space Systems, Missile and Space Division,
General Electric Company, Cincinnati, Ohio.
FORTRAN IV; CCC-124A: GE 635; CCC-124B: IBM 360
(References: GESP-226, XDC-59-6-220, XDC-60-3-68, GEMP-582,
GEMP-599, HW-71545)
- CCC-125: RSAC
RADIOLOGICAL SAFETY ANALYSIS CODE, contributed by Phillips
Petroleum Company, Atomic Energy Division, Idaho Falls, Idaho.
MAP; IBM 7044
(Reference: IDO-17151)
- CCC-126: ASOP
MULTIGROUP ONE-DIMENSIONAL DISCRETE ORDINATES TRANSPORT CODE FOR
SHIELD OPTIMIZATION, contributed by Neutron Physics Division,
ORNL, Oak Ridge, Tennessee.
FORTRAN IV; IBM 360
(Reference: CTC-INF-941)
- CCC-127: MORSE
A GENERAL PURPOSE MONTE CARLO MULTIGROUP NEUTRON AND GAMMA-RAY
TRANSPORT CODE, contributed by Neutron Physics Division, ORNL,
Oak Ridge, Tennessee.
CCC-127A: FORTRAN 63 and CODAP; CDC 1604. CCC-127B: FORTRAN IV;
IBM 360. CCC-127C: FORTRAN IV and COMPASS; CDC 6600.
(References: ORNL-4585, ORNL-TM-3203, ORNL-TM-2892, ORNL-TM-2879)
- CCC-128: 06R
A GENERAL-PURPOSE MONTE CARLO NEUTRON TRANSPORT CODE SYSTEM, con-
tributed by Neutron Physics Division, ORNL, Oak Ridge, Tennessee.
CCC-128A: FORTRAN 63 and CODAP; CDC 1604. CCC-128B: FORTRAN IV;
IBM 360
(References: ORNL-TM-3050, ORNL-3622, ORNL-3856, ORNL-TM-1552,
ORNL-TM-3051, ORNL-TM-2892, Informal Notes, ORNL-TM-3458)
- CCC-129: TWOTRAN
MULTIGROUP TWO-DIMENSIONAL DISCRETE ORDINATES TRANSPORT GENERAL
GEOMETRY CODE, contributed by Gulf General Atomic, San Diego,
California; Los Alamos Scientific Laboratory, Los Alamos, New
Mexico; Argonne National Laboratory, Argonne, Illinois.
FORTRAN IV; CCC-129A: CDC 6600; CCC-129B: IBM 360; CCC-129C:
CDC 6600
(References: LA-4432, LA-4058, GA-8747, ANL-7411)

- CCC-130: DTF 69
ONE-DIMENSIONAL MULTIGROUP PHOTON TRANSPORT DISCRETE ORDINATES
CODE, contributed by Sandia Laboratories, Albuquerque, New Mexico.
FORTRAN IV; CCC-130A: CDC 6600; CCC-130B: CDC 3600
(References: SC-RR-69-739, SC-RR-67-419, SC-RR-68-712)
- CCC-131: ANTE 2
ADJOINT MONTE CARLO TIME-DEPENDENT NEUTRON TRANSPORT CODE IN
COMBINATORIAL GEOMETRY, contributed by Mathematical Applications
Group, Inc. (MAGI), White Plains, New York, and Bell Telephone
Laboratories, Whippany, New Jersey.
FORTRAN IV; CCC-131A: CDC 6600; CCC-131B: IBM 360
(References: DASA-2396, MR-7003, MR-7004/2)
- CCC-132: ATOW
MULTIGROUP TWO-DIMENSIONAL REMOVAL-DIFFUSION (SPINNEY METHOD)
SHIELDING CODE, contributed by the Reactor Group, HQ, UKAEA,
Risley, Warrington, Lancs., England.
FORTRAN IV, MAP; IBM 7090/7094
(Reference: TRG 1466(R))
- CCC-133: UNC-SAM 3
MONTE CARLO THREE-DIMENSIONAL COMPLEX GEOMETRY SHIELDING CODE
SYSTEM WITH ENDT, contributed by United Nuclear Corporation,
White Plains, New York.
FORTRAN IV; CCC-133A: CDC 6600; CCC-133B: CDC 1604
(References: UNC-5157, UNC-5157 Supplement 1, UNC-5243)
- CCC-134: 2DBS
TWO-DIMENSIONAL MULTIGROUP NEUTRON DIFFUSION SHIELDING CODE,
contributed by Battelle Memorial Institute, Pacific Northwest
Laboratories, Richland, Washington.
FORTRAN IV; UNIVAC 1108
(References: BNWL-1291, BNWL-640)
- CCC-135: GAMMOM
GAMMA-RAY MOMENTS METHOD CODES - GRMM AND SPENCER, contributed
by Center for Radiation Research, National Bureau of Standards,
Washington, D.C., and Atomics International, Canoga Park,
California.
FORTRAN IV; IBM 360, CDC 1604
(References: NAA-SR-MEMO-11653 and Informal Notes)
- CCC-136: COLLIMATOR
MONTE CARLO CALCULATION OF THE SPECTRUM OF GAMMA RADIATION FROM
A COLLIMATED Co-60 SOURCE, contributed by the Nuclear Engineer-
ing Department, University of Illinois, Urbana.
FORTRAN IV and MAP; IBM 7090
(Reference: NRDL-TRC-68-6)

- CCC-137: RIBD
RADIOISOTOPE BUILDUP AND DECAY CODE AND DATA LIBRARY, contributed by Battelle Memorial Institute, Pacific Northwest Laboratory, Richland, Washington.
FORTRAN IV; UNIVAC 1108
(References: DUN-4136, BNWL-962, RL-NRD-610)
- CCC-138: PIGG
A MULTIGROUP ONE-DIMENSIONAL P-1 RADIATION TRANSPORT CODE, contributed by Institutt for Atomenergi, Kjeller Research Establishment, Kjeller, Norway, and ENEA Computer Programme Library, Ispra, Italy.
FORTRAN 63; CDC 3600
(Reference: KR-129)
- CCC-139: CONSTRIIP V
VERTICAL BARRIER - FINITE SOURCE PLANE GAMMA-RAY PENETRATION CODE, contributed by the Research Triangle Institute, Research Triangle Park, North Carolina.
FORTRAN IV; IBM 360
(References: R-OU-266 and R-OU-333)
- CCC-140: DIPHO
MONTE CARLO GAMMA-RAY CODE - INFINITE MEDIUM, MONOENERGETIC AND ISOTROPIC POINT SOURCE, contributed by Laboratoire Central et Ecoles de L'Armement, Ministere des Ormees, Fort de Montrouge Arcueil (Val de Marne), France.
FORTRAN IV; IBM 7040 and 360
(Reference: ORNL-tr-2349)
- CCC-141: RAC
SPINNEY REMOVAL-DIFFUSION CODE, ATTENUATION AND HEAT GENERATION IN A MULTIREGION SHIELD, contributed by Shielding Codes Group, Tokai Establishment, Japan Atomic Energy Institute (JAERI), Tokai, Ibaraki, Japan.
FORTRAN IV; IBM 360
(Reference: Informal Notes)
- CCC-142: MERCURE 3
GAMMA-RAY KERNEL INTEGRATION CODE - STRAIGHT-LINE ATTENUATION IN THREE-DIMENSIONAL GEOMETRY, contributed by Reactor Shielding Group, CEA/CEN Fontenay-aux-Roses Nuclear Research Center, France.
FORTRAN IV; IBM 360
(Reference: CEA-R-3264 (ORNL-tr-1812))
- CCC-143: GREAT-GRASS
MONTE CARLO RADIATION TRANSPORT CODES FOR FALLOUT SHIELDING, contributed by Radiation Research Associates, Fort Worth, Texas.
FORTRAN IV; CCC-143A: IBM 1130; CCC-143B: IBM 360
(References: RRA-T78, RRA-T79, RRA-M84)

- CCC-144: TIMOC
MONTE CARLO THREE-DIMENSIONAL NEUTRON TRANSPORT CODE, contributed
by CCR EURATOM, Ispra, Italy.
CCC-144A: FORTRAN II and FAP; IBM 7090 (TIMOC). CCC-144B:
FORTRAN IV; IBM 360 (CODAC).
(References: EUR-4519, EUR-4536, EUR-4521.3)
- CCC-145: SORS
MONTE CARLO NEUTRON AND PHOTON TWO/THREE DIMENSIONAL TRANSPORT
CODE SYSTEM, contributed by Lawrence Livermore Laboratory, Liver-
more, California.
FORTRAN IV; CDC 6600
(References: UCRL-50358, UCRL-50532)
- CCC-146: UNAMIT
ONE DIMENSIONAL SPHERICAL MULTILAYER REACTOR-SHIELD-WEIGHT
OPTIMIZATION CODE, contributed by NASA Lewis Research Center,
Cleveland, Ohio, and Columbia University, New York, N. Y.
CCC-146A: FORTRAN IV; IBM 7090/7094. CCC-146B: FORTRAN IV;
IBM 360. CCC-146C: FORTRAN 63; CDC 1604.
(Reference: NASA TM X-2048)
- CCC-147: EXDOSE
CALCULATION OF THE EXTERNAL GAMMA-RAY DOSE FROM AIRBORNE FISSION
PRODUCTS, contributed by Battelle Memorial Institute, Pacific
Northwest Laboratory, Richland, Washington.
FORTRAN IV and V; UNIVAC 1108
(Reference: BNWL-811)
- CCC-148: SPARES
SPACE RADIATION ENVIRONMENT AND SHIELDING SYSTEM, contributed by
the Aerospace Group, Boeing Company, Seattle, Washington.
FORTRAN IV; IBM 360
(Reference: AS-2807)
- CCC-149: GASOUT
CALCULATION OF GASEOUS FISSION PRODUCT RELEASE FOR A ZPR-6 AND
-9 DESIGN BASIS ACCIDENT, contributed by Applied Physics Division,
Argonne National Laboratory, Argonne, Illinois.
FORTRAN IV; IBM 360
(Reference: ANL-7534)
- CCC-150: MAP
KERNEL INTEGRATION CODE IN COMPLEX GEOMETRY WITH SPECIAL APPLI-
CATION TO SURFACE SOURCES DETERMINED BY DISCRETE ORDINATES CAL-
CULATIONS, contributed by Westinghouse Astronuclear Laboratory,
Pittsburgh, Pennsylvania.
FORTRAN IV; CCC-150A: CDC 6600; CCC-150B: IBM 360
(Reference: WANL-TME-2706)

- CCC-151: DOT2DB
TWO-DIMENSIONAL MULTIGROUP DISCRETE ORDINATES TRANSPORT/
DIFFUSION CODE WITH ANISOTROPIC SCATTERING, contributed by
General Electric Nuclear Energy Division, Sunnyvale, Calif.,
Battelle Northwest Laboratory, Richland, Washington, and
ORNL, Oak Ridge, Tenn.
FORTRAN IV; GE-635
(Reference: GEAP-13537)
- CCC-152: ALGAM
MONTE CARLO ESTIMATION OF INTERNAL DOSE FROM GAMMA-RAY SOURCES
IN A PHANTOM MAN, contributed by ORNL, Oak Ridge, Tennessee.
FORTRAN IV; IBM 360
(References: ORNL-TM-2250 and ORNL-4584)
- CCC-153: ASFIT
NEUTRON AND GAMMA-RAY TRANSPORT CODE FOR ONE-DIMENSIONAL FINITE
SYSTEMS, contributed by Bhabha Atomic Research Centre, Bombay,
India.
FORTRAN 63, CDC 3600
(Reference: Informal Notes)
- CCC-154: ANDYMG3
MONTE CARLO TIME-DEPENDENT PARTICLE AND PHOTON TRANSPORT CODE -
GENERAL GEOMETRY, contributed by Los Alamos Scientific Labora-
tory, Los Alamos, New Mexico.
FORTRAN IV; CDC 6600 and UNIVAC 1108
(Reference: LA-4539)
- CCC-155: ELTRAN
ONE-DIMENSIONAL MONTE CARLO ELECTRON TRANSPORT CODE, contributed
by Sandia Laboratories, Albuquerque, New Mexico, and Ion Physics,
Burlington, Massachusetts.
FORTRAN IV; CDC 3600
(Reference: SC-TM-68-713)
- CCC-156: MECC-3
MEDIUM-ENERGY INTRANUCLEAR CASCADE CODE SYSTEM, contributed by
Neutron Physics Division, ORNL, Oak Ridge, Tennessee.
FORTRAN IV; IBM 360
(References: ORNL-4564, Phys. Rev. 188, 1711)
- CCC-157: MEVDP
PRIMARY RADIATION TRANSPORT CODE - COMPLEX GEOMETRY - COMPUTER-
IZED ANATOMICAL MODEL MAN, contributed by Air Force Weapons
Laboratory, Kirtland Air Force Base, New Mexico; Space Division,
North American Rockwell Corporation, Downey, California; Martin
Marietta Corporation, Denver, Colorado.
FORTRAN IV; CDC 6600
(References: AFWL-TR-69-68 and AFWL-TR-69-161)

- CCC-158: MAGNA
MULTI-SOURCE GAMMA-RAY KERNEL INTEGRATION CODE, contributed by
Institut for Atomenergi, Kjeller Research Establishment, Kjeller,
Norway.
FORTRAN; CCC-158A: CDC 3600; CCC-158B: CDC 1604
(References: KR-111 and IFA-RPS Memo 82)
- CCC-159: ORPHEE VI
KERNEL INTEGRATION CODE - ATTENUATION OF FAST NEUTRONS IN
CYLINDRICAL LAYERS OF WATER AND DENSE MATERIAL, contributed
by CEA/CEN Fontenay-aux-Roses Nuclear Research Center, France.
FORTRAN IV; IBM 360
(Reference: CEA-N-1244 (ORNL-tr-2357))
- CCC-160: PICA
MONTE CARLO MEDIUM ENERGY PHOTON-INDUCED INTRANUCLEAR CASCADE
ANALYSIS CODE SYSTEM, contributed by the Neutron Physics Division,
ORNL, Oak Ridge, Tennessee.
FORTRAN IV; IBM 360
(References: ORNL-4687, ORNL-TM-2481, Phys. Rev. 188, 1711)
- CCC-161: NMTC
MONTE CARLO NUCLEON-MESON TRANSPORT CODE SYSTEM, contributed by
Neutron Physics Division, ORNL, Oak Ridge, Tennessee.
FORTRAN IV; IBM 360
(References: ORNL-4606, ORNL-3622)
- CCC-162: FPIP
FISSION PRODUCT INVENTORY CODE SYSTEM, contributed by Nuclear
Systems, Westinghouse Astronuclear Laboratory, Pittsburgh, Pa.
FORTRAN IV; CDC 6600
(Reference: WANL-TME-537 Rev. 1)
- CCC-163: FISSP-CLOUD
FISSION PRODUCT INVENTORY, RELEASE, TRANSPORT AND DOSE CALCU-
LATION, contributed by Sandia Laboratories, Albuquerque, New
Mexico.
FORTRAN IV; CCC-163A: CDC 6600; CCC-163B: IBM 360
(Reference: SC-RR-70-338)
- CCC-164: NAC
NEUTRON ACTIVATION ANALYSIS AND PRODUCT ISOTOPE INVENTORY CODE,
contributed by NASA Lewis Research Center, Cleveland, Ohio.
FORTRAN IV; CCC-164A: IBM 360; CCC-164B: IBM 7090
(Reference: NASA TM-X-52460)
- CCC-165: DOSE1
GAMMA-RADIATION DOSIMETRY FOR ARBITRARY SOURCE AND TARGET GEOM-
ETRY, contributed by the Physics Division, ORNL, Oak Ridge, Tenn.
FORTRAN IV; IBM 360
(Reference: ORNL-TM-3398)

- CCC-166: DAVE
MONTE CARLO GAMMA-RAY TRANSPORT CODE SYSTEM IN ONE-DIMENSIONAL
SPHERICAL GEOMETRY, contributed by Research Institute of the
Swedish National Defence, Stockholm, Sweden.
FORTRAN IV; IBM 360
(Reference: FOA-4C-4374-29)
- CCC-167: ELF
MONTE CARLO NEUTRON TRANSPORT CODE SYSTEM FOR CYLINDERS AND
SPHERES, contributed by CEA/CEN, Fontenay-aux-Roses Nuclear
Research Center, France.
FORTRAN IV; IBM 360
(References: CEA-Note N-1361 (ORNL-tr-2409), CEA Note N-1255
(ORNL-tr-2519), AWRE-0-70/63, AEEW-M 824, AEEW-M 802)
- CCC-168: FASTER III
MONTE CARLO NEUTRON AND GAMMA-RAY TRANSPORT CODE IN COMPLEX
GEOMETRIES, contributed by A. R. T. Research Corporation, Los
Angeles, California; NASA Lewis Research Center, Cleveland, Ohio.
FORTRAN IV; CCC-168A: UNIVAC 1108; CCC-168B: IBM 360
(Reference: ART-45, Vol. I & II, and WANL-PP(LL)-010, Vol. 9)

ABSTRACTS OF
RSIC
COMPUTER CODE PACKAGES

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RSIC CODE PACKAGE CCC-116

1. NAME AND TITLE OF CODE

TRECO: An Orbital Integration Estimation of Trapped Radiation.

TRECO is a version of an orbital integration code developed at Aerospace Corporation for use in conjunction with the generation of trapped radiation model environments. The model environment work is now being done at National Space Science Data Center (NSSDC), and the data is available from NSSDC.

2. CONTRIBUTOR

National Space Science Data Center, NASA Goddard Space Flight Center, Greenbelt, Maryland.

3. CODING LANGUAGE AND COMPUTER

CCC-116A: FORTRAN IV; IBM 360/75 and 360/91.

CCC-116B: FORTRAN IV; CDC 6600.

4. NATURE OF PROBLEM SOLVED

TRECO computes elliptic or circular orbits for earth satellites; or alternatively, it can read previously generated orbits stored on magnetic tape or punched cards. For each orbit, TRECO computes the daily fluence of geomagnetically trapped particles from model trapped radiation environments. Both the flux of electrons and protons can be computed by the code.

5. METHOD OF SOLUTION

To estimate the geomagnetically trapped radiation that a satellite would accumulate, the trajectory is computed as a function of time, given six initial conditions. These are (1) latitude of perigee (the point on an orbit of nearest approach to the earth), (2) initial longitude of perigee, (3) height of apogee (the point on an orbit farthest away from the earth), (4) height of perigee, (5) orbit inclination,

(6) quadrant of initial longitude of perigee or a direction for equatorial orbits.

Kepler's Laws and Kepler's Equation are used to determine the position of the orbiting satellite as a function of time in terms of geographic coordinates. The position in geographic coordinates, which is a three-dimensional representation, is mapped into a two-dimensional geomagnetic coordinate or the (B,L) coordinate system. In this system, B is the geomagnetic field strength and L is a parameter that is the physical analog of the geomagnetic equatorial distance to a field line in a dipole field and it is approximately constant along lines of force. The model trapped radiation environment is expressed in (B,L) coordinates. As each orbit (B,L) is determined, the flux is computed for each energy requested. Then each of these fluxes is multiplied by the time-interval in seconds, and the results are summed for each energy.

6. RESTRICTIONS OR LIMITATIONS

Maximum number of energy groups, 30. This can be increased easily by user.

7. TYPICAL RUNNING TIME

Estimated running time of the packaged sample problem on the IBM 360/91: 2 minutes.

8. COMPUTER HARDWARE REQUIREMENTS

Originally designed for the IBM 7094, the code is now operable on the CDC 6600, the IBM 360/75, and the 360/91.

9. COMPUTER SOFTWARE REQUIREMENTS

Originally programmed in FORTRAN II, TRECO has been converted to FORTRAN IV for the CDC 6600 and is also available on the IBM 360/75/91 Operating Systems, using OS-360 FORTRAN H Compiler.

Flux maps and spectrum data are required as input and a number of sets of these data are included in the code package. Updates are available from NSSDC.

Input variables are used to determine the number and mode of tapes that may be used as orbit packages from a previous calculation, or written by the current calculation.

10. REFERENCES

A. B. Lucero, "TRECO, An Orbital Integration Computer Program for Trapped Radiation," NSSDC 68-02 (January 1968).

James I. Vette, "Models of the Trapped Radiation Environment, Volume I: Inner Zone Protons and Electrons," NASA SP-3024 (1966).

James I. Vette, Antonio B. Lucero, and Jon A. Wright, "Models of the Trapped Radiation Environment, Volume II: Inner and Outer Zone Electrons," NASA SP-3024 (1966).

James I. Vette and Antonio B. Lucero, "Models of the Trapped Radiation Environment, Volume III: Electrons at Synchronous Altitudes," NASA SP-3024 (1967).

Joseph H. King, "Models of the Trapped Radiation Environment, Volume IV: Low Energy Protons," NASA SP-3024 (1967).

James P. Lavine and James I. Vette, "Models of the Trapped Radiation Environment, Volume V: Inner Belt Protons," NASA SP-3024 (1969).

James P. Lavine and James I. Vette, "Models of the Trapped Radiation Environment, Volume VI: High Energy Protons," NASA SP-3024 (1970).

11. CONTENTS OF THE CODE PACKAGE

Each package contains the following items:

- a. the referenced documents,
- b. a reel of magnetic tape on which is written in several files:
the source card deck, input for a sample problem, including flux and energy spectrum data, and the output listing of the sample problem.

12. HOW TO OBTAIN PACKAGE

Inquiries or requests for the code package may be mailed to

CODES COORDINATOR
Radiation Shielding Information Center
Oak Ridge National Laboratory
Post Office Box X
Oak Ridge, Tennessee 37830

or telephoned to

Area Code 615; 483-8611, extension 3-6944, or to
FTS xx-615-483-6944.

Persons requesting the code package should send a reel of magnetic tape to the above address, indicating how it should be written.

13. DATE OF ABSTRACT

August 1971.

RSIC CODE PACKAGE CCC-117

1. NAME AND TITLE OF CODE

BETA: Monte Carlo Bremsstrahlung and Electron Transport
Analysis Code in Complex Geometry.

2. CONTRIBUTORS

A.R.T. Research Corporation, Los Angeles, California.
Air Force Weapons Laboratory, Albuquerque, New Mexico.

3. CODING LANGUAGE AND COMPUTER

FORTRAN IV; IBM 7090, CDC 6600, and UNIVAC 1108. The packaged
version was run by RSIC only on the IBM 7090.

4. NATURE OF PROBLEM SOLVED

BETA is a computer procedure for bremsstrahlung and electron
transport analysis developed to augment experimental efforts. Gener-
alized three-dimensional geometry is included. Electron transport and
bremsstrahlung production and transport are treated explicitly for the
energy range from 0.1 to 15 MeV. Arbitrary sources are handled as
functions of energy, time, space, and angle. Point, surface, and/or
volume detectors are considered. The effects of electric field due
to charge buildup are treated.

Sample calculations made by BETA include energy deposition in
an aluminum slab, time dependence of energy deposition in tungsten,
energy deposition in metal discs, and charge trapping.

5. METHOD OF SOLUTION

BETA uses random sampling techniques to perform the integrations of
the order-of-scatter equations to give the unscattered and scattered com-
ponents of the flux. Electron transport considers electron-nucleus, elec-
tron-electron, and electron-bremsstrahlung scattering. Wide angle scatter-
ings are considered explicitly, while narrow angle scatterings are

grouped and the net effect lumped together. Klein-Nishina scattering is considered for photon transport. BETA will recognize simple surfaces such as planes, cones, elliptic cylinders and ellipsoids in specifying the geometry. Multiple energy and angular dependent sources in rectangular, cylindrical, and spherical geometry may be considered with the geometry of each source being superimposed over the various geometric regions. Flux estimation at points, surfaces, and in volumes is accomplished by summing contributions from all order-of-scatter components. Legendre moments of the angular flux are used to construct azimuthally-averaged differential angular fluxes. The time dependence is obtained by computing temporal moments and then analytically re-generating the temporal dependence from these moments.

6. RESTRICTIONS OR LIMITATIONS

None noted.

7. TYPICAL RUNNING TIME

No effort has been made by RSIC to establish typical running time. Estimated running time of the sample problem on the IBM 7090: 5 minutes.

8. COMPUTER HARDWARE REQUIREMENTS

The code is operable on the IBM 7090 and should be made operable easily on other 32K or larger computers. Five tape units are used.

9. COMPUTER SOFTWARE REQUIREMENTS

The packaged code has been compiled by RSIC on the IBM 7090 IBSYS Operating System using IJOB Processor. The OVERLAY feature and ALTIO are used in this version. The code should be compatible with other FORTRAN IV compilers.

10. REFERENCE

T. M. Jordan, "BETA, A Monte Carlo Computer Program for Bremsstrahlung and Electron Transport Analysis," AFWL-TR-68-111 (October 1968).

11. CONTENTS OF CODE PACKAGE

The package contains the following items:

- a. the referenced document,
- b. a reel of magnetic tape on which is written in 4 files: the source card deck, the object deck, BCD input for a sample problem, and the BCD output listing from the problem.

12. HOW TO OBTAIN PACKAGE

Inquiries or requests for the code package may be mailed to

CODES COORDINATOR
Radiation Shielding Information Center
Oak Ridge National Laboratory
P. O. Box X
Oak Ridge, Tennessee 37830

or telephoned to

Area Code 615; 483-8611, extension 3-6944, or to
FTS xx-615-483-6944.

Persons requesting the package should send a reel of magnetic tape to the above address, designating how the tape is to be written.

13. DATE OF ABSTRACT

August 1971.

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RSIC CODE PACKAGE CCC-118

1. NAME AND TITLE OF CODE

SIGMA: Space Radiation Dose Analysis Within Complex Configurations.

2. CONTRIBUTOR

McDonnell Douglas Astronautics Company, Western Division, Huntington Beach, California.

3. CODING LANGUAGE AND COMPUTER

FORTRAN IV; IBM 7090 and 7094.

4. NATURE OF PROBLEM SOLVED

SIGMA calculates space radiation doses at arbitrary points inside a space vehicle by performing a numerical angular integration of dose attenuation kernels about the dose points. The kernels are curve-fit functions constructed from input data. The geometry of the vehicle, equipment and supplies, and man models are described by quadric surfaces. Simultaneous dose calculations for multiple vehicle trajectories, each involving several radiation sources, may be performed for each specified dose point. The calculation may be a parametric study of dose as a function of shield thickness or an analysis of the dose received through designated outer sectors of the vehicle.

5. METHOD OF SOLUTION

SIGMA is usually used in conjunction with other codes such as CCC-70/CHARGE. The latter code computes basic dose transmission data through materials for idealized spherical or slab geometries of various thicknesses and for various types of radiations. The space vehicle dose calculation is performed by angular integration accomplished numerically by means of Simpson's rule. The polar and azimuthal angles about a dose point are divided into intervals, the endpoints of which

define directions for which ray tracing analysis through the space vehicle is performed. This analysis yields material thicknesses between dose point and vehicle exterior for each ray. From the source angular description, which may be anisotropic, the material thicknesses, and the basic dose transmission versus shield thickness (calculated, for example, by CHARGE) input, the dose at a given point is calculated. The source strength for a given ray direction is approximated by linear interpolation between source input value cosines and strengths. The dose transmissions for a given ray thickness is approximated by exponential interpolation between the basic dose transmission and shield thickness values.

Results are presented as either a printout of dose as a function of shield thickness or a printout of the dose received through designated outer sections of the vehicle.

6. RESTRICTIONS OR LIMITATIONS

Some dimensional limitations are as follows:

Maximum number of orbits	10
Maximum number of materials	25
Maximum number of quadric surfaces	100
Maximum number of material regions	100
Maximum number of shield locations	100
Maximum number of shield thicknesses	50
Maximum number of dose points	25
Maximum number of dose transmission values	20

7. TYPICAL RUNNING TIME

Estimated running time for the 3 packaged sample problems on the IBM 7090: 5 minutes.

8. COMPUTER HARDWARE REQUIREMENTS

The code was designed for the IBM 7094 with standard I-O and one tape unit.

9. COMPUTER SOFTWARE REQUIREMENTS

The code is operable on the IBM 7090/7094 IBSYS Operating System using IBJOB Processor. An option for plotting on the SC-4020 is included. Three subroutines for plotting are included as one file in the code package.

10. REFERENCES

T. M. Jordan, "SIGMA, A Computer Program for Space Radiation Dose Analysis Within Complex Configurations," DAC-60878 (November 1967).

11. CONTENTS OF CODE PACKAGE

The package contains the following items:

- a. the referenced document,
- b. a reel of magnetic tape on which is written in 5 files: the BCD source card decks, object deck, BCD input for a sample problem, and the BCD output from running the problem.

12. HOW TO OBTAIN PACKAGE

Inquiries or requests for the code package may be mailed to

CODES COORDINATOR
Radiation Shielding Information Center
Oak Ridge National Laboratory
P. O. Box X
Oak Ridge, Tennessee 37830

or telephoned to

Area Code 615; 483-8611, extension 3-6944, or to
FTS xx-615-483-6944.

Persons requesting the code package should send a reel of magnetic tape to the above address, designating how it should be written.

13. DATE OF ABSTRACT

August 1971.



RSIC CODE PACKAGE CCC-119

1. NAME AND TITLE OF CODE

ELBA: Electron and Bremsstrahlung Dose Rate Code.

2. CONTRIBUTOR

Space Sciences Laboratory, NASA George C. Marshall Space Flight Center, Huntsville, Alabama.

3. CODING LANGUAGE AND COMPUTER

FORTRAN IV; IBM 7090 and 7094.

4. NATURE OF PROBLEM SOLVED

ELBA takes an incident isotropic electron flux with a given differential energy spectrum and calculates the dose rate received from bremsstrahlung produced in a plane aluminum shield placed in front of the receiver. There is an option to also calculate the electron dose rate from the same source.

5. METHOD OF SOLUTION

The electron differential spectrum as a function of depth is inferred by assuming that electrons travel straight ahead and that distance travelled and energy are related by a range-energy relationship. The electron dose rate at a given depth is calculated by integrating, over energy and direction, the product of the electron flux, the stopping power, and the appropriate flux-to-dose rate conversion factor.

The bremsstrahlung source is assumed to be plane and isotropic at a given depth. This source is defined as the integral over energy and direction of the product of photon energy, the differential bremsstrahlung spectrum from electrons of a given energy, and the electron flux differential spectrum. The differential bremsstrahlung spectrum is derived from the Born approximation cross section multiplied by a

correction factor. The bremsstrahlung dose rate is obtained by integrating, over photon energy and slab volume, the product of the bremsstrahlung source, photon energy flux-to-dose rate conversion factor, buildup factor, and attenuation kernel. The buildup factor assumed is a plane isotropic buildup factor generated by Monte Carlo calculations. The integrations are performed by evaluating the integrand at the midpoint of each integration step, multiplying by the step width, and summing the result.

The incident electron spectrum, dose rate conversion factors, and range formula coefficients are input by the user. The buildup factor information is contained in three DATA statements in subroutine BURP.

6. RESTRICTIONS AND LIMITATIONS

There are limitations on the dimensions of certain arrays, but these dimensions probably can be increased to meet the user's requirements.

7. TYPICAL RUNNING TIME

Estimated running time of the packaged sample problem on the IBM 7090: 13.8 minutes.

8. COMPUTER HARDWARE REQUIREMENTS

The code was written for the IBM 7094 using standard I-O units.

9. COMPUTER SOFTWARE REQUIREMENTS

The code is operable on the IBM 7090/7094 IBSYS Operating System using IBJOB Processor.

10. REFERENCE

Martin O. Burrell, J. J. Wright, and John W. Watts, Jr., "The Calculation of Electron and Bremsstrahlung Dose Rates," NASA-SP-169 (ANS-SD-5) pp. 529-538 (1968).

Informal notes.

11. CONTENTS OF CODE PACKAGE

The package contains the following items:

- a. the referenced document,
- b. a reel of magnetic tape on which is written in 4 files: the BCD source card deck, BCD input for a sample problem and the BCD output from running the problem.

12. HOW TO OBTAIN PACKAGE

Inquiries or requests for the code package may be mailed to

CODES COORDINATOR
Radiation Shielding Information Center
Oak Ridge National Laboratory
Post Office Box X
Oak Ridge, Tennessee 37830

or telephoned to

Area Code 615; 483-8611, extension 3-6944, or to
FTS xx-615-483-6944.

Persons requesting the code package should send a reel of magnetic tape to the above address, designating how it should be written.

13. DATE OF ABSTRACT

August 1971.



RSIC CODE PACKAGE CCC-120

1. NAME AND TITLE OF CODE

SPACETRAN: Dose Calculations at Detectors at Various Distances from the Surface of a Cylinder.

2. CONTRIBUTOR

Neutron Physics Division, Oak Ridge National Laboratory, Oak Ridge, Tennessee.

3. CODING LANGUAGE AND COMPUTER

FORTRAN IV; IBM 360/75.

4. NATURE OF PROBLEM SOLVED

SPACETRAN is designed to calculate the energy-dependent total flux, or some proportional quantity such as kerma, due to the radiation leakage from the surface of a right-circular cylinder at detector positions located arbitrary distances from the surface. The assumptions are made that the radiation emerging from the finite cylinder has no spatial dependence and that a vacuum surrounds the cylinder.

5. METHOD OF SOLUTION

There are two versions of the program in the code package. SPACETRAN-I uses the surface angular fluxes calculated by the discrete ordinates S_n code, CCC-82/ANISN, as input. SPACETRAN-II assumes that the surface angular flux for all energies can be represented as a function $\cos^N \phi$, where ϕ is the angle between surface outward normal and radiation direction, and N is an integer specified by the user.

For both versions the energy group structure and the number and location of detectors is arbitrary. The flux (or response function) for a given energy group at some detection point are computed by summing the contributions from each surface area element over the entire surface. The surface area elements are defined by input data.

SPACETRAN-I handles contributions either from a cylinder "end" or "side," so the total contribution must be obtained by adding the results of separate end and side runs. ANISN angular fluxes are specified for discrete directions. In general, the direction between the detector and contributing area will not exactly coincide with one of these discrete directions. In this case, the ANISN angular flux for the "closest" discrete direction is used to approximate the contribution to the detector.

SPACETRAN-II handles contributions from both the side and end of a cylinder in a single run. Since the assumed angular distribution is specified by a continuous function, it is not necessary to perform the angle selection described above.

For each detector specified, both versions compute the flux and a response proportional to flux in each energy group and also compute the sum of these quantities over all energy groups.

6. RESTRICTIONS OR LIMITATIONS

There are limitations on the dimensions of certain arrays, but these dimensions can probably be increased somewhat to meet the user's requirements.

7. TYPICAL RUNNING TIME

Estimated running time of the packaged sample problem: 1 minute.

8. COMPUTER HARDWARE REQUIREMENTS

SPACETRAN was designed for an IBM 360/75 and uses standard I-O units.

9. COMPUTER SOFTWARE REQUIREMENTS

The code is operable on the IBM 360/75-91 Operating System using OS-360 FORTRAN H compiler.

10. REFERENCE

S. N. Cramer and M. Solomito, "SPACETRAN: A Code to Calculate Dose at Detectors at Various Distances from the Surface of a Cylinder," ORNL-TM-2592 (June 1969).

11. CONTENTS OF CODE PACKAGE

The package contains the following items:

- a. the referenced document,
- b. a reel of magnetic tape on which is written in 6 files: the BCD source card decks for versions I and II, BCD input for a sample problem for each version, and BCD output from running the problems.

12. HOW TO OBTAIN PACKAGE

Inquiries or requests for the code package may be mailed to
CODES COORDINATOR
Radiation Shielding Information Center
Oak Ridge National Laboratory
Post Office Box X
Oak Ridge, Tennessee 37830

or telephoned to

Area Code 615; 483-8611, extension 3-6944, or to
FTS xx-615-483-6944.

Persons requesting the code package should send a reel of magnetic tape to the above address, designating how it should be written.

13. DATE OF ABSTRACT

August 1971.

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RSIC CODE PACKAGE CCC-121

1. NAME AND TITLE OF CODE

SABINE: Spinney (Removal Diffusion) Shielding Code in
Complex Geometry.

AUXILIARY ROUTINE

SABLIB: Data Generator.

The code package is also available through the ENEA Computer
Programme Library, Abstract ENEA 218.

2. CONTRIBUTOR

EURATOM, Ispra (Varese) Italy.

3. CODING LANGUAGE AND COMPUTER

FORTRAN IV; IBM 360/65/75/91.

4. NATURE OF PROBLEM SOLVED

SABINE determines the spatial distribution of energy dependent
neutron and gamma-ray fluxes in a reactor shield. The neutron source
is a fission source arbitrarily distributed inside the core. The
gamma-ray source may include fission and fission product gamma rays
in the core, and neutron capture gamma rays in the core and shield.
Complex geometries may be treated.

The program is designed to solve a range of shielding problems
providing information on neutron and gamma-ray penetration, heat
deposition and reaction rates.

5. METHOD OF SOLUTION

The removal-diffusion method has been applied to calculate neu-
tron fluxes. The gamma-ray flux is the product of the uncollided flux
times a region dependent buildup factor, which is interpolated from
a table of values.

Other quantities calculated include any neutron response function, gamma-ray dose and energy deposition.

The group cross sections were calculated with GGC II and put into SABINE format by SABLIB. BIGGI 3 (CCC-66) was used to calculate build-up factors for 6 materials.

6. RESTRICTIONS OR LIMITATIONS

Provisions are made for the following: 2 core regions and up to 20 shielding regions; up to 35 neutron energy groups and 7 gamma-ray energy groups.

7. TYPICAL RUNNING TIME

Estimated running time of the 8 packaged sample problems on the IBM 360/91: 24 seconds.

8. COMPUTER HARDWARE REQUIREMENTS

The code was originally designed for the IBM 7090 and later converted to run on the IBM 360. Standard I-O and 5 additional tape units or direct access devices are required.

9. COMPUTER SOFTWARE REQUIREMENTS

The packaged code is operable on the IBM 360/75/91 Operating System using OS-360 FORTRAN H compiler and the OVERLAY feature.

10. REFERENCES

C. Ponti, H. Preusch and H. Schubart, "SABINE, A One-Dimensional Bulk Shielding Program," EUR 3636.e (1967).

C. Ponti, "SABINE, A One-Dimensional Bulk Shielding Program," EUR 3636.e Addendum (1969).

11. CONTENTS OF CODE PACKAGE

The package contains the following items:

- a. the referenced documents,
- b. a reel of magnetic tape on which is written in 9 files: the BCD source card decks, library data, overlay control cards, DD cards, BCD input data, and the BCD output listing of several sample problems.

12. HOW TO OBTAIN PACKAGE

Inquiries or requests for the code package may be mailed to

CODES COORDINATOR
Radiation Shielding Information Center
Oak Ridge National Laboratory
Post Office Box X
Oak Ridge, Tennessee 37830

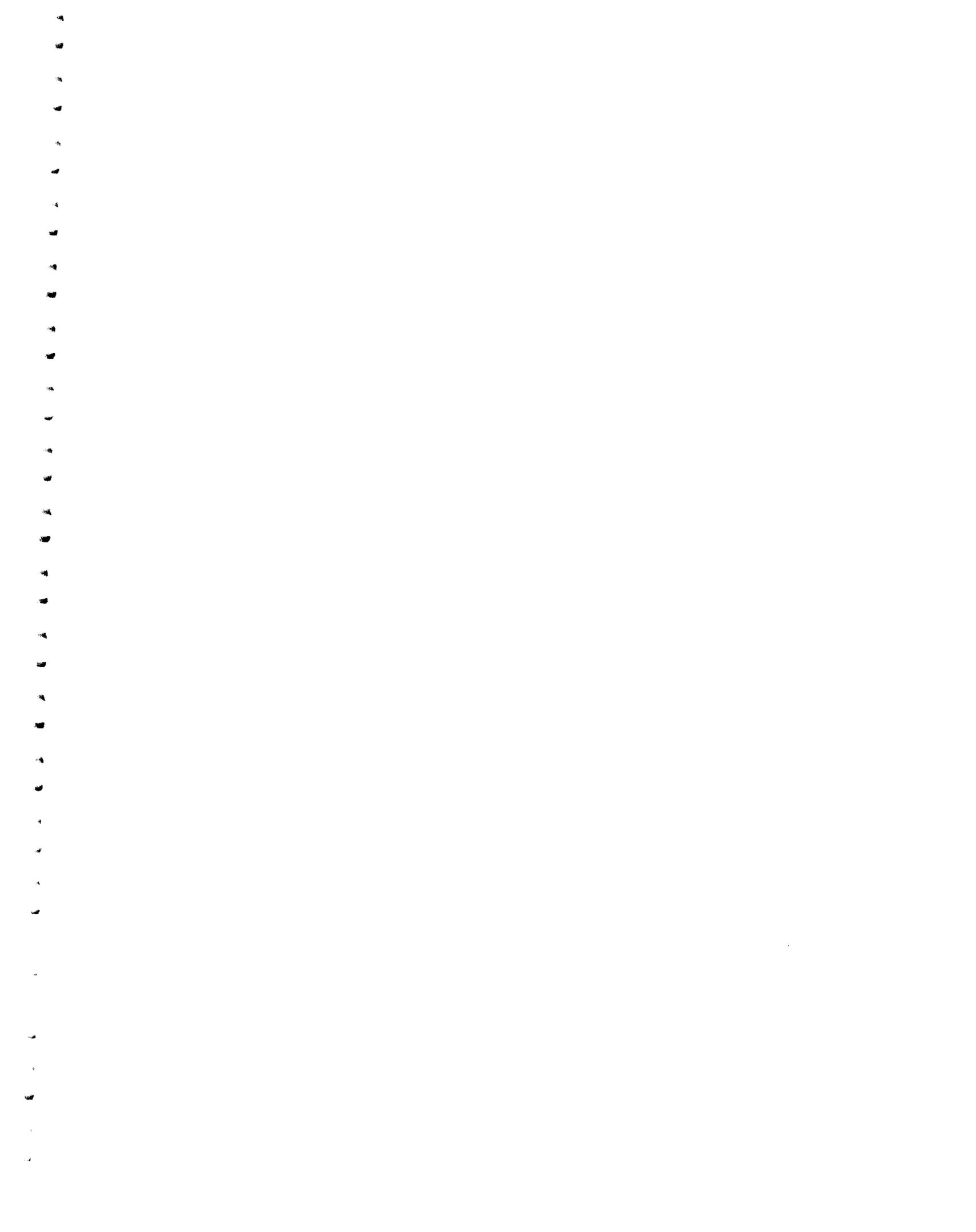
or telephoned to

Area Code 615; 483-8611, extension 3-6944, or to
FTS xx-615-483-6944.

Persons requesting the code package should send a reel of magnetic tape to the above address.

13. DATE OF ABSTRACT

August 1971.



RSIC CODE PACKAGE CCC-122

1. NAME AND TITLE OF CODE

RAD 2: Fission Product Radioactivities Calculation.

RAD 2 is also available through the Argonne Code Center, Abstract 231, and through the ENEA Computer Programme Library.

2. CONTRIBUTOR

General Atomic, San Diego, California.

3. CODING LANGUAGE AND COMPUTER

FORTRAN IV; IBM 7090.

4. NATURE OF PROBLEM SOLVED

RAD 2 was designed for the calculation of fission product activity distributions in a high temperature gas-cooled reactor system. It will perform a variety of bookkeeping calculations for radioactive product decay chains of any desired length.

5. METHOD OF SOLUTION

The program requires input information concerning the release of each fission product from the fuel material or, alternately, a "diffusion parameter" from which the program will calculate these release fractions, assuming diffusive release.

6. RESTRICTIONS OR LIMITATIONS

None noted.

7. TYPICAL RUNNING TIME

No effort has been made to ascertain a typical running time. The estimated running time of the packaged sample problem: less than one minute.

8. COMPUTER HARDWARE REQUIREMENTS

The code was designed for the IBM 7040/7044 and is operable on standard 32K IBM 7090 equipment with 3 tape units required for I-O and system.

9. COMPUTER SOFTWARE REQUIREMENTS

The IBM 7090 FORTRAN IV Monitor System was used to compile and execute the sample problem.

10. REFERENCES

F. E. Vanslager, "RAD 2, A Computer Program for Calculating Fission Product Radioactivities," GAMD-6519 (July 1965).

11. CONTENTS OF CODE PACKAGE

The package contains the following items:

- a. the referenced document,
- b. a reel of magnetic tape on which is written in 5 separate files: the BCD source card deck, BCD input for a sample problem which includes a library of permanent data, an object deck, and a BCD output listing from running the sample problem.

12. HOW TO OBTAIN PACKAGE

Inquiries or requests for the code package may be mailed to
CODES COORDINATOR
Radiation Shielding Information Center
Oak Ridge National Laboratory
Post Office Box X
Oak Ridge, Tennessee 37830

or telephoned to

Area Code 615; 483-8611, extension 3-6944, or to
FTS xx-615-483-6944.

Persons requesting the package should send a reel of magnetic tape to the above address.

13. DATE OF ABSTRACT

August 1971.

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RSIC CODE PACKAGE CCC-123

1. NAME AND TITLE OF CODE

XSDRN: Multigroup One-Dimensional Discrete Ordinates Spectral Averaging Neutron Transport Code.

AUXILIARY ROUTINES

BX: Produces XSDRN Library from GAM-II/THERMOS Libraries.

JUANITA: Cross Section Library Converter Code.

SUP: Cross Section Library Update Code.

SECS: Cross Section Library Edit Code.

2. CONTRIBUTORS

Mathematics, Neutron Physics and Reactor Divisions, Oak Ridge National Laboratory, Oak Ridge, Tennessee.

3. CODING LANGUAGE AND COMPUTER

FORTRAN IV; IBM 360/65/75/91.

4. NATURE OF PROBLEM SOLVED

XSDRN uses the Nordheim integral treatment, narrow resonance, or infinite mass approximation to process resonance data on a master cross section library and thus obtain microscopic fine-group cross sections for a large number of nuclides. The code will then use these cross sections in an independent calculation to solve for fluxes, eigenvalues, critical dimensions, etc., using discrete ordinates, diffusion, or an infinite medium theory calculation. The fine-group fluxes thus obtained can then be used to collapse the fine-group cross section data to a more tenable broad-group structure for use in several independent computer codes.

5. METHOD OF SOLUTION

The principal calculations performed by XSDRN (resonance calculation and flux calculation) both employ numerical finite-difference

techniques. For the resonance calculation, this involves a Simpson's integration to solve for the collision density in the resonance range. The flux calculations employ a multigroup energy structure, an arbitrary spatial structure and a mechanical angular quadrature, all of which must be used in the various integration and differencing schemes in the code.

The flexible dimensioning scheme employed by the code allows one to make optimal use of his core storage. A unique method of storing cross sections is employed which eliminates impossible and/or zero transfer cross sections.

The present master library tapes for XSDRN are produced by the BX code. Cross section tapes can be generated for ANISN, DOT, CITATION, ROD, or the EXTERMINATOR-II codes.

6. RESTRICTIONS OR LIMITATIONS

The principal restriction is the availability of adequate core storage to build required arrays. The code is flexibly dimensioned which means that array sizes are set for the particular problem at execution time. Present 123 group cross section library tapes need the 512K byte storage for efficient execution, but could be made to operate on a smaller machine through various out-of-core storage capabilities.

7. TYPICAL RUNNING TIME

Resonance calculations: Typical running times on the IBM 360/75 have been on the order of one half to one minute per nuclide. Flux calculation: It is extremely difficult, if not impossible, to assign accurate times for the flux calculation, since it depends on number of energy groups, number of space points, geometry, calculational option, cross section order, angular quadrature, convergence criteria, and even on out-of-core storage allocation. A typical problem (S_4P_3 , 25 space points, 123 energy groups, cylinder, k-calculation, reduce cross sections, 4 resonance nuclides) generally runs on the order of

ten to twelve minutes (total time). A fixed source calculation for the same system would take approximately the same time.

Estimated running time of the packaged sample problem on the IBM 360/91: 17 minutes.

8. COMPUTER HARDWARE REQUIREMENTS

XSDRN is designed to use computers in the IBM 360 series with a 512K byte directly-addressable storage. It may be run on an IBM 360 with approximately 90K words of directly addressable core storage available for the program. The various calculational options require from four to thirteen input-output devices, depending on the problem. At least one direct access device must be available.

9. COMPUTER SOFTWARE REQUIREMENTS

XSDRN is operable on the IBM OS 360 with FORTRAN H compiler.

The program consists of approximately 80 subroutines on 6600 source cards. The program is presently used in a four level overlay structure consisting of fourteen separate links. With this structure ~90K words are required on the IBM 360/75.

10. REFERENCES

N. M. Greene and C. W. Craven, Jr., "XSDRN: A Discrete Ordinates Spectral Averaging Code," ORNL-TM-2500 (July 1969).

11. CONTENTS OF CODE PACKAGE

The package contains the following items:

- a. the referenced document,
- b. a reel of magnetic tape on which is written in 8 separate files: the BCD source card decks for XSDRN and the auxiliary routines, input and output for sample problems,
- c. a library of cross sections (182,380 records written as card images on magnetic tape).

12. HOW TO OBTAIN PACKAGE

Inquiries or requests for the code package may be mailed to

CODES COORDINATOR
Radiation Shielding Information Center
Oak Ridge National Laboratory
Post Office Box X
Oak Ridge, Tennessee 37830

or telephoned to

Area Code 615; 483-8611, extension 3-6944, or to
FTS xx-615-483-6944.

Persons requesting the package should send reels of magnetic tape to the above address as follows: 1 reel for the program; 1 reel for 9-track, 800 bpi, or up to 7 reels for 7-track, 556 bpi, the number depending on blocking of the written data.

13. DATE OF ABSTRACT

August 1971.

RSIC CODE PACKAGE CCC-124

1. NAME AND TITLE OF CODE

KDLIBE: Kernel-Diffusion Shielding Analysis System.

AUXILIARY ROUTINES

QADRD: Kernel Integration Code - Uncollided Flux Calculation.

SURF: Conical and Plane Surface Scattering Code.

RAMP: Multigroup Source Term Data Generator.

GAMMIX: Microscopic Gamma Production Cross Section Generator.

ZIP III: Generalized Nuclear Analysis Code.

REORG: Secondary Gamma Source Data Generator.

QADRR: Kernel Integration Dose Rate Calculation.

UPDONC: Cross Section Update Code.

GEORGE: Multigroup One-Dimensional Diffusion Code (linkage of C-VII, Multigroup Constant Generator, and F-N, Multigroup, Multiregion One-Dimension Neutron Diffusion Code).

2. CONTRIBUTOR

Nuclear Systems Programs, Space Systems, Missile and Space Division, General Electric Company, Cincinnati, Ohio.

3. CODING LANGUAGE AND COMPUTER

124A: FORTRAN IV; GE 635.

124B: FORTRAN IV; IBM 360/91/75.

4. NATURE OF PROBLEM SOLVED

KDLIBE is an automated kernel-diffusion analysis sequence, with associated nuclear data, which were developed for shield nuclear analysis. The method is suitable for many preliminary and parametric shielding calculations. The sequence and data have been applied to the analysis of titanium hydride, and lithium hydride-Hevimet shields for space nuclear power reactors and lead-water shields for undersea nuclear reactors.

5. METHOD OF SOLUTION

KDLIBE is a kernel-diffusion code system composed of 9 separate codes, each of which may be run independently. Six of the codes may be run as a removal-diffusion sequence with certain data passed to the succeeding code by tape or disc. This sequence may be broken at any time and restarted on output card option. SURF, ZIP III, and UPDONC are independent, QARDR, RAMP, GEORGE, GAMMIX, REORG, and QADRR make up the removal-diffusion sequence.

The basic idea of the removal-diffusion sequence is essentially that used in codes such as CCC-54/NRN. A reasonably accurate flux calculation can be made by using the transport kernel to calculate the uncollided flux due to the primary source and diffusion theory to calculate the collided flux, where the source for the diffusion theory calculation is obtained from the uncollided flux and the zeroeth moment scatter-transfer matrix in each material.

QARDR combines the point-to-point attenuation function with summation over source regions. It is derived from QAD-P5 (CCC-48) by replacing all optional material attenuation functions by a simple exponential function having an energy-dependent neutron removal cross section. Uncollided (removal) neutron fluxes are computed and stored on tape or disc for use in RAMP's multigroup external source calculation to get input for GEORGE and for combination with GEORGE diffusion fluxes in REORG. The removal fluxes may be optionally punched on cards.

RAMP combines the uncollided multigroup flux from QARDR with multigroup scatter transfer cross sections to produce multigroup (external) source terms for input to GEORGE multigroup diffusion calculation.

GEORGE is a linkage of 2 codes, each of which may be run independently. C-VII performs a normal mode (space-independent buckling), continuous slowing down calculation using 19 non-thermal group cross sections and one thermal group. Multigroup cross sections in 1 to 19 groups for each of several reactor regions are produced for use in the F-N part of GEORGE, which solves the 1-dimensional

multigroup diffusion equation with transverse buckling to account for leakage perpendicular to the direction of computation. Output includes reactivity, flux, and power distributions. C-VII employs an analytical solution.

GAMMIX processes input microscopic gamma production cross sections into macroscopic (mixed) composition cross sections for input to REORG. Up to 20 incident neutron flux groups and 30 gamma source groups are allowed, and a total of 20 microscopic cross section sets may be entered from which an indefinite number of composition cross sections may be computed. The mixed data is weighted by atom densities and g-factors.

REORG combines the collided (diffusion) fluxes from GEORGE with the uncollided fluxes from QADRD and uses these total fluxes (in the diffusion energy group structure) to compute secondary gamma sources for use in QADRD. Compatibility of lattice points and group structure between the collided and uncollided is achieved through interpolation/extrapolation and collapsing.

QADRR, the final code in the kernel-diffusion sequence, was developed from converting QAD-P5 (CCC-48) to FORTRAN IV and incorporating the material attenuation functions from 14-0 (CCC-1) for the calculation of neutron and gamma-ray dose and energy absorption rates. The point kernel method combines the use of point-to-point attenuation functions with summation over source regions.

SURF calculates neutron and gamma-ray dose rates and flux distribution in energy at unshielded receiver points due to direct beam and once-scattered radiation from angularly anisotropic point sources.

ZIP III contains most of the same algorithms in GEORGE and uses the same data library, and is specially designed for cylindrical calculations limited to 2 or 3 diffusion groups. Its principal use is for reactor survey and parametric calculations and for determining the bare-equivalent transverse dimensions for more detailed 1-dimensional cylindrical calculations using GEORGE or multigroup transport. Output includes matched savings, reactivity, and radial and axial power distributions.

UPDONC prepares, duplicates, updates by correction, addition or deletion, punches, graphs, and edits the nuclear data tape (NDT) used by GEORGE and ZIP III.

6. RESTRICTIONS OR LIMITATIONS

Limits implied by core storage and code dimensions are outlined in the description of the input of each part of the code system.

7. TYPICAL RUNNING TIME

Estimated running time of the packaged sample problems and core storage required (IBM 360/91) are tabled below:

CODE	REGION SIZE (K)	RUNNING TIME (SEC) (GO STEP)	COMMENT
UPDONC	118	.60	
QADRD	128	.80	
RAMP	192	2.2	
GAMMIX	130	.37	
GEORGE	260	2.73	Run as an overlay
REORG	206	2.18	
QADRR	172	12.8	
SURF	116	6.56	
ZIP III	236	15.16	

8. COMPUTER HARDWARE REQUIREMENTS

The system was designed for the GE 635 and can also be run remotely using the GERTS system with a teletype as the remote terminal. Version B is operable on the IBM 360.

Maximum number of tape units or direct access devices: 3 plus input-output.

9. COMPUTER SOFTWARE REQUIREMENTS

The GE 635 standard operating system may be used for Version A. Version B program is operable on the IBM 360/75/91 Operating System using OS-360 FORTRAN H compiler.

The GEORGE codes require the overlay feature. UPDONC requires special cross sections which are included in the package.

10. REFERENCES

W. B. Henderson and W. E. Edwards, "NSP Kernel-Diffusion Library, KDLIBE, NS0910 (User's Manual)," GESP-226 (1970).

P. G. Fischer, F. D. Wenstrup, and R. A. Pastore, "Program C₅ - Direct and Adjoint Bare Reactor Program - Multigroup Constant Generator," XDC-59-6-220 (May 1959).

P. G. Fischer, "Multi-group, Multi-region, One Space Dimension Neutron Diffusion Theory Calculation - Program F-N (ANP 308)," XDC-60-3-68 (January 1960).

J. E. MacDonald, "Conical and Plane Surface Scattering Program - SURF," GEMP-582 (February 1968).

M. R. Edwards, "The UPDONC Code," GEMP-456 (December 1966).

W. E. Edwards, "Shield Kernel-Diffusion Analysis," GEMP-599 (March 1968).

J. R. Lilley, "Computer Code HFN - Multigroup, Multiregion Neutron Diffusion Theory in One Space Dimension," HW-71545 (November 1961).

11. CONTENTS OF CODE PACKAGE

The package contains the following items:

- a. the referenced documents,
- b. a reel of magnetic tape on which is written in separate files as indicated:

CCC-124A

12 files: the source card deck for each code in the package, BCD input data for a sample problem for each code, a BCD library of nuclear data and a BCD library of gamma-ray production cross sections.

CCC-124B

31 files: includes above items and, in addition, BCD output listings from running each of the sample problems.

12. HOW TO OBTAIN PACKAGE

Inquiries or requests for the code package may be mailed to

CODES COORDINATOR
Radiation Shielding Information Center
Oak Ridge National Laboratory
Post Office Box X
Oak Ridge, Tennessee 37830

or telephoned to

Area Code 615; 483-8611, extension 3-6944, or to
FTS xx-615-483-6944.

Persons requesting the code package should send a reel of magnetic tape to the above address.

13. DATE OF ABSTRACT

August 1971.

RSIC CODE PACKAGE CCC-125

1. NAME AND TITLE OF CODE

RSAC: Radiological Safety Analysis Code.

AUXILIARY ROUTINE

CURIE: Fission Product Inventory Code.

2. CONTRIBUTOR

Phillips Petroleum Company, Atomic Energy Division, Idaho Falls, Idaho.

3. CODING LANGUAGE AND COMPUTER

MAP; IBM 7044.

4. NATURE OF PROBLEM SOLVED

RSAC was designed to compute the potential radiological doses resulting from either a continuous or instantaneous release of radioactive fission products to the atmosphere. The doses considered are cloud gamma rays, deposition gamma rays, ingestion, and inhalation and are based upon the release of up to 200 isotopes, the relative abundance of each isotope being dependent upon the operating history of the nuclear reactor under consideration.

5. METHOD OF SOLUTION

The fission product inventory is computed by the CURIE portion of RSAC for any combination of steady-state, transient, and/or cyclic operations, and for refueling. The rate of fission product release, dispersion, and depletion is described by the various equations making up the concentration function. The diffusion equation, the leakage rate function, and the cloud depletion equations are integrated numerically, as are the dose equations.

6. RESTRICTIONS OR LIMITATIONS

The following limits are noted: maximum number of decay chains which can be considered, 100; maximum number of isotopes, 450 (for deposition gamma dose, 330); maximum number of downwind and crosswind positions, 25 each; the effects of neutron activation are neglected.

7. TYPICAL RUNNING TIME

No study of typical running time has been made.

8. COMPUTER HARDWARE REQUIREMENTS

The code was designed for the IBM 7040 and is also operable on the IBM 7044. Only standard hardware is used.

9. COMPUTER SOFTWARE REQUIREMENTS

Standard software for the IBM 7040/7044 may be used. Non-standard library routine, EXTERN, is packaged.

10. REFERENCES

R. L. Coates and N. R. Horton, "RSAC - A Radiological Safety Analysis Computer Program," IDO-17151 (May 1966).

11. CONTENTS OF CODE PACKAGE

The package contains the following items:

- a. the referenced document,
- b. a reel of magnetic tape on which is written in 5 separate files: the source card deck and the object deck, for the main program and for each auxiliary routine.

12. HOW TO OBTAIN PACKAGE

Inquiries or requests for the code package may be mailed to

CODES COORDINATOR
Radiation Shielding Information Center
Oak Ridge National Laboratory
Post Office Box X
Oak Ridge, Tennessee 37830

or telephoned to

Area Code 615; 483-8611, extension 3-6944, or to
FTS xx-615-483-6944.

Persons requesting the package should send a reel of magnetic tape to
the above address.

13. DATE OF ABSTRACT

August 1971.

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RSIC CODE PACKAGE CCC-126

1. NAME AND TITLE OF CODE

ASOP: Multigroup One-Dimensional Discrete Ordinates Transport Code for Shield Optimization.

The ASOP code is also known as the "ANISN Shield Optimization Program.

2. CONTRIBUTOR

Neutron Physics Division, Oak Ridge National Laboratory, Oak Ridge, Tennessee.

3. CODING LANGUAGE AND COMPUTER

FORTRAN IV; IBM 360.

4. NATURE OF PROBLEM SOLVED

ASOP is a shield optimization program based on the one-dimensional discrete ordinates transport code ANISN (CCC-82). It has been used to design optimum shields for space applications of SNAP zirconium-hydride-uranium-fueled reactors and uranium-oxide fueled thermionic reactors and to design beam stops for the ORELA facility.

5. METHOD OF SOLUTION

ASOP generates coefficients of linear equations describing the logarithm of the dose and dose-weight derivatives as functions of position from data obtained in an automated sequence of ANISN calculations. With the dose constrained to a design value and all dose-weight derivatives required to be equal, the linear equations may be solved for a new set of shield dimensions. Since changes in the shield dimensions may cause the linear functions to change, the entire procedure is repeated until convergence is obtained.

The detailed calculations of the radiation transport through shield configurations for every step in the procedure distinguish

ASOP from other shield optimization programs which rely on multiple component sources and attenuation coefficients to describe the transport.

6. RESTRICTIONS OR LIMITATIONS

Problem size is limited only by machine size.

7. TYPICAL RUNNING TIME

Estimated running time for the packaged sample problem on the IBM 360/91: 30 minutes.

8. COMPUTER HARDWARE REQUIREMENTS

The code was designed for the IBM 360. Standard I-O and a maximum of seven tapes or direct access devices are used.

9. COMPUTER SOFTWARE REQUIREMENTS

The code is operable on the IBM 360/75/91 Operating System using OS-360 FORTRAN H compiler. The overlay feature is utilized and 434K of memory is required.

Subroutine ITIME is supplied on master tape in HEX.

10. REFERENCES

Ward W. Engle, Jr., "A Users Manual for ASOP, ANISN Shield Optimization Program," CTC-INF-941 (September 1969).

11. CONTENTS OF CODE PACKAGE

The package contains the following items:

- a. the referenced document,
- b. a reel of magnetic tape on which is written in 6 files: the BCD source card deck, a nonstandard subroutine deck, BCD input for a sample problem and BCD output from running the problem.

12. HOW TO OBTAIN PACKAGE

Inquiries or requests for the code package may be mailed to

CODES COORDINATOR
Radiation Shielding Information Center
Oak Ridge National Laboratory
Post Office Box X
Oak Ridge, Tennessee 37830

or telephoned to

Area Code 615; 483-8611, extension 3-6944, or to
FTS xx-615-483-6944.

Persons requesting the code package should send a reel of magnetic tape to the above address.

13. DATE OF ABSTRACT

August 1971.

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RSIC CODE PACKAGE CCC-127

1. NAME AND TITLE OF CODE

MORSE: A General Purpose Monte Carlo Multigroup Neutron and
and Gamma-Ray Transport Code.

AUXILIARY ROUTINES

SAMBO: Collision Analysis Code.

PICTURE: GEOM Input Diagnostic Code.

MORSE was originally programmed for the CDC 1604 and was later modified and extended for the IBM 360. The original version was packaged (A, January 1970) but is not kept up-to-date by the originators.

2. CONTRIBUTORS

Neutron Physics Division, Oak Ridge National Laboratory, Oak Ridge, Tennessee.

Sandia Corporation, Albuquerque, New Mexico (C Version).

3. CODING LANGUAGE AND COMPUTER

(A) FORTRAN 63 and CODAP; CDC 1604

(B) FORTRAN IV and Assembly Language; IBM 360/75/91

(C) FORTRAN IV and COMPASS; CDC 6600

(D) FORTRAN IV; UNIVAC 1108

4. NATURE OF PROBLEM SOLVED

The MORSE code is a multipurpose neutron and gamma-ray transport Monte Carlo Code. It has been designed as a tool for solving most shielding problems. Through the use of multigroup cross sections, the solution of neutron, gamma-ray, or coupled neutron-gamma-ray problems may be obtained in either the forward or adjoint mode. Time dependence for both shielding and criticality problems is provided. General three-dimensional geometry, as well as specialized one-dimensional geometry descriptions, may be used with an albedo

option available at any material surface. Isotropic or anisotropic scattering up to a P_{16} expansion of the angular distribution is allowed.

5. METHOD OF SOLUTION

Monte Carlo methods are used to solve the forward and the adjoint transport equations. Quantities of interest are then obtained by summing the contributions over all collisions, and frequently over most of phase space.

Standard multigroup cross sections such as those used in discrete ordinates codes may be used as input; either CCC-82/ANISN, CCC-42/DTF-IV, or CCC-89/DOT cross section formats are acceptable. Anisotropic scattering is treated for each group-to-group transfer by utilizing a generalized Gaussian quadrature technique.

The MORSE code is organized into functional modules with simplified interfaces such that new modules may be incorporated with reasonable ease. The modules are (1) random walk, (2) cross section, (3) geometry, (4) analysis, and (5) diagnostic.

While the basic MORSE code assumes the analysis module is user-written, a general analysis package, SAMBO, has been developed. SAMBO handles most of the drudgery associated with the analysis of random walks and minimizes the amount of user-written coding. An arbitrary number of detectors, energy-dependent response functions, energy bins, time bins, and angle bins are allowed. Analysis is divided for each detector as follows: uncollided and total response, fluence versus energy, time-dependent response, fluence versus time and energy, and fluence versus angle and energy. Each of these quantities is listed as output. The diagnostic module provides an easy means of printing out, in useful form, the information in the various labelled COMMON and any part of blank COMMON. This module is very useful to debug a problem and to gain further insight into the physics of the random walk.

6. RESTRICTIONS OR LIMITATIONS

None noted.

7. TYPICAL RUNNING TIME

No effort has been made to estimate typical running time.

Estimated running times for the packaged sample problems on the IBM 360/91: (1) 0.42 minutes and (2) 1.16 minutes.

8. COMPUTER HARDWARE REQUIREMENTS

MORSE was originally written for the CDC 1604, was converted to the IBM 360, revised considerably, and the latter version is also operable on the CDC 6600. Standard hardware equipment is used in each version, with standard I-O devices required.

9. COMPUTER SOFTWARE REQUIREMENTS

Standard software for each of the versions may be used. Any non-standard library routines which were used have been included in the code package, or a full description is given for the user's information.

The OS-360 FORTRAN H compiler was used for B version on the IBM 360/75/91 Operating System. In the GO Step: problem 1 used 300K and problem 2, 306K of storage.

10. REFERENCES

E. A. Straker, P. N. Stevens, D. C. Irving, and V. R. Cain, "The MORSE Code - A Multigroup Neutron and Gamma-Ray Monte Carlo Transport Code," ORNL-4585 (September 1970).

V. R. Cain, "SAMBO, A Collision Analysis Package for Monte Carlo Doses," ORNL-TM-3203 (September 1970).

D. C. Irving and G. W. Morrison, "PICTURE: An Aid in Debugging GEOM Input Data," ORNL-TM-2892 (May 1970).

D. C. Irving, "The Adjoint Boltzmann Equation and Its Simulation by Monte Carlo," ORNL-TM-2879 (May 1970).

11. CONTENTS OF CODE PACKAGE

The package contains the following items:

- a. the referenced documents,
- b. a reel of magnetic tape on which is written in several separate files: the BCD source card decks for all codes in the package, BCD input for sample problems for each code, and the BCD output listing from running each of the problems.

12. HOW TO OBTAIN PACKAGE

Inquiries or requests for the code package may be mailed to

CODES COORDINATOR
Radiation Shielding Information Center
Oak Ridge National Laboratory
Post Office Box X
Oak Ridge, Tennessee 37830

or telephoned to

Area Code 615; 483-8611, extension 3-6944, or to
FTS xx-615-483-6944.

Persons requesting the package should send a reel of magnetic tape to the above address.

13. DATE OF ABSTRACT

August 1971.

RSIC CODE PACKAGE CCC-128

1. NAME AND TITLE OF CODE

O6R: A General-Purpose Monte Carlo Transport Code System.

PROTOTYPES

O6R-O5RRED-DUMCOLL: Neutron Transport, Analysis, and Edit Codes (Version A).

O6R-ACT1FK: Simultaneous Neutron Transport and Analysis Code (Version B).

AUXILIARY ROUTINES

XSECT: Cross Section Handling Code Package - O5R-type Format.

COD678: Cross Section Handling Code Package - ENDF to O5R-type Format.

CODE 5: Cross Section Arithmetic Code.

NUSECT: Cross Section Handling Code Package - ENDF Format.

EDIT: Data Converter - ENDF Format - BCD to Binary Mode.

PICTURE: Geometry Input Diagnostic Code.

The O6R code system represents later developments of the Oak Ridge Monte Carlo Code series, which began with O5R (CCC-17). Version A, O6R-O5RRED-DUMCOLL prototype above, was written for the CDC 1604 and packaged in February 1970.

2. CONTRIBUTOR

Neutron Physics Division, Oak Ridge National Laboratory, Oak Ridge, Tennessee.

3. CODING LANGUAGE AND COMPUTER

CCC-128A: FORTRAN 63 and CODAP; CDC 1604.

CCC-128B: FORTRAN IV and machine language; IBM 360/75/91.

4. NATURE OF PROBLEM SOLVED

The O6R code system consists of major modifications and extensions made to CCC-17/05R, designed to calculate any quantity related to neutron transport in reactor or shielding problems. The O6R code system solves the integral Boltzmann equation in general phase space for the collision density of neutrons in the system. By using the appropriate analysis routines, quantities of interest such as the flux, absorption rate, neutron lifetime, and slowing down density can be estimated. A wide variety of problems in the areas of radiation shielding, reactor analysis and design, criticality safety, and neutron scattering experiments can be solved with the O6R system. To demonstrate the applicability of O6R to some of these areas a series of test problems have been included in the package.

The features of prototype O6R-O5RRED-DUMCOLL (A) are demonstrated through several packaged sample problems. These include a simple one-velocity k calculation with isotropic lab scattering in a sphere with several variations illustrating cross section handling, geometry features, source treatment and other options. The problem is then modified to test the albedo option. More complicated problems (a total of 7) test Russian roulette, splitting, path stretching, history tape generation, and several cross section options. A special-purpose analysis is included for this version, ANAL.

The O6R-ACT1FK prototype permits simultaneous neutron transport and analysis, the specific features slanted toward the application to calculations of shields rather than reactors. It embodies all the features of O5R and ACT1FK-analysis codes with substantial improvements.

5. METHOD OF SOLUTION

O6R solves the integral form of the Boltzmann equation by the Monte Carlo method. Both version A and B of O6R are revisions and updates of the CCC-17/05R General Purpose Monte Carlo code, and they employ the same general method of solution. However, many new features have been incorporated which improve the performance of the code.

Features Common to Version A & B

Rather than having cross-section information for only one supergroup in core, O6R allows information for as many supergroups as can be accommodated by computer memory. In addition, in-core analysis of neutron histories can be performed. Version A does this by means of a user-written subroutine ANAL while version B utilizes O6R-ACT1FK routines.

A parameter, AGE, is computed and stored for use in the calculation of time-dependent problems.

An option is available which allows a medium to be treated as an albedo medium. A user routine is called to pick a new energy and direction.

Path stretching, a variance reduction technique equivalent to the exponential transform, has been made a standard feature.

Version A uses KINNY and version B uses NONLAS, both user-written routines, to incorporate inelastic scatter.

ENDF/B formatted cross-section data can be used to generate cross section input to O6R. Version A utilizes COD678 for this purpose while version B employs NUSECT. Both versions also utilize the O5R library, which has not been revised in several years and is probably out of date.

Special Features of Each Version

Gamma-ray generation during neutron transport is available with version B, the IBM 360 O6R code. Subroutine GAMMA generates a tape containing appropriate parameters. The user supplies subroutines SETGAM and GAMGEN.

Variable supergroup energy boundaries, rather than the fixed boundaries of O5R, are available with version A, the CDC 1604 O6R code. Also, the user may specify an energy for each scatterer below which scattering is isotropic in the center of mass system. This by-passes the anisotropic treatment and saves time.

Also with version A, k_{eff} may be calculated. In addition, the treatment of fission calculation was changed, allowing the user to select renormalization as an option. The method of renormalization was improved.

6. RESTRICTIONS AND LIMITATIONS

The O6R system is presently limited to 16 media in the cross section representation. This can be increased by changing a dimension statement. The order of the scattering is limited to a P_{16} Legendre expansion.

7. TYPICAL RUNNING TIME

Running time for O6R on various types of problems varies extremely from a few seconds up to an hour on the IBM 360/91.

Estimated IBM 360 CPU time of the packaged sample problems (B version):

EDIT,	8.43 seconds
NUSECT,	44.50 seconds
O6R,	51.50 seconds
PICTURE,	8.44 seconds

8. COMPUTER HARDWARE REQUIREMENTS

Uses standard I-O and a maximum of 3 tapes or direct access devices on either the IBM 360 or the CDC 1604.

Maximum region size used is 474K.

9. COMPUTER SOFTWARE REQUIREMENTS

The A version may be compiled and executed on the FORTRAN 63 COOP Monitor or other compatible system.

The B version is operable on the IBM 360/91 operating system using the OS-360 FORTRAN H (Level 18) compiler.

A cross section library is required as input to the EDIT code. The cross sections for H, C₁₂ and O₁₆, used in the sample problems, are included in this code package.

10. REFERENCES

C. L. Thompson and E. A. Straker, "O6R-ACT1FK, Monte Carlo Neutron Transport Code," ORNL-TM-3050 (August 1969).

R. R. Coveyou, J. G. Sullivan, H. P. Carter, D. C. Irving, R. M. Freestone, Jr., and F. B. K. Kam, "O5R, A General-Purpose Monte Carlo Neutron Transport Code," ORNL-3622 (February 1965).

F. B. K. Kam and K. D. Franz, "ACT1FK, A General Analysis Code for O5R," ORNL-3856 (September 1966).

E. A. Straker and V. R. Cain, "A Note on Subroutine CEASE (O5R:EVAP), A Modification of EVAP in O5R," ORNL-TM-1552 (August 1966).

C. E. Burgart, "Treatment of ${}^6\text{Li}(n,dn)\alpha$, ${}^7\text{Li}(n,tn)\alpha$ Reactions in O6R Random Walks and Analysis," ORNL-TM-3051 (April 1970).

D. C. Irving and G. W. Morrison, "PICTURE: An Aid in Debugging GEOM Input Data," ORNL-TM-2892 (May 1970).

Informal Notes - Description of the O6R-ACT1FK Sample Problem and of EDIT.

D. C. Irving, "Description of the CDC-1604 Version of the O6R Neutron Monte Carlo Transport Code," ORNL-TM-3458 (May 1971).

11. CONTENTS OF CODE PACKAGE

The package contains the following items:

- a. the referenced documents,
- b. a reel of magnetic tape for each code version on which is written in several files: the source card decks, BCD input for sample problems, and BCD output listings from running the sample problems.

12. HOW TO OBTAIN PACKAGE

Inquiries or requests for the code package may be mailed to

CODES COORDINATOR
Radiation Shielding Information Center
Oak Ridge National Laboratory
Post Office Box X
Oak Ridge, Tennessee 37830

or telephoned to

Area Code 615; 483-8611, extension 3-6944, or to
FTS xx-615-483-6944.

Persons requesting the package should send a reel of magnetic tape to the above address, specifying which prototype and how the tape should be written.

13. DATE OF ABSTRACT

August 1971.

RSIC CODE PACKAGE CCC-129

1. NAME AND TITLE OF CODE

TWOTRAN: Multigroup Two-Dimensional Discrete Ordinates Transport General Geometry Code.

TWOTRAN was originally programmed for the UNIVAC 1108 and was modified for the CDC 6600. The latter version was converted for the IBM 360. The code package is also available from the Argonne Code Center and from the ENEA Computer Programme Library, Abstract 358.

2. CONTRIBUTORS

Gulf General Atomic, San Diego, California.

Los Alamos Scientific Laboratory, T Division, Los Alamos, New Mexico.

Argonne National Laboratory, Argonne, Illinois.

3. CODING LANGUAGE AND COMPUTER

CCC-129A: FORTRAN IV; CDC 6600.

CCC-129B: FORTRAN IV; IBM 360/75/91 (x,y only).

CCC-129C: FORTRAN IV; CDC 6600 (Extended Core Storage version).

In all versions, FORTRAN IV is used with a small amount of mixed integer-floating arithmetic, generalized subscripting, encode statements, and minor use of 10 H Hollerith formats.

4. NATURE OF PROBLEM SOLVED

Two dimensional particle transport problems in X-Y, R-Z, and R- θ geometries. Both direct and adjoint, homogeneous (k_{eff} or parametric eigenvalue searches) or inhomogeneous time-independent problems are solved subject to vacuum, reflective, white, periodic, or input specification of boundary flux conditions. Both anisotropic inhomogeneous problems and general anisotropic scattering problems are treated. Arbitrary numbers of groups of up or down scattering are allowed.

5. METHOD OF SOLUTION

Energy dependence is treated by the multigroup approximation and the angular dependence by a discrete ordinates approximation. Space dependence is approximated by the diamond difference scheme with a set-to-zero negative flux control. Anisotropic scattering and anisotropic inhomogeneous sources are represented by finite spherical harmonics expansions. Within-group iterations, upscattering iterations, k_{eff} iterations, and eigenvalue search iterations are accelerated by a coarse-mesh particle rebalancing algorithm.

Unusual features of the program include: coarse-mesh convergence acceleration; coarse-mesh spatial and angular organization to permit larger problems; general anisotropic scattering and inhomogeneous source option; input specification of top, bottom, or right boundary fluxes; built-in discrete ordinates constants (S_2, S_4, \dots, S_{16}); diamond difference scheme with negative flux fix up; detailed edit provisions; pointwise cross section density variation option; overlay program organization; and general dump and restart options.

6. RESTRICTIONS OR LIMITATIONS

The variable dimensioning capability of Fortran-IV is used so that any combination of problem parameters leading to a blank common vector length less than LENXCA can be used. For a 65K machine LENXCA can be greater than 35,000, depending on local system requirements. With a few exceptions, only within group problem data are stored in fast memory and data for all other groups are stored in auxiliary bulk memory such as extended core storage.

7. TYPICAL RUNNING TIME

The author of the program cites the following problems and estimated running times on the CDC 6600: (X,Y), seven-group, linear anisotropic, upscattering 40×40 , S_4 k_{eff} calculation - 64.73 minutes; (R, θ), nine-group, linear anisotropic, upscattering 42×10 , S_4 k_{eff} calculation - 12.83 minutes; and (R,Z), ten-group, P_2 scattering, 21×20 , S_8 , source problem - 21.34 minutes.

Estimated running times of the packaged 8 sample problems on the IBM 360/91: GO step CPU time: 7 minutes.

8. COMPUTER HARDWARE REQUIREMENTS

Five output units (disks, drums, or tapes) in addition to 3 system input/output units are required. Version C requires a CDC Extended Core Storage unit or a large bulk memory. Disks, drums or tapes can be substituted for this requirement. Region used in Version B GO Step: 442K.

9. COMPUTER SOFTWARE REQUIREMENTS

The CDC versions have been run on the CDC 6600 Scope 3.1 (LASL modified). This TWOTRAN uses a local cross section library (subroutine LAXS), an algorithm to reduce core storage (subroutine REDUCE), a special decimal dump for CDC mode errors, and special plotting routines for contour and projective flux displays. These features may be removed simply.

The IBM 360 version is operable on the IBM 360/75/91 Operating System using the OS 360 FORTRAN H compiler. A dummy CLOCK subroutine was used.

10. REFERENCES

K. D. Lathrop and F. W. Brinkley, "Theory and Use of the General Geometry TWOTRAN Program," LA-4432 (1970).

K. D. Lathrop, "User's Guide for the TWOTRAN (x,y) Program," LA-4058 (1969).

K. D. Lathrop, "TWOTRAN, A Fortran Program for Two Dimensional Transport," GA-8747 (1968).

M. Shapiro, "2DF," Reactor Code Abstract 173, ANL-7411 (1968).

11. CONTENTS OF CODE PACKAGE

The package contains the following items:

- a. the referenced documents,
- b. a reel of magnetic tape for each code version on which is written in several files: the source card decks, BCD input for sample problems and BCD output listings from running the problems.

12. HOW TO OBTAIN PACKAGE

Inquiries or requests for the code package may be mailed to
CODES COORDINATOR
Radiation Shielding Information Center
Oak Ridge National Laboratory
Post Office Box X
Oak Ridge, Tennessee 37830

or telephoned to

Area Code 615; 483-8611, extension 3-6944, or to
FTS xx-615-483-6944.

Persons requesting the package should send a reel of magnetic tape to the above address.

13. DATE OF ABSTRACT

August 1971.

RSIC CODE PACKAGE CCC-130

1. NAME AND TITLE OF CODE

DTF69: One Dimensional Multigroup Photon Transport Discrete Ordinates Code.

AUXILIARY ROUTINE

GAMLEG69: Cross Section Generator.

DTF69 is a modified version of DTF-IV (CCC-42), written at LASL for the solution of neutron transport problems. GAMLEG69 is an extensive modification of the LASL-written GAMLEG (also packaged in CCC-42).

2. CONTRIBUTOR

Sandia Laboratories, Albuquerque, New Mexico.

3. CODING LANGUAGE AND COMPUTER

CCC-130A: FORTRAN IV; CDC 6600.

CCC-130B: FORTRAN IV; CDC 3600.

4. NATURE OF PROBLEM SOLVED

DTF 69 applies the method of discrete ordinates to the gamma-ray transport problem. The major addition required to existing discrete ordinates programs used in neutron transport is a device for processing gamma-ray cross section data. This is provided in a subroutine called GAMLEG 69.

5. METHOD OF SOLUTION

The main body transport program DTF 69 is an evolutionary step beyond DTF IV which is well documented. Apart from its ability to deal with gamma rays, DTF 69 has the following distinguishing features.

1. Time dependence is handled by Laplace transform with a newly developed inversion technique.

2. A convergence acceleration scheme using Chebychev polynomials is included.
3. A buckling term to correct for transverse leakage is incorporated.
4. Provision is made for fluorescent and annihilation radiation. (These are particle production processes and ordinarily would introduce an essential complication.)
5. The source options available are a volume distributed isotropic source, a point isotropic source in the center of a spherically symmetric system, and an angle dependent current falling upon the right boundary.

The gamma ray processing subroutine GAMLEG 69 integrates basic gamma-ray data to produce for each energy group and material the energy deposition, total, absorption (photo electric plus pair production), secondary production cross sections (fluorescence and pair production), and group transfer cross sections representing the Klein-Nishina scattering process.

6. RESTRICTIONS OR LIMITATIONS

None noted.

7. TYPICAL RUNNING TIME

No study has been made by RSIC as to typical running time.

8. COMPUTER HARDWARE REQUIREMENTS

The codes were designed for the CDC 6600 and are also operable on the CDC 3600. Standard I-O equipment is used.

9. COMPUTER SOFTWARE REQUIREMENTS

The codes will compile and execute on the standard CDC FORTRAN IV Monitor System.

10. REFERENCES

James H. Renken and Kenneth G. Adams, "An Improved Capability for Solution of Photon Transport Problems by the Method of Discrete Ordinates," SC-RR-69-739 (December 1969).

James H. Renken and Kenneth G. Adams, "Application of the Method of Discrete Ordinates to Photon Transport Calculations," SC-RR-67-419 (June 1967).

K. G. Adams, "DTFALL: A Program to Run Photon Transport Problems by the Method of Discrete Ordinates," SC-RR-68-712 (October 1968).

11. CONTENTS OF CODE PACKAGE

The package contains the following items:

- a. the referenced documents,
- b. a reel of magnetic tape on which is written in several files: the source card decks for the two programs, a library of cross section data in BCD, input for sample problems, and an output listing from running the problems.

12. HOW TO OBTAIN PACKAGE

Inquiries or requests for the code package may be mailed to
CODES COORDINATOR
Radiation Shielding Information Center
Oak Ridge National Laboratory
Post Office Box X
Oak Ridge, Tennessee 37830

or telephoned to

Area Code 615; 483-8611, extension 3-6944, or to
FTS xx-615-483-6944.

Persons requesting the package should send a reel of magnetic tape to the above address.

13. DATE OF ABSTRACT

August 1971.

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RSIC CODE PACKAGE CCC-131

1. NAME AND TITLE OF CODE

ANTE 2: Adjoint Monte Carlo Time-Dependent Neutron Transport Code in Combinatorial Geometry.

AUXILIARY ROUTINE

PAT: Neutron Element Data Tape Generator (from ENDF format).

ADCROS: Adjoint Data Generator.

GPROC: Geometry Data Processor.

MAS: Monte Carlo Calculation.

Two versions are packaged; (A) ANTE 2 for the CDC 6600 and (B) ANTE-BELLM for the IBM 360.

2. CONTRIBUTORS

Mathematical Application Group, Inc. (MAGI), White Plains, New York.

Bell Telephone Laboratories, Whippany, New Jersey.

3. CODING LANGUAGE AND COMPUTER

CCC-131A: FORTRAN IV; CDC 6600.

CCC-131B: FORTRAN IV; IBM 360/75/91.

4. NATURE OF PROBLEM SOLVED

Given a finite neutron transmitting medium (hereafter called the object or transmission object) and given a neutron detector imbedded in that medium, what would be the response of the detector to an arbitrary neutron flux field in which the object is immersed? ANTE 2 computes an importance weighting function over a surface which contains the object. Thus any arbitrary flux field may be weighted by this function to determine the effect this field would produce upon the enclosed detector.

5. METHOD OF SOLUTION

Whereas conventional neutron Monte Carlo programs seek to generate flux distributions (which may be called the contravariant aspect) from specified sources and are usable in estimating effects upon arbitrary (within practical limits) detectors, this program deals with the covariant aspect and seeks to generate importance or adjoint distributions for definite detectors usable in estimating effects from arbitrary (within practical limits) sources.

There appear to be four principal features of the program.

1. Geometry - the following objects are building blocks of the geometry: sphere, right elliptical cylinder, truncated right angle cone, ellipsoid, or convex polyhedron of 4, 5, or 6 sides. These building blocks may be combined in arbitrary fashion - unions or intersections. An elegant Boolean notation is used to describe the combinations.
2. Scattering kernels - differential cross section data must be analyzed to construct probability distributions of the phase coordinates of a particle prior to a collision which produces some definite exit phase coordinates. This data preparation may be viewed as the heart of the calculation and may be usefully adapted to other Monte Carlo calculations of adjoints.
3. Standard Monte Carlo techniques are used in particle tracking and construction of histories.
4. Scoring - The transmission object is considered to be completely enclosed by a scoring surface which must be a rectangular parallelepiped or a sphere. Scoring bins represent meshes of time, energy, polar, and azimuthal angles. The surfaces of the rectangular parallelepiped may be further subdivided to provide spatial bins. In the case of spherical enclosure, no provision is made for surface subdivision or for recording azimuthal angles of escape. Hence, only spherically symmetric problems can be addressed with this option.

Included in the package are the following routines:

PAT abstracts required cross section data from ENDF/B files;

ADCROS receives cross section data from PAT, operates on it to provide scattering kernels and other basic data required for adjoint calculation and makes an ADCROS tape;

GPROC processes geometric description provided in input and places results on ADCROS tape; and

MAS performs the calculation after the above preparations are made.

6. RESTRICTIONS OR LIMITATIONS

None noted.

7. TYPICAL RUNNING TIME

Estimated running time for the sample problem on the IBM 360/75:

PAT, 1 minute

ADCROS, 5 minutes, and

MAS, 7 seconds.

8. COMPUTER HARDWARE REQUIREMENTS

The codes are operable on the CDC 6600 or the IBM 360/75/91 with standard I-O, and a maximum of 9 tape units or direct access devices. Approximately 400K is required in the GO Step.

9. COMPUTER SOFTWARE REQUIREMENTS

ANTE 2 was designed for the FORTRAN IV CDC 6600 Operating System. It is also operable on the IBM 360/75/91 Operating System using OS 360 Fortran H compiler.

10. REFERENCES

M. O. Cohen, "ANTE 2 - A FORTRAN Computer Code for the Solution of the Adjoint Neutron Transport Equation by the Monte Carlo Technique," DASA-2396 (January 1970).

M. O. Cohen and W. Guber, "ANTE-BELLM - A Computer Program for the Solution of the Adjoint Neutron Time Dependent Transport Equation by the Monte Carlo Technique," MR-7003 (March 1970).

Walter Guber, "The Combinatorial Geometry Technique for the Description and Computer Processing of Complex Three Dimensional Objects," MR-7004/2 (March 1970).

11. CONTENTS OF CODE PACKAGE

The package contains the following items:

- a. the referenced documents,
- b. a reel of magnetic tape on which is written in several files: the source card decks, BCD input for a sample problem, and a BCD output listing from running the sample problem for each version of the code. A library of cross sections for use in the sample problem has been especially prepared from ENDF/B-Version I data files and is included in the package.

12. HOW TO OBTAIN PACKAGE

Inquiries or requests for the code package may be mailed to

CODES COORDINATOR
Radiation Shielding Information Center
Oak Ridge National Laboratory
Post Office Box X
Oak Ridge, Tennessee 37830

or telephoned to

Area Code 615; 483-8611, extension 3-6944, or to
FTS xx-615-483-6944.

Persons requesting the code should send a reel of magnetic tape to the above address.

13. DATE OF ABSTRACT

August 1971.

RSIC CODE PACKAGE CCC-132

1. NAME AND TITLE OF CODE

ATTOW: Multigroup Two-Dimensional Removal-Diffusion (Spinney Method) Shielding Code.

AUXILIARY ROUTINE

ATTOWLIB: Cross Section Handling Code and Library.

ATTOW is also available in the ENEA Computer Programme Library, listed as Abstract ENEA 186.

2. CONTRIBUTOR

The Reactor Group, HQ, UKAEA, Risley, Warrington, Lancs.

The code was placed in the RSIC collection through the services of the ENEA Computer Programme Library, Ispra (Varese), Italy.

3. CODING LANGUAGE AND COMPUTER

FORTRAN IV and MAP; IBM 7090 and 7094.

4. NATURE OF PROBLEM SOLVED

ATTOW is designed to solve the multigroup diffusion equations, reinforced where necessary by removal theory, for non-multiplying media in two dimensions of either cylindrical or rectangular coordinates. It is used for calculating neutron flux distributions, activations of reactor materials and of test samples, generation of heat in the shields, neutron currents and other quantities.

ATTOW was thus designed specifically as a 2-dimensional shielding program with special facilities for source problems. The one-dimensional problem may be solved by COMPRASH (CCC-72).

5. METHOD OF SOLUTION

The Spinney removal-diffusion method of calculating neutron transport is described in the abstract of CCC-72/COMPRASH.

In ATTOW, the finite difference expression is solved by iteration using accelerated successive line over-relaxation. The removal source expression is handled by Gaussian integration using a stored table of zeros and weights.

6. RESTRICTIONS AND LIMITATIONS

There is no restriction on the number of neutron groups. One hundred differently valued source areas are allowed, and 10,000 mesh points.

7. TYPICAL RUNNING TIME

Typical running time depends on the individual case, with the following cited as reasonable estimates: one minute per group for 1,000 mesh points; 15-30 minutes per group for 10,000 mesh points.

8. COMPUTER HARDWARE REQUIREMENTS

The code was designed for the IBM 7090 with a 32K core and 12 tape units.

9. COMPUTER SOFTWARE REQUIREMENTS

The code may be compiled and executed on the FORTRAN IV IBJOB Monitor (Version 13) in the IBSYS Operating System on the IBM 7090 and 7094. Standard I-O, systems tapes and scratch tapes are assigned.

10. REFERENCE

W. D. Collier and G. C. Curtis, "ATTOW, A Two-Dimensional Shielding Program," TRG Report 1466 (R) (1967).

11. CONTENTS OF CODE PACKAGE

The package contains the following items:

- a. the referenced document,

- b. a reel of magnetic tape on which is written in several files: the source card decks, BCD input for a sample problem, a BCD library of cross section data, and an output listing from running the sample problem.

12. HOW TO OBTAIN PACKAGE

Inquiries or requests for the code package may be mailed to

CODES COORDINATOR
Radiation Shielding Information Center
Oak Ridge National Laboratory
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Oak Ridge, Tennessee 37830

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13. DATE OF ABSTRACT

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RSIC CODE PACKAGE CCC-133

1. NAME AND TITLE OF CODE

UNC-SAM-3: Monte Carlo Three-Dimensional Complex Geometry
Shielding Code System with ENDT.

AUXILIARY ROUTINE

ENDT: ENDF-to-UNC Format Data Generator (Element Data Tape).
TUNC: Overlay Control Code.
BAND: Problem Data Generator (Organized Data Tape).
BEDIT: Organized Data Tape Editor.
GEOM: Geometry Data Expander.
INPUTD: Problem Specification Input.
MONTE: Transport Code.
PEDIT: Monte Carlo Editor.

2. CONTRIBUTOR

United Nuclear Corporation, White Plains, New York.

3. CODING LANGUAGE AND COMPUTER

133A: FORTRAN IV; CDC 6600 (ENDT).
133B: FORTRAN 63; CDC 1604 (UNC-SAM-3).

4. NATURE OF PROBLEM SOLVED

UNC-SAM-3 follows a series of programs (CCC-13/ADONIS, CCC-81/
UNC-SAM and UNC-SAM-2) designed to calculate the solution to the
Boltzmann transport equation in complex three-dimensional geometry.
With ENDT, the program is able to make use of the ENDF Master Files
of cross section data (Version I).

5. METHOD OF SOLUTION

UNC-SAM-3 is a chained series of independent programs which
process cross section and geometry data, do the transport problem by
Monte Carlo methods, and edit the results. The methods are well

documented in CCC-81/UNC-SAM and are not repeated here.

6. RESTRICTIONS OR LIMITATIONS

Certain limits are imposed because of computer memory size. These are itemized in the input description for each program in the series. ENDT, as packaged, reads only Version I ENDF/B data.

7. TYPICAL RUNNING TIME

No study has been made by RSIC as to typical running time.

8. COMPUTER HARDWARE REQUIREMENTS

ENDT was designed for and is operable on the CDC 6600. The other codes run on the CDC 1604. A 32K core and from two to eight tape units are required.

9. COMPUTER SOFTWARE REQUIREMENTS

ENDT will compile and execute on the CDC 6600 FORTRAN IV Operating System. It produces a compatible tape for input to TUNC. The chained series of programs were designed for the CDC 1604 FORTRAN 63 COOP Monitor System.

10. REFERENCES

E. S. Troubetzkoy, "UNC-SAM-2: A FORTRAN Monte Carlo Program Treating Time-Dependent Neutron and Photon Transport Through Matter," UNC-5157 (September 1966).

E. S. Troubetzkoy, "Modification of UNC-SAM-2 to UNC-SAM-3," UNC-5157 (Supplement 1) (DASA-2338) (January 1970).

S. Kellman, "ENDT: A FORTRAN Program to Prepare Cross-Section Data for UNC-SAM-3 from ENDF/B Tapes," UNC-5243 (DASA-2337) (January 1970).

11. CONTENTS OF CODE PACKAGE

The package contains the following items:

- a. the referenced documents and
- b. a reel of magnetic tape on which is written in 10 files:
the source card decks for each code and input data for the packaged sample problems, including the ENDF formatted data required by the sample problem.

12. HOW TO OBTAIN PACKAGE

Inquiries or requests for the code package may be mailed to
CODES COORDINATOR
Radiation Shielding Information Center
Oak Ridge National Laboratory
Post Office Box X
Oak Ridge, Tennessee 37830

or telephoned to

Area Code 615; 483-8611, extension 3-6944, or to
FTS xx-615-483-6944.

Persons requesting the package should send a reel of magnetic tape to the above address.

13. DATE OF ABSTRACT

August 1971.

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RSIC CODE PACKAGE CCC-134

1. NAME AND TITLE OF CODE

2DBS: Two-Dimensional Multigroup Neutron Diffusion Shielding Code.

AUXILIARY ROUTINE

ANISN Library Routine.

2DBS Plot Routine.

2DBS is a revision and extension of 2DB, a fast reactor analysis code.

2. CONTRIBUTOR

Battelle Memorial Institute, Pacific Northwest Laboratories, Richland, Washington.

3. CODING LANGUAGE AND COMPUTER

FORTRAN IV; UNIVAC 1108.

4. NATURE OF PROBLEM SOLVED

2DBS is a flexible, two-dimensional (XY, RZ, $R\theta$ geometry) diffusion code designed for calculating the performance of in-vessel shields typical of reactors in the LMFBR program. The compactness of such cores often necessitates the 2-dimensional treatment and diffusion theory is deemed adequate because the exclusion of homogeneous media from the liquid metal environment relaxes the dependence of the total flux on the high energy scattering treatment where anisotropy is so important.

Included in the output is pointwise convergence information, reaction or activation rates; and from a coupled routine, Calcomp plots of the flux, power, fission rates, or spectra.

5. METHOD OF SOLUTION

Standard source-iteration techniques are used in 2DBS. Group rebalancing and successive over-relaxation with line inversion are used to accelerate convergence. Special features are built in to accelerate flux convergence, particularly for points far from the source. Cross section libraries in the ANISN (CCC-82) format may be used.

6. RESTRICTIONS OR LIMITATIONS

Since variable dimensioning is used, no simple bounds can be stated. The current version, however, is normally restricted to 50 energy groups in a 65K memory.

7. TYPICAL RUNNING TIME

No study of typical running time has been made by RSIC.

8. COMPUTER HARDWARE REQUIREMENTS

2DBS was designed for and is operable on the UNIVAC 1108. A 65K memory and three peripheral storage devices are required.

9. COMPUTER SOFTWARE REQUIREMENTS

The code was programmed in FORTRAN IV and may be compiled and executed on a compatible FORTRAN IV Monitor System. The packaged version has been run on the UNIVAC 1108.

10. REFERENCES

D. R. Marr, "A User's Manual for 2DBS, A Diffusion Theory Shielding Code," BNWL-1291 (February 1970).

W. W. Little, Jr. and R. W. Hardie, "2DB, A Two-Dimensional Diffusion-Burnup Code for Fast Reactor Analysis," BNWL-640 (January 1968).

11. CONTENTS OF CODE PACKAGE

The package contains the following items:

- a. the referenced documents, and
- b. a reel of magnetic tape on which is written in 8 separate files: the source card decks for each code in the package, input data for a sample problem, including the necessary cross sections, and a BCD output listing of the sample problem.

12. HOW TO OBTAIN PACKAGE

Inquiries or requests for the code package may be mailed to

CODES COORDINATOR
Radiation Shielding Information Center
Oak Ridge National Laboratory
Post Office Box X
Oak Ridge, Tennessee 37830

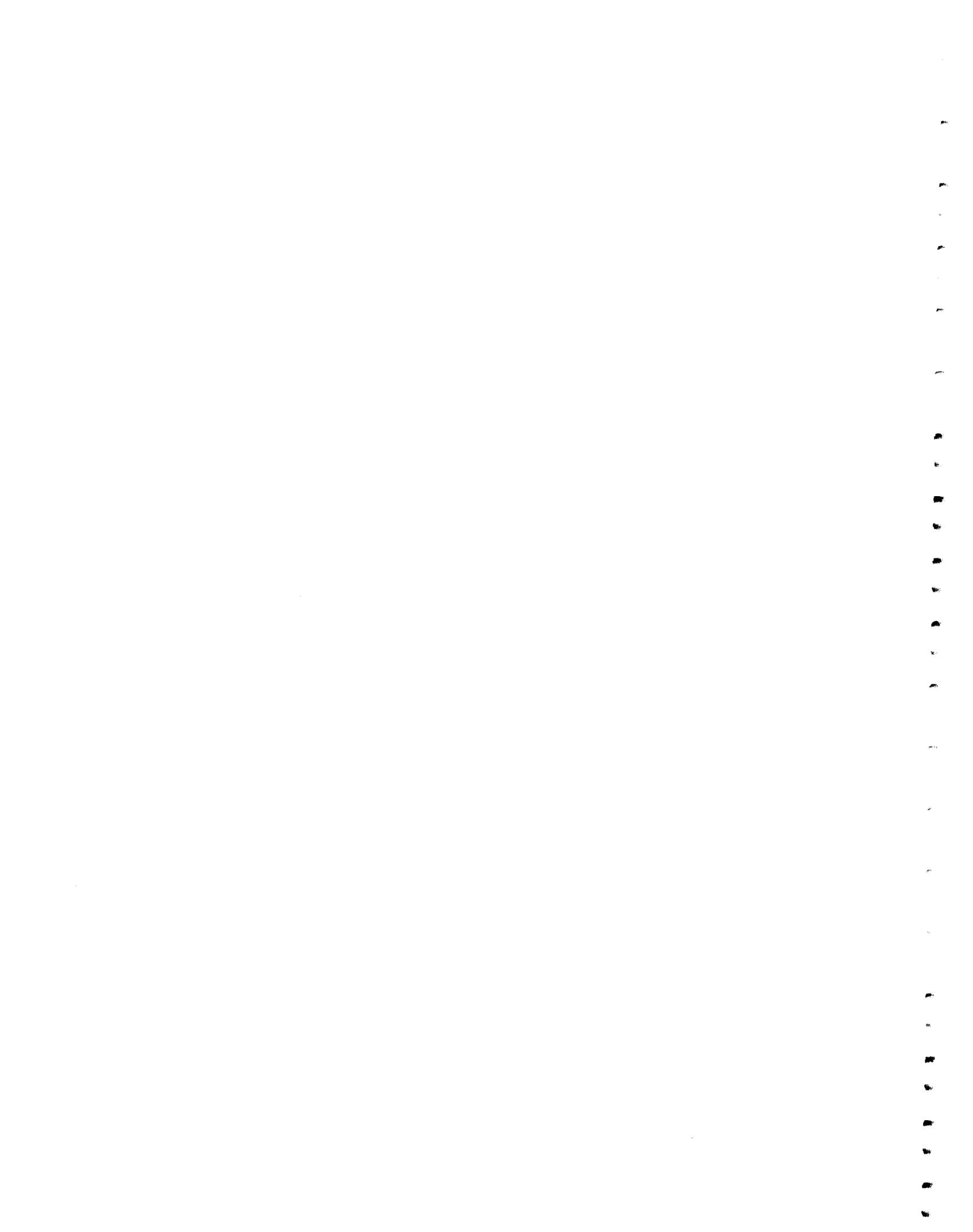
or telephoned to

Area Code 615; 483-8611, extension 3-6944, or to
FTS xx-615-483-6944.

Persons requesting the package should send a reel of magnetic tape to the above address.

13. DATE OF ABSTRACT

August 1971.



RSIC CODE PACKAGE CCC-135

1. NAME AND TITLE OF CODE

GAMMOM: Gamma Ray Moments Method Codes - GRMM and SPENCER.

GRMM is a version frozen since 1966, SPENCER since 1968.

2. CONTRIBUTORS

Center for Radiation Research, National Bureau of Standards,
Washington, D. C. (GRMM).

Atomics International, Canoga Park, California (SPENCER).

3. CODING LANGUAGE AND COMPUTER

FORTRAN IV; IBM 360, CDC 1604.

4. NATURE OF PROBLEM SOLVED

The code GRMM calculates the moments of the flux from a plane isotropic monoenergetic source in an infinite homogeneous medium.

SPENCER was designed to approximately reconstitute a function from a finite number of its moments.

5. METHOD OF SOLUTION

The directional dependence of quantities in the Boltzmann equation can be removed by making a spherical harmonic - or, in sufficiently symmetric cases a Legendre polynomial expansion, of all the terms of the equation. The GRMM code restricts its notice to those symmetric cases where the Legendre expansion suffices. This expansion reduces the Boltzmann equation to an infinite set of coupled equations for the Legendre coefficients, which are functions of energy and distance (only one spatial variable is allowed.)

If the j^{th} equation of the expanded set is multiplied term by term by the m^{th} power of the space variable, Z^m , and integrated term by term over the whole space domain, there results a doubly indexed set of equations for the quantities $M_{m,j}$.

We shall call $M_{m,j}$ the m th moment of the j^{th} Legendre expansion coefficient.

The equations for the quantities $M_{m,j}$ are integral equations in energy of the Volterra type.

A moments calculation is not complete without a means of reconstruction of functions from the moments. The companion program, SPENCER, performs this function.

Given a finite number of the moments

$$M_n = \int u^n f(u) du$$

of some function $f(u)$, we seek to construct from them a function $g(u)$ which is a good approximation to $f(u)$. SPENCER can be used whenever this general problem arises.

In GRMM, calculation of the moments appears to be reasonably straightforward and in many ways similar to previous schemes. As is customary, energy is transformed into photon wave length λ . If the source wave length is λ_0 , there is a well-known discontinuity in the kernel (which corresponds to a scattering kernel) at $\lambda_0 + 2$. Some care has been taken in this program to avoid significant error originating in this discontinuity.

The reconstruction of a function from its moments is perhaps the most challenging aspect of a "moments method" calculation. It does not proceed by well-defined routine operations. The finite number of moments available grossly underspecify the function. It is essential to bring to the problem insight concerning the functional characteristics and asymptotic behavior, and, by some means to incorporate this insight into the inversion process. If the "insight" is erroneous, disaster is wrought. In SPENCER the function $f(u)$ is written as

$$\begin{aligned} f(u) &= f_e(u) + f_o(u) \\ f_e(u) &= 1/2 (f(u) + f(-u)) \\ f_o(u) &= 1/2 (f(u) - f(-u)) \end{aligned}$$

A function $h(u)$ is constructed which reflects whatever properties the insight of the constructor suggests are in $f(u)$. Further

$$h(u) = h_e(u) + h_o(u)$$

$$h_e(u) = 1/2 (h(u) + h(-u))$$

$$h_o(u) = 1/2 (h(u) - h(-u))$$

Now the following two approximations are made:

$$f_e(u) = \sum_{j=1}^J \frac{A_{ej}}{\sqrt{b_{ej}}} h_e\left(\frac{u}{\sqrt{b_{ej}}}\right)$$

$$f_o(u) = \sum_{j=1}^J \frac{A_{oj}}{\sqrt{b_{oj}}} h_o\left(\frac{u}{\sqrt{b_{oj}}}\right)$$

With the further definitions

$$M_n = \int_{-\infty}^{\infty} u^n f(u) du$$

$$M_{en} = \int_{-\infty}^{\infty} u^{2m} f_e(u) du \quad n = 2m$$

$$M_{on} = \int_{-\infty}^{\infty} u^{2m-1} f_o(u) du \quad n = 2m-1$$

$$M_{kn} = \int_{-\infty}^{\infty} u^n h_k(u) du \quad k = e \text{ or } o$$

The coefficients A_{kj} , b_{kj} are determined from the following set of relations:

$$\mu_{km} = \sum_{j=1}^J A_{rj} (b_{kj})^m$$

$$\mu_{km} = M_{km} / M_{km}$$

$$b_{kj}^J = - \sum_{n=0}^{J-1} C_n b_{kj}^n$$

$$\sum_{n=0}^{J-1} \mu_{io+i+n} C_n = - \mu_{io+i+j}; \quad 0 \leq i \leq j-1;$$

$$io \geq 0$$

6. RESTRICTIONS OR LIMITATIONS

None noted.

7. TYPICAL RUNNING TIME

No study has been made by RSIC as to typical running time.

8. COMPUTER HARDWARE REQUIREMENTS

The codes are operable on the IBM 360/75 equipment with standard I-Ø devices.

9. COMPUTER SOFTWARE REQUIREMENTS

The FORTRAN IV Operating System of the IBM 360/75 may be used with the packaged codes.

10. REFERENCES

W. R. Kimel, Editor-in-Chief, Radiation Shielding, TR 40 (1966);
L. V. Spencer, "Gamma Ray Shielding Theory," Chapter I, Pages 132-145.

Bruce D. O'Reilly and Richard C. Lewis, "SPENCER, A Fortran Program to Approximately Reconstitute a One-Dimensional Function from a Finite Number of Its Moments," NAA-SR-Memo-11653 (November 1965).

11. CONTENTS OF CODE PACKAGE

The package contains the following items:

- a. the referenced documents,
- b. a reel of magnetic tape on which is written in several files:
the source card decks for GRMM and SPENCER, BCD input for a sample problem for each, and a BCD output listing from running the sample problems.

12. HOW TO OBTAIN PACKAGE

Inquiries or requests for the code package may be mailed to

CODES COORDINATOR
Radiation Shielding Information Center
Oak Ridge National Laboratory
Post Office Box X
Oak Ridge, Tennessee 37830

or telephoned to

Area Code 615; 483-8611, extension 3-6944, or to
FTS xx-615-483-6944.

Persons requesting the package should send a reel of magnetic tape to
the above address.

13. DATE OF ABSTRACT

February 1972.



RSIC CODE PACKAGE CCC-136

1. NAME AND TITLE OF CODE

COLLIMATOR: Monte Carlo Calculation of the Spectrum of Gamma Radiation from a Collimated Co-60 Source.

2. CONTRIBUTOR

Nuclear Engineering Department, University of Illinois, Urbana, Illinois.

3. CODING LANGUAGE AND COMPUTER

FORTRAN IV and MAP; IBM 7090.

4. NATURE OF PROBLEM SOLVED

COLLIMATOR performs a calculation of gamma-ray transport in a finite, multi-region medium having cylindrical symmetry. The source may be described as a volume of point sources of various energies. The code has been utilized to calculate the energy spectrum emitted by a collimated Co-60 source.

5. METHOD OF SOLUTION

The Monte Carlo method is used by COLLIMATOR. Splitting is utilized as well as importance sampling of source directions. In addition, a procedure to facilitate the treatment of boundaries is included in which a fictitious interaction is employed. In effect, this allows interaction coefficients for one medium to be used for sampling path lengths in two media. This eliminates the need for calculating coordinates of interface crossing points.

6. RESTRICTIONS OR LIMITATIONS

None noted.

7. TYPICAL RUNNING TIME

Estimated running time of packaged sample problem: one minute on the IBM 360.

8. COMPUTER HARDWARE REQUIREMENTS

The code is operable on the IBM 360 and uses standard equipment.

9. COMPUTER SOFTWARE REQUIREMENTS

Any FORTRAN Compiler may be used. The IBM 360 FORTRAN IV compiler was used to prepare the code package.

10. REFERENCE.

E. E. Morris and A. B. Chilton, "Monte Carlo Calculation of the Spectrum of Gamma Radiation from a Collimated Co-60 Source," NRDL-TRC-68-6 (December 1967).

11. CONTENTS OF CODE PACKAGE

The package contains the following items:

- a. the referenced document, including sample problem output, and
- b. a reel of magnetic tape on which is written in 3 files: the source card deck, BCD input for a sample problem, and the object deck.

12. HOW TO OBTAIN PACKAGE

Inquiries or requests for the code package may be mailed to

CODES COORDINATOR
Radiation Shielding Information Center
Oak Ridge National Laboratory
Post Office Box X
Oak Ridge, Tennessee 37830

or telephoned to

Area Code 615; 483-8611, extension 3-6944, or to
FTS xx-615-483-6944.

Persons requesting the package should send a reel of magnetic tape to the above address.

13. DATE OF ABSTRACT

August 1971.



RSIC CODE PACKAGE CCC-137

1. NAME AND TITLE OF CODE

RIBD: Radioisotope Buildup **and** Decay Code and Data Library.

An abbreviated version of RIBD is built into the CCC-79D/ISOSHLD Code.

2. CONTRIBUTOR

Battelle Memorial Institute, Pacific Northwest Laboratory,
Richland, Washington.

3. CODING LANGUAGE AND COMPUTER

FORTRAN IV; UNIVAC 1108.

4. NATURE OF PROBLEM SOLVED

RIBD is a radioisotope buildup and decay code designed to analyze the fission product content of irradiated reactor fuel in terms of potential biological hazards and heating effects accompanying radioactive decay.

5. METHOD OF SOLUTION

RIBD is a grid processor which calculates isotopic concentrations resulting from two fission sources with normal down-chain decay by beta emission and isomeric transfer and inter-chain coupling resulting from n, γ reactions. The calculations follow the irradiation history through an unlimited number of step changes of unrestricted duration and variability including shutdown periods, restarts at different power levels and/or any other level changes.

A nuclear data library has been generated for use with RIBD to calculate fission product inventories and decay heat rates associated with fuels irradiated in fast reactor environments.

6. RESTRICTIONS OR LIMITATIONS

None noted.

7. TYPICAL RUNNING TIME

No study of typical running time has been made by RSIC.

8. COMPUTER HARDWARE REQUIREMENTS

RIBD was designed to run on the UNIVAC 1108 computer with standard input-output devices.

9. COMPUTER SOFTWARE REQUIREMENTS

The packaged code version was compiled and executed on the UNIVAC 1108 FORTRAN compiler.

10. REFERENCES

R. O. Gumprecht, "Mathematical Basis of Computer Code RIBD," DUN-4136 (June 1968).

L. D. O'Dell and W. L. Bunch, "Revised Fast Reactor Library for Use with RIBD," BNWL-962 (April 1969).

J. L. Rash, "Use of Computer Code RIBD for Fission Product Analysis," RL-NRD-610 (November 1965).

11. CONTENTS OF CODE PACKAGE

The package contains the following items:

- a. the referenced documents, and
- b. a reel of magnetic tape on which is written in 3 separate files: the source card decks, system routines, and a BCD library of data.

12. HOW TO OBTAIN PACKAGE

Inquiries or requests for the code package may be mailed to

CODES COORDINATOR
Radiation Shielding Information Center
Oak Ridge National Laboratory
Post Office Box X
Oak Ridge, Tennessee 37830

or telephoned to

Area Code 615; 483-8611, extension 3-6944, or to
FTS xx-615-483-6944.

Persons requesting the package should send a reel of magnetic tape to
the above address.

13. DATE OF ABSTRACT

February 1972.



RSIC CODE PACKAGE CCC-138

1. NAME AND TITLE OF CODE

PIGG: A Multigroup One-Dimensional P-1 Radiation Transport Code.

AUXILIARY ROUTINE

BIGLIB: Cross Section Handling Code and Data Library.

PIGG is also available from the ENEA Computer Programme Library, abstracted as ENEA 276.

2. CONTRIBUTORS

Institutt for Atomenergi, Kjeller Research Establishment, Kjeller, Norway.

European Nuclear Energy Agency's Computer Programme Library, Ispra, Italy.

3. CODING LANGUAGE AND COMPUTER

FORTRAN 63; CDC 3600.

4. NATURE OF PROBLEM SOLVED

PIGG solves the P_1 equations in a Greuling-Goertzel approximation for the spatially dependent multigroup neutron flux in both slab and cylindrical geometry. The resonance absorption and fission are calculated using single-pin resonance and fission integrals, Dancoff factors, and resonance-distribution functions. The thermal region can be treated either as one group or as few groups to which diffusion equations are applied. Logarithmic boundary conditions are applied at the external boundaries. The output is very extensive and includes the criticality factor, point-wise group spectra and macrogroup constants, region-averaged macrogroup constants, and epithermal reaction rates for selected isotopes in different mesh points and macrogroups. Interesting applications include calculation

of space-dependent epithermal spectra in BWR-fuel elements, spectrum transients near core-reflector interfaces, and effective macrogroup boundary conditions on control absorbers.

5. METHOD OF SOLUTION

Three point difference equations for the group flux are obtained using box integration. The equations are solved using power iteration accelerated by Chebyshev extrapolation of the normalized source. Matrix factorization technique is used for solving the inhomogeneous equations.

BIGLIB is a data library containing macrogroup lethargy structure and Watt's normalized fission spectrum, resonance integral distribution functions and cross sections for 28 materials and a program to update the library tape.

6. RESTRICTIONS OR LIMITATIONS

The following limitations apply: 39 epithermal and 6 thermal groups; 100 mesh points in 20 regions, and 10 homogeneous compositions; 28 different materials may be used for calculation of macroscopic cross sections in 6 macrogroups. For special calculations of reaction rates, 10 selected nuclides in 6 selected mesh points may be used.

7. TYPICAL RUNNING TIME

Required to compute each iteration on the CDC 3600: 10 - 20 seconds.

8. COMPUTER HARDWARE REQUIREMENTS

The code was designed for a 32K CDC 3600 computer with standard input-output devices, 4 scratch units on drums and up to 3 tape units.

9. COMPUTER SOFTWARE REQUIREMENTS

The packaged version was compiled and executed on the CDC 3600 FORTRAN 63 Monitor System, DRUM SCOPE V2.0.

10. REFERENCE

J. O. Berg, G. E. Fladmark, T. Kulikowska, and O. P. Tverbakk, "PIGG, A Multigroup One-Dimensional P-1 Code," KR-129 (May 1968).

11. CONTENTS OF CODE PACKAGE

The package contains the following items:

- a. the referenced document,
- b. a reel of magnetic tape on which is written in 5 files: the BCD source card decks, BCD input for two sample problems, source deck for the auxiliary code and a BCD library of data.

12. HOW TO OBTAIN PACKAGE

Inquiries or requests for the code package may be mailed to

CODES COORDINATOR
Radiation Shielding Information Center
Oak Ridge National Laboratory
Post Office Box X
Oak Ridge, Tennessee 37830

or telephoned to

Area Code 615; 483-8611, extension 3-6944, or to
FTS xx-615-483-6944.

Persons requesting the package should send a reel of magnetic tape to the above address.

13. DATE OF ABSTRACT

August 1971.



RSIC CODE PACKAGE CCC-139

1. NAME AND TITLE OF CODE

CONSTRIP V: Vertical Barrier - Finite Source Plane Gamma-Ray Penetration Code.

The CONSTRIP V code is the successor to CONSTRIP II which was developed by C. Eisenhauer and E. E. Morris of the National Bureau of Standards, and the CONSTRIP III code developed subsequently by the Research Triangle Institute.

2. CONTRIBUTOR

Research Triangle Institute, Research Triangle Park, North Carolina.

3. CODING LANGUAGE AND COMPUTER

FORTRAN IV; IBM 360/75/91.

4. NATURE OF PROBLEM SOLVED

The problem solved is that of the dose rate to a detector located behind a wall, or adjacent to the front surface of a wall, due to direct and wall scattered gamma radiation from source planes of various sizes and shapes located in front of the wall. Contributions from the source plane through various wall segments to the detector are determined through numerical integration of the dose rate arising from segments of the source field. CONSTRIP V can calculate the dose rates at the detector location due to gamma radiation which is singly scattered, multiply scattered, or that (direct) radiation which passes unscattered through the wall. No edge effects are considered and all geometrical considerations are made relating to the external face of the wall.

5. METHOD OF SOLUTION

The single scatter contribution is treated exactly with the Klein Nishina differential scatter cross section.

The multiple scatter calculation makes use of differential Monte Carlo dose transmission coefficients. These data are based upon the assumption of a uniform monodirectional flux incident on the wall. They are tabulated as a function of incident obliquity cosines, emergent obliquity cosines, and azimuthal angles of scatter. These data are three-way interpolated for the multiple scatter calculation. Both multiple scatter and total scatter Monte Carlo data are available. When total scatter data are supplied to CONSTRIP V, the single scatter calculation is bypassed. The dose transmission data are interpolated also for thickness and for energy to obtain results for arbitrary thicknesses and source energies bounded by the available Monte Carlo data.

Either Monte Carlo total scatter data, or multiple scatter data are required as input. Data suitable for this input have been generated by M. J. Berger, Charles Eisenhauer, and E. E. Morris of the National Bureau of Standards for wall thicknesses of 0.25 to 4 mean free paths of concrete for source energies of 0.2, 0.4, 0.66, 1.25, 5, and 10 MeV. These data are given as dose transmission functions for 11 source obliquity cosine base points, 11 detector obliquity cosine base points, and seven detector azimuthal angle base points.

6. RESTRICTIONS OR LIMITATIONS

The only restrictions on the calculational technique are those imposed by availability of (1) Monte Carlo data spanning the energy and wall thickness under consideration, and (2) buildup factors for the source energy employed. Currently, the buildup factor in the program is that for the 1.25-MeV Cobalt-60 gamma. This factor must be replaced in the code in subroutine BILDUP if another one is desired.

7. TYPICAL RUNNING TIME

For source fields in front of a shielding wall with the detector located behind the wall, typical running times are from 15 to 30 seconds per case on the IBM 360/75. When the fields are adjacent to the front of the wall and extend past either end of the wall, running times are highly dependent on the fractional integration accuracy requested.

8. COMPUTER HARDWARE REQUIREMENTS

The program is designed for the IBM 360/75 machine with 300K fast core. It is also adaptable to the CDC 3600, 65K core providing the interpolation routines for thickness and energy are not simultaneously required. In this case, the interpolation routine for either thickness or energy must be separated from the code before loading. Auxiliary equipment required is limited to an input device and a printer. The code may be loaded directly from cards, tapes, or other auxiliary storage. Problems are loaded from cards. The clock is not sampled by the program.

9. COMPUTER SOFTWARE REQUIREMENTS

CONSTRIP V may be compiled and executed on the FORTRAN IV Monitor System of the IBM 360/75/91 Operating System.

10. REFERENCES

W. O. Doggett and F. A. Bryan, Jr., "CONSTRIP V, A Computer Program for the Vertical Barrier - Finite Source Plane Gamma Ray Penetration Problem," NRDL-TRC-68-55 Final Report R-OU-333 (part 1 of 2) (April 1969).

W. O. Doggett and F. A. Bryan, Jr., "Radiological Recovery Requirements, Structures and Operations Research, Volume I: Computational Technique for Determining Importance of Limited Strip Decontamination Procedures," R-OU-266 (May 1967).

11. CONTENTS OF CODE PACKAGE

The package contains the following items:

- a. the referenced documents and
- b. a reel of magnetic tape on which is written in 5 files:
two sets of BCD data, the source card deck, BCD input for a sample problem, and a BCD output listing from running the sample problem.

12. HOW TO OBTAIN PACKAGE

Inquiries or requests for the code package may be mailed to

CODES COORDINATOR
Radiation Shielding Information Center
Oak Ridge National Laboratory
Post Office Box X
Oak Ridge, Tennessee 37830

or telephoned to

Area Code 615; 483-8611, extension 3-6944, or to
FTS xx-615-483-6944.

Persons requesting the package should send a reel of magnetic tape to the above address.

13. DATE OF ABSTRACT

February 1972.

RSIC CODE PACKAGE CCC-140

1. NAME AND TITLE OF CODE

DIPHO: Monte Carlo Gamma-Ray Code - Infinite Medium, Mono-energetic and Isotropic Point Source.

DIPHO is also available in the European Nuclear Energy Agency Computer Programme Library (ENEA CPL), Ispra (Varese), Italy, listed under Abstract ENEA 165.

2. CONTRIBUTOR

Laboratoire Central et Ecoles de L'Armement, Ministere des Armees, Fort de Montrouge, Arcueil (Val de Marne), France through the ENEA CPL.

3. CODING LANGUAGE AND COMPUTER

FORTRAN IV; IBM 7040 and 360/65/75/91.

4. NATURE OF PROBLEM SOLVED

DIPHO is designed as a differential energy flux calculation in a photon scattering problem. The point photon source is monoenergetic and isotropic. The absorbing medium is assumed to be infinite and homogeneous.

5. METHOD OF SOLUTION

The solution to the Boltzmann equation is determined by the Monte Carlo method with exponential transformation.

6. RESTRICTIONS OR LIMITATIONS

None noted.

7. TYPICAL RUNNING TIME

Estimated running time of the packaged sample problem on the IBM 360/65: from one half to two hours for particles depending on the value of C.

8. COMPUTER HARDWARE REQUIREMENTS

DIPHO was originally designed for the IBM 7040. The packaged version is operable on the IBM 360 with standard equipment. Only 16K of memory is required.

9. COMPUTER SOFTWARE REQUIREMENTS

The code may be compiled and executed on the IBM 360/65/75/91 FORTRAN IV Monitor System with standard input-output devices.

10. REFERENCE

"Photon Scattering - Monoenergetic Isotropic Point Source - Infinite Medium," Note 9, Nuclear Shielding Calculations Study No. 07-1, Defense Ministry, Ministerial Defense Delegation, Technical Direction of Ground Defense, Central Defense Laboratory (November 1965) (ORNL-tr-2349).

11. CONTENTS OF CODE PACKAGE

The package contains the following items:

- a. the referenced document,
- b. a reel of magnetic tape on which is written in 4 files: the source card deck, BCD input for a sample problem, punched card output, and a BCD output listing from running the sample problem.

12. HOW TO OBTAIN PACKAGE

Inquiries or requests for the code package may be mailed to

CODES COORDINATOR
Radiation Shielding Information Center
Oak Ridge National Laboratory
Post Office Box X
Oak Ridge, Tennessee 37830

or telephoned to

Area Code 615; 483-8611, extension 3-6944, or to
FTS xx-615-483-6944.

Persons requesting the package should send a reel of magnetic tape to the above address.

13. DATE OF ABSTRACT

August 1971.

RSIC CODE PACKAGE CCC-141

1. NAME AND TITLE OF CODE

RAC: Spinney Removal-Diffusion Code - Attenuation and Heat Generation in a Multiregion Shield.

RAC is also available in the European Nuclear Energy Agency's Computer Programme Library (ENEA CPL), listed under Abstract ENEA 122.

2. CONTRIBUTOR

Shielding Codes Group, Tokai Establishment, Japan Atomic Energy Research Institute (JAERI), Tokai, Ibaraki, Japan through the ENEA CPL.

3. CODING LANGUAGE AND COMPUTER

FORTRAN II; IBM 7090.

4. NATURE OF PROBLEM SOLVED

RAC was designed for calculating neutron and gamma-ray attenuation and heat generation in a multiregion shield. The following information is provided as a function of the distance from the core center: neutron fluxes for 6 diffusion groups and 18 removal energy groups, total neutron and gamma-ray dose-rate in each region, approximate neutron and gamma-ray spectra, and nuclear heat generation by induced radiation in each region in the shield.

5. METHOD OF SOLUTION

The Spinney method (removal-diffusion) is used. The high energy neutrons are treated using the energy dependent removal theory, and the epithermal and thermal neutrons are treated with few group diffusion theory. The attenuation of the primary gamma rays and the thermal neutron capture gamma rays are calculated by using buildup factors in the Taylor form.

The code consists of a main program and 4 routines. The REMOVAL routine calculates the uncollided neutron flux and dose. DIFFUSION

calculates the slowing down and diffusion neutron flux and dose from removal source calculated in REMOVAL. SEC.GAMMA calculates the secondary gamma-ray flux produced in the shield region. PRY.GAMMA calculates the primary gamma-ray flux and dose. The control program calculates the total dose and total heat generation in the regions from data supplied by the routines above.

6. RESTRICTIONS OR LIMITATIONS

The following limits apply:

Maximum number of shield regions, 15.

Maximum number of meshpoints for each region, 20.

Maximum number of energy groups for removal flux, 18.

Maximum number of energy groups for diffusion flux, 6.

Maximum numbers of regions and mesh points in each region which give the dose distribution as the output, 15 and 5, respectively.

Maximum numbers of regions and mesh points in each region which give the distribution of heat generation as the output, 10 and 20, respectively.

7. TYPICAL RUNNING TIME

Ten minutes is estimated to be an average running time on the IBM 7090.

Estimated running time of the packaged sample problem: 12 minutes.

8. COMPUTER HARDWARE REQUIREMENTS

A 32K IBM 7090 computer with standard equipment and at least 3 tape units are required.

9. COMPUTER SOFTWARE REQUIREMENTS

RAC may be compiled and executed under the IBM 7090 IBSYS Operating System (Version II and FMS).

10. REFERENCE

"A User's Manual for the RAC Radiation Attenuation Code," JAERI Informal Memorandum (July 1966).

11. CONTENTS OF CODE PACKAGE

The package contains the following items:

- a. the referenced document,
- b. a reel of magnetic tape on which is written in 2 files: the source card decks, BCD input for a sample problem, and a BCD output listing from running the sample problem.

12. HOW TO OBTAIN PACKAGE

Inquiries or requests for the code package may be mailed to
CODES COORDINATOR
Radiation Shielding Information Center
Oak Ridge National Laboratory
Post Office Box X
Oak Ridge, Tennessee 37830

or telephoned to

Area Code 615; 483-8611, extension 3-6944, or to
FTS xx-615-483-6944.

Persons requesting the code package should send a reel of magnetic tape to the above address.

13. DATE OF ABSTRACT

August 1971.



RSIC CODE PACKAGE CCC-142

1. NAME AND TITLE OF CODE

MERCURE 3: Gamma-Ray Kernel Integration Code - Straight-Line Attenuation in Three-Dimensional Geometry.

MERCURE 3 is also available in the European Nuclear Energy's Computer Programme Library (ENEA CPL), listed under Abstract ENEA 194.

2. CONTRIBUTOR

Reactor Shielding Group, CEA Centre d'Etudes Nucleaires de Fontenay-aux-Roses, France through the ENEA CPL.

3. CODING LANGUAGE AND COMPUTER

FORTRAN IV and machine language; IBM 360/75/91.

4. NATURE OF PROBLEM SOLVED

MERCURE 3 treats the straight-line attenuation of particles in heterogeneous media bounded by planes and quadrics, and to assemble the cases of volume sources encountered in practice (geometry, spatial distribution, spectrum) in a single code.

The code may be used to calculate the dose, heating, flux and energy transport due to a volume source of gammas, or to calculate any quantity which is possible to treat by the point kernel and the line of sight method provided kernels are supplied. The code has built in gamma-ray attenuation kernels and buildup factors for a number of materials. The geometry is three-dimensional and defined by plane and quadric surfaces. Sources are isotropic and can be defined in cylindrical, cartesian, or spherical coordinate systems.

The program uses a nuclear data library for gamma-ray calculations (total and energy absorption coefficients, coefficients for the dose, energy absorption, and energy buildup).

5. METHOD OF SOLUTION

The geometry is decomposed in homogeneous and analytical convex mesh such that to each mesh point on a limiting surface there corresponds only one neighbor mesh point. The integration of the point kernel is carried out in the coordinate system utilized for the definition of the source. In cylindrical or spherical geometry, the quadrature in both angular coordinates may be defined by the user or calculated by the program in such a way that the resulting volumes are effectively cubical.

6. RESTRICTIONS OR LIMITATIONS

The following limits apply:

- 25 energy groups
- 40 geometrical configurations or compositions
- 10 conversion factors
- 100 geometrical mesh points
- 50 quadrix
- 100 planes
- 10 volume sources
- 5 spectra
- 50 point doses.

7. TYPICAL RUNNING TIMES

The calculation time varies appreciably with the geometrical complexity. For a given geometry it is approximately proportional to the number of volume elements used in the communication. The time corresponding to the calculation of the contribution from an integration volume element is of the order of some milliseconds on the IBM 360/75.

Estimated running time of the packaged sample problem: 58.6 seconds on the IBM 360/75.

8. COMPUTER HARDWARE REQUIREMENTS

The program was originally designed to run on the IBM 7094 with standard configuration. The packaged version was made operable on the IBM 360 and may be run on Models 65, 75, or 91 with standard input-output devices.

9. COMPUTER SOFTWARE REQUIREMENTS

Designed for the IBM 7094 FORTRAN IV IBSYS Operating System, the current program will compile and execute on the FORTRAN IV Monitor System of the IBM 360-series computers.

10. REFERENCE

Christian Devillers, "The MERCURE 3 Program: Straight-Line Attenuation in a Three-Dimensional Geometry," CEA-R-3264 (ORNL-tr-1812) (1967).

11. CONTENTS OF CODE PACKAGE

The package contains the following items:

- a. the referenced document, and
- b. a reel of magnetic tape on which is written in 10 files:
all source card decks, BCD input for a sample problem, and
the BCD output listing from running the sample problem.

12. HOW TO OBTAIN PACKAGE

Inquiries or requests for the code package may be mailed to

CODES COORDINATOR
Radiation Shielding Information Center
Oak Ridge National Laboratory
Post Office Box X
Oak Ridge, Tennessee 37830

or telephoned to

Area Code 615; 483,8611, extension 3-6944, or to
FTS xx-615-483-6944.

Persons requesting the package should send a reel of magnetic tape to the above address.

13. DATE OF ABSTRACT

August 1971.

RSIC CODE PACKAGE CCC-143

1. NAME AND TITLE OF CODE

GREAT - GRASS: Monte Carlo Radiation Transport Codes for
Fallout Shielding.

2. CONTRIBUTOR

Radiation Research Associates, Inc., Fort Worth, Texas.

3. CODING LANGUAGE AND COMPUTER

CCC-143A: FORTRAN IV; IBM 1130.

CCC-143B: FORTRAN IV; IBM 360/75/91.

4. NATURE OF PROBLEM SOLVED

GREAT is designed to calculate the scattered gamma-ray environment in an air-ground geometry resulting from a plane-isotropic monoenergetic gamma-ray source. Statistical estimation is used to determine the flux energy and angle distribution at a point detector in air above a finite annular source located on or near the ground surface. Ground roughness may be simulated by placing the source below the ground surface.

GRASS was written to calculate gamma-ray attenuation in simple cylindrical structures. GRASS calculates the structure-scattered gamma-ray flux energy distribution and exposure angle distribution at a point on the centerline of an upright cylindrical barrier exposed to gamma rays from a plane source located on, or near, the ground surface.

5. METHOD OF SOLUTION

GREAT is a Monte Carlo code designed to calculate the energy and angle distribution of the scattered gamma-ray flux at a point detector in an air-ground geometry resulting from a plane-isotropic monoenergetic gamma-ray source parallel to the air-ground interface.

The source, which is finite and annular in shape, may be positioned below the smooth ground surface to simulate the effects of a source located on rough ground. The point detector is located on the vertical axis of the annular source. Initial photon parameters, path lengths, interactions and scattering angles are obtained by random sampling of appropriate probability distributions. Each photon is traced as it scatters within the defined geometry and estimates are made of the flux contribution at the detector from each collision. Portions of the procedure were taken from COHORT, a general purpose Monte Carlo Code.

Gamma-ray interactions considered in GREAT include absorption, Compton scattering and pair production. The scattering angle distributions are obtained from the Klein-Nishina formula. Statistical estimation is used to compute the flux contribution at the detector resulting from each collision. A gamma-ray history may be terminated on energy, weight or collision number.

The Monte Carlo method is also used in GRASS. The energy and angle distribution of the flux incident upon the structure is calculated by a related Monte Carlo procedure and is input in a fixed form which describes (1) "uncollided" gamma rays which reach the barrier before having a collision in the air or ground and (2) "scattered" gamma rays incident on the barrier after having at least one collision in the air or ground. GRASS employs a pseudo source based on probability distributions derived from the incident flux distributions. The source is located on the outer lateral surface of the barrier. The intensity of the pseudo source representing the uncollided incident flux may be varied in up to 3 steps as a function of height along with barrier wall, while the pseudo source used to describe the scattered incident flux is assumed to have the same intensity at all heights. Much of the versatility found in larger Monte Carlo procedures was sacrificed for speed and compactness, allowing GRASS to be compatible with the storage available in the IBM 1130 computer and perform gamma-ray scattering calculations in cylindrical barriers with greater efficiency.

6. RESTRICTIONS OR LIMITATIONS

The size of the IBM 1130 computer imposed certain limitations, requiring care in programming and considerable use of auxiliary hardware equipment.

7. TYPICAL RUNNING TIME

None noted.

8. COMPUTER HARDWARE REQUIREMENTS

Both programs were designed for the IBM 1130 system with core and disk storage capacities of 8192 and 512K 16-bit words respectively. These are packaged as CCC-143A.

Both programs were converted by RSIC to run on the IBM 360/75/91 System with standard input-output devices. This package is CCC-143B.

9. COMPUTER SOFTWARE REQUIREMENTS

Standard compilers and operating systems were used in both packaged versions.

10. REFERENCES

J. H. Price, "GREAT, A Monte Carlo Procedure for Calculating Gamma-Radiation Environments Above Terrain," RRA-T78 (NRDL TRC-68-11) (December 1967).

J. H. Price, "GRASS, A Monte Carlo Procedure for Calculating Gamma-Ray Attenuation of Simple Structures," RRA-T79 (NRDL TRC-68-10) (December 1967).

J. H. Price, "Monte Carlo Programming for Small Computers," RRA-M84 (June 1968).

11. CONTENTS OF CODE PACKAGE

The package contains the following items:

- a. the referenced documents,

- b. a reel of magnetic tape on which is written in 4 files (CCC-143A): the 1130 FORTRAN source card decks and BCD input data for a sample problem, and
- c. a reel of magnetic tape on which is written in 8 separate files (CCC-143B): the BCD source card decks for both codes, BCD input for a sample problem for each code, and a BCD output listing from running the sample problem on each program.

12. HOW TO OBTAIN PACKAGE

Inquiries or requests for the code package may be mailed to

CODES COORDINATOR
Radiation Shielding Information Center
Oak Ridge National Laboratory
Post Office Box X
Oak Ridge, Tennessee 37830

or telephoned to

Area Code 615; 483-8611, extension 3-6944, or to
FTS xx-615-483-6944.

Persons requesting the code package should send a reel of magnetic tape to the above address, indicating which code version is desired.

13. DATE OF ABSTRACT

February 1972.

RSIC CODE PACKAGE CCC-144

1. NAME AND TITLE OF CODE

TIMOC: Monte Carlo Three-Dimensional Neutron Transport Code.

AUXILIARY ROUTINE

CODAC: ENDF Format to TIMOC Format Data Generator.

TIMOC and CODAC are also available from the European Nuclear Energy Agency's Computer Programme Library (ENEA CPL), Ispra, Italy, listed under Abstracts numbered ENEA 270 and ENEA 275 respectively.

2. CONTRIBUTOR

CCR Euratom, Ispra, Italy through the ENEA CPL.

3. CODING LANGUAGE AND COMPUTER

FORTRAN II and FAP; IBM 7090/94 (TIMOC).

FORTRAN IV; IBM 360 (CODAC).

4. NATURE OF PROBLEM SOLVED

TIMOC solves by the Monte Carlo technique the energy- and time-dependent (or stationary) homogeneous or inhomogeneous neutron transport equation in three-dimensional geometries. The program can treat absorption and all commonly used scattering kernels, such as: fission, isotropic and anisotropic scattering, level excitation, the evaporation model, and the energy transfer matrix model, which includes (n,2n) reactions. The exchangeable geometry routines at present consist of:

- a. periodical multilayered slab, spherical and cylindrical lattices;
- b. an elaborate three-dimensional cylindrical geometry, which allows all kinds of subdivisions; and
- c. the very flexible O5R geometry routine which is able to describe any body or body combinations with surfaces of second order.

The program calculates the stationary or time-, energy-, and region-dependent fluxes as well as the transmission ratios between geometrical regions and the following integral quantities or eigenvalues: the leakage rate, the slowing down density, the production to source ratio, the multiplication factor based on flux and collision estimator, the mean production time, the mean destruction time, time distribution of production and destruction, the fission rates, the energy dependent absorption rates, and the energy deposition due to elastic scattering for the different geometrical regions.

A special option allows the calculation of geometrical perturbation effects. In such calculations the differential effect does not depend on the total variance of the considered quantity. The sampling of these differential effects is based on the method of similar flight paths.

CODAC is a nuclear data processing code. It converts ENDF/B data into group averaged cross sections in the form needed by Monte Carlo codes. CODAC generates the mean values of σ_{EXP} , σ_{el} , σ_{in} , σ_{fiss} , and ν for any group structure by using specified weighting spectra. In the case of anisotropic elastic scattering either the average cosine ($\bar{\mu}$) or the angular distribution function $d\sigma(\mu)/d\mu$ is calculated for each energy group. The inelastic scattering is described by a transfer matrix which can also include (n,2n) reactions.

5. METHOD OF SOLUTION

TIMOC is a Monte Carlo program and uses several partially optional variance reducing techniques, such as: the method of expected values (weight factor), Russian roulette, the method of fractional generated neutrons, double sampling, semi-systematic sampling, and the method of expected leakage probability. The neutron histories are assigned a discrete energy value after each collision process. The nuclear data input is done, however, by group averaged cross sections.

The program can generate the neutron fluxes either resulting from an external source or in the form of fundamental mode distributions by a special source iteration procedure. In this latter case, the calculations of eigenvalues are based on the life-cycle concept.

The program can be linked to the ENDF/B data file via the CODAC code. CODAC generates for any desired group structure the nuclear parameters and group averaged cross sections needed by TIMOC. Averaging is done by using a weighting spectrum to be input. Group averaged cross sections are calculated by summing up the smooth contributions and the contributions of the resolved and unresolved resonances, using the methods of the ETOG-ETOM codes and of MC² at zero temperature. Anisotropic elastic secondary angular distribution is calculated optionally as $\bar{\mu}_{\text{Lab}}$, Legendre expansion, or point by point along the μ -axis. The inelastic secondary energy distribution is computed as transfer matrix.

Anisotropic elastic scattering secondary angle distribution can optionally be calculated in three different ways: as Legendre expansion in the c.m. system, point by point along the μ -axis in the c.m. system, as averaged cosine in the lab system. In calculating, nearly isotropic scattering is assumed in and below the resonant region.

6. RESTRICTIONS OR LIMITATIONS

The following limits apply.

In TIMOC: number of energy groups ≤ 50 , number of isotopes ≤ 20 , number of mixtures ≤ 20 .

CODAC will handle any number of ENDF/B materials during one run. The number of energy groups is limited to 50. The output formats of CODAC correspond to the input formats of TIMOC.

7. TYPICAL RUNNING TIME

The running time of TIMOC depends very much on the problem treated and to some extent on the options specified. A complete eigenvalue and flux analysis in an unreflected highly enriched system requires for a probable error of $\pm 0.5\%$ in k_{eff} 6 minutes of computing time on an IBM 7090. A reflected system, consisting of 85 geometrical regions, and 18 isotope mixtures using 26 energy groups needs for obtaining the same probable error 45 minutes.

Estimated running time of the packaged sample CODAC problem: GO STEP on the IBM 360/75, 85 seconds.

8. COMPUTER HARDWARE REQUIREMENTS

TIMOC was designed for the IBM 7090/94 with a 32K Memory, 2 channels and 3 additional tape units.

CODAC is operable on the IBM 360/75/91 hardware with standard I-O and punch equipment. The GO STEP uses 190K of core memory. The code is built in an overlay structure where it requires 155,600 bytes (38,900 words) on the 360/65. Without overlay 221,400 bytes (55,350 words) are necessary. CODAC requires two I/O devices for reading and printing, one I/O device for the ENDF/B library tape and one on which the TIMOC library is written or punched. One additional I/O device is necessary if the program is loaded from tape.

9. COMPUTER SOFTWARE REQUIREMENTS

TIMOC will run under the FORTRAN II, Version II Monitor System.

CODAC is operable on the IBM 360/75/91 Operating System using OS-360 FORTRAN H Compiler and the Overlay feature. A library of ENDF Cross Sections is required. The package includes only those elements needed by the sample problem; data taken from the ENDF Master Files.

10. REFERENCES

H. Rief and H. Kschwendt, "A Monte Carlo Approach to the Calculation of Characteristic Reactor Parameters in Three-Dimensional Assemblies," EUR-4519 (May 1965).

R. J. Jaarsma and H. Kschwendt, "Geometry Routine for Use in TIMOC," EUR-4536 (1970).

H. Krainer, "CODAC, A FORTRAN IV Programme to Process a TIMOC Library from the ENDF/B File," EUR-4521.3 (1970).

11. CONTENTS OF CODE PACKAGE

The package contains the following items:

- a. the referenced documents,
- b. a reel of magnetic tape on which is written in 9 files:
the TIMOC source card decks (FORTRAN II and FAP), BCD input for a sample problem, and a BCD output listing from running the problem;
- c. a reel of magnetic tape on which is written in 7 files:
the ENDF Cross Sections required for the sample problem (BCD), the CODAC source card deck, BCD input for a sample problem and output from running the sample problem.

12. HOW TO OBTAIN PACKAGE

Inquiries or requests for the code package may be mailed to

CODES COORDINATOR
Radiation Shielding Information Center
Oak Ridge National Laboratory
Post Office Box X
Oak Ridge, Tennessee 37830

or telephoned to

Area Code 615; 483-8611, extension 3-6944, or to
FTS xx-615-483-6944.

Persons requesting the package should send two reels of magnetic tape to the above address.

13. DATE OF ABSTRACT

February 1972.

RSIC CODE PACKAGE CCC-145

1. NAME AND TITLE OF CODE

SORS: Monte Carlo Neutron and Photon Two/Three Dimensional
Transport Code System.

PROTOTYPES

CCC-145A/SORS: Photon Code.

CCC-145B/SORS: Neutron Code.

2. CONTRIBUTOR

Lawrence Livermore Laboratory, Livermore, California.

3. CODING LANGUAGE AND COMPUTER

FORTRAN IV; CDC 6600.

4. NATURE OF PROBLEM SOLVED

The A version is designed to calculate the transport of photons in the energy range from 1.0 keV to 15.0 MeV through complex media. Only planes parallel to the xy, xz, and yz coordinate planes are allowed. Closed quadratic surfaces may be placed arbitrarily. Infinite quadratic surfaces have their local axis of symmetry parallel to the z axis. Output includes special tally options, and the energy deposition determined for each zone and each time interval.

The B version calculates the transport of neutrons through complex media. It calculates the energy deposition, the number of inelastic, capture and fission events per zone in as many as five time intervals. It can be used to calculate specific neutron reactions, surface flux and biological phase in specified zones. Energy range for neutrons: 18 MeV - thermal.

The two codes are the same in every respect with the exception of the physics packages. They are flexible, able to provide point

and volume sources with energy and angle spectrum using a table input. Fission gamma-ray and neutron sources are provided. Both have a time dependent capability.

5. METHOD OF SOLUTION

SORS simulates the neutron and the photon trajectory by use of the Monte Carlo method. The only variance reduction techniques are splitting and Russian roulette. The kinematic used in the codes are elastic scattering, scattering from a discrete level and a statistical model.

Both neutron and gamma-ray cross sections in the package are from sets derived at the Lawrence Livermore Laboratory.

6. RESTRICTIONS OR LIMITATIONS

The following restrictions and limitations apply to the neutron version:

- 110 zones
- 100 quadratic surfaces
- 75 planes
- 9 boundaries per zone
- 5 time intervals
- 15 isotopes per material
- 20 materials
- 14.6 MeV maximum input energy
- 2.0×10^{-8} MeV maximum input energy
- 66 output tally groups
- 19 angular bins
- 21 zones in which angular information is lattical
- 6 energy bins for angular information

7. TYPICAL RUNNING TIME

It takes 37 minutes of CDC 6600 computer time to obtain a 10% Std. deviation at 22 mean free paths at 14.6 MeV from a point source in homogeneous air. Further fluences were determined at each mean free path between the source and the 22.0 mean free path distance.

8. COMPUTER HARDWARE REQUIREMENTS

The codes were designed to run on the CDC 6600 computer. A 75K memory and 3 tape units are required.

9. COMPUTER SOFTWARE REQUIREMENTS

The codes were designed to operate on the CDC 6600 FORTRAN IV standard operating system.

10. REFERENCES

John Kimlinger, Ernest F. Plechaty, and John R. Terrall, "SORS Monte Carlo Photon-Transport Code for the CDC 6600," UCRL-50358 (December 1967).

John Kimlinger and Ernest F. Plechaty, "SORS Monte Carlo Neutron-Transport Code for the CDC 6600," UCRL-50532 (October 1968).

11. CONTENTS OF CODE PACKAGE

The package contains the following items:

- a. the referenced documents,
- b. a reel of magnetic tape on which is written in several separate files for each code version: BCD source card decks, BCD input for a sample problem, and a library of cross section data written in BCD.

12. HOW TO OBTAIN PACKAGE

Inquiries or requests for the code package may be mailed to

CODES COORDINATOR
Radiation Shielding Information Center
Oak Ridge National Laboratory
Post Office Box X
Oak Ridge, Tennessee 37830

or telephoned to

Area Code 615; 485-8611, extension 3-6944, or to
FTS xx-615-483-6944.

Persons requesting the code package should send a reel of magnetic tape to the above address.

13. DATE OF ABSTRACT

February 1972.

RSIC CODE PACKAGE CCC-146

1. NAME AND TITLE OF CODE

UNAMIT: One-Dimensional Spherical Multilayer Reactor-Shield-
Weight Optimization Code.

2. CONTRIBUTORS

NASA Lewis Research Center, Cleveland, Ohio.
Columbia University, New York, New York.

3. CODING LANGUAGE AND COMPUTER

CCC-146A: FORTRAN IV; IBM 7090 and 7094.
CCC-146B: FORTRAN IV; IBM 360.
CCC-146C: FORTRAN 63; CDC 1604 (listings only).

4. NATURE OF PROBLEM SOLVED

UNAMIT generates 4π multilayer spherical reactor shields of minimum weight subject to a fixed shield surface dose rate constraint.

5. METHOD OF SOLUTION

The code uses an exponential attenuation-secondary-production analytic model. Methods are presented for determining required input parameters for attenuation of primary neutrons, primary gamma rays, and secondary gamma rays, and for production of secondary gamma rays.

6. RESTRICTIONS OR LIMITATIONS

None noted.

7. TYPICAL RUNNING TIME

Estimated running time of the packaged sample problem on the IBM 360/91: 1/2 second, and on the IBM 7090/94: 1-2 minutes.

8. COMPUTER HARDWARE REQUIREMENTS

The code may be run on the IBM 360, 7090, and 7094 and the CDC 1604 with standard I-O equipment.

9. COMPUTER SOFTWARE REQUIREMENTS

The code was designed for the IBM 7090/7094 IBSYS Operating System using IJOB Processor, and for the CDC 1604 COOP system. It was made operable by RSIC on the IBM 360/75/91 Operating System using OS-360 Fortran H Compiler.

The FORTRAN 63 source program is listed in the documentation and is not packaged on tape.

10. REFERENCE

Eugene S. Troubetzkoy and Millard L. Wohl, "UNAMIT - A One-Dimensional 4π Spherical Multilayer Reactor-Shield-Weight Optimization Code," NASA TM X-2048 (July 1970).

11. CONTENTS OF PACKAGE

The package contains the following items:

- a. the referenced document,
- b. a reel of magnetic tape on which is written in 2 files:
the source card deck which has been run on the IBM 360
and BCD input for a sample problem.

12. HOW TO OBTAIN PACKAGE

Inquiries or requests for the code package may be mailed to

CODES COORDINATOR
Radiation Shielding Information Center
Oak Ridge National Laboratory
Post Office Box X
Oak Ridge, Tennessee 37830

or telephoned to

Area Code 615; 483-8611, extension 3-6944, or to
FTS xx-615-483-6944.

Persons requesting the code package should send a reel of magnetic
tape to the above address.

13. DATE OF ABSTRACT

February 1972.

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RSIC CODE PACKAGE CCC-147

1. NAME AND TITLE OF CODE

EXDOSE: Calculation of the External Gamma-Ray Dose from
Airborne Fission Products.

RIBD, the reactor inventory, burnup, and decay code (CCC-137),
has been adapted and used as Link 1 in EXDOSE.

2. CONTRIBUTOR

Battelle Memorial Institute, Pacific Northwest Laboratory,
Richland, Washington.

3. CODING LANGUAGE AND COMPUTER

FORTRAN IV and V; UNIVAC 1108.

4. NATURE OF PROBLEM SOLVED

EXDOSE calculates the external gamma radiation dose to the body
from a half-infinite cloud of fission products. The code first
calculates a fission product inventory according to specified input
parameters. The fission products are then released to the atmosphere
in a prescribed manner and the resulting dose to individuals is
calculated for specified meteorological conditions.

5. METHOD OF SOLUTION

The fission product inventory is calculated by an adaptation of
a chain grid processor (RIBD). The time integral of photon energy
release is calculated by Simpson's rule.

6. RESTRICTIONS OR LIMITATIONS

A maximum of 350 cases per run may be made. Each case gives
one gamma-ray dose calculation for one set of variables.

7. TYPICAL RUNNING TIME

Typical running times are given for each portion of the program. Fission product inventory, 30-40 seconds; energy release integral calculation, 15-30 seconds/leak rate; and gamma-ray dose calculation, one second.

8. COMPUTER HARDWARE REQUIREMENTS

The code was designed to run on the UNIVAC 1108. 32K memory is required. An auxiliary tape unit or drum device is necessary for library data input.

9. COMPUTER SOFTWARE REQUIREMENTS

The source program is FORTRAN IV. The relocatable elements and the absolute elements ABEXDO and ABEXD were compiled on the FORTRAN V system. The packaged version is operable on the UNIVAC 1108 operating system.

10. REFERENCE

M. M. Hendrickson, "EXDOSE, A Computer Program for Calculating the External Gamma Dose from Airborne Fission Products," BNWL-811 (September 1968).

11. CONTENTS OF CODE PACKAGE

The package contains the following items:

- a. the referenced document,
- b. a reel of magnetic tape on which is written in 12 files:
the source card decks for the main code and several routines, and BCD libraries of cross section data.

12. HOW TO OBTAIN PACKAGE

Inquiries or requests for the code package may be mailed to

CODES COORDINATOR
Radiation Shielding Information Center
Oak Ridge National Laboratory
Post Office Box X
Oak Ridge, Tennessee 37830

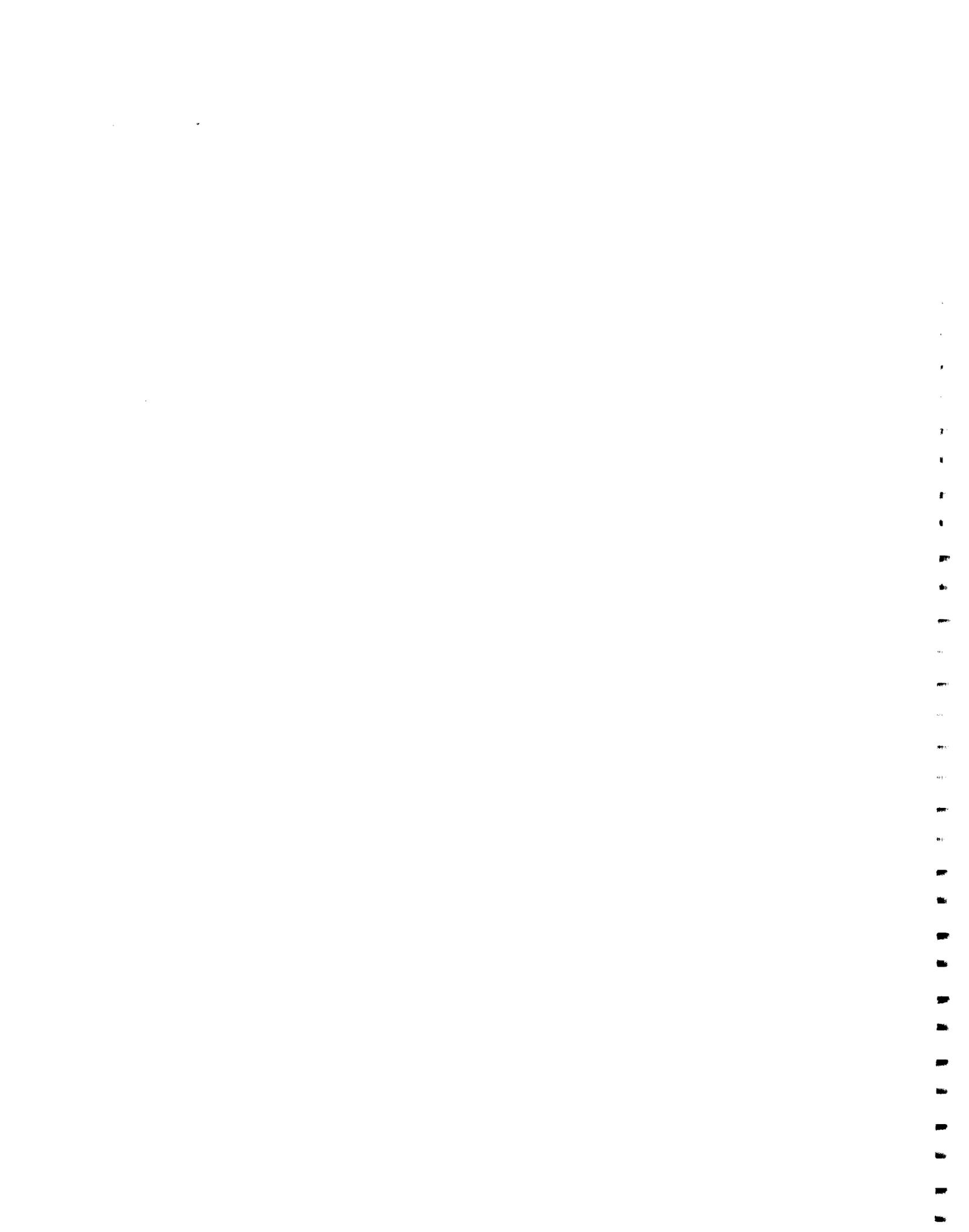
or telephoned to

Area Code 615; 483-8611, extension 3-6944, or to
FTS xx-615-483-6944.

Persons requesting the code package should send a reel of magnetic
tape to the above address.

13. DATE OF ABSTRACT

February 1972.



RSIC CODE PACKAGE CCC-148

1. NAME AND TITLE OF CODE

SPARES: Space Radiation Environment and Shielding System.

AUXILIARY ROUTINES

- (A) ELMC: Electron Monte Carlo Code.
- (B) EPEN: Electron Penetration Code.
- (C) BREMS: Bremsstrahlung Code.
- (D) HEVPART: Heavy Particle Penetration Code.
- (E) SECPRO: Secondary Proton Penetration Code.
- (F) TANDE: Trajectory and Environment Code.

2. CONTRIBUTOR

Aerospace Group, The Boeing Company, Seattle, Washington.

3. CODING LANGUAGE AND COMPUTER

FORTRAN IV; IBM 360/75/91.

Random Number Generator in CCC-148A (ELMC) is in assembly language..

4. NATURE OF PROBLEM SOLVED

(A) The ELMC Code calculates the electron number, energy, and angular fluence resulting from the incidence of a specified initial spectrum on a multilayered one-dimensional shield.

(B) Data from the ELMC Electron Monte Carlo Code have been used in EPEN to formulate analytic expressions to describe electron number penetration and the penetrating energy spectrum. Dose and spectral data are obtained for a set of initial energies, and the results are then weighted by the incident spectra of interest and summed for the final solution.

(C) The bremsstrahlung dose resulting from electrons incident on a shield is calculated in BREMS. Either one-dimensional, multi-layer slab geometries or three-dimensional geometries can be treated.

(D) HEVPART calculates the penetrating energy spectrum, LET spectrum, and absorbed dose in multilayered slabs resulting from a fluence of protons, He, or heavy ions. Results for three-dimensional geometries can also be obtained to describe space vehicle structures.

(E) The penetrating proton energy spectrum and the resulting secondary protons, neutrons, and gamma rays are calculated in SECPRO for multilayered shields. Dose and LET spectral data are also given. The recoiling nuclei dose resulting from the penetrating proton and neutron spectra are also given.

(F) TANDE was designed to calculate the Van Allen belt electron and proton fluxes and fluences encountered in or near earth trajectories.

5. METHOD OF SOLUTION

(A) ELMC employs the Monte Carlo method with angular scattering treated by the method of Leiss, Penner, and Robinson. Energy loss is treated by the continuous slowing down approximation, and energy straggling is not treated. The energy dose and angular deflections are calculated in path length segments of Δx , where Δx can be adjusted by input data and made proportional to particle energy if desired.

(B) The EPEN code calculates the absorbed dose at a point of interest caused by electrons penetrating a shielding system. The penetrating electron energy spectrum is also calculated. Multi-layer shields can be treated.

(C) The bremsstrahlung differential energy spectrum produced in a material is estimated in BREMS by an expression given by Wyard (Ref. 9). The photon energy spectrum is then transported through the remaining shielding material by the use of ray theory plus buildup factors. Two basic calculational modes are available. In the surface production option, the bremsstrahlung is all produced at the surface of the shield. In the volume production option, the attenuation of the electron spectrum is considered, and the bremsstrahlung source is volume distributed.

(D) The straight ahead approximation is used in HEVPART and nuclear interactions are neglected to provide a rapid solution of the heavy ion transport problem. The range-energy and stopping power tables of Barkas and Berger are used. Low energy correction factors are employed to describe the changes in stopping power resulting from electron capture.

(E) The first collision approximation and the straight ahead approximation are employed in SECPRO to simplify the cascade transport problem. Neutron induced protons are also calculated to refine the neutron dose estimate. The code employs the tabulated Barkas and Berger range energy data and the secondary particle production data of Bertini for numerical integration of the primary and secondary particle fluxes.

(F) The user supplies to TANDE description of a vehicle trajectory and radiation-environment data. The program calculates electron or proton flux rate and time-integrated flux along the trajectory. The general procedure is to give as input or calculate trajectory points and then compute radiation flux at these points.

Given a description of the orbit and the point of injection, subject trajectory points are calculated as a function of time, using orbital flight equations. The trajectory points are converted to McIlwain's geomagnetic coordinates (B,L, and R, λ).

Proton or electron flux at each point is determined by a table lookup and interpolation. Numerical integration (in conjunction with an interpolation scheme on B and L) gives a time-integrated flux for each point. A table lookup and interpolation on an array of spectral coefficients determines the spectral coefficients for the point. The flux at the point, dose-conversion factors, and the spectral coefficients are then used to determine dose rate and total dose at the point.

Angular distribution is determined for each trajectory point by solution of a pitch angle distribution function.

The code is designed so that new experimental data on the radiation environment and on the interaction of radiation with matter can be accepted.

The following general methods are followed:

1. calculation of the spacecraft trajectory in B, L, and t coordinates,
2. devising a mathematical representation of the space-radiation environment, including geomagnetically trapped radiation (Van Allen belts), solar particle event radiation, and galactic cosmic radiation,
3. determination of the radiation flux and energy spectra encountered in a given space mission.

6. RESTRICTIONS OR LIMITATIONS

(A) Validity of results in ELMC is dependent on the choice of Δx , and the number of histories.

(B) The basic accuracy of EPEN is determined by the Monte Carlo data. In addition, the analytic fits developed have ranges of validity.

(C) The volume source options in BREMS must be carefully chosen to match the electron energy and shield configuration.

(D) As secondary interactions are neglected in HEVPART, the shield should then be compared to the mean free path of the ion.

(E) In SECPRO secondary data is provided only for aluminum and H₂O. The shield thickness must be smaller than a proton mean free path to remain within the valid range of the first collision approximation.

(F) The principle restriction in the use of TANDE is that the trajectories selected for analysis can only be evaluated at B, L points described by the flux maps.

7. TYPICAL RUNNING TIME

No statistics are available to determine typical running time. Estimated running time for the sample problems are tabulated below.

	Code	Core Size (GO STEP)	Running Time (Sec) IBM 360/91
(A)	ELMC	246 K	49.54
(B)	EPEN	162 K	0.96
(C)	BREMS	184 K	0.72
(D)	HEVPART	186 K	0.82
(E)	SECPRO	296 K	3.11
(F)	TANDE	314 K	5.09

8. COMPUTER HARDWARE REQUIREMENTS

The codes were designed to run on the IBM 360 with standard I-O and a maximum of 3 tape units or direct access devices.

9. COMPUTER SOFTWARE REQUIREMENTS

The codes are operable on the IBM 360/75/91 Operating System using OS-360 FORTRAN H Compiler.

A random number generator is required in ELMC and is included on the master tape.

10. REFERENCES

1. Packaged:

Paul G. Hahn, "Space Radiation Environment System," AS 2807 (1969).

2. Background information:

John A. Barton, B. W. Mar, G. L. Keister, W. R. Doherty, J. R. Benbrook, W. R. Sheldon, J. R. Thomas, K. Moriyasu, and M. C. Wilkinson, "Computer Codes for Space Radiation Environment and Shielding," Volume I and II (August 1964).

11. CONTENTS OF CODE PACKAGE

The package contains the following items:

- a. the first referenced document,
- b. a reel of magnetic tape on which is written in 19 files:
the BCD source card deck, BCD input for a sample problem, and
output from the sample problem for each code in the package -
A through F.

12. HOW TO OBTAIN PACKAGE

Inquiries or requests for the code package may be mailed to

CODES COORDINATOR
Radiation Shielding Information Center
Oak Ridge National Laboratory
Post Office Box X
Oak Ridge, Tennessee 37830

or telephoned to

Area Code 615; 483-8611, extension 3-6944, or to
FTS xx-615-483-6944.

Persons requesting the code package should send a reel of magnetic
tape to the above address.

13. DATE OF ABSTRACT

February 1972.

RSIC CODE PACKAGE CCC-149

1. NAME AND TITLE OF CODE

GASOUT: Calculation of Gaseous Fission Product Release for a ZPR-6 and -9 Design Basis Accident.

The code was used in preparing the Plutonium Safety Analysis Report for Argonne National Laboratory's ZPR-6 and -9 critical facilities (Ref. 2).

2. CONTRIBUTOR

Applied Physics Division, Argonne National Laboratory, Argonne, Illinois.

3. CODING LANGUAGE AND COMPUTER

FORTRAN IV; IBM 360.

4. NATURE OF PROBLEM SOLVED

GASOUT calculates the amount of noble gas and radioiodine fission products expelled into the atmosphere following a "Design Basis Accident" (DBA) in a ZPR reactor cell. The DBA is an extremely severe nuclear accident which is postulated and analyzed to determine the containment capability of the reactor cell.

The code considers a simplified model of the reactor facility consisting of a containment cell and an exhaust system including a filter and stack. Heat generated in the cell by burning fuel and sodium following the excursion and the pressure dependent flow rate of cell air through the exhaust system are supplied as input to the code. The code assumes that all fissions giving rise to fission product release occur at the same time and that these fission products are spread uniformly throughout the cell as soon as the fuel begins to burn. A time dependent calculation of cell pressure and temperature determines the flow of air from the cell. The fission products are released from the cell along with the air.

5. METHOD OF SOLUTION

Small time increments are used for the calculation. A constant volume process using the perfect gas law determines the pressure and temperature rise due to heat addition. Following the removal of part of the air through the exhaust system, isothermal expansion of the remaining air causes the pressure to decrease. The process is repeated for successive time increments.

The number of noble gas or radioiodine fission product and precursor atoms in the cell during each time increment is obtained by interpolation from data listing fission product activities versus decay time following fission (Ref. 3).

Options in the code allow the user to consider or ignore the holdup of precursor atoms in the filter and to specify instantaneous release of all fission products or a constant rate of release for one hour.

6. RESTRICTIONS OR LIMITATIONS

6 isotopes per fission product decay chain, and
999 time increments.

7. TYPICAL RUNNING TIME

A typical running time is approximately 1 second per fission product isotope (CPU time).

Estimated running time of the sample problem on the IBM 360:
0.5 seconds.

8. COMPUTER HARDWARE REQUIREMENTS

The code uses 250K bytes of the IBM 360 CPU. No auxiliary storage is required.

9. COMPUTER SOFTWARE REQUIREMENTS

The IBM FORTRAN IV (level H) Compiler for the 360.

10. REFERENCES

C. D. Swanson and E. M. Bohn, "GASOUT - The Code Used to Calculate Gaseous Fission Product Release for a ZPR-6 and -9 Design Basis Accident," ANL-7534 (January 1970).

W. Y. Kato et al., "Final Safety Analysis Report on the Use of Plutonium in ZPR-6 and -9," ANL-7442 (March 1969).

D. B. Kochendorfer, "Calculated Activities of ^{235}U Fission Products for Very Short Nuclear Reactor Operation," Vol. II, USNRDL-TR-757 (June 1964).

11. CONTENTS OF CODE PACKAGE

The package contains the following items:

- a. the first referenced document,
- b. a reel of magnetic tape on which is written in 3 files: the source card deck, input for a sample problem, and an output listing from running the problem.

12. HOW TO OBTAIN PACKAGE

Inquiries or requests for the code package may be mailed to

CODES COORDINATOR
Radiation Shielding Information Center
Oak Ridge National Laboratory
Post Office Box X
Oak Ridge, Tennessee 37830

or telephoned to

Area Code 615; 483-8611, extension 3-6944, or to
FTS xx 615-483-6944.

Persons requesting the code package should send a reel of magnetic tape to the above address.

13. DATE OF ABSTRACT

February 1972.



RSIC CODE PACKAGE CCC-150

1. NAME AND TITLE OF CODE

MAP: Kernel Integration Code in Complex Geometry With Special Application to Surface Sources Determined by Discrete Ordinates Calculations.

Input to MAP may be supplied by CCC-89/DOT and by GAMLEG, packaged in CCC-42/DTF-IV.

2. CONTRIBUTOR

Westinghouse Astronuclear Laboratory, Pittsburgh, Pennsylvania.

3. CODING LANGUAGE AND COMPUTER

CCC-150A: FORTRAN IV; CDC 6600.

CCC-150B: FORTRAN IV; IBM 360/75/91.

4. NATURE OF PROBLEM SOLVED

MAP solves for the last flight radiation transport in an r, z, θ geometry. The MAP code solves for the nuclear radiation transport to a detector surface from an energy- and angular-dependent surface source defined by the surface leakage data of a DOT-IIW discrete ordinate transport solution in r, z geometry. During the numerical integration over the surface source, the radiation transport can be through a void or, at option, radiation transport through a three-dimensional geometry described by intersecting quadratic surfaces can be solved. Uncollided energy- and angular-dependent, neutron or photon flux results at a detector plane are provided with the optional use of buildup factor techniques to estimate multiple scattering of photons available. Cross section data for use in the radiation transport can be obtained from (1) internal calculations of point value data from Klein-Nishina relationships and tables for photons, or (2) input values of macroscopic cross sections for materials in zones. General discrete ordinate quadrature data can be used as surface source input with two techniques available to calculate the angular-dependent source data from discrete ordinate data.

5. METHOD OF SOLUTION

A variable-interval numerical integration technique is used to integrate the visible surface angular leakage flux at each detector point. Energy dependence is treated as the multigroup data supplied to the code. The code is applicable to neutron, photon, or coupled neutron-photon analysis; the numerical integration accuracy is dependent upon the input discrete ordinate data.

The code employs the r,z geometry discrete ordinate transport leakage angular flux data from the DOT-IIW code to calculate neutron and photon transport in a void or in a purely attenuating geometry to provide energy and angular dependent data at a detector surface. The coupling of DOT-IIW calculations through voids is handled by the MAP code. Photon absorption cross sections may be supplied by a GAMLEG-W library tape. Discrete ordinate transport angular leakage flux data are supplied by the DOT-IIW code as the scalar flux output tape.

6. RESTRICTIONS OR LIMITATIONS

The MAP code uses complete, flexible dimensioning to facilitate dynamic core storage allocation at execution time and during various phases of the calculation. Because of the use of a flexible dimensioning technique for each array, during a specific phase of the calculation, no size restriction is imposed for a given array. The only restriction is the size of the sum of all array storage required during a specific phase of the calculation. The required storage size for a given problem may be exactly computed as indicated in the documentation. In using the code on the CDC 6600, the total amount of core storage for a given problem may be specified at execution time. For the IBM 360/75 and UNIVAC 1108 computers, the size of BLANK COMMON must be compiled with a fixed dimension in the main subroutine.

7. TYPICAL RUNNING TIME

Running time requirements for the CDC 6600 version of the MAP code can be approximated by the following relationship:

$$t \text{ (second CPU)} = \frac{\text{no. of source-to-detector point calculations}}{250 \text{ to } 600 \text{ calculations/second}}$$

Estimated running time of the sample problem on the IBM 360/91: 6.5 seconds.

8. COMPUTER HARDWARE REQUIREMENTS

MAP is operable on both CDC 6600 and IBM 360 computers. The source program requires 22K decimal locations; the remaining locations are used for problem data storage. Up to six tape or disk devices are required in addition to input, output, and punch disks.

9. COMPUTER SOFTWARE REQUIREMENTS

The MAP code is written in standard USASI FORTRAN IV. With minor change the code can be used on the CDC 6600, IBM 360/75, or UNIVAC 1108 computers. It is operational under the CDC 6600 SCOPE 3.2 Monitor System, the IBM 360/75 Release 18 Monitor System and the UNIVAC 1108 EXEC8 Monitor System.

10. REFERENCES

R. K. Disney, R. G. Soltesz, J. Jedruch, and S. L. Zeigler, "Code Description and User's Manual for the MAP Radiation Transport Computer Code," WANL-TME-2706 (August 1970).

R. G. Soltesz, R. K. Disney, and G. Collier, "User's Manual for the DOT-IIW Discrete Ordinates Transport Computer Code," WANL-TME-1982 (December 1969).

F. R. Mynatt, F. J. Muckenthaler, and P. N. Stevens, "Development of Two-Dimensional Discrete Ordinates Transport Theory for Radiation Shielding," CTC-INF-952 (August 1969).

R. G. Soltesz, R. K. Disney, and S. L. Zeigler, "Cross Section Generation and Data Processing Techniques," WANL-PR(LL)-034, Volume 3 (August 1970).

11. CONTENTS OF CODE PACKAGE

The package contains the following items:

- a. the first referenced document,
- b. a reel of magnetic tape on which is written in several files:
the source card deck, input for a sample problem and an output listing from running the sample problem for each version of the code.

12. HOW TO OBTAIN PACKAGE

Inquiries or requests for the code package may be mailed to

CODES COORDINATOR
Radiation Shielding Information Center
Oak Ridge National Laboratory
Post Office Box X
Oak Ridge, Tennessee 37830

or telephoned to

Area Code 615; 483-8611, extension 3-6944, or to
FTX xx-615-483-6944.

Persons requesting the code package should send a reel of magnetic tape to the above address.

13. DATE OF ABSTRACT

February 1972.

RSIC CODE PACKAGE CCC-151

1. NAME AND TITLE OF CODE

DOT2DB: Two-Dimensional Multigroup Discrete Ordinates Transport/Diffusion Code with Anisotropic Scattering.

DOT2DB is a conglomerate of the diffusion subroutines of the BNWL computer code 2DB and the ORNL transport code DOT (CCC-89).

2. CONTRIBUTORS

General Electric Company, (BRD), Sunnyvale, California.
Battelle Northwest Laboratory, Richland, Washington.
Oak Ridge National Laboratory, Oak Ridge, Tennessee.

3. CODING LANGUAGE AND COMPUTER

FORTTRAN IV; GE-635.

4. NATURE OF PROBLEM SOLVED

DOT2DB solves both the multigroup discrete ordinates transport theory and the multigroup diffusion theory equations in two dimensions. Anisotropic scattering of any order Legendre expansion is allowed in the transport theory option. Anisotropic scattering in the diffusion theory option is treated with the "transport" approximation, using the P_1 scattering matrix, when provided, to calculate the transport cross sections. Options include solutions in (x,y), (r,z), (r, θ), and, in the diffusion theory option, triangular geometries. Both direct and adjoint fluxes may be computed for fixed volume-distributed source, multiplication constant iteration, time absorption iteration, concentration search, zone thickness search, and fixed boundary source problems. In addition to the fixed boundary source problem, options include vacuum, reflection, periodic and white boundary conditions. Cross sections may be entered from cards or from tape in the DTF format.

Activities for any material in the system may be output by interval (optional) and zone. Other output includes the interval fluxes and sources and a reaction summary table for each zone and for the system.

5. METHOD OF SOLUTION

DOT2DB has three iteration levels. The innermost iteration level computes the spatial flux distribution within an energy group. Convergence at this level may be accelerated by extrapolation of the scalar flux (and, in the transport theory option, the flux moments) by an input factor. The second iteration level computes the energy spectrum of the flux and the multiplication constant or time absorption eigenvalues. Convergence at this level may be accelerated by extrapolation of the fission source distribution by an input factor. The outermost iteration level searches for the material concentration or zone thickness eigenvalues of the problem.

6. RESTRICTIONS OR LIMITATIONS

Variable dimensioned arrays are used exclusively within the program. Hence, the only problem size restriction is dictated by the amount of available computer memory.

7. TYPICAL RUNNING TIME

Running time varies widely, depending on the problem size and the options chosen. An S_6 , one group, 10×10 spatial mesh, multiplication constant iteration problem required 212 seconds of computer processor time. The same problem in diffusion theory required 13 seconds.

8. COMPUTER HARDWARE REQUIREMENTS

The code is designed for the GE-635 computer and can be made to run in various memory sizes due to its variable dimensioning. The

code when operating at 45K handles a majority of desired problem sizes.

Seven scratch files are required and optionally one file tape. Each scratch file consists of approximately 80,000 to 100,000 words. Two output files are used (06,33), requiring the use of a printer.

9. COMPUTER SOFTWARE REQUIREMENTS

DOT2DB runs under the GECOS-III operating system which precludes any operator intervention. The code requires segmentation (chaining - linking). It operates under time surveillance with the following subroutines which are written in GMAP, the GE-635 assembly language.

CLOCK - Returns current processor time in seconds.

MYTIME - Returns user time limits in hundredths of an hour.

LIMRIN - When run time is exceeded, this GE routine will allow the user to obtain five more minutes of processor time.

10. REFERENCES

R. Protsik, "User's Manual for DOT2DB: A Two-Dimensional Multi-group Discrete Ordinates Transport/Diffusion Code with Anisotropic Scattering," GEAP-13537 (September 1969).

W. W. Little, Jr., and R. W. Hardie, "2DB, A Two-Dimensional Diffusion Burnup Code for Fast Reactor Analysis," BNWL-640 (January 1968).

W. W. Little, Jr., and R. W. Hardie, "2DB User's Manual," BNWL-831 (July 1968).

W. W. Little, Jr., and R. W. Hardie, "2DB User's Manual - Revision 1," BNWL-831 Rev. 1 (February 1969).

F. R. Mynatt, "A User's Manual for DOT - A Two-Dimensional Discrete Ordinates Transport Code with Anisotropic Scattering," K-1694 (January 1967).

11. CONTENTS OF CODE PACKAGE

The package contains the following items:

- a. the first referenced document,
- b. a reel of magnetic tape on which is written in 2 files:
the BCD source card decks and BCD input for a sample
problem.

12. HOW TO OBTAIN PACKAGE

Inquiries or requests for the code package may be mailed to

CODES COORDINATOR
Radiation Shielding Information Center
Oak Ridge National Laboratory
Post Office Box X
Oak Ridge, Tennessee 37830

or telephoned to

Area Code 615; 483-8611, extension 3-6944, or to
FTS xx-615-483-6944.

Persons requesting the code package should send a reel of magnetic
tape to the above address.

13. DATE OF ABSTRACT

February 1972.

RSIC CODE PACKAGE CCC-152

1. NAME AND TITLE OF CODE

ALGAM: Monte Carlo Estimation of Internal Dose from Gamma-Ray Sources in a Phantom Man.

2. CONTRIBUTOR

Oak Ridge National Laboratory, Oak Ridge, Tennessee.

3. CODING LANGUAGE AND COMPUTER

FORTRAN IV, Assembly Language; IBM 360/75/90.

4. NATURE OF PROBLEM SOLVED

ALGAM computes dose deposited by gamma rays from an external or internal source in various sections of a phantom approximating the body of a man.

5. METHOD OF SOLUTION

The body is considered to be made up of three kinds of tissue: soft tissue, bone, and lung. The phantom is divided into a number of sections and each is given a reasonably typical tissue composition. The user can specify a source arbitrarily through a subroutine. Monte Carlo calculations using the OGRE System (see CCC-46) are employed for particle transport. In addition to options available in OGRE, Woodcock's scheme of the artificial "nothing" cross section is employed to make the total cross section a constant throughout the phantom.

Doses are determined from photon energy deposited at each interaction point.

6. RESTRICTIONS OR LIMITATIONS

The user must provide his own SOURCE routine; he must provide a pseudo random number generator, FUNCTION FLTRNF (dummy), appropriate to his own computer.

7. TYPICAL RUNNING TIME

To run histories of 50,000 photons would take between 3 minutes and 15 minutes on a IBM 360/91 computer depending upon the initial energy of the photons. The code occupies about 185,000 bytes of storage in a IBM 360/91.

Estimated running time of the packaged sample problem on the IBM 360/91: 4 minutes.

8. COMPUTER HARDWARE REQUIREMENTS

IBM 360, Standard I-O and 1 tape/disk unit.

9. COMPUTER SOFTWARE REQUIREMENTS

The code is operable on the IBM 360/75/91 Operating System using OS-360 Fortran H Compiler. A random number generator is required and is supplied in Assembly Language. A table of cross sections are required and DLC-7C is included on the master tape.

10. REFERENCES

G. G. Warner and A. M. Craig, Jr., "ALGAM, A Computer Program for Estimating Internal Dose for Gamma-Ray Sources in a Man Phantom," ORNL-TM-2250 (June 1968).

M. J. Cook, W. S. Snyder, and G. G. Warner, "Estimates of Dose to Red Bone Marrow From Monoenergetic Sources of Photons in Lungs and Other Organs," reprinted from Health Physics Division Annual Progress Report for Period Ending July 31, 1970, ORNL-4584.

W. S. Snyder, H. L. Fisher, Jr., Mary R. Ford, and G. G. Warner, "Estimates of Absorbed Fractions for Monoenergetic Photon

Sources Uniformly Distributed in Various Organs of the Heterogeneous Phantom," MM/MIRD Phamphlet No. 5 (Aug. 1969, Vol. 10).

11. CONTENTS OF CODE PACKAGE

The package contains the following items:

- a. the referenced documents,
- b. a reel of magnetic tape on which is written in 6 files:
DLC-7C/HPIC cross sections in BCD, the source card decks,
input cards for a sample problem and the output listing
from running the problem.

12. HOW TO OBTAIN PACKAGE

Inquiries or requests for the code package may be mailed to

CODES COORDINATOR
Radiation Shielding Information Center
Oak Ridge National Laboratory
Post Office Box X
Oak Ridge, Tennessee 37830

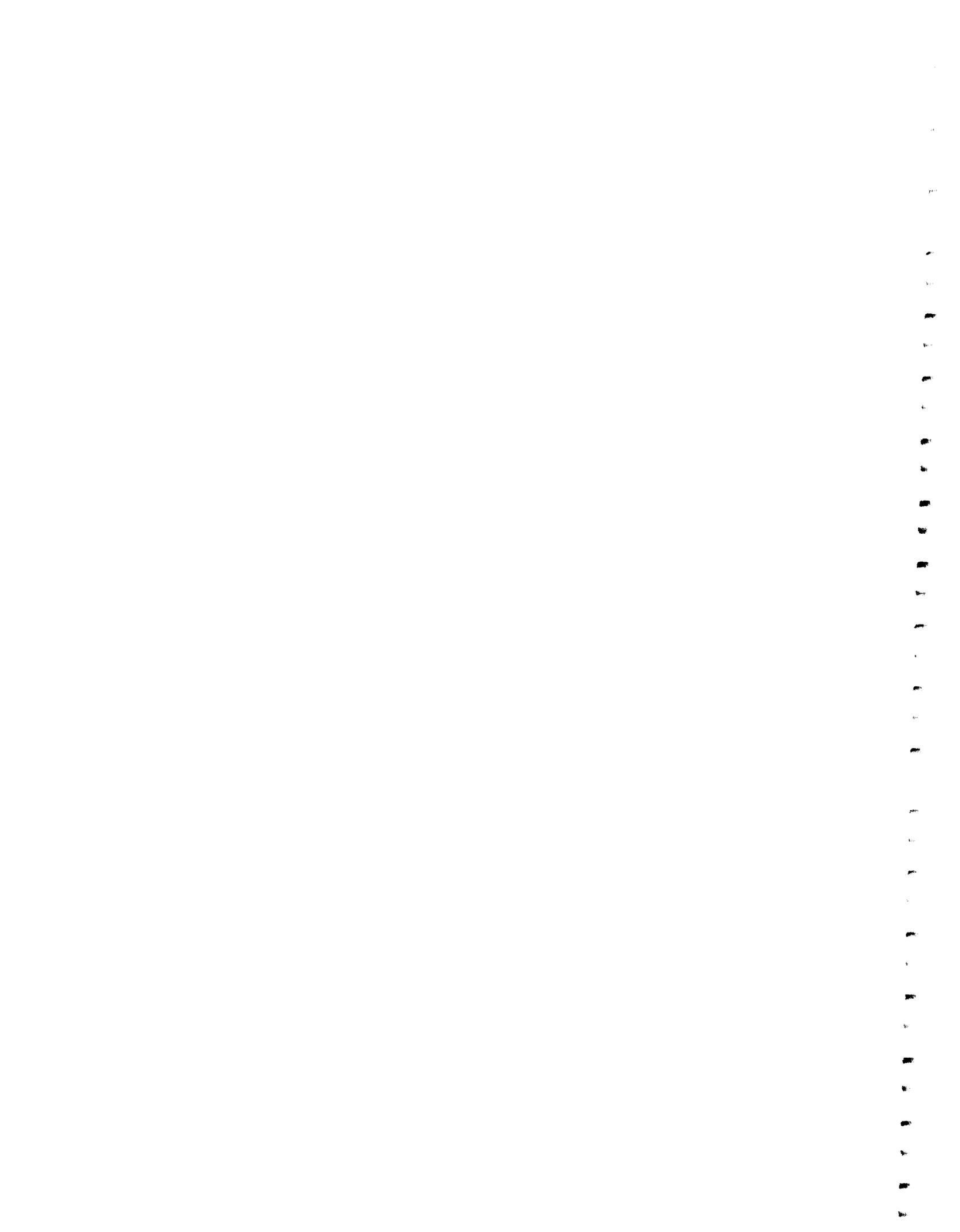
or telephoned to

Area Code 615; 483-8611, extension 3-6944, or to
FTS xx-615-483-6944.

Persons requesting the code package should send a reel of magnetic tape to the above address.

13. DATE OF ABSTRACT

February 1972.



RSIC CODE PACKAGE CCC-153

1. NAME AND TITLE OF CODE

ASFIT: Neutron and Gamma-Ray Transport Code for One-Dimensional Finite Systems.

2. CONTRIBUTOR

Bhabha Atomic Research Center, Bombay, India.

3. CODING LANGUAGE AND COMPUTER

FORTRAN 63; CDC 3600.

4. NATURE OF PROBLEM SOLVED

ASFIT was designed to solve problems of neutron and gamma-ray transport in one-dimensional finite systems. The method is applicable to multivelocidity, multiregion systems with arbitrary degree of anisotropy.

5. METHOD OF SOLUTION

A semi-analytical technique is used. The transport equation is written in the form of coupled integral equations separating the spatial and energy-angular transmissions. Legendre polynomial approximation in the direction cosine, and discrete ordinate representation in energy and spatial domain are used for radiation source and flux. The space and energy-angle transmission kernels are evaluated analytically and the integral equations are then solved by a fast-converging iterative technique. For a plane parallel beam of radiation incident on a slab, the virgin and the first collision flux are not amenable to polynomial expansion due to the singularities. For such a case, up to second collision, source is computed analytically and then recourse is taken to polynomial approximation.

6. RESTRICTIONS OR LIMITATIONS

None noted.

7. TYPICAL RUNNING TIME

No study of typical running time has been made by RSIC.

8. COMPUTER HARDWARE REQUIREMENTS

The code was designed for the CDC 3600 with 32K memory.

9. COMPUTER SOFTWARE REQUIREMENTS

The code may be compiled and executed in the standard CDC 3600 operating system.

10. REFERENCE

D. V. Gopinath and K. Santhamam, "Radiation Transport in One-Dimensional Finite Systems, Part I, Development of the Anisotropic Source-Flux Interaction Technique," BARC unpublished Ms (1971).

11. CONTENTS OF CODE PACKAGE

The package contains the following items:

- a. the referenced document,
- b. a reel of magnetic tape on which is written in one file:
the source card deck and an output listing of a sample problem.

12. HOW TO OBTAIN PACKAGE

Inquiries or requests for the code package may be mailed to

CODES COORDINATOR
Radiation Shielding Information Center
Oak Ridge National Laboratory
Post Office Box X
Oak Ridge, Tennessee 37830

or telephoned to

Area Code 615; 483-8611, extension 3-6944, or to
FTS xx-615-483-6944.

Persons requesting the code package should send a reel of magnetic
tape to the above address.

13. DATE OF ABSTRACT

February 1972.

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RSIC CODE PACKAGE CCC-154

1. NAME AND TITLE OF CODE

ANDYMG3: Monte Carlo Time-Dependent Particle and Photon
Transport Code - General Geometry.

ANDYMG3 is the basic program of a series, each of which is
intended to be small, fast, and system-independent.

2. CONTRIBUTOR

Los Alamos Scientific Laboratory, Los Alamos, New Mexico.

3. CODING LANGUAGE AND COMPUTER

FORTRAN IV; CDC 6600 and UNIVAC 1108.

4. NATURE OF PROBLEM SOLVED

Particle or photon transport is calculated in a general
geometry using group cross sections in usual format. Output in-
cludes tallies of the weight of particles colliding or crossing
surfaces in bins indexed by time, energy group, and order number.
An arbitrary tally array is also provided. Source emission may be
delayed.

Any complex geometry formed by segments of planes, spheres,
cylinders, ellipsoids, and cones can be treated.

5. METHOD OF SOLUTION

The Monte Carlo method is employed with splitting permitted.

6. RESTRICTIONS OR LIMITATIONS

Cross section expansion $\leq P_3$.

7. TYPICAL RUNNING TIME

CDC 6600: 0.5-2 msec/collision. Estimated running time of the sample problem on the CDC 6600: 61 seconds, CP time.

8. COMPUTER HARDWARE REQUIREMENTS

The code was designed for CDC 6600 and UNIVAC 1108 standard hardware.

9. COMPUTER SOFTWARE REQUIREMENTS

Intended to be system-independent, ANDYMG3 may be compiled and executed in standard FORTRAN IV operating systems.

10. REFERENCES

D. R. Harris, "ANDYMG3, The Basic Program of a Series of Monte Carlo Programs for Time-Dependent Transport of Particles and Photons," LA-4539 (October 1970).

D. R. Harris, "ANDY1G2 and ANDY1R2, Monte Carlo Programs for Time-Dependent Monoenergetic Particle Transport in General Geometries and Repeating Arrays," LA-4410 (August 1970).

11. CONTENTS OF CODE PACKAGE

The package contains the following items:

- a. the first referenced document,
- b. a reel of magnetic tape on which is written in 2 separate files: the source card deck, BCD input for a sample problem, and a BCD library of cross sections.

12. HOW TO OBTAIN PACKAGE

Inquiries or requests for the code package may be mailed to

CODES COORDINATOR
Radiation Shielding Information Center
Oak Ridge National Laboratory
Post Office Box X
Oak Ridge, Tennessee 37830

or telephoned to

Area Code 615; 483-8611, extension 3-6944, or to
FTS xx-615-483-6944.

Persons requesting the package should send a reel of magnetic tape
to the above address.

13. DATE OF ABSTRACT

February 1972.



RSIC CODE PACKAGE CCC-155

1. NAME AND TITLE OF CODE

ELTRAN: One-Dimensional Monte Carlo Electron Transport Code.

ELTRAN was originally developed at the Boeing Company, Seattle, was extended by Avco Corporation, and evolved into this version.

2. CONTRIBUTOR

Sandia Laboratories, Albuquerque, New Mexico.

Ion Physics, Burlington, Massachusetts.

3. CODING LANGUAGE AND COMPUTER

FORTRAN IV; CDC 3600.

4. NATURE OF PROBLEM SOLVED

ELTRAN is designed to compute one-dimensional, electron deposition, in semi-infinite composite slabs. It has the capability of computing either the energy deposition profile distribution of stopped electrons or the spectral properties of electrons transmitted by a thin slab, which may be solid, liquid, or gas. The incident electrons may be monoenergetic or they may form a spectrum; they may be incident perpendicular or at an angle to the slab.

5. METHOD OF SOLUTION

ELTRAN employs a quasi Monte Carlo scheme which involves calculation of the trajectory of an individual electron. This calculation is repeated until a sufficient number of electrons or case histories has been accumulated so that the user obtains the desired degree of statistical precision. Instead of considering each of the tens of thousands of individual collisions, the electron is given an incremental displacement, and the small energy loss and small effective scattering angle are computed. The electron is given additional incremental displacements until the electron is

absorbed (stopped), backscattered or transmitted. Again this incrementation is repeated for additional electrons until the desired number of case histories is run.

The energy loss is computed using the continuous slowing-down approximation, with the effects of density, range straggling, and bremsstrahlung included. The scattering angle is computed by means of the Moliere multiple scattering angle distribution.

6. RESTRICTIONS OR LIMITATIONS

None noted.

7. TYPICAL RUNNING TIME

No study of typical running time has been made by RSIC.

8. COMPUTER HARDWARE REQUIREMENTS

The code was designed for the CDC 3600 with standard equipment.

9. COMPUTER SOFTWARE REQUIREMENTS

The code is operable on the CDC 3600 FORTRAN operating system.

10. REFERENCE

F. H. Brittain, "A Description of ELTRAN: An Electron Deposition Code," SC-TM-68-713 (January 1969).

11. CONTENTS OF CODE PACKAGE

The package contains the following items:

- a. the referenced document,
- b. a reel of magnetic tape on which is written in 2 files:
the BCD source card deck and the input cards for a sample problem.

12. HOW TO OBTAIN PACKAGE

Inquiries or requests for the code package may be mailed to

CODES COORDINATOR
Radiation Shielding Information Center
Oak Ridge National Laboratory
Post Office Box X
Oak Ridge, Tennessee 37830

or telephoned to

Area Code 615; 483-8611, extension 3-6944, or to
FTS xx-615-483-6944.

Persons requesting the code package should send a reel of magnetic tape to the above address.

13. DATE OF ABSTRACT

February 1972.



RSIC CODE PACKAGE CCC-156

1. NAME AND TITLE OF CODE

MECC-3: Medium-Energy Intranuclear Cascade Code System.

AUXILIARY ROUTINES

I4C: Cascade Data Analysis Code, combined form of 4 codes:
Analysis Code I, Analysis Code II, Evaporation Code, and
Angular Momentum Code.

Nuclear Configuration Code: Update Code for Nuclear Configura-
tion-Cross Section Library.

SPECIAL LIBRARIES OF DATA

Nuclear Configuration Data and Cross Sections for MECC-3.

Evaporation Table for EVAP.

2. CONTRIBUTOR

Neutron Physics Division, Oak Ridge National Laboratory, Oak
Ridge, Tennessee.

3. CODING LANGUAGE AND COMPUTER

FORTRAN IV; IBM 360/75/91. A few subroutines are in Assembly
Language.

4. NATURE OF PROBLEM SOLVED

MECC-3 calculates the results of nuclear reactions caused by a
medium-high energy particle colliding with a nucleus. The incident
particles may be protons or neutrons with energies from about 100
to 2500 MeV or charged pions with energies from about 100 to 1500
MeV. Target nuclei may be any element heavier than carbon. MECC-3
writes a history tape containing data on the properties of the
particles escaping from the nucleus as a result of the particle-
nucleus collision. The data consist of the type of escaping parti-
cles, their energies, and angles of emission.

The associated analysis code I4C utilizes the data on the MECC-3 history tape to calculate particle multiplicities and various cross sections, such as the nonelastic cross section or the doubly-differential cross section for energy-angle correlated distributions. I4C also carries the nuclear reaction through an additional phase, that of evaporation, and calculates evaporation residual nuclei (radiochemical) cross sections and the particle multiplicities and energy spectra of particles "boiled off" from the nucleus after the cascade has stopped.

5. METHOD OF SOLUTION

The code system is based on the assumption that nuclear reactions involving high-energy particles can be described in terms of particle-particle collisions within the nucleus. The life history of each individual particle is traced as the incident particle, and the subsequent generations of particles involved in collisions, wind their way through the nucleus. The point of collision, the type of collision, the momentum of the struck nucleon, and the scattering angles for each collision are determined by statistical sampling techniques. Free-particle experimental data are used whenever cross-section data are required. Cross sections and distributions resulting from the nuclear reaction are calculated in the associated analysis code I4C by taking the average value of many results.

6. RESTRICTIONS OR LIMITATIONS

There are no known restrictions implied by storage allocation. Target nuclei may be any element from ${}^4\text{He}$ to ${}^{239}\text{Pu}$. Incident particle energies must be 1 MeV or greater. Incident nucleon energies must be less than 3500 MeV, and incident pion energies must be less than 2500 MeV. The results are expected to be valid only over the ranges described in section 4. The maximum possible number of incident particle histories is 999,999.

7. TYPICAL RUNNING TIME

The approximate running times on the IBM 360/91 per 1000 incident particles range from 0.26 minutes for a 1-GeV particle on oxygen to 3.3 minutes for the same energy particle on lead, and from 0.4 minutes for a 2.5-GeV particle on oxygen to 6.2 minutes for the same energy particle on lead. The I4C analysis code with all options takes approximately 1/5 of the MECC-3 running time.

Estimated running times for the packaged sample problems on the IBM 360/75, GO STEP: Configuration Code, 14 seconds; MECC-3, 12 minutes; I4C, 3 minutes.

8. COMPUTER HARDWARE REQUIREMENTS

The codes were designed for the IBM 360/75/91 with standard I-O and a maximum of 6 tape units or direct access devices. Maximum core size required \sim 1020K.

9. COMPUTER SOFTWARE REQUIREMENTS

The codes operate on the IBM 360/75/91 Operating System using OS-360 FORTRAN H Compiler.

I4C uses the overlay feature and was compiled with OPT = 2.

10. REFERENCES

H. W. Bertini, M. P. Guthrie, and O. W. Hermann, "Instructions for the Operation of Codes Associated with MECC-3, A Preliminary Version of an Intranuclear-Cascade Calculation for Nuclear Reactions," ORNL-4564 (May 1971).

Hugo W. Bertini, Phys. Rev. 188, 1711 (1969).

11. CONTENTS OF CODE PACKAGE

The package contains the following items:

- a. the referenced documents,

- b. a reel of magnetic tape on which is written in 19 files:
the data libraries in BCD, the BCD source card decks, BCD
input cards for a sample problem for each code and a BCD out-
put listing from running the sample problems.

12. HOW TO OBTAIN PACKAGE

Inquiries or requests for the code package may be mailed to

CODES COORDINATOR
Radiation Shielding Information Center
Oak Ridge National Laboratory
Post Office Box X
Oak Ridge, Tennessee 37830

or telephoned to

Area Code 615; 483-8611, extension 3-6944, or to
FTS xx-615-483-6944.

Persons requesting the code package should send a reel of magnetic
tape to the above address.

13. DATE OF ABSTRACT

February 1972.

RSIC CODE PACKAGE CCC-157

1. NAME AND TITLE OF CODE

MEVDP: Primary Radiation Transport Code - Complex Geometry -
Computerized Anatomical Model Man.

MEVDP is a modification of EVDP, a code developed to calculate primary radiation doses from protons and alpha particles associated with solar-flare particle events and the earth's trapped radiation for deep space and earth orbital missions.

AUXILIARY DATA

Standing Model Man.

Seated Model Man.

2. CONTRIBUTORS

Air Force Weapons Laboratory, Albuquerque, New Mexico.

Space Division, North American Rockwell Corp., Downey, California.

Martin Marietta Corporation, Denver, Colorado.

3. CODING LANGUAGE AND COMPUTER

FORTRAN IV; CDC 6600.

4. NATURE OF PROBLEM SOLVED

MEVDP generates ordered path-length areal densities for primary electron, electron-bremsstrahlung, and secondary particle radiation transport calculations. The code also generates standard-material areal-density distribution functions for proton and heavy ionizing nuclear radiation. The primary and secondary areal-density functions can be used for particle transport calculations to compute emergent fluxes and energy deposition. The code has been successfully run with the complex Apollo command and service modules and the lunar module, which are represented by 1000 elemental volume shield configurations.

The Computerized Anatomical Model Man is a detailed representation of the radiation transport properties of the human body. It is to be used for computation of the areal density distribution of specified locations in the body. This information is applicable to dose calculations in natural, weapon, reactor, and other radiation environments. The model has two configurations -- standing and seated. Over 2200 individual geometrical shapes have been used to depict the external conformation, the skeleton, and the principal organs. The exterior dimensions are those of the 50th percentile Air Force man; the skeleton and organs were scaled from life-size models to conform to the exterior. The model includes variations of material density and fractional composition by weight due to the principal chemical elements contained in muscle, bone, bone marrow, and organ tissue.

5. METHOD OF SOLUTION

MEVDP generates spatially oriented arrays of material type and associated thicknesses which are encountered by radiation traveling toward a dosimeter point within any complex geometrical shielding configuration. These thickness arrays are available in the order in which the incident rays encounter the different materials. The program computes the standard-material areal-density distribution function versus fractional solid angle for heavy charged particle dose computations. The code contains the source ray selection and geometrical options of the original EVDP, including the composite shield routine. These features have been modified and augmented by a subroutine to order the computed nuclear transport parameters. The ordered arrays can be used for electron and secondary transport calculations with the straight-ahead approximation.

To interface with MEVDP, the Computerized Anatomical Model Man conforms to two principal requirements. First, the geometrical configuration of the model has been established with combinations of the geometrical shapes recognized by MEVDP. These shapes are the hexahedron - a figure comprised of six planar surfaces: right circular cylinder; sphere; hemisphere; right circular cone, truncated right circular cone, and the ellipsoid - truncated by two or fewer parallel

planes. Second, the two punched card decks that represent the standing and seated positions of the man are in the format required for operation with MEVDP. In addition to geometrical data, these cards contain the identity of the shield - a four-digit number, a three-digit number identifying the shield material, the material density, and a number indicating the number of individual shields comprising a composite shield.

6. RESTRICTIONS OR LIMITATIONS

None noted.

7. TYPICAL RUNNING TIME

No study of typical running time has been made by RSIC.

8. COMPUTER HARDWARE REQUIREMENTS

The code was designed for the IBM 360. The packaged version is operable on the CDC 6600, using a maximum of 8 tape units or other direct access devices.

9. COMPUTER SOFTWARE REQUIREMENTS

Standard FORTRAN IV compilers may be used.

10. REFERENCES

B. Liley and S. C. Hamilton (North American Rockwell), "Modified Elemental Volume Dose Program (MEVDP)," AFWL-TR-69-68 (March 1971).

Paul G. Kase (Martin Marietta), "Computerized Anatomical Model Man," AFWL-TR-69-161 (January 1970).

11. CONTENTS OF CODE PACKAGE

The package contains the following items:

- a. the referenced documents,

- b. a reel of magnetic tape on which is written in 3 separate files: the MEVDP source card decks, sample data for the Standing Model Man and sample data for the Seated Model Man.

12. HOW TO OBTAIN PACKAGE

Inquiries or requests for the code package may be mailed to

CODES COORDINATOR
Radiation Shielding Information Center
Oak Ridge National Laboratory
Post Office Box X
Oak Ridge, Tennessee 37830

or telephoned to

Area Code 615; 483-8611, extension 3-6944, or to
FTS xx-615-483-6944.

Persons requesting the package should send a reel of magnetic tape to the above address.

13. DATE OF ABSTRACT

February 1972.

RSIC CODE PACKAGE CCC-158

1. NAME AND TITLE OF CODE

MAGNA: Multi-Source Gamma-Ray Kernel Integration Code.

MAGNA is also available from the ENEA Computer Programme Library (ENEA CPL), listed under Abstract ENEA 163.

2. CONTRIBUTORS

Institut for Atomenergi, Kjeller Research Establishment,
Kjeller, Norway.

ENEA CPL, Ispra (Varese), Italy.

3. CODING LANGUAGE AND COMPUTER

CCC-158A: FORTRAN; CDC 3600.

CCC-158B: FORTRAN; CDC 1604.

4. NATURE OF PROBLEM SOLVED

Gamma-ray dose rates are computed for chosen points in and around a complex arrangement of point or cylindrical gamma-ray sources and slab, cylindrical or cylindrical shell shields.

5. METHOD OF SOLUTION

Dose rates are calculated by point kernel integration using a multi-layer buildup factor. Spatial integration is accomplished by summing multiple point sources. Self shielding (attenuation in the source region) and shadow shielding are included. The buildup factor function is either the Taylor form or the Russian (Broder) form which accounts for multiple layers.

6. RESTRICTIONS OR LIMITATIONS

The following limits are noted:

7 gamma-ray energies
10 different materials
350 sources
500 cylinders (sources and/or shields) and/or cylinder
shells
6 slab shields

7. TYPICAL RUNNING TIME

Typical running time is half a minute for one reasonably sized cylindrical source. Running time is roughly proportional to the number of shields and total number of non-point sources, and also dependent upon source volume and self-absorption coefficient or source-dose point distance.

8. COMPUTER HARDWARE REQUIREMENTS

The code is operable on CDC computers with standard equipment.

9. COMPUTER SOFTWARE REQUIREMENTS

Versions are packaged which have been compiled and executed on the FORTRAN 3600 Monitor System and on the FORTRAN 63 COOP System.

10. REFERENCES

Ulf Tveten, "MAGNA, A Multi-Source Gamma Shielding Programme," KR-111 (June 1966).

Ulf Tveten, "MAGNA, A Gamma Computer Programme," IFA-RPS Memo 82.

11. CONTENTS OF CODE PACKAGE

The package contains the following items:

- a. the referenced documents,
- b. a reel of magnetic tape on which is written in several separate files: the BCD source card decks, BCD input for a sample problem, and an output listing from running the sample problem for each code version.

12. HOW TO OBTAIN PACKAGE

Inquiries or requests for the code package may be mailed to

CODES COORDINATOR
Radiation Shielding Information Center
Oak Ridge National Laboratory
Post Office Box X
Oak Ridge, Tennessee 37830

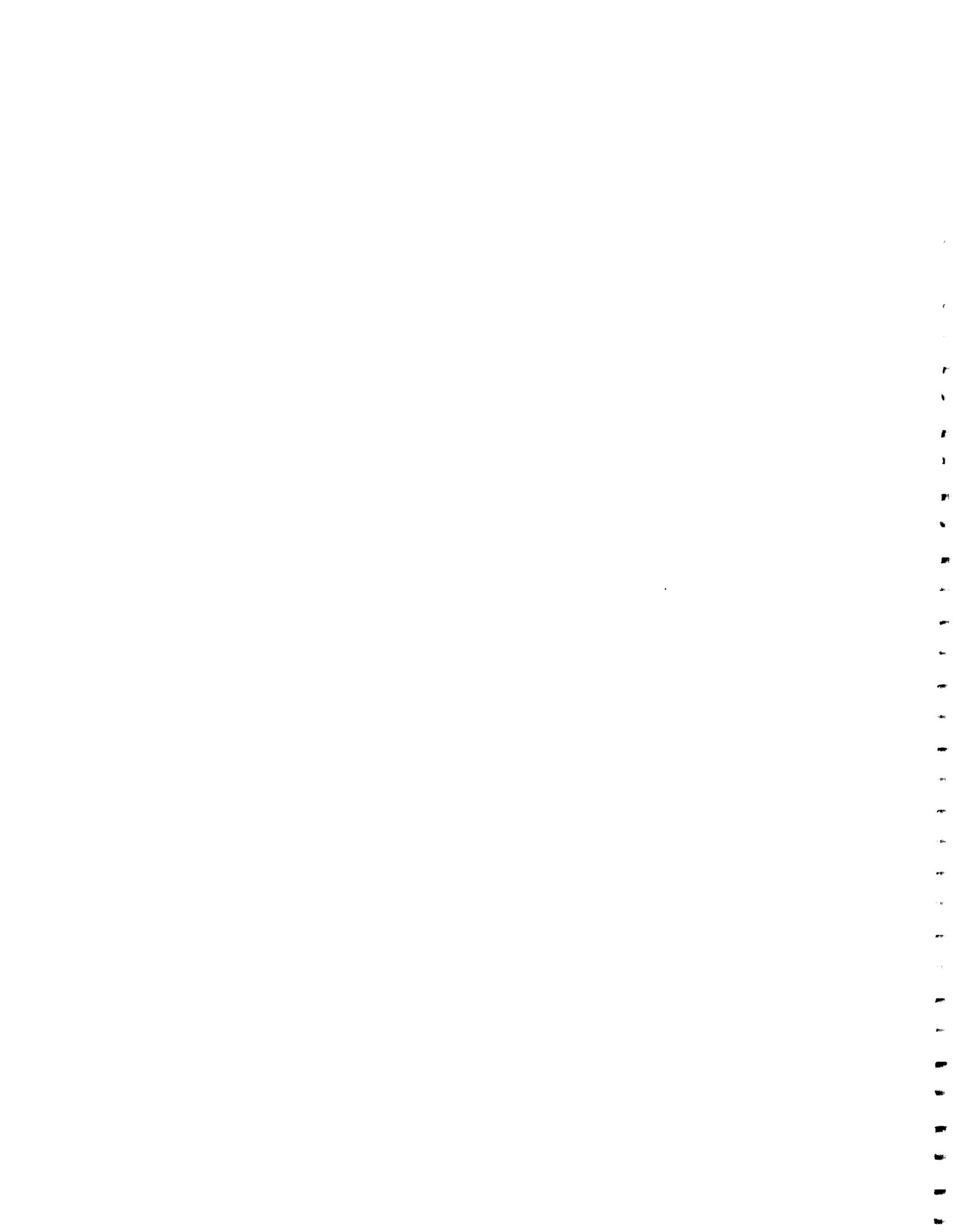
or telephoned to

Area Code 615; 483-8611, extension 3-6944, or to
FTS xx-615-483-6944.

Persons requesting the package should send a reel of magnetic tape
to the above address.

13. DATE OF ABSTRACT

June 1972.



RSIC CODE PACKAGE CCC-159

1. NAME AND TITLE OF CODE

ORPHEE VI: Kernel Integration Code - Attenuation of Fast Neutrons in Cylindrical Layers of Water and Dense Material.

2. CONTRIBUTOR

CEA/CEN Fontenay-aux-Roses Nuclear Research Center, France.

3. CODING LANGUAGE AND COMPUTER

FORTRAN IV and some Assembly language; IBM 360.

4. NATURE OF PROBLEM SOLVED

The attenuation of fast neutrons is calculated by an empirical kernel. The geometry is a succession of heavy media separated by thin layers of water in a cylindrical geometry. The description of the source by bundles renders it particularly suited to light water power reactors. The fast neutron flux is given for each group of cores and for each height of rods.

5. METHOD OF SOLUTION

The coefficients of the empirical kernel, including a buildup factor and removal cross sections, are determined from either Monte Carlo or discrete ordinates calculations. The kernel is integrated over a power distribution given by the product of radial and vertical distributions $S(x,y) f(z)$ where $f(z)$ is a cosine function.

6. RESTRICTIONS OR LIMITATIONS

The limits on input parameters are documented.

7. TYPICAL RUNNING TIME

No study of typical running time has been made by RSIC. The author indicates that ORPHEE VI considers the given radial distribution, element by element, and the actual axial distribution in about 1 minute per point on the IBM 360/50.

8. COMPUTER HARDWARE REQUIREMENTS

The code was designed for the IBM 360/50. The packaged version is operable on the IBM 360/50/75/91 computer configurations.

9. COMPUTER SOFTWARE REQUIREMENTS

The IBM 360 FORTRAN IV compilers may be used. A few routines are written in assembly language.

10. REFERENCES

Michel Simon, "ORPHEE VI Program, Attenuation of Fast Neutrons in a Structure Consisting of Layers of Water and Dense Material," CEA-N-1244 (January 1970) (ORNL-tr-2357).

11. CONTENTS OF CODE PACKAGE

The package contains the following items:

- a. the referenced document,
- b. a reel of magnetic tape on which is written in 5 separate files: the source card deck, several supplementary routines in card image, input cards for a sample problem, BCD output cards, and a BCD output listing from running the sample problem.

12. HOW TO OBTAIN PACKAGE

Inquiries or requests for the code package may be mailed to

CODES COORDINATOR
Radiation Shielding Information Center
Oak Ridge National Laboratory
Post Office Box X
Oak Ridge, Tennessee 37830

or telephoned to

Area Code 615; 483-8611, extension 3-6944, or to
FTS xx-615-483-6944.

Persons requesting the package should send a reel of magnetic tape
to the above address.

13. DATE OF ABSTRACT

June 1972.

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RSIC CODE PACKAGE CCC-160

1. NAME AND TITLE OF CODE

PICA: Monte Carlo Medium-Energy Photon-Induced Intranuclear Cascade Analysis Code System.

AUXILIARY ROUTINES

CON: Data Conversion Code.

PIC: Cascade Code.

MECCAN: Analysis Code.

EVAP: Evaporation Code.

NUC: Nuclear Configuration Code.

2. CONTRIBUTOR

Neutron Physics Division, Oak Ridge National Laboratory, Oak Ridge, Tennessee.

3. CODING LANGUAGE AND COMPUTER

FORTRAN IV; IBM 360/75/91.

Random number generators are written in Assembly Language for the IBM 360.

4. NATURE OF PROBLEM SOLVED

PIC calculates the results of nuclear reactions caused by the collision of medium-energy photons with nuclei. The photon energy range in which the calculations are applicable is $30 \lesssim E_{\gamma} \lesssim 400$ MeV. All target nuclei with mass numbers > 4 are possible. The program PIC can accommodate incident monoenergetic photons as well as thin-target bremsstrahlung spectra, thin-target bremsstrahlung difference spectra and thick-target bremsstrahlung spectra. For the last type of spectra the user must furnish the photon spectral data. PIC writes a history tape containing data on the properties of the particles (protons, neutrons, or pions) escaping from the nucleus. The data consists of the types of escaping particles and their

energies and angles of emission. The associated analysis code MECCAN utilizes the data on the PIC history tape to calculate cross sections such as the nonelastic cross section or the doubly differential cross section for energy-angle correlated distributions. EVAP then carries the nuclear reaction through an additional phase, that of evaporation, and calculates the energy spectra of particles (protons, neutrons, deuterons, tritons, ^3He , and alpha particles) "boiled off" from the nucleus after the cascade has stopped, evaporation particle multiplicities, and evaporation residual nuclei (radio-chemical) cross sections.

5. METHOD OF SOLUTION

The interaction of high-energy photons with nuclei is described by using the intranuclear-cascade and evaporation models. Monte Carlo methods are employed to provide a detailed description of each interaction. The initial interaction of the photon with the nucleus is obtained from the quasi-deuteron model of Levinger, that is, photon absorption by a neutron-proton pair moving within the nucleus or from one of the four pion-nucleon states formed in the photon-nucleon interaction. The effect of secondary nucleon-nucleus and/or pion-nucleus interactions following the photon absorption is accounted for by utilizing the intranuclear-cascade concept of high-energy particle-nucleus reactions. Each particle involved in a collision is traced through the nucleus using the appropriate particle-particle cross sections until the particle escapes from the nucleus or is captured by the nucleus. In all parts of the calculation, the Fermi momentum of the struck particle, the exclusion principle, and the nonuniform density distribution are taken into account. Following the cascade phase, the nucleus is usually in a state of high excitation. This excitation energy can be dissipated through particle emission. This de-excitation is handled by the evaporation model.

The Nuclear Configuration Code (NUC) creates a modified nuclear configuration data input tape for PIC. In PIC three nuclear

regions are used to approximate a continuous nucleon-density distribution in the nucleus, and the radii of these regions can be changed by using NUC.

PIC uses an exact sampling technique to determine the collision site and the types of particles in the reaction.

6. RESTRICTIONS OR LIMITATIONS

The range of validity of PIC is from 30 to 400 MeV for the energy of the incident photon.

7. TYPICAL RUNNING TIMES

The approximate machine time to obtain reasonable statistical accuracy is 30 to 60 minutes on the IBM 360/75 computer.

Estimated running times of the packaged sample problem: CON, one minute; PIC, 2 minutes; MECCAN, one second; EVAP, 5 seconds; and NUC, one second.

8. COMPUTER HARDWARE REQUIREMENTS

The codes were designed for the IBM 360/75/91 computer configuration, requiring a maximum of 3 tape units or direct access devices in addition to I-O. Storage (GO STEP) required: CON, 15K; PIC, 300K; MECCAN, 400K; EVAP, 250K; and NUC, 160K.

9. COMPUTER SOFTWARE REQUIREMENTS

The codes packaged are operable on the IBM 360/75/91 Operating System using OS-360 Level 18 FORTRAN H Compiler.

10. REFERENCES

T. A. Gabriel, M. P. Guthrie, and O. W. Hermann, "Instructions for the Operation of the Program Package PICA, An Intranuclear-Cascade Calculation for High-Energy (30 to 400 MeV) Photon-Induced Nuclear Reactions," ORNL-4687 (September 1971).

T. A. Gabriel and R. G. Alsmiller, Jr., "Photonuclear Disintegration at High (< 350 MeV) Energies" (Thesis), ORNL-TM-2481 (February 1969).

Hugo W. Bertini, Phys. Rev. 188, 1711 (1969).

11. CONTENTS OF CODE PACKAGE

The package contains the following items:

- a. the referenced documents,
- b. a reel of magnetic tape on which is written in 16 separate files:
2 libraries of data in EBCDIC, source card images of each of the auxiliary codes, input data for a sample problem for each of the codes, and a BCD output listing from running the sample problem.

12. HOW TO OBTAIN PACKAGE

Inquiries or requests for the code package may be mailed to

CODES COORDINATOR
Radiation Shielding Information Center
Oak Ridge National Laboratory
Post Office Box X
Oak Ridge, Tennessee 37830

or telephoned to

Area Code 615; 483-8611, extension 3-6944, or to
FTS xx-615-483-6944.

Persons requesting the package should send a reel of magnetic tape to the above address.

13. DATE OF ABSTRACT

June 1972.

RSIC CODE PACKAGE CCC-161

1. NAME AND TITLE OF CODE

NMTC: Monte Carlo Nucleon-Meson Transport Code System.

AUXILIARY ROUTINES

COPY Code.

PUNCRØSS: Cross Section Handling Code.

NMT: Nucleon-Meson Transport Code.

NMT ANALYSIS: Data Processing Code.

ENMT: Edit Code.

XSECT: O5R Cross Section Handling Code.

O5R: Modified O5R Monte Carlo Code.

O5R ANALYSIS: Data Processing Code.

The NTC Code (CCC-7) was used as a starting point for developing NMTC, and many of the NTC concepts and techniques have been retained. A modified version of O5R (CCC-17) is included in the system.

2. CONTRIBUTOR

Neutron Physics Division, Oak Ridge National Laboratory, Oak Ridge, Tennessee.

3. CODING LANGUAGE AND COMPUTER

FORTRAN IV, some assembly language; IBM 360/75/91.

4. NATURE OF PROBLEM SOLVED

NMTC computes the transport of nucleons below 3.5 GeV and muons and charged pions below 2.5 GeV. Monte Carlo methods are employed to provide a detailed description of the transport process. Virtually arbitrary geometries may be specified.

The source-particle description is specified in a user-written subroutine and may be arbitrarily distributed in energy, direction, and space, provided the maximum energies mentioned above are not exceeded. Proton, neutron, π^+ , π^- , μ^+ , and μ^- sources are allowed.

The code stores on magnetic tape a complete description of each "event" (nuclear interaction, geometry boundary crossing, pion decay, etc.) which occurs during the transport process. This information is then read and processed by user-written analysis programs to obtain results of interest for a particular problem.

5. METHOD OF SOLUTION

NMTC consists of two basic transport codes: NMT and a modified version of O5R. NMT transports particles in the energy range from the source-particle energy down to a specified cutoff energy, which, for nucleons, is usually chosen between 15 and 50 MeV. Neutrons produced in NMT below the cutoff energy are transported via the O5R code.

The description of nonelastic-collision products in NMT is obtained using the intranuclear-cascade-evaporation model. At each nonelastic collision, a calculation is performed using subprogram versions of Bertini's intranuclear-cascade program and Guthrie's evaporation program to determine the energy and direction of emitted cascade nucleons and pions and evaporated nucleons, deuterons, tritons, He's and alphas. Nonelastic collisions in O5R are treated using the evaporation model in conjunction with experimental cross sections. Experimental data are used for elastic-collision cross sections.

Charged-pion and muon decay in flight and at rest are taken into account using known lifetimes. Negative-pion capture at rest is treated via the intranuclear-cascade-evaporation model.

Differential cross sections for nucleon and pion production from nucleon-nucleus and pion-nucleus nonelastic collisions are not required as input since they are, in effect, computed in the course of the transport calculation using the intranuclear-cascade-evaporation model.

6. RESTRICTIONS OR LIMITATIONS

Present dimensions restrict the number of different media to 15 or less and the number of nuclei types per medium to 11 or less.

7. TYPICAL RUNNING TIME

Running time is extremely problem-dependent. A sample problem included with the code documentation requires approximately 10 minutes on the IBM 360/91.

Estimated running times of the packaged sample problems: PUNCROSS, 1 second; XSECT, 1 minute; O5R, 4 minutes; O5R ANALYSIS, 15 seconds; NMT, 6 minutes, NMT ANALYSIS, 11 seconds; and ENMT, 1/2 second.

8. COMPUTER HARDWARE REQUIREMENTS

The codes were designed for the IBM 360 equipment, requiring a maximum of 610K (NMT) of core storage, and a maximum of 4 tape units or direct access devices, in addition to I-Ø devices.

9. COMPUTER SOFTWARE REQUIREMENTS

The codes are operable on the IBM 360/75/91 Operating System using OS-360 FORTRAN H Compiler.

10. REFERENCES

W. A. Coleman and T. W. Armstrong, "The Nucleon-Meson Transport Code NMTC," ORNL-4606 (October 1970).

D. C. Irving, R. M. Freestone, Jr., and F. B. K. Kam et al., "O5R, A General-Purpose Monte Carlo Neutron Transport Code," ORNL-3622 (February 1965).

11. CONTENTS OF CODE PACKAGE

The package contains the following items:

- a. the referenced documents,
- b. a reel of magnetic tape on which is written in 28 separate files:
2 BCD libraries of data, the source card decks for all codes in the system, input data for a sample problem for each code, and a BCD output listing from running the problems.

12. HOW TO OBTAIN PACKAGE

Inquiries or requests for the code package may be mailed to

CODES COORDINATOR
Radiation Shielding Information Center
Oak Ridge National Laboratory
Post Office Box X
Oak Ridge, Tennessee 37830

or telephoned to

Area Code 615; 483-8611, extension 3-6944, or to
FTS xx-615-483-6944.

Persons requesting the package should send a reel of magnetic tape to the above address.

13. DATE OF ABSTRACT

June 1972.

RSIC CODE PACKAGE CCC-162

1. NAME AND TITLE OF CODE

FPIP: Fission Product Inventory Code System.

FPIP originated at Westinghouse Astronuclear Laboratory in 1963. The packaged version was revised extensively in 1967.

AUXILIARY ROUTINES

FPIPLTG: Library Tape Generator.

FPIP: Main Program.

2. CONTRIBUTOR

Nuclear Systems, Westinghouse Astronuclear Laboratory, Pittsburgh, Pennsylvania.

3. CODING LANGUAGE AND COMPUTER

FORTRAN IV; CDC 6600.

4. NATURE OF PROBLEM SOLVED

A calculation of the fission product abundances and beta and gamma-ray source strengths, using information on reactor history and requested decay times after reactor shutdown.

5. METHOD OF SOLUTION

The Library Tape Generator generates and stores in machine language form on tape data required by FPIP. The main program calculates the energy contribution from each fission product nuclide in each selected energy grouping, and decay chains and subchains for each nuclide. Fission product abundances and beta and gamma-ray sources are then calculated.

6. RESTRICTIONS OR LIMITATIONS

A maximum of 50 decay time points can be calculated per run. A maximum of 6 separate operating periods can be used to describe the reactor operating history. Output is limited to a maximum of 7 gamma and 5 beta energy groups. The library accommodates 254 nuclides.

7. TYPICAL RUNNING TIME

No study of typical running time has been made by RSIC.

8. COMPUTER HARDWARE REQUIREMENTS

The code is operable on the CDC 6600 computer.

9. COMPUTER SOFTWARE REQUIREMENTS

The code is designed to run on the CDC 6600 FORTRAN IV compiler.

10. REFERENCES

W. S. Brown and D. W. Call, "Revised Fission Product Inventory Program (FPIP-REV)," WANL-TME-537, Revision 1 (May 1967).

11. CONTENTS OF CODE PACKAGE

The package contains the following items:

- a. the referenced document,
- b. a reel of magnetic tape on which is written in 5 separate files: the source card decks, a BCD library of data, and BCD input data for a sample problem.

12. HOW TO OBTAIN PACKAGE

Inquiries or requests for the code package may be mailed to

CODES COORDINATOR
Radiation Shielding Information Center
Oak Ridge National Laboratory
Post Office Box X
Oak Ridge, Tennessee 37830

or telephoned to

Area Code 615; 483-8611, extension 3-6944, or to

FTS xx-615-483-6944.

Persons requesting the package should send a reel of magnetic tape to the above address.

13. DATE OF ABSTRACT

June 1972.

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RSIC CODE PACKAGE CCC-163

1. NAME AND TITLE OF CODE

FISSP-CLOUD: Fission Product Inventory, Release, Transport
and Dose Calculation.

AUXILIARY ROUTINE

DELTA T: Input Data Generator.

2. CONTRIBUTOR

Sandia Laboratories, Albuquerque, New Mexico.

3. CODING LANGUAGE AND COMPUTER

CCC-163A: FORTRAN IV; CDC 6600.

CCC-163B: FORTRAN IV; IBM 360.

4. NATURE OF PROBLEM SOLVED

FISSP calculates the fission product inventory in a (^{235}U fueled) reactor for a specified power history. In addition, following the specified power history, FISSP computes an instantaneous point (puff) release of the fission products to the atmosphere, and continues to calculate the radioactivity history of each released fission product.

CLOUD calculates the external gamma and beta dose, and the internal dose to the lung, thyroid, gastro-intestine, bone, and whole body resulting from inhalation of the passing radioactive cloud.

5. METHOD OF SOLUTION

The method of linear chain resolution by England is used in FISSP to calculate the fission product atom density of 326 separate nuclides and metastable states for a specified series of (stepwise) power histories. The same equations are solved for zero power

following the instantaneous release to the atmosphere of specified fractions of the noble gases, halogens, high temperature volatiles, and other fission products.

In CLOUD, the released nuclides are allowed to drift downwind at the average wind speed while diffusing in three dimensions as determined by the appropriate coefficients in the Sutton atmospheric diffusion equation. The external gamma and beta dose received at the dose point during cloud passage is taken to be (after Eckert) proportional to the activity concentration of each nuclide integrated over the effective exposure time and summed over all nuclides. The inhalation dose to the lung, thyroid, gastro-intestine, bone, and whole body is calculated by assuming a modified exponential body model. In this method (after Valentine) the dose to an internal organ from each nuclide is taken to be proportional to the time integral of the organ burden due to inhalation. Total organ dose is obtained by summing over-all nuclides.

6. RESTRICTIONS OR LIMITATIONS

FISSP may be used separately to calculate fission product inventories, but CLOUD requires the released nuclide radioactivity history generated by FISSP.

FISSP is restricted to 20 or less stepwise power histories by dimension statements.

Care should be taken in setting up the time base origin between FISSP and CLOUD. CLOUD requires 360 properly located values of the fission product activities from FISSP to correctly calculate the doses. Thus it is important to specify the correct final power history in FISSP.

7. TYPICAL RUNNING TIME

FISSP running time on the CDC-6600 for 3 power histories and 360 print intervals is 700 seconds. CLOUD running time on the CDC-6600 is 175 seconds.

Estimated running times of the packaged sample problems in CCC-163B, GO STEP: FISSP, 2 minutes and CLOUD, 4 minutes.

8. COMPUTER HARDWARE REQUIREMENTS

CCC-163A/FISSP-CLOUD were written for the CDC-6600, and use one tape unit (or disc). FISSP requires an 80K (octal) memory. CLOUD requires a 65K (octal) memory.

Version B is operable on the IBM 360/75/91 hardware configuration. This conversion was made by RSIC. Core size used: FISSP, 140K and CLOUD, 126K.

9. COMPUTER SOFTWARE REQUIREMENTS

Version A uses CDC 6600 FORTRAN IV compiler. For Version B, the codes are operable on the IBM-360/75/91 Operating System using OS-360 FORTRAN H Compiler. Standard input, output and logical unit 7 tape assignments are used.

10. REFERENCES

Lloyd L. Bonzon and Joseph B. Rivard, "Computational Method for Calculation of Radiological Dose Resulting from Hypothetical Fission Product Release," SC-RR-70-338 (July 1970).

Background Material

T. R. England, "Cinder - A One Point Depletion and Fission Product Program, U. W. Version," Bettis Atomic Power Laboratory, WAPD-TM-334 (1962, revised 1968).

T. R. England, "An Investigation of Fission Product Behavior and Decay Heating in Nuclear Reactors," Ph.D. Thesis, University of Wisconsin (1969).

O. G. Sutton, Micrometeorology, New York, McGraw-Hill Book Co., Inc., 1953.

R. J. Eckert, "A Fortran Program for the Calculation of the Potential Hazards from an Accidental Fission-Product Release - M0142," Bettis Atomic Power Laboratory, WAPD-TM-348 (1964).

D. R. Connors, H. J. Litke, and D. S. Rampolla, "Identification of Important Fission Product Nuclides for Radiological Dose Calculations Following a Uranium 235 - Fueled Reactor Accident," Bettis Atomic Power Laboratory, WAPD-TM-599 (1967).

11. CONTENTS OF CODE PACKAGE

The package contains the following items:

- a. the first referenced document,
- b. a reel of magnetic tape on which is written in several files: the source card decks, 3 sets of library data, and input for sample problems for each code version. Output from running the sample problems is included in Version B.

12. HOW TO OBTAIN PACKAGE

Inquiries or requests for the code package may be mailed to

CODES COORDINATOR
Radiation Shielding Information Center
Oak Ridge National Laboratory
Post Office Box X
Oak Ridge, Tennessee 37830

or telephoned to

Area Code 615, 483-8611, extension 3-6944, or to
FTS xx-615-483-6944.

Persons requesting the code package should send a reel of magnetic tape to the above address.

13. DATE OF ABSTRACT

June 1972.

RSIC CODE PACKAGE CCC-164

1. NAME AND TITLE OF CODE

NAC: Neutron Activation Analysis and Product Isotope Inventory Code.

NAC is a subset of NAP (CCC-101).

2. CONTRIBUTOR

National Aeronautics and Space Administration, Lewis Research Center, Cleveland, Ohio.

3. CODING LANGUAGE AND COMPUTER

CCC-164A: FORTRAN IV; IBM 360.

CCC-164B: FORTRAN IV; IBM 7090/7094.

4. NATURE OF PROBLEM SOLVED

NAC is a computer code designed to predict the neutron-induced gamma-ray radioactivity for a wide variety of composite materials. The NAC output includes the input data, a list of all reactions for each constituent element, and the end-of-irradiation disintegration rates for each reaction. The code also compiles a product isotope inventory containing the isotope name, the disintegration rate, the gamma-ray source strength, and the absorbed dose rate at 1 meter from an unshielded point source. The induced activity is calculated as a function of irradiation and decay times; the effect of cyclic irradiation can also be calculated.

5. METHOD OF SOLUTION

The code uses the standard neutron activation and decay equations.

6. RESTRICTIONS OR LIMITATIONS

Each composite material may consist of up to 20 different elements and up to 20 different decay times may be included. Both limits may be increased by the user by increasing the appropriate items in the DIMENSION statement.

7. TYPICAL RUNNING TIME

Execution time for 3 sample problems (a: 9 elements, 4 decay times; b: 6 element, 1 decay time; c: 6 elements, 1 decay time) was 0.14 minutes on the IBM 7090/94; on the IBM 360, CPU: 4 seconds.

8. COMPUTER HARDWARE REQUIREMENTS

The code was designed to run on the IBM 7094 with standard I-O equipment.

It was converted by RSIC to be operable on the IBM 360/75/91.
Core size required: 130K.

9. COMPUTER SOFTWARE REQUIREMENTS

The code is operable on the IBM 7090/7094 IBSYS Operating System using IJOB Processor, and the IBM 360/75/91 Operating System using OS-360 FORTRAN H Compiler.

A Data Library is required and is included on the master tape.

10. REFERENCES

Suzanne T. Weinstein, "NAC: Neutron Activation Code," NASA TM X-52460 (1968).

11. CONTENTS OF CODE PACKAGE

The package contains the following items:

- a. the referenced document,
- b. a reel of magnetic tape on which is written in several files: the source card deck, a BCD library of data, BCD input for a sample problem, and a BCD output listing from running the problem for each version of the code.

12. HOW TO OBTAIN PACKAGE

Inquiries or requests for the code package may be mailed to

CODES COORDINATOR
Radiation Shielding Information Center
Oak Ridge National Laboratory
Post Office Box X
Oak Ridge, Tennessee 37830

or telephoned to

Area Code 615; 483-8611, extension 3-6944, or to

FTS xx-615-483-6944.

Persons requesting the code package should send a reel of magnetic tape to the above address.

13. DATE OF ABSTRACT

June 1972.



RSIC CODE PACKAGE CCC-165

1. NAME AND TITLE OF CODE

DOSE1: Gamma-Radiation Dosimetry for Arbitrary Source and Target Geometry.

This code was originally developed¹ in the pre-1968 formalism of radiobiologic dosimetry² in the Biological and Medical Research Division of Argonne National Laboratory. It was specialized for several specific problems and some computational results were published.^{3,4} The present version is in the currently accepted formalism⁵ of absorbed fractions. It is intended for general application or as a base for specialized programs. The name of the program signifies that the result (which is the first collision absorbed fraction) can be used to calculate the first collision dose.

2. CONTRIBUTOR

Physics Division, Oak Ridge National Laboratory, Oak Ridge, Tennessee.

3. CODING LANGUAGE AND COMPUTER

FORTRAN IV; IBM 360.

4. NATURE OF PROBLEM SOLVED

DOSE 1 calculates the first collision absorbed fraction (i.e., the average fraction of the energy lost by all emitted photons on their first collision) and point specific absorbed fractions for a monoenergetic, gamma-ray source. Arbitrary placements of source(s), attenuating material and absorbing material is possible.

To the extent that the first collision absorbed fraction is desired, the only approximation in the calculation arises from the division of the space into small Cartesian units.

5. METHOD OF SOLUTION

The space is subdivided into rectangular parallelepipeds each of which is considered uniform in attenuation, while the absorption and the activity are considered to reside at the center of the parallelepiped. The attenuation is calculated for each ray according to the length of the ray within each parallelepiped. The contribution from all source points is integrated for each parallelepiped.

6. RESTRICTIONS OR LIMITATIONS

The space is divided into 15 subunits in each of the three dimensions. This limits the detail available for spacial accuracy. This can easily be changed; however, the general problem can easily become excessive in computer time. Specialized problems in which some conditions (such as a point source, or uniform attenuation) can be applied, can be programmed to run much faster.

7. TYPICAL RUNNING TIME

For arbitrary source and target volumes execution time will vary approximately as $N^{7/3}$ where N is the number of elementary subvolumes. For small values of N (i.e., $N \lesssim 50$) execution times are a few seconds on the IBM 360/91.

8. COMPUTER HARDWARE REQUIREMENTS

DOSE 1 is operable on the IBM 360/75 and 91 systems. It should run without major modifications on most computers using FORTRAN IV or similar FORTRAN. It requires about 26,000 (32 bit) words on the 360/75 or 91. The standard reader and punch are used and no auxiliary storage is needed. The clock is not sampled.

9. COMPUTER SOFTWARE REQUIREMENTS

The code is written in FORTRAN and the statements have been chosen for compatibility with other computer hardware. The standard monitor and only common FORTRAN supplied subprograms are used (e.g., ALØG, SQRT).

10. REFERENCES

Lincoln B. Hubbard, "A FORTRAN Program for Gamma-Radiation Dosimetry for Arbitrary Source and Target Geometry," ORNL-TM-3398 (April 1971).

Background material:

(1) L. B. Hubbard and F. S. Williamson, Argonne National Laboratory Biological and Medical Research Division Annual Report, ANL-7409, 1969, p. 250.

(2) ICRU report 10a and 10e, Handbook 88 (National Bureau of Standards, Washington, D. C.), 1962.

(3) L. B. Hubbard and F. S. Williamson, Physics in Medicine and Biology 14, 255 (1969).

(4) L. B. Hubbard, Radiation Research 44, 4 (1970).

(5) R. Loevinger and M. Berman, Physics in Medicine and Biology 13, 205 (1968).

11. CONTENTS OF CODE PACKAGE

The package contains the following items:

- a. the first referenced document which includes output from the sample problem,
- b. a reel of magnetic tape on which is written in 2 separate files: the source card deck and BCD input cards for a sample problem.

12. HOW TO OBTAIN PACKAGE

Inquiries or requests for the code package may be mailed to

CODES COORDINATOR
Radiation Shielding Information Center
Oak Ridge National Laboratory
Post Office Box X
Oak Ridge, Tennessee 37830

or telephoned to

Area Code 615; 483-8611, extension 3-6944, or to
FTS xx-615-483-6944.

Persons requesting the package should send a reel of magnetic tape to the above address.

13. DATE OF ABSTRACT

June 1972.

RSIC CODE PACKAGE CCC-166

1. NAME AND TITLE OF CODE

DAVE: Monte Carlo Gamma-Ray Transport Code System in One-Dimensional Spherical Geometry.

AUXILIARY ROUTINES

DAVE I: Collision Tape Generator.

DAVE II: Analysis Code.

DAVE is a modification of SALOMON (CCC-33).

2. CONTRIBUTOR

The Research Institute of the Swedish National Defence, Stockholm, Sweden.

3. CODING LANGUAGE AND COMPUTER

FORTRAN IV, Assembler Language; IBM 360/75/91.

4. NATURE OF PROBLEM SOLVED

DAVE solves the time-dependent Boltzmann equation for gamma radiation in spherical geometry, and computes the densities, average radial velocities and radial current of the Compton electrons generated by the gammas. All computed data are time dependent. Monoenergetic gamma source.

5. METHOD OF SOLUTION

DAVE is a Monte Carlo code using weighted averages to reduce the number of random variables at the calculation of velocities and currents.

6. RESTRICTIONS OR LIMITATIONS

The program is limited to 1 material, 20 radial points and 300 time intervals.

7. TYPICAL RUNNING TIME

With one radial point 3-4 mfp from source and 20000 histories, a typical time is 5 minutes on the IBM 360/75.

The estimated running times for the packaged sample problems: DAVE I - GO STEP, 8 seconds, and DAVE-II - GO STEP, 8 seconds.

8. COMPUTER HARDWARE REQUIREMENTS

The codes were designed for the IBM 360 with standard I-O and one tape unit.

The core size required: DAVE I approximately 85K, and DAVE II approximately 140K.

9. COMPUTER SOFTWARE REQUIREMENTS

The code is operable on the IBM 360/75/91 System using OS-360 FORTRAN H Compiler. A random number generator routine is required and is provided in assembly language.

10. REFERENCES

G. Engstrom, "A User's Manual for a Computer Code Calculating Densities and Velocities of Compton Electrons Generated by Gammas - DAVE (IBM 360 FORTRAN)," FOA-4C-4374-29 (October 1968).

11. CONTENTS OF CODE PACKAGE

The package contains the following items:

- a. the referenced document,
- b. a reel of magnetic tape on which is written in 8 files: the source card deck, input for a sample problem, and an output listing from running the sample problem for each of the auxiliary routines.

12. HOW TO OBTAIN PACKAGE

Inquiries or requests for the code package may be mailed to

CODES COORDINATOR
Radiation Shielding Information Center
Oak Ridge National Laboratory
Post Office Box X
Oak Ridge, Tennessee 37830

or telephoned to

Area Code 615; 483-8611, extension 3-6944, or to
FTS xx-615-483-6944.

Persons requesting the code package should send a reel of magnetic tape to the above address.

13. DATE OF ABSTRACT

June 1972.



RSIC CODE PACKAGE CCC-167

1. NAME AND TITLE OF CODE

ELF: Monte Carlo Neutron Transport Code System for Cylinders and Spheres.

AUXILIARY ROUTINE

LEC-AWRE: Cross Section Handling Code.

ELF-NEUTRON: Monte Carlo Neutron Transport Code.

ELF-VARI-PYRO: Analysis Code.

UKNLD: Data Library - 91 elements.

2. CONTRIBUTOR

CEA/CEN Fontenay-aux-Roses Nuclear Research Center, France.

3. CODING LANGUAGE AND COMPUTER

FORTRAN IV; IBM 360.

4. NATURE OF PROBLEM SOLVED

ELF was designed to study the propagation of neutrons in cylindrical and spherical geometries using AWRE cross sections. In cylindrical geometry the external contour can be circular or polygonal, the limiting conditions at this contour being a vacuum or total reflection.

5. METHOD OF SOLUTION

The Monte Carlo method is used to compute the radiation transport. Auxiliary programs are used for the preparation of cross sections, the study of variance, and to further analyze the ELF collision results.

ELF-VARI-PYRO calculates the spectrum and detector responses from ELF-NEUTRON collision results. The calculation of the mean value is associated with a statistical error bar and a normality test.

The UK-AWRE data library is updated periodically and the update is available from CCDN, Saclay, France.

6. RESTRICTIONS OR LIMITATIONS

The following limits apply for 265K storage: 10 elements, 20 materials, 30 meshes, 150 energy intervals, 200 neutron emission energies, and 100 points defining the radial distribution.

7. TYPICAL RUNNING TIME

No study of typical running time has been made by RSIC.

Estimated running time of the sample problem on the IBM 360/91: 5 minutes.

8. COMPUTER HARDWARE REQUIREMENTS

The code system is operable on the IBM 360/75/91 hardware configuration. 256K of core storage is required.

9. COMPUTER SOFTWARE REQUIREMENTS

Standard IBM 360 OS Compilers may be used.

10. REFERENCES

Francois Gervaise, "ELF Program-Description of Utilization," CEA-Note N-1361 (ORNL-tr-2409) (1970).

Francois Gervaise, "ELF-NEUTRON Program," CEA-Note N-1255 (ORNL-tr-2519) (1970).

K. Parker, "The Aldermaston Nuclear Data Library as at May 1963," AWRE-O-70/63 (September 1963).

D. S. Norton, "The UKAEA Nuclear Data Library," AEEW-M824 (February 1968).

D. S. Norton and J. S. Story, "UKAEA Nuclear Data Library, January, 1967," AEEW-M802.

P. J. Hemmings and Susan M. Offord, "Photon Interaction Data in the UKAEA Nuclear Data Library," AHSB(S)R 109 (1966).

11. CONTENTS OF CODE PACKAGE

The package contains the following items:

- a. the referenced documents,
- b. a reel of magnetic tape on which is written in several separate files: the source card decks, the UKNLD library in BCD, input for sample problems and a BCD output listing from running the sample problems.

12. HOW TO OBTAIN PACKAGE

Inquiries or requests for the code package may be mailed to

CODES COORDINATOR
Radiation Shielding Information Center
Oak Ridge National Laboratory
Post Office Box X
Oak Ridge, Tennessee 37830

or telephoned to

Area Code 615; 483-8611, extension 3-6944, or to
FTS xx-615-483-6944.

Persons requesting the package should send one reel of magnetic tape to the above address (9 track, 800) or 3 reels (7 track).

13. DATE OF ABSTRACT

June 1972.

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RSIC CODE PACKAGE CCC-168

1. NAME AND TITLE OF CODE

FASTER III: Monte Carlo Neutron and Photon Transport Code in Complex Geometries.

FASTER III accepts neutron and secondary production cross sections in the ANISN-DOT format, e.g., output from POPOP4 (PSR-11). It incorporates numerous additions and modifications made to the FASTER program (CCC-98).

2. CONTRIBUTORS

A.R.T. Research Corporation, Los Angeles, California, and NASA Lewis Research Center, Cleveland, Ohio.

3. CODING LANGUAGE AND COMPUTER

CCC-168A: FORTRAN IV, UNIVAC 1108.

CCC-168B: FORTRAN IV, IBM 360.

4. NATURE OF PROBLEM SOLVED

FASTER III calculates neutron and/or photon transport for point, surface and/or volume detectors. Program capabilities include the use of multigroup or point value neutron cross sections, an approximate slowing down kernel for neutrons, calculation of secondary photon (or neutron) transport, multiple complex source descriptions, quadric, helical and/or toroidal surface geometries; simple input for spherical, cylindrical and regular three-dimensional geometries, a spherical exponential atmosphere model, correlated calculations with source and/or material changes; flux edits for birth region, order of scatter, scattering region, boundary crossing, and initial source group; angular fluxes by Legendre moments and/or solid angle bins; temporal fluxes by temporal moments and/or time bins; derivatives of fluxes with respect to region dimension changes; optimum shield mass distribution for multiple response constraints,

arbitrary rotation and/or translation of sources and/or geometry, and numerous importance sampling options.

5. METHOD OF SOLUTION

FASTER III utilizes the Monte Carlo method for the generation and tracking of particles. Importance sampling is used in all of the decision processes. Usually, the entire spectrum of particle energies is treated simultaneously. Optionally, the program will treat a limited portion of the spectrum on each history.

6. RESTRICTIONS OR LIMITATIONS

Dynamic storage allocation is used. The limitation on problem size is based on available core.

7. TYPICAL RUNNING TIME

Typical running time: 0.1 second per collision for an energy-dependent particle.

Estimated running time of the packaged sample problem on the IBM 360/91: 24 seconds.

8. COMPUTER HARDWARE REQUIREMENTS:

FASTER III hardware requirements follow.

- a. Designed for machine-independent operation with approximately 60K (decimal) core (240K bytes) required for large, complex problems.
- b. Operates on various IBM 360, UNIVAC 1108, and CDC 6600 computers. By judicious elimination of portions of the program, it will operate on the IBM 7094 with limited problem solving capabilities.
- c. Auxiliary storage is used only as required for particular problems, e.g., cross section library tape (optional), restart tape (optional), secondary source files.

- d. Channel configuration is set by user through control cards and input data specifying logical designations to the program.
- e. Auxiliary equipment is normally limited to the printer with optional use of a card punch and/or Calcomp plotter.

9. COMPUTER SOFTWARE REQUIREMENTS

All parameters which may need changing for system compatibility can be changed in the main program. The program functions under standard operating systems, i.e., 360/ØS, 1108/EXECII and 8, 6600/RUN compile, 7094/IBSYS. Only standard library routines are used.

Channel assignments are variable. A special Map routine should be used on the IBM 7094 to eliminate buffer assignments for unused units.

10. REFERENCES

T. M. Jordan, "FASTER-III, A Generalized-Geometry Monte Carlo Computer Program for the Transport of Neutrons and Gamma Rays," ART-45, Volumes I and II (December 1970).

T. M. Jordan, "FASTER, A FORTRAN Analytic Solution of the Transport Equation by Random Sampling," WANL-PP(LL)-010, Volume 9 (June 1967).

11. CONTENTS OF CODE PACKAGE

The package contains the following items:

- a. the referenced documents,
- b. a reel of magnetic tape on which is written in several files: the source card decks, BCD input for a sample problem and a BCD output listing from running the sample problem for each version of the code.

12. HOW TO OBTAIN PACKAGE

Inquiries or requests for the code package may be mailed to

Codes Coordinator
Radiation Shielding Information Center
Oak Ridge National Laboratory
Post Office Box X
Oak Ridge, Tennessee 37830

or telephoned to

Area Code 615; 483-8611, extension 3-6944, or to

FTS xx-615-483-6944.

Persons requesting the code package should send a reel of magnetic tape to the above address.

13. DATE OF ABSTRACT

June 1972.