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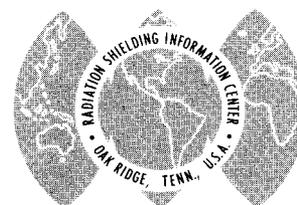
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## ABSTRACTS OF DIGITAL COMPUTER CODE PACKAGES ASSEMBLED BY THE RADIATION SHIELDING INFORMATION CENTER

Betty F. Maskewitz

Note:  
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RADIATION SHIELDING INFORMATION CENTER



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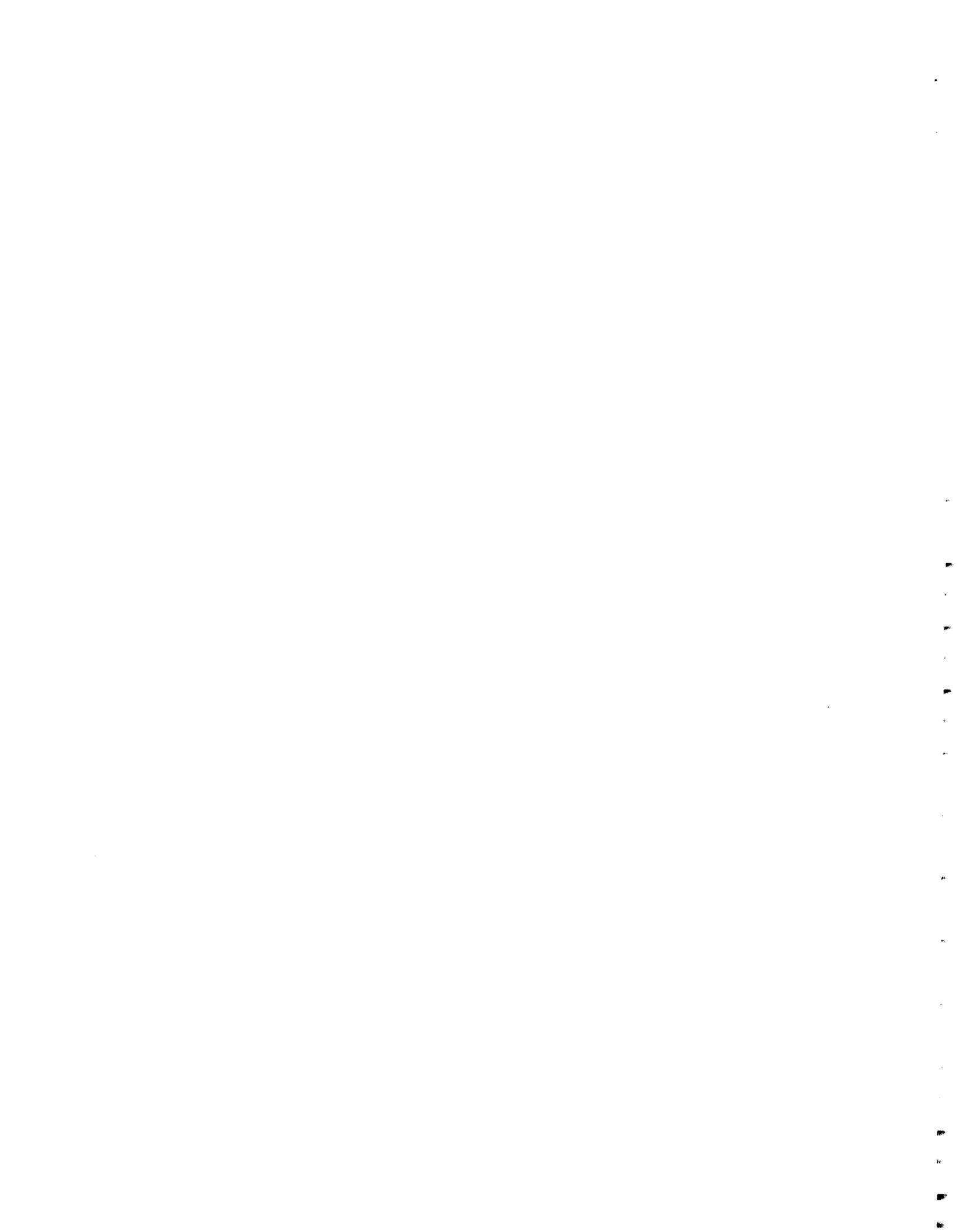
## PREFACE

It was necessary to issue a second volume of ORNL-RSIC-13, "Abstracts of Digital Computer Code Packages Assembled by the Radiation Shielding Information Center." Volume I contains abstracts for code packages 1 - 58, a total of 250 pages. For continuity, paging for the abstracts in Volume II begins at 251.

The TABLE OF CONTENTS in each volume will denote the abstracts in that particular volume. The alphabetic and numeric index, category list, and code package list will be updated periodically to include the latest abstracts issued. Two sets will be distributed so that each volume will contain such information for the entire code collection. Paging on the following pages is indicated for insertion in either Volume I or Volume II. The volume is indicated by the prefix.

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 next logical step before requesting the complete package.

The RSIC staff is pleased to note the increasing number of requesters who visit the Center and assist in the checkout of computer codes in which they have an interest. Such participation is encouraged as being profitable to all participants. The code packaging process has been expedited by the increasing assistance given by the code contributors in the writing of abstracts. A copy of the abstract format is included in the introduction to Volume I as a guide to the abstracts and also as a guide to the writing of additional ones.



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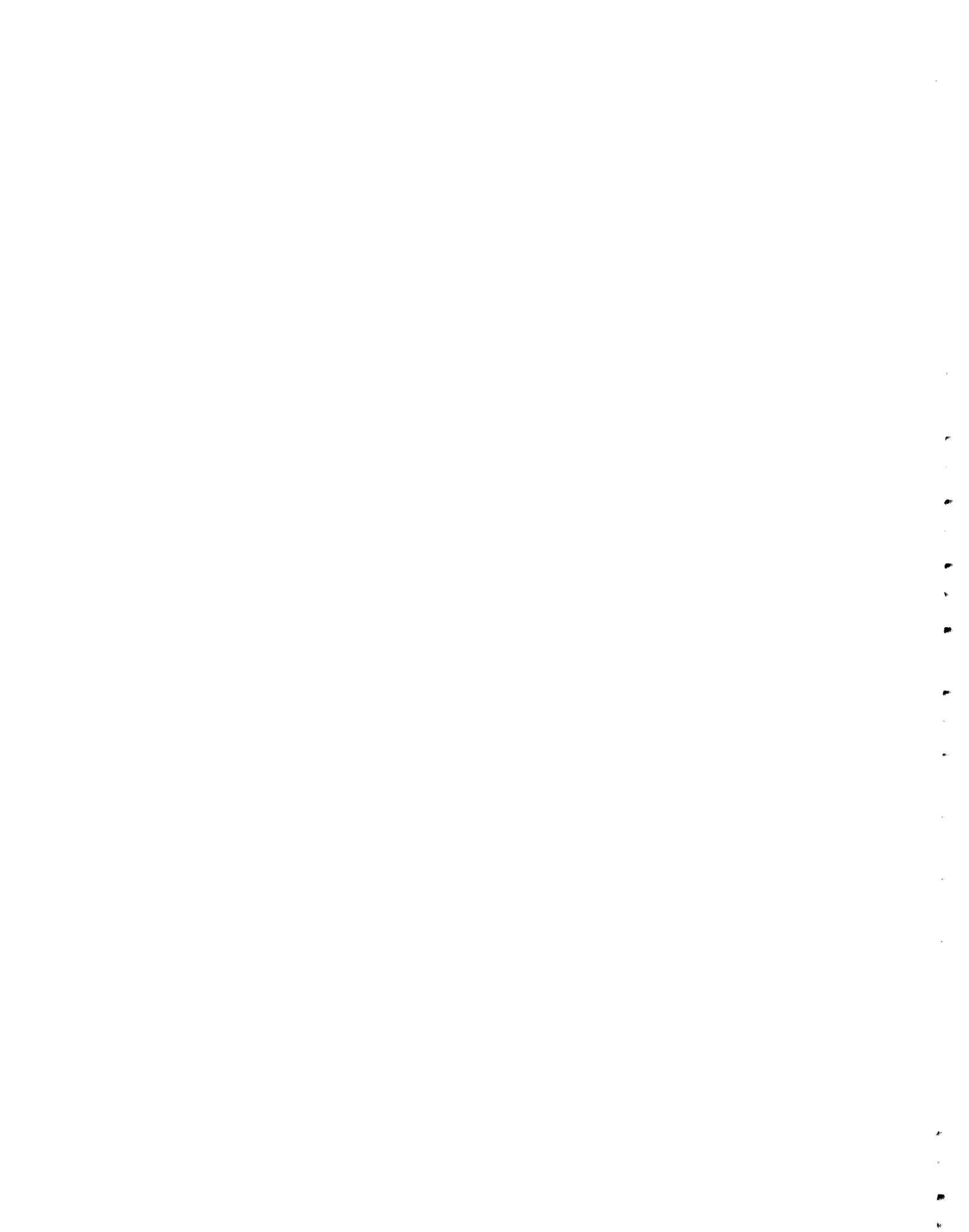
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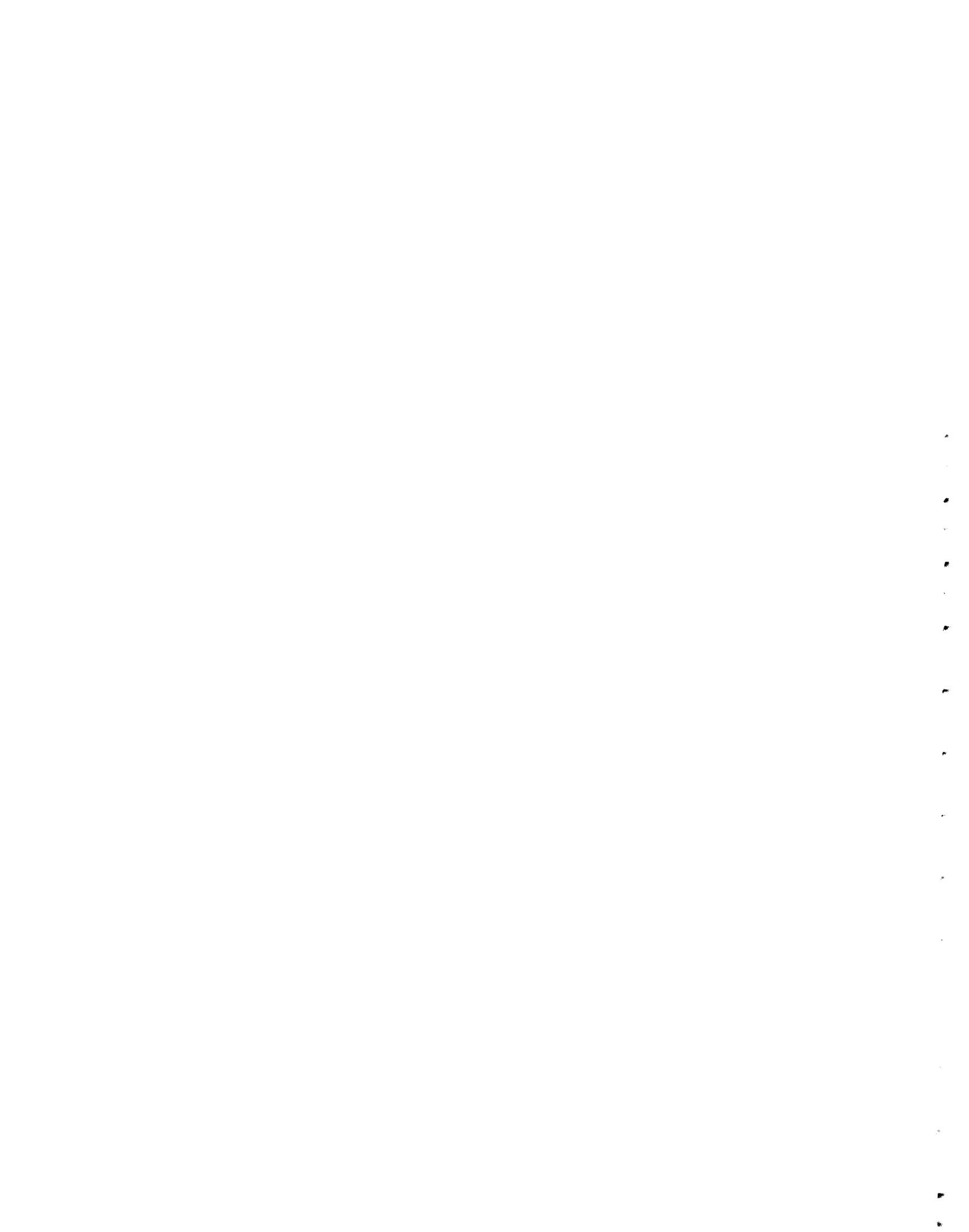
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LIST OF RSIC CODE PACKAGES, TITLES,  
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- CCC-1: 14-0 and 14-3, and Auxiliary Routine  
 KERNEL INTEGRATION CODE - CALCULATED SOURCES DESCRIBED IN  
 CYLINDRICAL COORDINATE SYSTEM, contributed by Nuclear Materials  
 and Propulsion Operation, General Electric, Cincinnati, Ohio.  
 FAP; IBM 704 and 7090  
 (References: XDC 59-2-16 and XDC 59-3-52)
- CCC-2: 14-1 and 14-3, and Auxiliary Routine  
 KERNEL INTEGRATION CODE - INPUT SOURCES DESCRIBED IN  
 CYLINDRICAL COORDINATE SYSTEM, contributed by Nuclear Materials  
 and Propulsion Operation, General Electric, Cincinnati, Ohio.  
 FAP; IBM 704 and 7090  
 (References: XDC 59-2-16 and XDC 59-3-52)
- CCC-3: 14-2 and 14-3, and Auxiliary Routine  
 KERNEL INTEGRATION CODE - SOURCES DESCRIBED IN RECTANGULAR  
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 Propulsion Operation, General Electric, Cincinnati, Ohio.  
 FAP; IBM 704 and 7090  
 (References: XDC 59-6-173 and XDC 59-3-52)
- CCC-4 15-2  
 MONTE CARLO CALCULATION - GAMMA-RAY SCATTERING IN AIR,  
 contributed by Nuclear Materials and Propulsion Operation,  
 General Electric, Cincinnati, Ohio.  
 FAP; IBM 704 and 7090  
 (References: XDC 61-5-1 and DC 60-10-150)
- CCC-5 C-17 and Auxiliary Routine  
 KERNEL INTEGRATION CODE - FRUSTRA OF RECTANGULAR PYRAMIDS AND  
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 Facility, General Dynamics, Fort Worth, Texas.  
 FAP, FORTRAN; IBM 704, 709, 7090, and 7094  
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- CCC-6: L-63 and Auxiliary Routine  
 KERNEL INTEGRATION CODE - CYLINDERS, SPHERES, AND COMPLEX  
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- CCC-7: NTC and Auxiliary Routines  
 MONTE CARLO HIGH ENERGY NUCLEON TRANSPORT CODE INCORPORATING  
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 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: FORTRAN II 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, Atomics 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 Atomics  
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  
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 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 DISTRIBU-  
 TIONS 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)

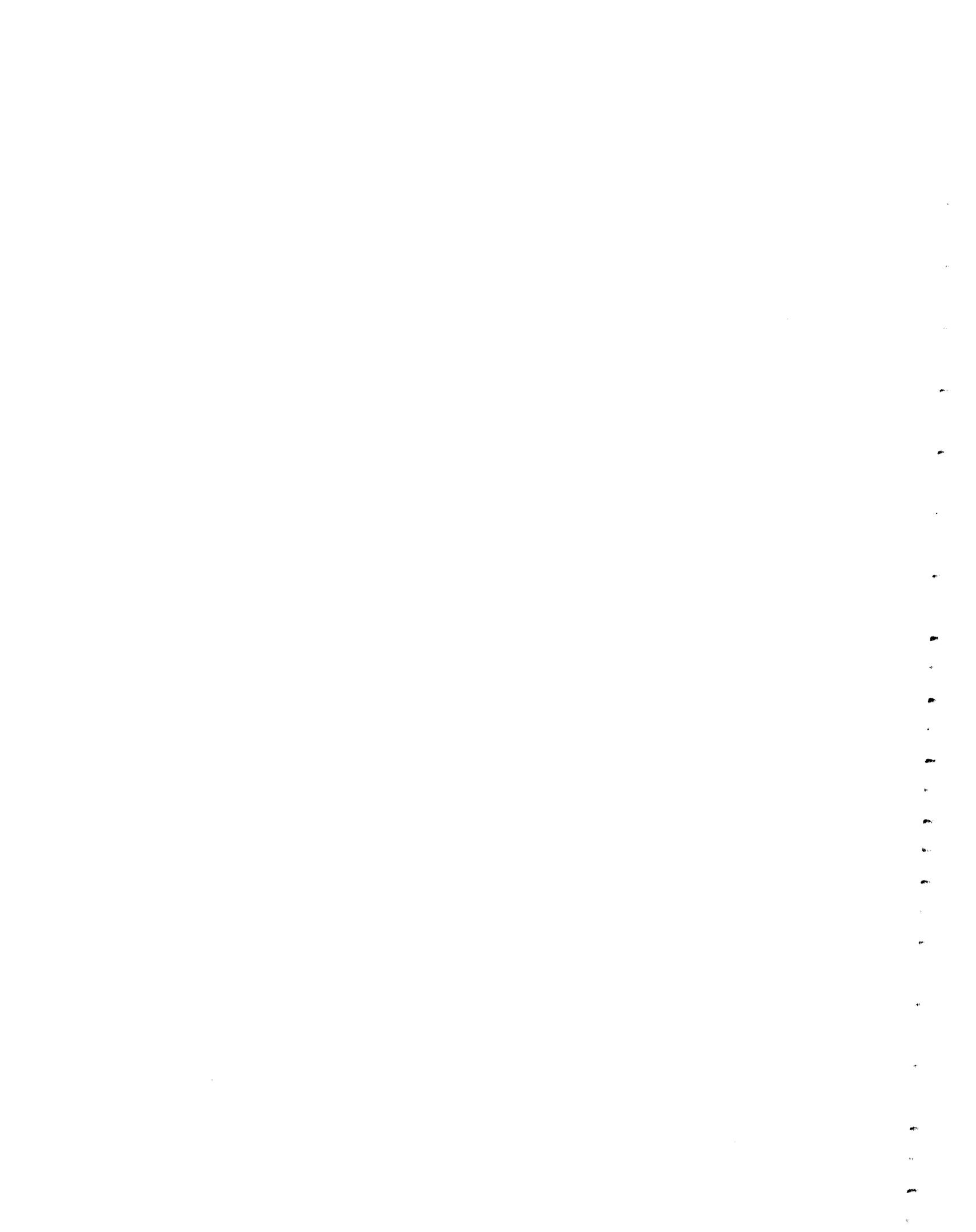
- CCC-91: NEFIRS  
MULTIGROUP SPINNEY METHOD REMOVAL-DIFFUSION CODE FOR NEUTRONS,  
contributed by Gulf General Atomic, San Diego, California.  
FORTRAN IV and MAP; IBM 360/75  
(Reference: GA-8069)
- CCC-92: SAP - N and G  
NEUTRON AND GAMMA-RAY ALBEDO MODEL SCATTER SHIELD ANALYSIS CODE,  
contributed 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/ETRAN 15: UNIVAC 1108  
 FORTRAN IV; CCC-107B/ETRAN 16 and CCC-107C/ETRAN 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)

ABSTRACTS  
OF  
RSIC  
COMPUTER CODE PACKAGES



## RSIC CODE PACKAGE CCC-59

## 1. NAME AND TITLE OF CODE

COMBINE: Kernel Integration of Transmission and Reflection Probabilities for Shielded Compartments.

COMBINE represents an extensive revision and extension of an auxiliary routine, R-35, packaged in CCC-10/C-18. The procedure consists of the following codes named by the GD/FW internal accounting designations.

COMBINE I: Transmission Calculation (E91 and E20, differing only in coding language).

COMBINE II: Reflection Calculation (F29).

## AUXILIARY ROUTINE

F 57: Data Generator.

## 2. CONTRIBUTORS

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.

## 3. CODING LANGUAGE AND COMPUTER

FORTRAN II (E91 and F29); FORTRAN IV (E20); IBM 7090 and 7094.

## 4. NATURE OF PROBLEM SOLVED

COMBINE is a set of integration procedures designed to calculate the radiation flux and dose at one or more detector points inside a shielded compartment by combining (1) the radiation incident on the shielded compartment, (2) the slab transmission and reflection probabilities generated by SPARC (CCC-58) or similar Monte Carlo codes, and (3) the definition of the geometry of the shielded compartment.

## 5. METHOD OF SOLUTION

The COMBINE procedure consists of two separate calculations: transmission, using E91 or E20, and reflection, using F29.

The task is accomplished by associating the proper SPARC slab transmission and reflection data with each segment of the compartment, sampling the energy and angular distribution of the radiation environment, and integrating over the interior surfaces while considering slab transmission and reflection scattering angles.

## 6. RESTRICTIONS OR LIMITATIONS

E91 and E20: Up to 10 detectors, 20 regions, and 6 incident energy groups may be considered in one problem. Edge effects of adjoining slabs are not considered.

F29: No more than one detector, 20 regions, and 6 incident energy groups may be considered in one problem. Edge effects of adjoining slabs are not considered.

## 7. TYPICAL RUNNING TIME

No studies have been made to indicate typical running time. Estimated times to run the sample problems are as follows:

E91 - 5 minutes, 30 seconds; F29 - .09 hours; and E20 - time not currently available.

## 8. COMPUTER HARDWARE REQUIREMENTS

The codes were designed for the 32K IBM 7090 and 7094. Twelve tape units are required for E20 and 8 tape units for the others.

## 9. COMPUTER SOFTWARE REQUIREMENTS

The IBM FORTRAN II Monitor System may be used for E91 and F29. E20 is operable in the IBM IBJOB Operating System on the FORTRAN IV IBSYS Monitor.

## 10. REFERENCES

R. F. Falkenbury, J. L. Hunter, and J. R. Stokes, "Radiological Armor Program, Methods and Data Compilation, Vol. 5B. Computer Procedures: COMBINE Transmission Calculation (E20)", FZK-200-5B (June 1965).

J. L. Hunter, "Radiological Armor Program, Methods and Data Compilation, Vol. 5B. Computer Procedures: COMBINE Transmission Calculation (E20), Programmers Manual Supplement", FZK-200-5B Supp (June 1965).

R. F. Falkenbury, J. L. Hunter, and J. R. Stokes, "Computer Procedure: COMBINE Transmission Calculation (E91)", FZK-235 (June 1965).

J. L. Hunter, C. W. Austin, and J. B. Wyatt, "Computer Procedure: COMBINE Transmission Calculation (E91), Programmers Manual Supplement", FZK-235 Supp (June 1965).

R. F. Falkenbury and J. L. Hunter, "Radiological Armor Program Methods and Data Compilation, Volume 5C. Computer Procedures: COMBINE Reflection Calculation (F29)", FZK-200-5C (June 1965).

J. L. Hunter, C. W. Austin, D. H. Huckaby, and J. R. Stokes, "Radiological Armor Program Methods and Data Compilation, Volume 5C. Computer Procedures: COMBINE Reflection Calculation (F29), Programmers Manual Supplement", FZK-200-5C Supp (June 1965).

## 11. CONTENTS OF CODE PACKAGE

The package contains the following items:

- a. the referenced documents, and
- b. a reel of magnetic tape for each of the three code versions on which is written in separate files: the BCD source card decks, the binary card decks, and BCD input and output 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, indicating which code version is desired.

13. DATE OF ABSTRACT

January 1968.

## RSIC CODE PACKAGE CCC-60

## 1. NAME AND TITLE OF CODE

SDC: Kernel Integration Shield Design Code for Radioactive Fuel Handling Facilities.

## AUXILIARY ROUTINES

TABLES: Data Edit.

PHOEBE: Source Strength Generator.

The PHOEBE code is a revised and expanded version of the Internuclear Company EAPRR Code.

The code package is also available in the ENEA Computer Programme Library.

## 2. CONTRIBUTOR

Chemical Technology Division, Oak Ridge National Laboratory, Oak Ridge, Tennessee.

## 3. CODING LANGUAGE AND COMPUTER

(A) FORTRAN II; IBM 7090.

(B) FORTRAN IV; IBM 7090.

(C) FORTRAN IV; IBM 360.

(D) FORTRAN 63; CDC 1604.

## 4. NATURE OF PROBLEM SOLVED

SDC is designed to calculate the gamma-ray shielding requirements for chemical processing, fabrication, or fuel handling facilities. It will handle 13 source geometries (point, line, disk, plane, slab, cylinder with shield at side, cylinder with shield at end, sphere, ring, rod cluster, skew line, annular cylinder with shield at side, and annular cylinder with shield at end) either unshielded or shielded by slab shields. Materials of construction for shield, cladding, or source volume may be selected from a list of 17. As many as 12 gamma-ray energy groups, covering an energy range of 0.10 to 10 Mev, with corresponding source

strengths, may be used to describe the gamma-ray spectrum.

The PHOEBE code was developed to calculate the decay properties of mixed fission products resulting from the fission of  $^{235}\text{U}$ . Fission product beta-ray and gamma-ray activity and spectra are the primary quantities calculated.

#### 5. METHOD OF SOLUTION

Integration of the basic exponential attenuation point kernel over the various geometries provides the uncollided gamma-ray flux in SDC. Speed is achieved by utilizing many of the integrations found in Rockwell.\* Biological dose rate was obtained by multiplying this uncollided flux by the product of a flux-weighted buildup factor and a dose-conversion factor. Two major options in the code permit calculation of (1) required shield thickness when a dose-rate level is specified, or (2) dose rate when the shield thickness is given. Calculation of dose rates from unshielded sources as well as surface intensities for cylinders and spheres is also included.

Results from PHOEBE are given in terms of photon-emission rates for each group, gamma-ray energy emission rates for each group, total gamma-ray emission rate, total gamma-ray energy emission rate, beta particle emission, total average beta-ray energy emission rate, and total energy release rate.

#### 6. RESTRICTIONS OR LIMITATIONS

12 energy groups, 0.1 to 10 Mev.

17 materials.

#### 7. TYPICAL RUNNING TIME

Average running time for SDC: 0.01 hour per case. No study has been made for typical running time for PHOEBE. Estimated running times for sample problems: SDC, 8 minutes; PHOEBE, 2 minutes; and TABLES, 2 minutes.

#### 8. COMPUTER HARDWARE REQUIREMENTS

The code is operable on the IBM 7090, IBM 360, and the CDC 1604. Only 3 tape units are required.

## 9. COMPUTER SOFTWARE REQUIREMENTS

The following software may be used:

- (A) The IBM FORTRAN II Monitor System.
- (B) The IBM FORTRAN IV IJOB Monitor System.
- (C) The IBM FORTRAN (level H) Compiler for the 360.
- (D) The FORTRAN 63 COOP Monitor System.

Normally, only systems, input and output tapes are used. A scratch tape is used by TABLES.

## 10. REFERENCES

E. D. Arnold and B. F. Maskewitz, "SDC, A Shielding Design Calculation Code for Fuel-Handling Facilities", ORNL-3041 (March 1966).

E. D. Arnold, "PHOEBE - A Code for Calculating Beta and Gamma Activity and Spectra for <sup>235</sup>U Fission Products", ORNL-3931 (March 1966).

\*T. Rockwell, "Reactor Shielding Design Manual", D. Van Nostrand Company (1956).

## 11. CONTENTS OF CODE PACKAGE

The package contains the following items:

- a. the first two referenced documents, and
- b. a reel of tape for each of the different versions(A-D) on which is written in 10 separate files: the BCD source card decks, the binary card decks set up with BCD input for a sample problem and output from 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, indicating which version (A through D) is wanted.

13. DATE OF ABSTRACT  
January 1968.

## RSIC CODE PACKAGE CCC-61

## 1. NAME AND TITLE OF CODE

CEP: Monte Carlo Calculation of Neutron First-Flight Escape Probabilities for Finite Cylinders.

## 2. CONTRIBUTOR

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

## 3. CODING LANGUAGE AND COMPUTER

MAP; IBM 7040.

## 4. NATURE OF PROBLEM SOLVED

CEP is a Monte Carlo program for computing the first-flight escape probability of neutrons from finite cylinders. Either one cylinder or two concentric cylinders (of which the inner may be empty) of the same height may be used.

## 5. METHOD OF SOLUTION

Each cylinder is subdivided into smaller cylindrical volumes, or cells, for a uniform neutron source distribution. Or, alternately, a non-uniform source distribution or weight may be read in as a function of radius and height for each cell.

A representative particle from each cell is followed along randomly generated direction cosines until it escapes from the cylindrical system, producing a probability of escape for the particle in that cell. The sum of the resulting probabilities, weight-averaged over all the cells, yields the total neutron escape probability for the cylinder(s). A Student's t-distribution statistical analysis is computed for the probability of escape.

## 6. RESTRICTIONS OR LIMITATIONS

The maximum number of cells into which the cylinders may be subdivided is 19,500.

The code has restart capabilities.

## 7. TYPICAL RUNNING TIME

Running times are dependent upon the statistical accuracy requested by the user in the input.

In a single cylinder about 12,700 cells per minute can be processed by the IBM 7040. For two cylinders the rate is about 9750 cells per minute, and for an annulus, about 8260 cells per minute.

Estimated running time of the sample problem on the IBM 7040: 3 minutes.

## 8. COMPUTER HARDWARE REQUIREMENTS

The code was designed for the IBM 7040 with 32K core. Standard input and output equipment is used. No special tape units are needed.

Information is taken from the time clock.

## 9. COMPUTER SOFTWARE REQUIREMENTS

The code is operable on the standard IBM 7040 IBSYS Monitor System.

A square root subroutine has been made a part of the code package.

## 10. REFERENCE

W. K. Foell, C. W. Berner, and S. Tong, "CEP, A Monte Carlo Program for the Calculation of First-flight Escape Probabilities for Finite Cylinders," IDO-17075 (September 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 2 files: the source card deck, the binary card deck and input for a sample problem, and
- c. an output listing from 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

January 1968.



## RSIC CODE PACKAGE CCC-62

## 1. NAME AND TITLE OF CODE

K009: Solid Angle Integration Charged Particle Penetration Code.

## 2. CONTRIBUTOR

NASA Manned Spacecraft Center, Houston, Texas.

## 3. CODING LANGUAGE AND COMPUTER

FORTRAN II; IBM 7090.

## 4. NATURE OF PROBLEM SOLVED

K009 was designed to evaluate space radiation hazards to man and equipment by performing a dose analysis on a cylindrical phantom man located inside a solid-angle-defined-spacecraft and by generating tables of dose values depending on spherical shell thickness.

## 5. METHOD OF SOLUTION

The K009 code uses a ray tracing method to calculate the point dose and/or dose to a cylindrical phantom man from primary charged particles inside a solid-angle-defined-spacecraft or shielding configuration. These calculations are made by calculating the attenuation of the particles through the shield material and tissue along the ray to the dose point and integrating numerically the dose over incident energy. Calculation of dose as a function of thickness is made in the same manner, using  $4\pi$  solid angle shields. The Gaussian numerical integration procedure is so formulated that the dose contribution from various incident energy increments (input as integration limits) can be determined.

Two subroutines, designed to be written by the user, are incorporated in K009. These subroutines, DFLUX and CHANGE, are not a fixed part of the program and are included to make it more flexible. However, operations performed within or by them must

comply with the requirements of the main code. Subroutine DFLUX is designed to input the differential incident particle energy spectrum into the main code. The spectrum can be defined in any form, within program requirements, that may apply to the specific needs of the user. Subroutine CHANGE is designed to make the program more flexible in the use of solid angle data. It is used, primarily, to make temporary corrections or revisions in the data without repunching cards. However, it can be used as an input routine.

Dose or dose rate values, depending on the type of input spectrum, are calculated in units of rad (1 rad = 100 ergs./gm) and rem (dose in rem = product of dose in rad and the Relative Biological Effectiveness), and may be printed out for each solid angle if desired.

#### 6. RESTRICTIONS OR LIMITATIONS

The following restrictions apply:

number of particle spectra  $\leq 20$   
number of energy increments  $\leq 100$   
number of solid angles  $\leq 1000$   
number of dose points  $\leq 50$

Storage requirements: extreme cases required less than 20,000 locations.

#### 7. TYPICAL RUNNING TIME

No study has been made to determine typical running time. Estimated running time of the two sample problems: 4 minutes each.

#### 8. COMPUTER HARDWARE REQUIREMENTS

The code was designed for the IBM 7090 computer with 3 tape units being utilized.

#### 9. COMPUTER SOFTWARE REQUIREMENTS

The code was designed for and is operable on the standard IBM FORTRAN II Monitor System. Only input, output, and system tapes are assigned.

## 10. REFERENCE

Alva C. Hardy, "Program K009, MSC Solid Angle Radiation Dose Program", NASA Project No. 3208P (January 1966).

## 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 5 separate files: the source card decks, the binary card decks, BCD input for two sample problems and a BCD output listing from 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

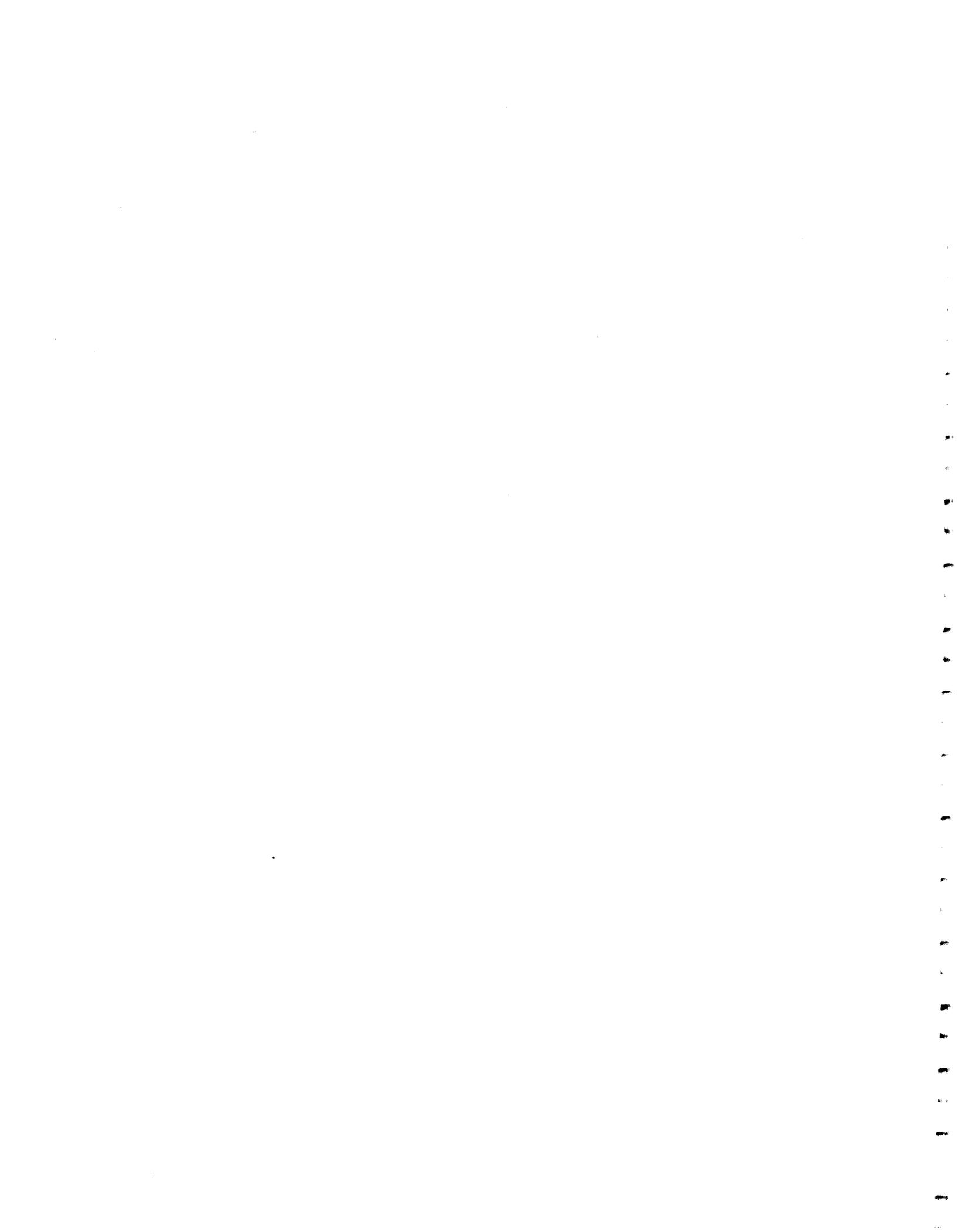
or telephoned to

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

January 1968.



## RSIC CODE PACKAGE CCC-63

## 1. NAME AND TITLE OF CODE

OPEX: Shield Weight Optimization Code.

## 2. CONTRIBUTOR

Atomics International, Canoga Park, California.

## 3. CODING LANGUAGE AND COMPUTER

FORTRAN II; IBM 7090.

## 4. NATURE OF PROBLEM SOLVED

OPEX is designed to find the minimum weight configuration for laminated shields of up to 10 slabs subject to the constraint of a constant total dose rate at some selected detector point.

## 5. METHOD OF SOLUTION

The code uses the method of steepest descent to minimize shield weight. The dose rate is approximated by an exponential fit to data (supplied by the user) obtained from experiments or analytical calculations.

## 6. RESTRICTIONS OR LIMITATIONS

The following limits apply:

number of regions  $\leq 10$

number of iterations permitted  $\leq 50$

number of terms in the expression for dose rate  $\leq 10$ .

## 7. TYPICAL RUNNING TIME

Estimated running time for sample problem: 2 minutes.

## 8. COMPUTER HARDWARE REQUIREMENTS

The code was designed for the IBM 7090 with 3 tape units.

## 9. COMPUTER SOFTWARE REQUIREMENTS

The code will compile and execute in the IBM FORTRAN II Monitor System, with input, output, and system tape assignments

made.

The AI input routine, FORED, is used and is packaged with the code.

10. REFERENCE

R. L. Bernick, "The OPEX Shield Optimization Code",  
NAA-SR-TDR-11516 (July 1965).

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 3 separate files: the source card deck, the binary card deck set up with BCD input for a sample problem, and BCD output for 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

January 1968.

## RSIC CODE PACKAGE CCC-64

## 1. NAME AND TITLE OF CODE

LPSC: Proton Penetration Code - Multilayer Slab Geometry.  
AUXILIARY ROUTINE  
TAPFIX: Data Generator.

## 2. CONTRIBUTOR

NASA Lewis Research Center, Cleveland, Ohio.

## 3. CODING LANGUAGE AND COMPUTER

FORTRAN IV; IBM 7090 and 7094.

## 4. NATURE OF PROBLEM SOLVED

LPSC calculates particle spectra and dose behind multilayer, infinite slabs for any given energy spectrum of protons impinging normally on the shield. Incident primary protons are considered, as well as the following types of secondary radiation: cascade protons, cascade neutrons, and evaporation neutrons.

## 5. METHOD OF SOLUTION

The straightahead approximation technique is used in treating the high-energy "cascade" particles produced from nonelastic nuclear collisions - that is, it is assumed that when a nonelastic collision occurs, the high-energy secondary particles are emitted in the direction of the incident particle. The code assumes that the low-energy "evaporation" secondary neutrons from nonelastic nuclear collisions are emitted isotropically and takes into account this angular dependence. Data developed by Bertini\* is employed for particle production from high-energy nonelastic collisions.

Spectra and doses are calculated inside the shield and at intermediate thicknesses through the shield.

## 6. RESTRICTIONS OR LIMITATIONS

The approximation is assumed to be valid for protons with energies  $\geq 100$  MeV.

## 7. TYPICAL RUNNING TIME

No studies have been made to determine typical running time. Estimated running time for the sample problem on the IBM 7090: TAPFIX - .01 hour; LPSC - .09 hour.

## 8. COMPUTER HARDWARE REQUIREMENTS

The code was designed for the IBM 709<sup>4</sup> and is also operable on the IBM 7090.

A 32,768 core memory and a maximum of 5 tape units are required.

## 9. COMPUTER SOFTWARE REQUIREMENTS

The codes are operable in the FORTRAN IV IJOB Monitor System.

The TAPFIX code is processed first to generate a library (binary) tape to be used as an input tape for LPSC.

## 10. REFERENCES

R. I. Hildebrand and H. E. Renkel, "The Lewis Proton Shielding Code", NASA TM X-52166 (1966).

\*Hugo W. Bertini, "Low Energy Intranuclear Cascade Calculations", Phys. Rev. 131, 1801 (1963).

## 11. CONTENTS OF CODE PACKAGE

The package contains the following items:

- a. the first referenced document, and
- b. a reel of magnetic tape on which is written in 6 files: the source card decks, the binary card decks set up with input for a sample problem, and output from 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

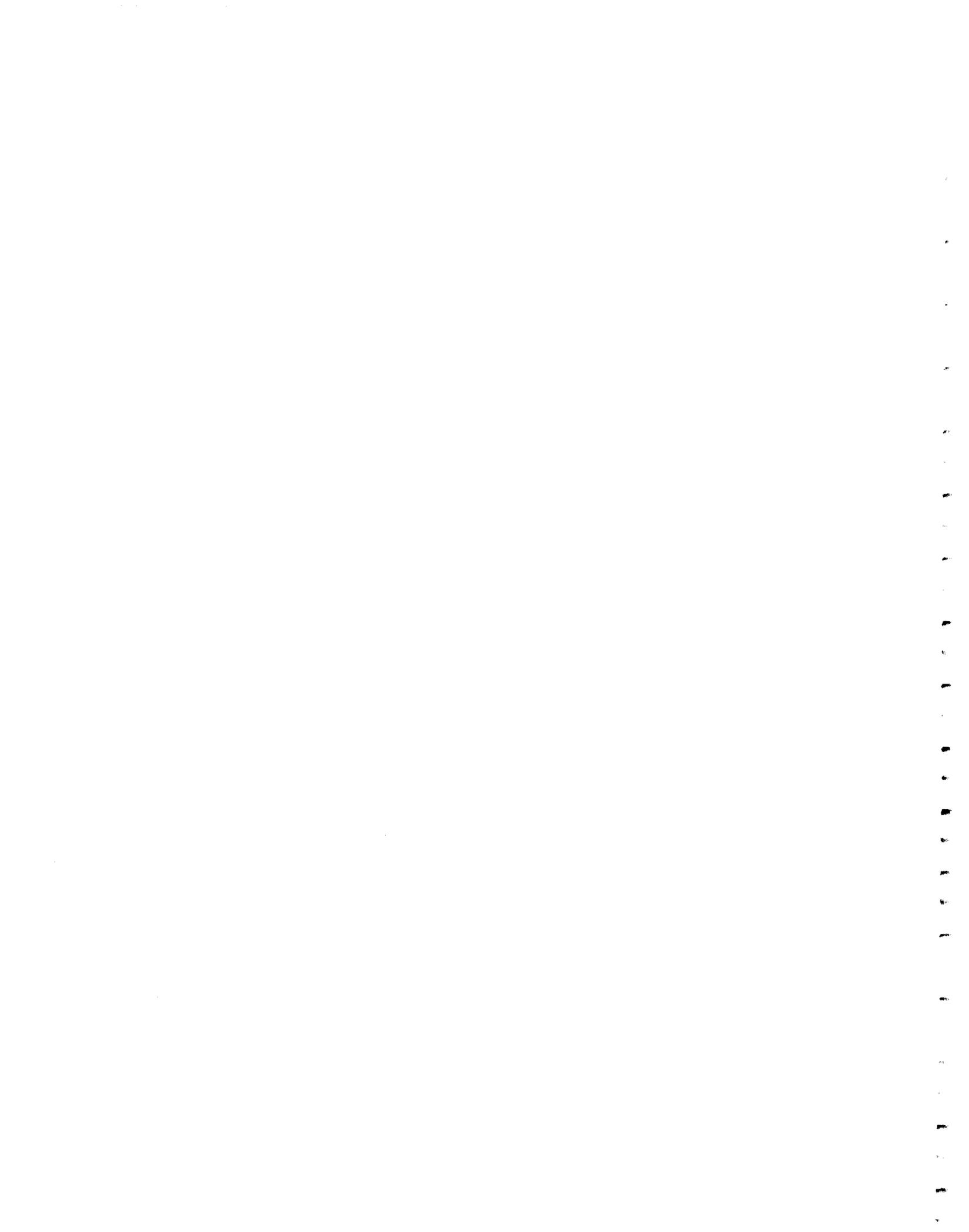
or telephoned to

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

January 1968.



## RSIC CODE PACKAGE CCC-65

## 1. NAME AND TITLE OF CODE

TDSN: Two-Dimensional Multigroup Discrete Ordinates Neutron Transport Code.

## 2. CONTRIBUTOR

NASA Lewis Research Center, Cleveland, Ohio.

## 3. CODING LANGUAGE AND COMPUTER

FORTRAN IV and MAP; IBM 7090 and 7094.

## 4. NATURE OF PROBLEM SOLVED

TDSN solves the linear, time-independent, Boltzmann equation for particle transport for the energy, space, and angular dependence of the neutron distribution in multigroup two-dimensional complex geometric configurations. The code will solve either one-dimensional (slab, cylinder, or sphere) or two-dimensional (x,y, or r,z) problems with either zero-return-current or reflective boundary conditions. The reflective condition for curved boundaries can be either mirror reflection from a plane surface or isotropic reflection.

## 5. METHOD OF SOLUTION

The transport equation is solved by the discrete angular segmentation (Carlson  $S_N$ ) method, a numerical, iterative difference method in which the continuous angular distribution of neutron velocities is represented by considering discrete angular directions. Scaling and over-relaxation methods are used to accelerate the convergence of the neutron fluxes.

Geometric symmetries are used to reduce the number of mesh intervals. The cross sections can be either  $P_1$  or transport corrected  $P_0$  with full up- and down-scattering matrices. Provisions are made to stop and restart a problem after a specified running time or inner iteration count. On restart, the acceleration parameters, convergence criteria, and output options can

be changed.

Output options include activity tables, flux-weighted cross-section averaging, collapsed-group cross sections, disadvantage factors, and edits over specified groups and mesh intervals.

#### 6. RESTRICTIONS OR LIMITATIONS

The array dimensions are flexible. Approximately 15,000 storages are required for the unsubscripted variables and the program, leaving 17,500 storages available for subscripted variables on a 32K computer.

#### 7. TYPICAL RUNNING TIME

Variable. Estimated running time for two sample problems: 0.09 hour.

#### 8. COMPUTER HARDWARE REQUIREMENTS

The code was designed for a 32K IBM 7090 and 7094 with 4 tape units.

#### 9. COMPUTER SOFTWARE REQUIREMENTS

The code is operable in the IBM FORTRAN IV IBSYS Operating System. Tapes assigned are those for the system, input, output, and auxiliary output (required for restart procedure).

#### 10. REFERENCE

Clayton E. Barber, "A FORTRAN IV Two-Dimensional Discrete Angular Segmentation Transport Program", NASA TN D-3573 (August 1966).

#### 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 6 separate files: the source card decks, the binary card decks, BCD input for two sample problems, output listing from the sample problems and sample punched card output for a restart.

## 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

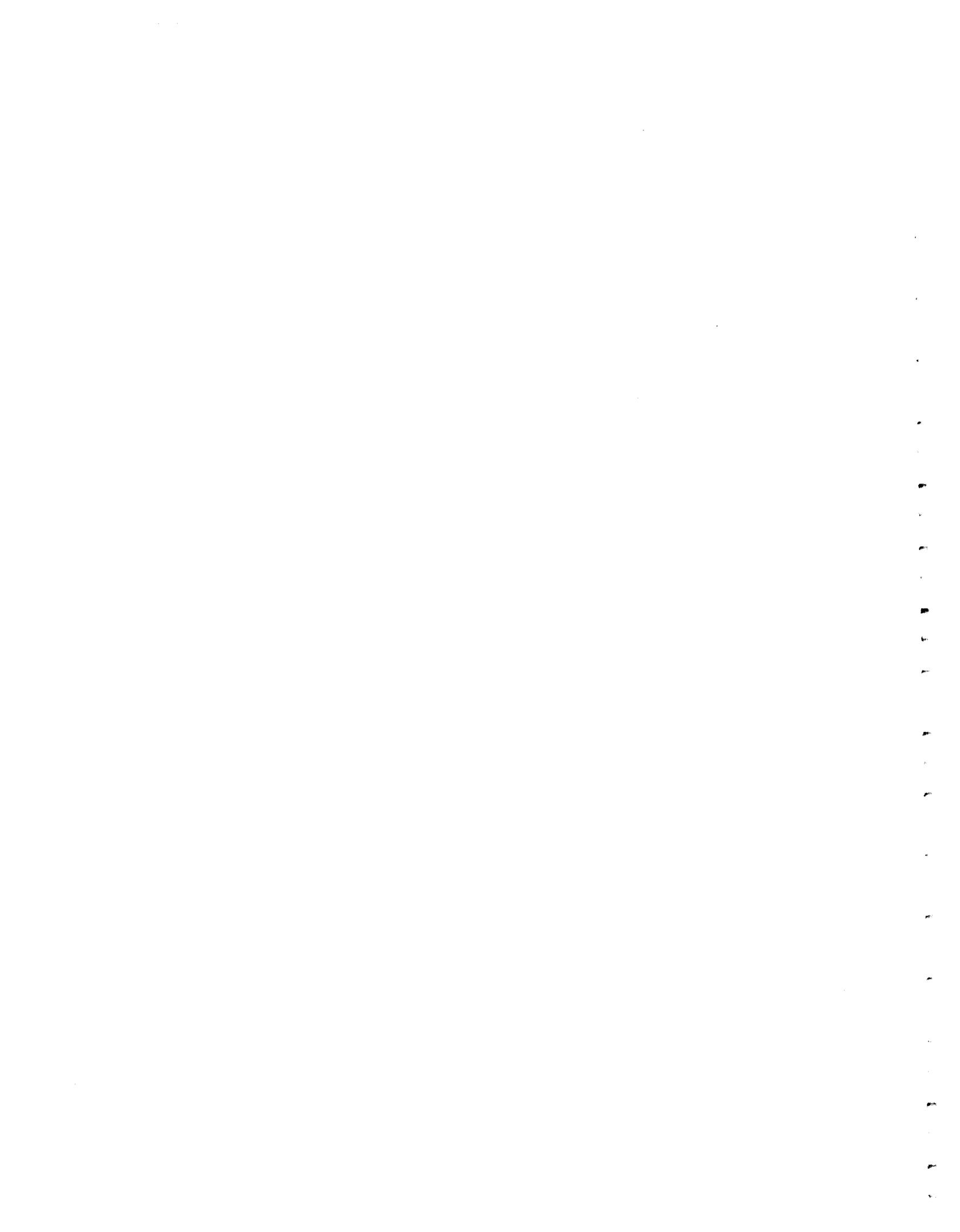
or telephoned to

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.

## 13. DATE OF ABSTRACT

January 1968.



## RSIC CODE PACKAGE CCC-66

## 1. NAME AND TITLE OF CODE

BIGGI 3P: Numerical Gamma-Ray Transport Code for Plane Multilayer Geometry.

The code BIGGI 3P is the first of the BIGGI series to be published. BIGGI stands for "Boltzmanns Integralgleichung fuer Gamma-Intensitaeten (Boltzmanns Integral Equation for Gamma Intensities)", 3 for the third version, and P means that the pair production process can be included, if wanted. The next version, BIGGI 4T, is to be published, together with EUR 3555 e report describing both of them. The first work on the code was done in 1962-63 at Reaktorstation Geesthacht (Elbe), Western Germany.

The code is also available in the ENEA Computer Programme Library, abstracted as E147.

## 2. CONTRIBUTOR

EURATOM, Ispra (Varese) Italy.

## 3. CODING LANGUAGE AND COMPUTER

FORTRAN; IBM 7090.

## 4. NATURE OF PROBLEM SOLVED

BIGGI 3P solves the Boltzmann transport equation in plane multilayer geometry. It computes gamma angular fluxes, spectra, buildup factors and albedos. The sources must be monoenergetic and located on one outer boundary. Their angular distribution can be isotropic or collimated.

## 5. METHOD OF SOLUTION

BIGGI 3P integrates the Boltzmann equation numerically. The basis is the pair of coupled integral equations, discussed (for the

case of neutrons) in Weinberg and Wigner\*. Discrete ordinate meshes are defined in all the three concerned dimensions (angle, space, and gamma wave length), and the integrals figuring in the transport equation are approximated by sums. The program solves the integral equations without iteration, since they belong to the Volterra type (as long as only energetic downscattering is assumed). The gamma cross section (in Thomson Units per electron) of each considered slab must be given in input. The contribution of the low-energetic tail below the cutoff energy to the four buildup factors (energy and particle flux, dose and energy absorption rate) and the two albedos (energy and particle current) is estimated. An exponential transformation allows rather great spatial integration steps, up to 2 or 3 mfp.

#### 6. RESTRICTIONS OR LIMITATIONS

The program allows a maximum of 5 slabs, 8 angular, 26 spatial, and 71 wavelength mesh points.

#### 7. TYPICAL RUNNING TIME

Typical running time on the IBM 7090 is in the order of 20 sec. per spatial point, (somewhat less in strong gamma absorbers as Pb or Sn with a high energy cutoff, somewhat more in weak gamma absorbers as H<sub>2</sub>O with a low energy cutoff). If 2 mfp are taken as the integration step, a 20-mfp shield can be calculated in about 3 or 4 minutes.

Estimated running time of the sample problem: 2 minutes, 10 seconds.

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\*"The Physical Theory of Neutron Chain Reactors", Chicago 1959, p. 228 f. ("third form of the Boltzmann equation").

## 8. COMPUTER HARDWARE REQUIREMENTS

Execution requires a 32K IBM 7090. Tape units are not needed explicitly, with the exception of the standard input, output, and systems procedures.

## 9. COMPUTER SOFTWARE REQUIREMENTS

The code is designed for the FORTRAN II (version 2 or 3) Monitor System of the IBM 7090.

## 10. REFERENCES

H. Penkuhn, "A Numerical Solution of the Boltzmann Equation Applied to Concrete Slabs", EUR 2488.e; ANL 7050, p. 113; and EUR 1643.e, p. 92.

H. Penkuhn, "User's Manual for the Gamma Transport Codes BIGGI 3P and BIGGI 4T", to be published as EUR report.

## 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 files: the source card deck, the binary card deck set up with a sample problem, and output from 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  
Post Office Box X  
Oak Ridge, Tennessee 37830

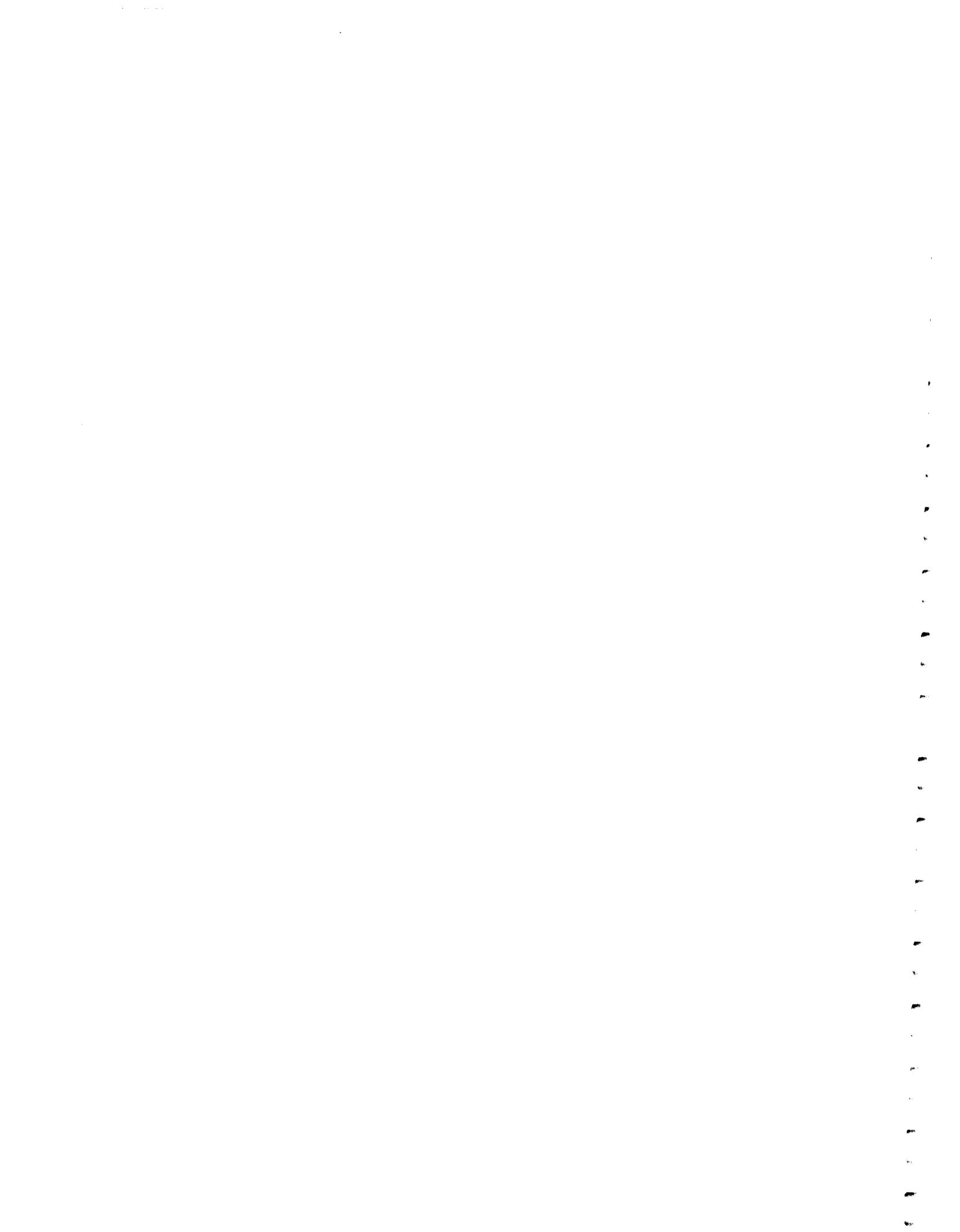
or telephoned to

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

January 1968.



## RSIC CODE PACKAGE CCC-67

## 1. NAME AND TITLE OF CODE

STORM: Solar Flare Radiation Hazard to Earth Orbiting  
Vehicles.

## AUXILIARY ROUTINES

OSO ORBIT: Trajectory Calculation.

SOLAR FLARE: Radiation Environment Code.

DOSE: Dose Rate Calculator.

## 2. CONTRIBUTORS

NASA, Langley Research Center, Hampton, Virginia.

Aerospace Systems and Services, Republic Aviation Corporation,  
Farmingdale, Long Island, New York.

## 3. CODING LANGUAGE AND COMPUTER

FORTRAN IV; IBM 7090 and 7094.

## 4. NATURE OF PROBLEM SOLVED

The primary radiation incident on earth-orbiting vehicles during a solar flare is calculated. The effects of the earth's geomagnetic field have been taken into account, as well as those of a perturbing field due to the geomagnetic storm associated with a solar flare. Simple earth shadow effects have also been considered.

## 5. METHOD OF SOLUTION

The solar flare radiation hazard is assumed to be due to increased solar cosmic radiation and the geomagnetic storm associated with a flare. The effect of the former is obvious. The latter is caused by a change in the earth's magnetic field, resulting in a change in the natural magnetic shielding that permits more particles of greater or lesser energy to arrive than during the "quiet" magnetic state. It is further assumed that the distribution of particles a large distance from the earth's field is isotropic, and an application of Liouville's Theorem indicates

that the intensity of particles in any allowed direction is the same as at their starting point. It therefore suffices to determine the allowed directions at the point in question. In the absence of magnetic effects other than that due to the earth's dipole, it is assumed that these directions are given by the classical Störmer theory, neglecting earth's shadow effect. In the presence of a magnetic disturbance the allowed Störmer cone is modified. In calculating this modification it is assumed that the cylindrical symmetry of the problem is not broken, thus enabling the modified Störmer integral of motion to be solved.

#### 6. RESTRICTIONS OR LIMITATIONS

No information on restrictions or limitations is available at this time.

#### 7. TYPICAL RUNNING TIME

No studies have been made to determine typical running time. Estimated running times for the sample problem on the IBM 7090 in hours: OSO ORBIT, .02; SOLAR FLARE, .04; and DOSE, .02.

#### 8. COMPUTER HARDWARE REQUIREMENTS

The codes were designed for the IBM 709<sup>4</sup> and are compatible with the IBM 7090. A maximum of 5 tape units are required.

#### 9. COMPUTER SOFTWARE REQUIREMENTS

The FORTRAN IV IJOB Monitor System with standard systems, input, and output tapes is required. In addition, the following tapes are required:

1. OSO ORBIT, one pool tape which is written to be used as input to SOLAR FLARE,
2. SOLAR FLARE, tape in 1) above and another pool tape which is written to be used as input to DOSE, and
3. DOSE, tape in 2) above.

## 10. REFERENCES

E. Kuhn, W. T. Payne, and F. E. Schwamb, "Solar Flare Hazard to Earth-Orbiting Vehicles", RAC 1395-1 (July 1964).

E. Kuhn, W. T. Payne, and D. A. Levine, "Solar Flare Hazard to Earth-Orbiting Vehicles", RAC 1395-2 (January 1965).

"Solar Flare Hazard to Earth-Orbiting Vehicles", FHR-1395-3 (April 1966).

## 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 9 files: the source card decks, the binary card decks set up with a sample problem, and output from 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  
P. O. Box X  
Oak Ridge, Tennessee 37830

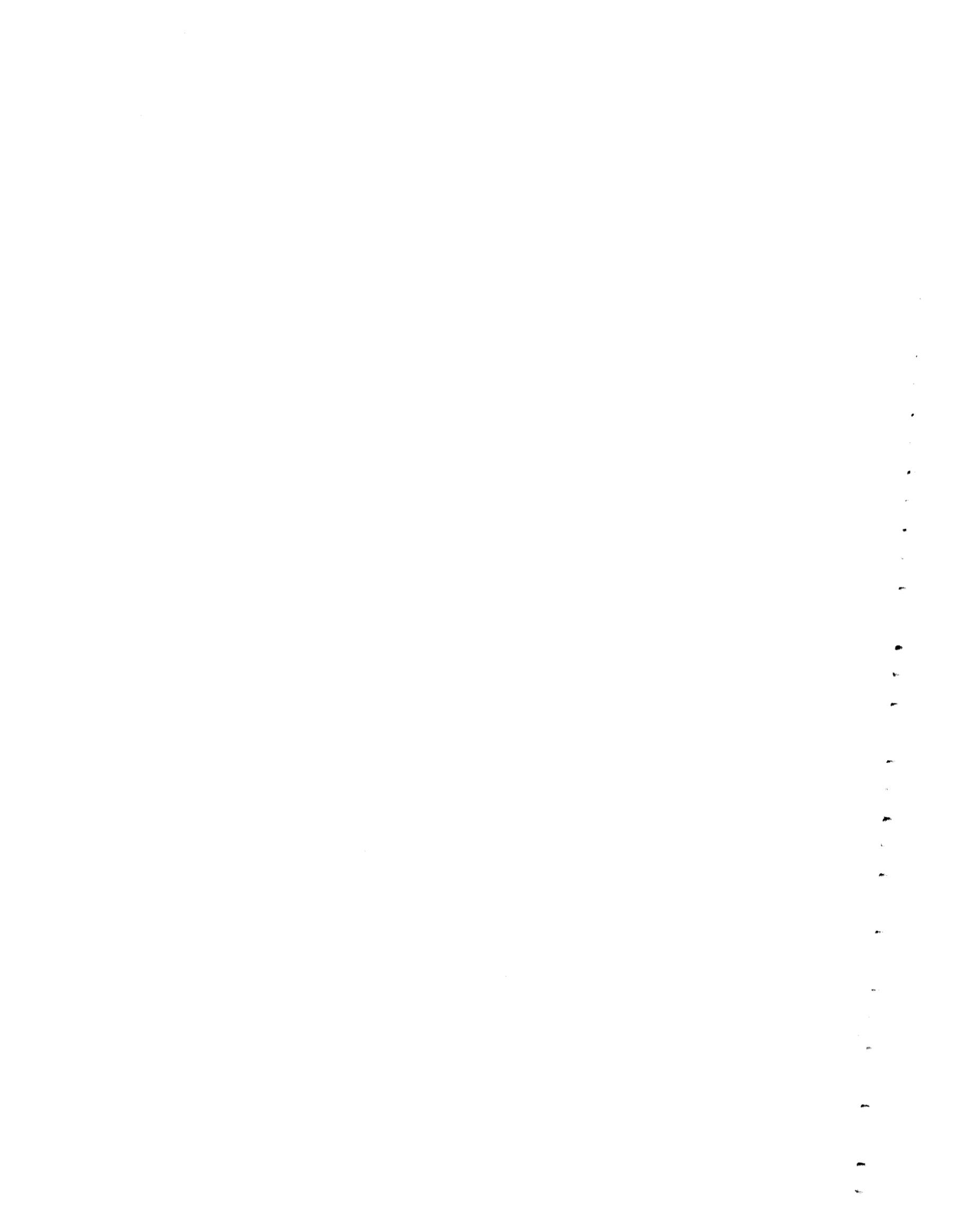
or telephoned to

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.

## 13. DATE OF ABSTRACT

January 1968.



## RSIC CODE PACKAGE CCC-68

## 1. NAME AND TITLE OF CODE

TYCHE III: Monte Carlo Code - Neutron Slowing Down Moments  
in an Infinite Homogeneous Medium.

TYCHE III is also carried in the Argonne Code Center as Reactor  
Code Abstract 149, and in the ENEA Computer Programme Library as  
N2-44.

## 2. CONTRIBUTOR

Atomics International, Canoga Park, California.

## 3. CODING LANGUAGE AND COMPUTER

FORTRAN and FAP; IBM 7090 and 7094.

## 4. NATURE OF PROBLEM SOLVED

TYCHE evaluates the second, fourth, and sixth moments of the  
neutron slowing down density distribution in an infinite homogeneous  
medium.

## 5. METHOD OF SOLUTION

TYCHE is a Monte Carlo code which uses analytic techniques to  
reduce the number of random variables and recursive relations to  
generate the moments. Neutron weights are used to avoid the termin-  
ation of a history by absorption and minimize the running time.

Cross sections and anisotropic scattering coefficients for ten  
elements are packaged with the code.

Provisions are made for restart of non-converged problems,  
graphical displays of the moments and average fission energy as a  
function of the number of sets of histories and calculation of the  
correction to flux moments.

## 6. RESTRICTIONS OR LIMITATIONS

No more than 1200 points can be used for the energy grid, five  
elements and 150 sets of anisotropic scattering coefficient sets

per element.

#### 7. TYPICAL RUNNING TIME

One may expect to run approximately 2000 histories per minute in predominantly homogeneous media and about 350 in carbon on an IBM 7094. Estimated running time for three sample problems: 4 minutes and 30 seconds.

#### 8. COMPUTER HARDWARE REQUIREMENTS

The code was designed for the IBM 32K 7090 and 7094 with seven tape units.

#### 9. COMPUTER SOFTWARE REQUIREMENTS

The code is operable in the FORTRAN II Monitor System, using systems, input, output, chain, and 3 pool tapes.

Nonstandard library routines have been packaged. The main code was altered by RSIC to bypass plotting in running the sample problems. Both versions are packaged.

#### 10. REFERENCES

R. A. Blaine, "TYCHE, A Monte Carlo Slowing Down Code", NAA-SR-7357 (June 1962).

R. A. Blaine, "Improvements to the TYCHE Moments Code and Operating Instructions for TYCHE III", NAA-SR-MEMO-9802 (May 1964).

R. A. Blaine, "ISRCH, A Binary Search FORTRAN Function Sub-program", NAA-SR-MEMO-9721 (March 1964).

M. Hoffman and W. A. Rhoades, "AICRT 3 - A General Code for Display of Digital Data", NAA-SR-MEMO-9069 (October 1963).

#### 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 6 files: the BCD source card decks, the binary card decks set up with BCD input for three sample problems, the BCD output listing for the sample problems and the punched card

restart output. The library of cross sections is included with the problem 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  
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.

13. DATE OF ABSTRACT

January 1968.



## RSIC CODE PACKAGE CCC-69

## 1. NAME AND TITLE OF CODE

CURIE-DOSE-THUNDERHEAD: Calculation of External and Internal Dose from a Radioactive Cloud.

The code is also listed in the Argonne Code Center collection as Reactor Code Abstract 196, and in the ENEA Computer Programme Library as USCC-196.

## 2. CONTRIBUTOR

Atomics International, Canoga Park, California.

## 3. CODING LANGUAGE AND COMPUTER

FORTRAN II and FAP; IBM 7090 and 7094.

## 4. NATURE OF PROBLEM SOLVED

CURIE calculates the fission product inventory produced in a reactor during constant power operation and releases different percentages of the noble gases, halogens, and particulates to the atmosphere at reactor shutdown. DOSE calculates the total dose to 13 internal body organs resulting from inhalation of the passing radioactive cloud. THUNDERHEAD calculates the external cloud gamma-ray exposure from the released fission products.

## 5. METHOD OF SOLUTION

Differential equations relating the number of atoms of a fission product produced to the reactor power and operating time are solved. The same equations are solved repeatedly after reactor shutdown with the power set equal to zero and the operating time replaced by the decay time, or time since reactor shutdown. The particular fraction of the fission product inventory released to the atmosphere at reactor shutdown is allowed to drift downwind at the average wind speed while diffusing in the off-axis and vertical directions as determined by the Sutton atmospheric diffusion equation. A receptor located downwind inhales a portion of the passing

radioactive cloud and receives a small dose to each internal body organ from the isotopes inhaled. The total dose is the sum of the individual doses. The external cloud gamma exposure dose is obtained by considering the cloud to be an expanding eclipse of unit thickness whose semi-major and semi-minor axes are determined by the horizontal and vertical atmospheric diffusion parameters,  $C(Y)$  and  $C(Z)$ , respectively. This cloud activity is distributed throughout the cloud as several unshielded point sources. The cloud is positioned by the code and the incremental dose rate from each source is determined. This is repeated for several cloud positions, and the total external cloud exposure is obtained by Gaussian integration.

Different percentages of noble gases, halogens, cesium, and the rest of the particulates, respectively, may be released at reactor shutdown. Variable power histories may be approximated by replacing the actual history with a series of step-functions and considering the shutdown fission product inventory of one step to be the initial inventory of the next one. The receptor can be located anywhere downwind in DOSE and anywhere in space in THUNDERHEAD.

#### 6. RESTRICTIONS OR LIMITATIONS

Decay times should be entered in ascending order with the first time being 0 seconds if a receptor position for THUNDERHEAD is less than 300 meters downwind of the release point. The last decay time should represent a cloud position at least 300 meters downwind of any of the receptor points.

#### 7. TYPICAL RUNNING TIME

Typical running time: 5 minutes for 17 decay times.

Estimated running time for sample problem: 0.13 hour on the IBM 7090.

#### 8. COMPUTER HARDWARE REQUIREMENTS

The program was designed for the IBM 7094, using 11 tape units. A 32K memory is required.

## 9. COMPUTER SOFTWARE REQUIREMENTS

The program may be compiled and executed on the FORTRAN II Monitor System. It is a 4-link chain job and uses 8 pool tapes in addition to standard I-O and Systems tapes.

Non-standard library routines are included in the code package.

## 10. REFERENCE

G. P. Kenfield, W. R. Lahs, W. B. Sayer, R. G. MacAdams, N. A. Harris, "CURIE-DOSE-THUNDERHEAD - A Digital Computer Program for External and Internal Radiation Dose Calculations", NAA-SR-8884 (June 1965).

## 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 3 files: the BCD source card deck, the binary card deck, a library of permanent data, BCD input for a sample problem, and output 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  
Post Office Box X  
Oak Ridge, Tennessee 37830

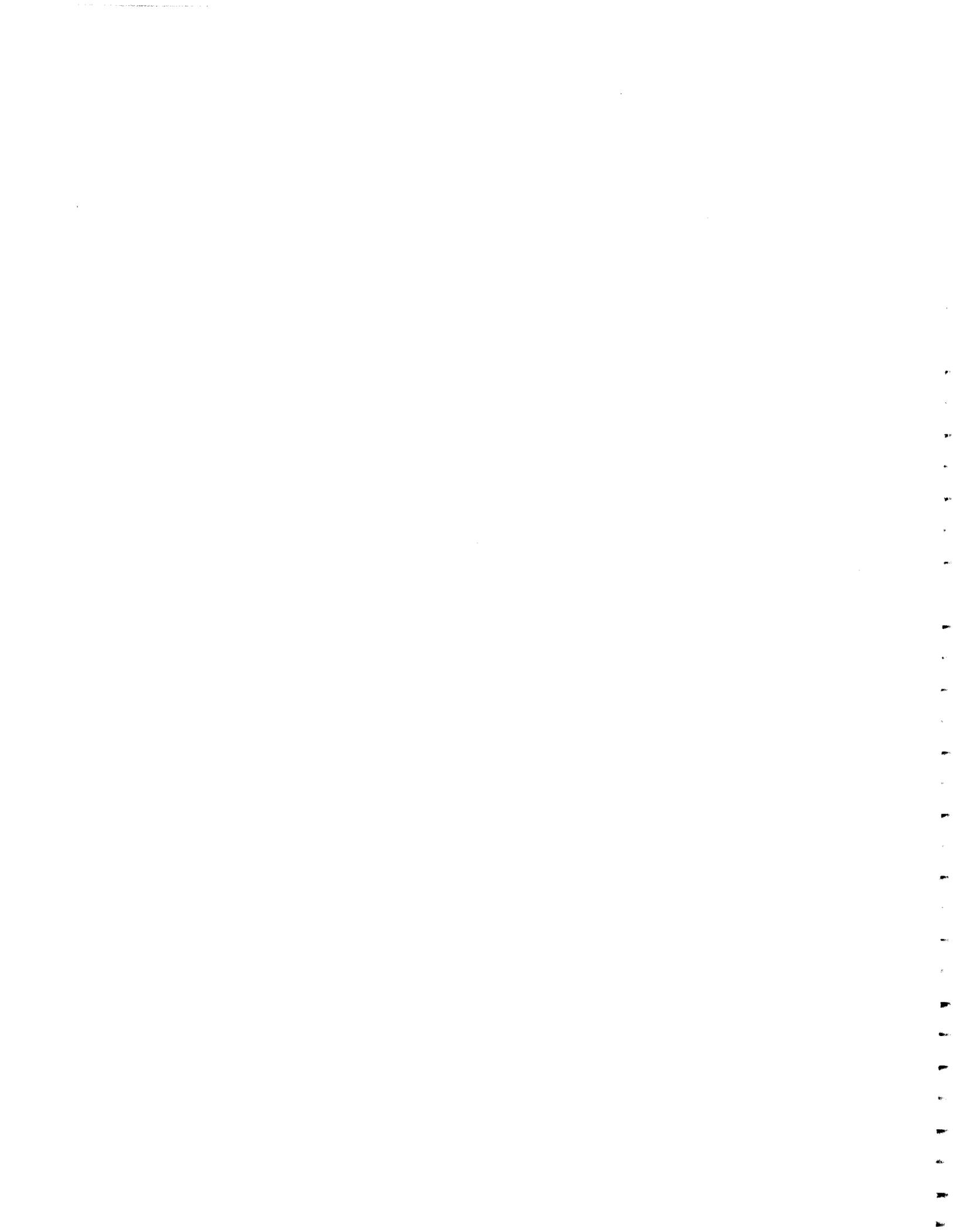
or telephoned to

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

January 1968.



## RSIC CODE PACKAGE CCC-70

## 1. NAME AND TITLE OF CODE

CHARGE: Space Radiation Shielding Code - Proton and Electron Penetration of Multilayered Slabs and Spheres.

## 2. CONTRIBUTOR

Missile and Space Systems Division, Douglas Aircraft Company, Inc., Santa Monica, California.

## 3. CODING LANGUAGE AND COMPUTER

FORTRAN IV; IBM 7090 and 7094.

## 4. NATURE OF PROBLEM SOLVED

STORM was designed to compute the dose rates and energy fluxes behind multilayered shields exposed to electrons and/or protons. Also, the doses due to electron, primary proton, electron bremsstrahlung, secondary proton, and secondary neutron radiations are calculated as a function of penetration into the shield.

## 5. METHOD OF SOLUTION

For protons the straightahead approximation technique is used in treating the high-energy "cascade" particles produced from non-elastic nuclear collisions. The code assumes that the low-energy "evaporation" secondary neutrons from nonelastic nuclear collision are emitted isotropically and takes into account this angular dependence. Data developed by Bertini is employed for particle production from high-energy nonelastic collisions.

For electrons the range-energy calculation assumes the same functional relationship between penetration and energy as was used for protons.

Empirically corrected Born-approximation cross-sections for electron bremsstrahlung production calculations have been employed within the CHARGE code.

## 6. RESTRICTIONS OR LIMITATIONS

Restrictions and limitations of major consequence are not known.

Number of regions in shield  $\leq 100$ .

## 7. TYPICAL RUNNING TIME

No study of typical running time for CHARGE problems has been made by RSIC. Estimated running time for sample problem: 0.06 hour.

## 8. COMPUTER HARDWARE REQUIREMENTS

The code was designed for the 32K IBM 7090 and 7094 computers with provision to handle systems, input, and output.

## 9. COMPUTER SOFTWARE REQUIREMENTS

The code may be compiled and executed on the FORTRAN IV IBJOB Monitor in the IBSYS Operating System with standard I-O, systems, and 2 scratch tapes assigned.

A special FORTRAN IV-written subroutine has been packaged with double precision and input capabilities. Plotting routines (SC 4020) are included but may be bypassed by an input parameter. Provisions are also included to bypass clock sampling and date routines.

## 10. REFERENCE

J. R. Lilley and W. R. Yucker, "CHARGE, A Space Radiation Shielding Code", SM-46335 (April 1965).

## 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 3 files: the source card deck, the binary card deck set up with BCD sample problem data, and a BCD output listing from 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  
Oak Ridge, Tennessee 37830

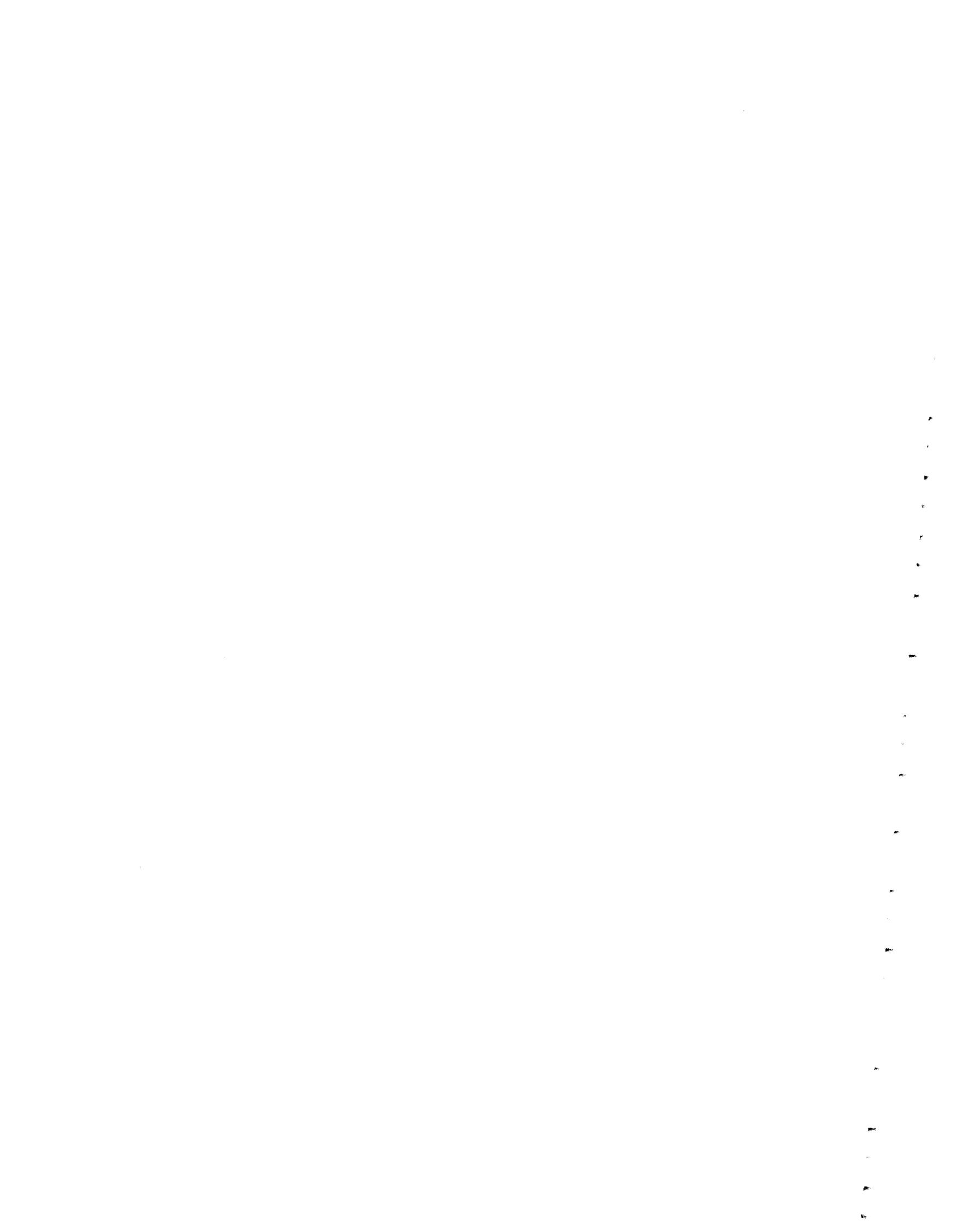
or telephoned to

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

January 1968.



## RSIC CODE PACKAGE CCC-71

## 1. NAME AND TITLE OF CODE

MIST: Multigroup Discrete Ordinates Transport Code for Slab Geometry.

The FORTRAN II version is listed in the Argonne Code Center as Reactor Code Abstract 59 and in the ENEA Computer Programme Library as USCC-59.

## 2. CONTRIBUTOR

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

## 3. CODING LANGUAGE AND COMPUTER

FORTRAN II and IV; IBM 7090.

## 4. NATURE OF PROBLEM SOLVED

MIST obtains the solution to the one-dimensional Boltzmann equation in slab geometry.

## 5. METHOD OF SOLUTION

The numerical approximation used is a linear one that can be described as an extension and generalization of the  $S_N$  approximation. The equations are formulated in terms of a double  $S_N$  approximation and the solution is obtained by a direct method (one requiring no inner iterations).

The boundary conditions for each group may be independently specified and permit very general specifications with respect to

- (a) Perfect mirror reflection or symmetry (by input of mirror albedos).
- (b) Anisotropic diffuse sources (by input of Legendre polynomial coefficients up to  $\lambda = 9$ , or a short table describing a known angular distribution of the flux).
- (c) Isotropic (Lambert surface) reflection.

Isotropic volume sources in each group may also be independently specified. The scattering from one group to another is assumed to be isotropic, but the scattering function within each group can be a second order Legendre polynomial series.

## 6. RESTRICTIONS OR LIMITATIONS

In order to maintain the maximum flexibility, the MIST program is divided into four separate codes. The limit on the number of spatial and mesh points for each code is as follows:

<u>Code</u>	<u>Maximum Number Angular Intervals</u>	<u>Space Points</u>
MIST 4	4	250
MIST 6	6	150
MIST 8	8	100
MIST 10	10	70
Maximum number of energy groups	- 6	
Maximum number of regions	- 40	
Maximum number of materials	- 40	
Maximum number of downscatter groups	- 5	

## 7. TYPICAL RUNNING TIME

Typical running times for a 2 group, 69 point problem requiring six outer iterations to pointwise convergence of 0.001

<u>Angular Intervals</u>	<u>Time (min.)</u>
2	0.58
4	1.03
6	1.68
8	2.52
10	3.56

Problems which require no outer iterations will take a maximum of about one minute.

Estimated running time for the MIST 4 sample problem: 0.03 hour.

## 8. COMPUTER HARDWARE REQUIREMENTS

IBM 7090 with 32K core and 6 tape units.

## 9. COMPUTER SOFTWARE REQUIREMENTS

The codes were designed for the IBM 7090 FORTRAN II Monitor System and have also been made operable on the FORTRAN IV IBSYS Monitor System. Input, output, systems and 3 pool tapes are assigned.

## 10. REFERENCE

G. E. Putnam and D. M. Shapiro, "MIST (Multigroup Inter-nuclear Slab Transport)", IDO-16856 (May 1963).

## 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 the first 4 files: the source card decks, the binary card decks set up with input for a sample problem, and output from the sample problem, for the FORTRAN IV version of the code; and the same information for the FORTRAN II version on files 5 through 8.

## 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

January 1968.



## RSIC CODE PACKAGE CCC-72

## 1. NAME AND TITLE OF CODE

COMPRASH: Spinney Removal-Diffusion Shielding Code.

The code was originally programmed under the name RASH B for the Ferranti Mercury computer and has undergone a series of developmental changes, each carrying the RASH name.

The code package is also available in the ENEA Computer Programme Library, abstracted as E126.

## 2. CONTRIBUTOR

Shielding Group, Reactor Physics Division (AEEW), Atomic Energy Research Establishment, UKAEA, Harwell, England.

## 3. CODING LANGUAGE AND COMPUTER

FORTRAN II and FAP; IBM 7090.

## 4. NATURE OF PROBLEM SOLVED

Fast neutron spectra, thermal neutron flux densities, and the secondary gamma-ray dose rates are calculated for reactor shields in slab geometry.

## 5. METHOD OF SOLUTION

The Spinney removal-diffusion method of calculating neutron transport is a two-step model. The basic assumption is that the penetrating component from a point source can be calculated by a kernel given by

$$\varphi_0(E, R) = \frac{S f(E) e^{-\Sigma_r(E)R}}{4\pi R^2}$$

where  $\varphi_0(R)$  = "removal" flux density

$f(E)$  = fission neutron spectrum

$S$  = source normalization

$\Sigma_r(E)$  = energy dependent "removal cross section"

$R$  = distance from source.

The removal source function

$$S(R,E) = \Sigma_r(E) \varphi_0(E,R)$$

is then made the source term in a conventional multigroup age-diffusion calculation to calculate the diffusing neutron spectra, including a thermal neutron group.

The value of the removal cross section is taken to be the usual transport cross section which is justified by the remarkable success of the model in predicting the thermal neutron flux densities measured in bulk concrete shields.

The secondary gamma ray transport is calculated with the use of an analytic form of the buildup factor.

#### 6. RESTRICTIONS OR LIMITATIONS

There are no known restrictions implied by storage allocation.

#### 7. TYPICAL RUNNING TIME

No study has been made to determine typical running time. Estimated running time for sample problem: 12 minutes.

#### 8. COMPUTER HARDWARE REQUIREMENTS

The code was designed for and is operable on the IBM 32K 7090 computer with 9 tape units.

#### 9. COMPUTER SOFTWARE REQUIREMENTS

The code will compile and operate in the IBM FORTRAN II Monitor System. A special expanded I-O table and several routines from the Risley library have been packaged with the code.

#### 10. REFERENCES

J. Butler, "The Status of Theoretical Methods for Reactor Shield Design", AEEW-R361 (March 1964).

A. F. Avery, J. Clarke, Mrs. A. Hartley, "COMPRASH - (A Preliminary Users Guide Prepared for the ENEA Seminar-Workshop on Shielding Programmes Held at Ispra, April 1966)", AEEW-M 648 (March 1966).

## 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 BCD source card decks, the binary card decks, set up  
with BCD input for a sample problem, a library of cross  
sections in binary, and BCD output from 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  
P. O. Box X  
Oak Ridge, Tennessee 37830

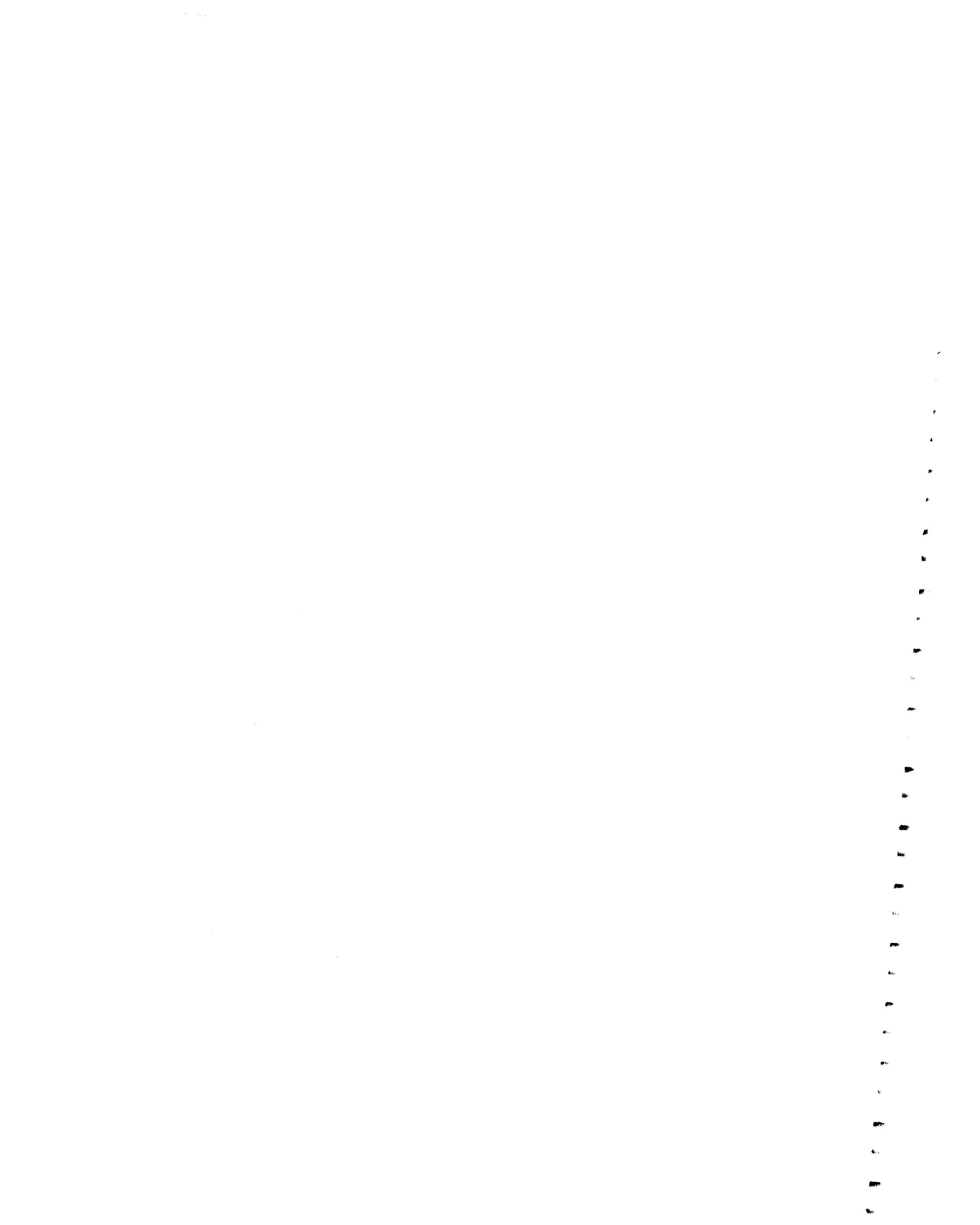
or telephoned to

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.

## 13. DATE OF ABSTRACT

January 1968.



## RSIC CODE PACKAGE CCC-73

## 1. NAME AND TITLE OF CODE

ASTROS: Calculation of Primary and Secondary Proton Dose Rates in Spheres and Slabs of Tissue.

## AUXILIARY ROUTINES

TAPE GEN: Input Data Generator.

## 2. CONTRIBUTOR

Lawrence Radiation Laboratory, Berkeley, California.

## 3. CODING LANGUAGE AND COMPUTER

FORTRAN IV and MAP; IBM 7090 and 7094.

## 4. NATURE OF PROBLEM SOLVED

ASTROS was developed for the depth-dose relation in spheres of tissue due to primary protons and to cascade, evaporation, and hydrogen elastically scattered secondary protons.

The code was designed to calculate depth-dose profiles from isotropic fluxes of monoenergetic protons in mammals from mouse size to man size.

## 5. METHOD OF SOLUTION

The treatment of primary protons is simplified by neglecting straggling and approximating the range-energy relation by  $R = pE^q$ , where  $p$  and  $q$  are constants over each of five energy intervals and  $E$  and  $R$  are respectively the energy and corresponding range of the protons.

Three classes of secondary protons are considered: cascade, evaporation and recoil protons, and for all three classes a Monte Carlo code based on a nuclear model of Metropolis has been used to estimate the energy spectrum and number of secondary protons created.

## 6. RESTRICTIONS OR LIMITATIONS

The code may be used to calculate doses in spheres from 2 to 25 cm. in radius.

## 7. TYPICAL RUNNING TIME

No study has been made by RSIC as to typical running time. Estimated running time of sample problem: TAPE GEN, 0.16 hours, and ASTROS, 0.05 hour.

## 8. COMPUTER HARDWARE REQUIREMENTS

The code was originally designed for the IBM 7044 computer. It is now operable on the IBM 7090 and 7094 with 6 tape units.

## 9. COMPUTER SOFTWARE REQUIREMENTS

The code may be compiled and executed on the FORTRAN IV IJOB Monitor in the IBSYS Operating System on the IBM 7090 and 7094. Standard I-O and systems tapes and three scratch tapes are assigned.

## 10. REFERENCES

Roger Wallace, Palmer G. Steward, and Charles Sondhaus, "Primary and Secondary Proton Dose Rates in Spheres and Slabs of Tissue", UCRL-10980 Rev. (July 1964).

Palmer G. Steward, "Results of Computations of Depth Dose in Tissue Irradiated by Protons", UCRL-16154 (May 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 7 files:  
the source card decks, the binary card decks set up with BCD input cards for a sample problem for each code in the package, and a BCD output listing 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  
P. O. Box X  
Oak Ridge, Tennessee 37830

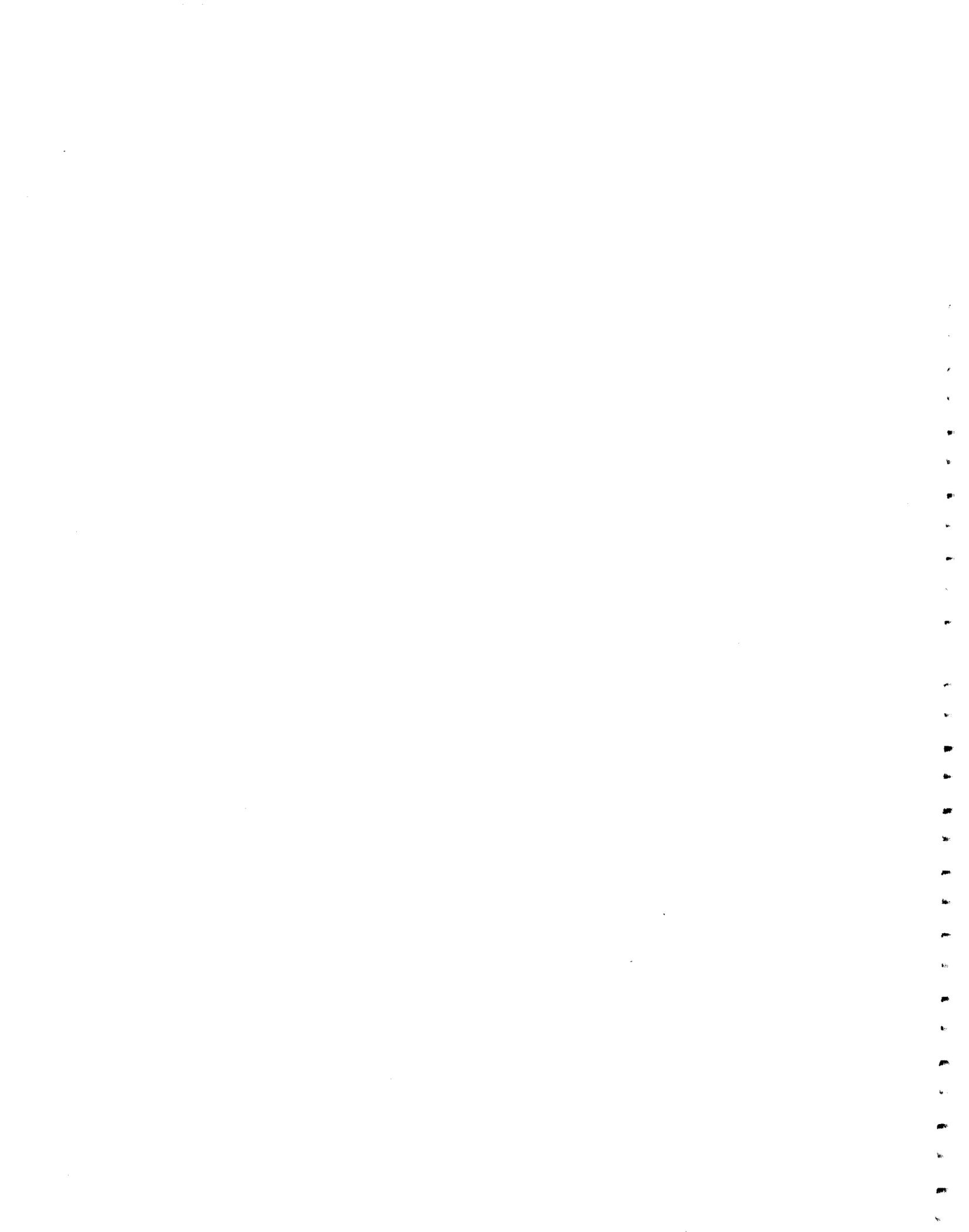
or telephoned to

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.

13. DATE OF ABSTRACT

January 1968.



## RSIC CODE PACKAGE CCC-74

## 1. NAME AND TITLE OF CODE

CAPS-2: Analysis of Structures for Fallout Radiation  
Shielding.

## AUXILIARY ROUTINE

PREPARE: Data Generator.

The CAPS-2 code is one of a series of CAPS (Computer Analysis for Protective Structures) developed by OCD. Originally developed by the A and E firm of Praeger-Kavanaugh-Waterbury, extensive modification has been made to the code by OCD.

## 2. CONTRIBUTOR

Office of Civil Defense, Office of the Secretary of the Army,  
Department of the Army, Washington, D. C.

## 3. CODING LANGUAGE AND COMPUTER

FORTRAN 63; CDC-1604.

## 4. NATURE OF PROBLEM SOLVED

CAPS-2 was designed to analyze structures for fallout radiation shielding by computing protection factors and for studying the sensitivity of results to varying shielding parameters or other basic input data.

## 5. METHOD OF SOLUTION

The code represents the mechanization of what has come to be called the "Engineering Manual Method" of shelter design and fallout shielding analysis. The method is based largely on the data and analysis described by Spencer<sup>1</sup> and prescribed in detail in a number of OCD publications<sup>2-4</sup> by Eisenhower and Fitz-Simons. Most of the basic data used are results of moments method calculations. The analysis takes the form of determining and combining barrier attenuation factors, geometry factors (in which are incorporated integrations over angular distributions), albedo

factors, and other modifying factors.

6. RESTRICTIONS OR LIMITATIONS

There are no known restrictions or limitations.

7. TYPICAL RUNNING TIME

The code will process a variety of structures (with up to 10 detector points on any designated story) at an average speed of less than a second per shelter area on the CDC-3600.

Estimated running time for the sample problems on the CDC 1604A: PREPARE, 30 seconds; COMPRAD #1, 40 seconds, and COMPRAD #2, 1 minute, 3 seconds.

8. COMPUTER HARDWARE REQUIREMENTS

The code was designed for the CDC 3600 and is also operable on the CDC 1604 with 3 tape units.

9. COMPUTER SOFTWARE REQUIREMENTS

The code may be compiled and executed on the FORTRAN 63 COOP Monitor or other compatible system. Tape assignments are made for input, output, system and one pool tape for a library.

10. REFERENCES

a. Background material:

- (1) L. V. Spencer, "Structure Shielding Against Fallout Radiation from Nuclear Weapons", NBS Monograph 42 (1962).
- (2) OCD Publication, "Design and Review of Structures for Protection from Fallout Gamma Radiation", PM 100-1 (February 1965).
- (3) C. Eisenhower, "Design and Review of Structures for Protection from Fallout Gamma Radiation", PM 100-1 Supplement (1964).

b. Code package material:

- (4) OCD Publication, "Shelter Design and Analysis", Vol.1, Fallout Protection TR-20 (Vol.1)(May 1964).
- (5) John L. Dirst, "CAPS-2 - A Computerized Method of Analyzing Structures for Radiation Shielding", (January 1966).

## 11. CONTENTS OF CODE PACKAGE

The package contains the following items:

- a. the referenced documents (4) and (5), and
- b. a reel of magnetic tape on which is written in 9 files:  
the BCD source card decks, the binary card decks, BCD  
input for two sample problems and BCD output listings for  
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

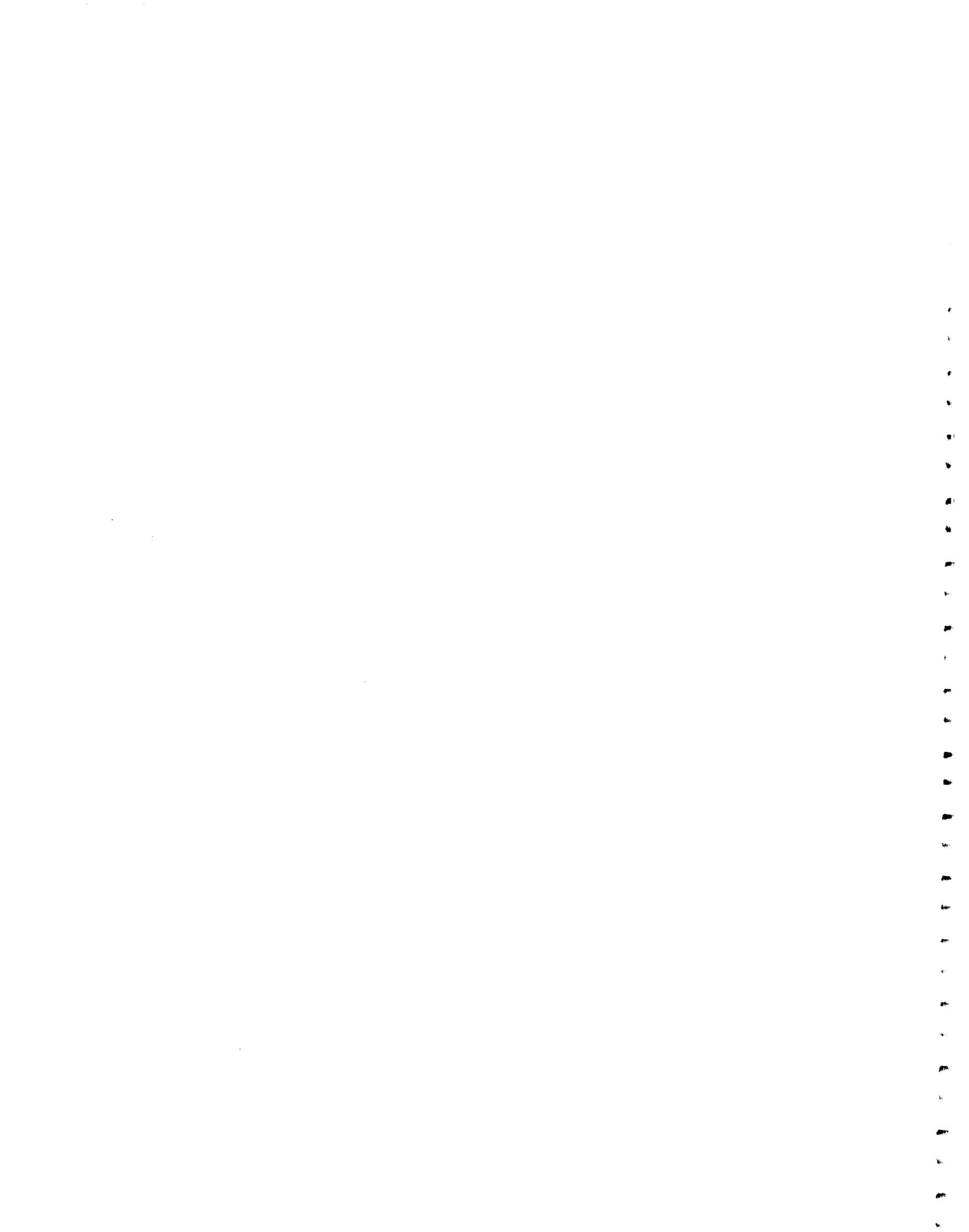
or telephoned to

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.

## 13. DATE OF ABSTRACT

January 1968 .



## RSIC CODE PACKAGE CCC-75

## 1. NAME AND TITLE OF CODE

G-33: Kernel Integration Code - Multigroup Gamma Ray Scattering.

The original version of G-33, written at LASL, was known as GAS. With revision and a name change, the code was given to AGN, who contributed the referenced document for the FORTRAN II version. The FORTRAN IV version, some informal notes, and an abstract were contributed by NASA LRC.

## 2. CONTRIBUTORS

Los Alamos Scientific Laboratory, Los Alamos, New Mexico.  
NASA Lewis Research Center, Cleveland, Ohio.  
Aerojet-General Nucleonics, San Ramon, California.

## 3. CODING LANGUAGE AND COMPUTER

G-33A FORTRAN II and FAP; IBM 7090.  
G-33B FORTRAN IV; IBM 7090, 7094, and 7094 II.

## 4. NATURE OF PROBLEM SOLVED

G-33 calculates flux and dose (or other response) due to a point source (in geometry describable by CCC-48/QAD geometry specifications) from singly scattered photons. For comparison, uncollided, uncollided with buildup on line of sight, uncollided and single scattered, uncollided and single scattered with buildup are also calculated.

## 5. METHOD OF SOLUTION

An orthogonal scattering geometry specification is superimposed on the general QAD geometry; the midpoint of the orthogonal cube is ascertained, the QAD region in which this point is located is determined, and the entire scatter volume is assumed to be of the ascertained QAD material and concentrated at the midpoint. G-33 traces a ray from the source point to each scatter

point, calculates uncollided photon flux at that point, and (from the Klein-Nishina differential cross-section and knowledge of the electron density at the scatter point) determines contribution from each scatter point to the detector. The library of photon cross sections and buildup factors used is the same as that used by QAD-HD (NASA-TM-X-1397).

#### 6. RESTRICTIONS OR LIMITATIONS

The following limits apply:

- 20 x 20 x 20 scatter grid,
- 50 QAD boundaries and regions, and
- 30 energy groups.

#### 7. TYPICAL RUNNING TIME

G-33 will process 2000 to 5000 geometry rays (source-to-scatterer or scatterer-to-detector) per minute depending on complexity of QAD geometry for simple problems:

$$\text{running time, minutes} = \frac{\text{LU} * \text{MU} * \text{NU} *}{5000} * (1 + \text{number of detectors})$$

where LU, MU, and NU are the number of divisions in the scatter grid.

Estimated running times for sample problem on an IBM 7090 in hours: 0.02 (G-33A) and 0.01 (G-33B).

#### 8. COMPUTER HARDWARE REQUIREMENTS

The codes may be run on a 32K IBM 7090. G-33B will also run on the IBM 7094 and 7094 II.

Two peripheral storage devices (tape units or disks) are required.

#### 9. COMPUTER SOFTWARE REQUIREMENTS

G-33A will compile and execute in the FORTRAN II Monitor System; G-33B in the FORTRAN IV IJOB Monitor under the IBSYS Operating System.

Standard systems and I-O assignments are used, and two scratch tapes or other peripheral storage devices may be used.

G-33 has a library of cross section data contained in a FAP subroutine.

G-33B has a library in the form of BLOCK DATA. Both are packaged.

#### 10. REFERENCES

G. H. Anno and J. K. Witthaus, "G-33, Code for Computing Gamma Ray Scattering", EAD-119, AN-COMP-196 (Feb. 1964).

G. P. Lahti, Informal notes (July 1966).

The document package from CCC-48/QAD may give helpful information: LA-3573 and NASA TM-X-1397.

#### 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 6 files: the source card decks, the binary card decks, BCD input for sample problems, and BCD listing of output from the sample problems for both G-33A and 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

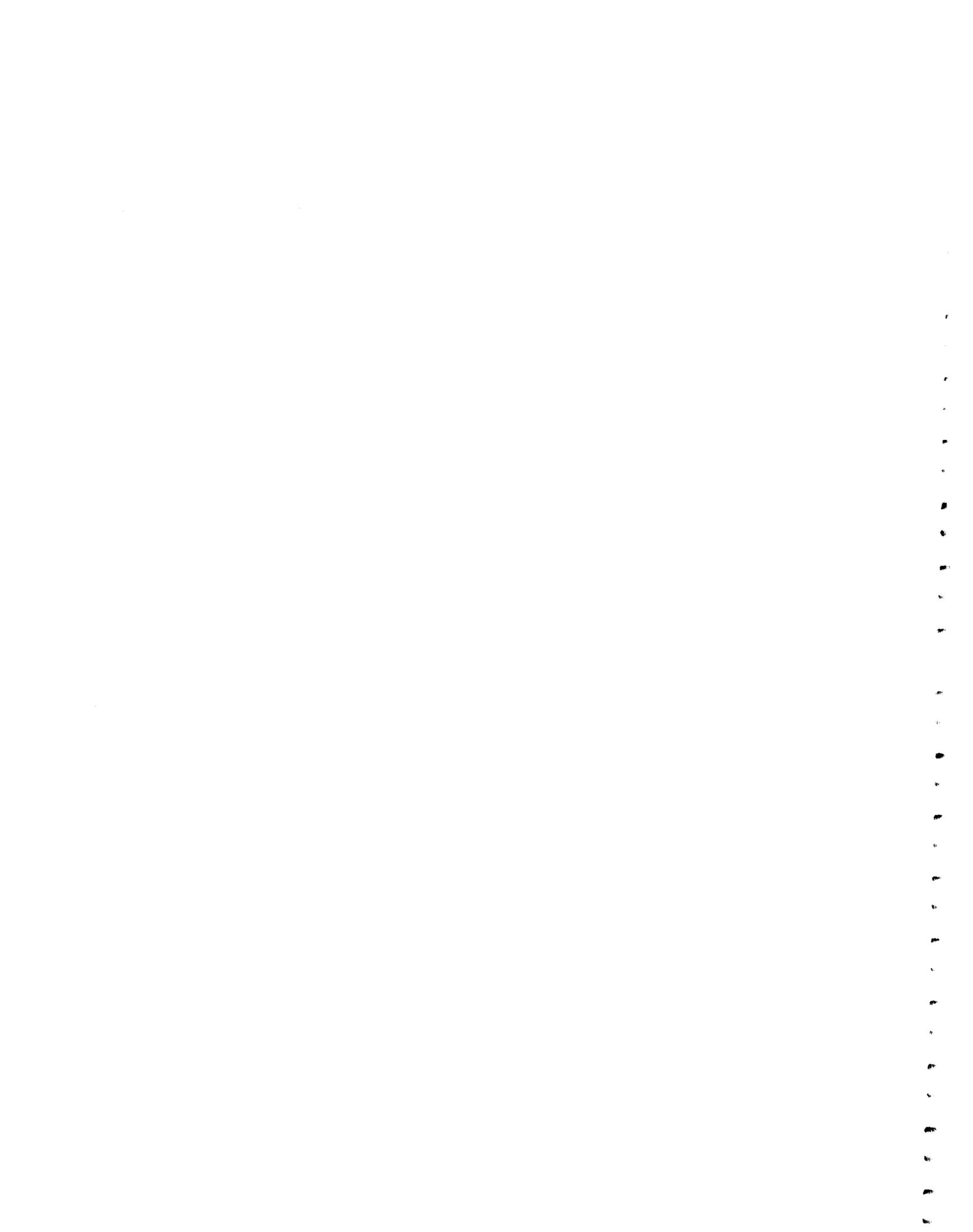
or telephoned to

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

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

#### 13. DATE OF ABSTRACT

January 1968.



## RSIC CODE PACKAGE CCC-76

## 1. NAME AND TITLE OF CODE

BPPC: Proton Penetration Codes for Space Vehicles.

## AUXILIARY ROUTINES

Three codes are included in the proton penetration package:

- (A) Secondary Proton Code.
- (B) Primary Proton Dose Through Layered Materials.
- (C) Proton Spectrum Through Layered Materials.

## 2. CONTRIBUTORS

NASA Langley Research Center, Hampton, Virginia.

Nuclear and Space Physics, Aerospace Division, The Boeing Company, Seattle, Washington.

Air Force Weapons Laboratory, Kirtland Air Force Base, New Mexico.

## 3. CODING LANGUAGE AND COMPUTER

- (A) FORTRAN IV; IBM 7090 and 7094.
- (B) FORTRAN II; IBM 7090 and 7094.
- (C) FORTRAN IV; IBM 7090 and 7094.

## 4. NATURE OF PROBLEM SOLVED

The proton penetration program was developed to predict radiation doses encountered in space missions. It may be used to determine the optimum vehicle structure and arrangement of internal equipment to provide maximum radiation shielding. It can determine the dose received by an astronaut at various body points when the vehicle is subjected to the incident radiations of the space environment.

The program calculates the primary proton dose and the dose from secondary protons, neutrons, and gamma rays produced by the interaction of the primary protons with shield materials within spherical multilayered shields.

## 5. METHOD OF SOLUTION

A straightforward approximation technique is used in the proton transport. Basically the method of calculation consists of a set procedure for calculating the spectrum as it emerges from a given shield material thickness. In the case of several layers of different types of material, this procedure is repeated until every layer has been traversed. At this point the dose is computed by forming the integral of the flux spectrum multiplied by the energy loss rate over all proton energies. This set of calculations is repeated for each solid angular section of the vehicle, and the weighted sum of these doses is then the total dose at the point of interest.

The procedure used for each type of secondary radiation is basically the same. Each shield layer is divided into a set of thin sublayers. For each sublayer, the spectrum of secondary particles or photons generated in that sublayer is first determined. This spectrum is then attenuated through the remainder of the shield layers to the point of interest or dose point. The spectrum of each radiation from the shield layers is then the sum of the attenuated spectra, originating in each thin sublayer.

## 6. RESTRICTIONS OR LIMITATIONS

There are several limitations arising from memory storage limitations. These include the following restrictions on array sizes.

- (1) Maximum number of materials considered is 12. This is also the restriction on shield layers in any sector.
- (2) Maximum number of solid angle sectors is 350.
- (3) Not more than 50 points may be given for the proton energy spectrum.
- (4) Not more than 10 angles of incidence may be specified.
- (5) Only one spectrum may be considered in one computer run.
- (6) In the arrays which specify the differential energy spectrum of secondary protons and neutrons as a function of primary proton energy, up to 50 primary energies

and 25 secondary energies may be specified. (These arrays have the code symbols XPMULT and XNMULT.)

(7) Only 10 gamma energy points are considered.

In addition to these limits, the program calculates tissue dose only.

#### 7. TYPICAL RUNNING TIME

The time is dependent upon the number of material layers and solid angles, and it is not possible to give a running time for cases in general. However, for sample problem (A) (two material thicknesses and two solid angle sectors) a running time of 0.02 hours was necessary; for that for (B) and (C), 0.01 hours each.

#### 8. COMPUTER HARDWARE REQUIREMENTS

All routines operate on the 32K IBM 7090 and 7094 computers. Standard input, output, and punch tapes, plus an on-line printer for operator instructions are required.

#### 9. COMPUTER SOFTWARE REQUIREMENTS

The codes (A) and (C) can be compiled and executed on the IBM IJOB Monitor in FORTRAN IV within the IBM IBSYS Operating System. Code (B) requires the IBM FORTRAN II Monitor System.

#### 10. REFERENCES

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", WL-TDR-64-71, Volume I and II (August 1964).

John A. Barton and G. L. Keister, "Symposium on Space Radiation Environment", D2-90684-1 (April 1965).

#### 11. CONTENTS OF CODE PACKAGE

The package contains the following items:

- a. the pertinent referenced documents, and
- b. a reel of magnetic tape on which is written in 9 files:

the BCD source card deck, the binary card deck, BCD input for a sample problem, and a BCD output listing from the problem for each of the codes packaged.

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.

13. DATE OF ABSTRACT

January 1968.

## RSIC CODE PACKAGE CCC-77

## 1. NAME AND TITLE OF CODE

BEBC: Electron Bremsstrahlung Penetration Code for Space Vehicles.

## 2. CONTRIBUTORS

NASA Langley Research Center, Hampton, Virginia.

Air Force Weapons Laboratory, Kirtland Air Force Base, New Mexico.

The Boeing Company, Seattle, Washington.

## 3. CODING LANGUAGE AND COMPUTER

FORTRAN IV; IBM 7090 and 7094.

## 4. NATURE OF PROBLEM SOLVED

BEBC calculates the Bremsstrahlung dose received through multilayered slabs of shield material. Bremsstrahlung is an x-ray created when high energy electrons are slowed down by a material.

Required inputs to this program are:

1. the incident electron spectrum,
2. photon attenuation coefficients,
3. build-up coefficients,
4. shield materials and thicknesses, and solid angles, and
5. flux to dose conversion factors.

## 5. METHOD OF SOLUTION

The Bremsstrahlung is assumed to be generated on the surface of the outermost shield layer and thus is treated as a parallel beam source of infinite area with semi-infinite slab shielding. It is assumed that the electrons do not penetrate the surface. To account for the additional dose from scattered radiation, empirical buildup factors are used. Angular distribution of the source is accomplished with a weighting factor. For successive shielding layers, the dose buildup is assumed to be multiplicative.

## 6. RESTRICTIONS OR LIMITATIONS

The incident electron spectrum must be analytically defined as a function of electron energy. Only 12 or fewer shield materials may be considered at one time. A polyhedron of no more than 350 sides may be used to represent a vehicle.

## 7. TYPICAL RUNNING TIME

No study has been made by RSIC as to typical running time. Estimated running time for the sample problem: 0.03 hour.

## 8. COMPUTER HARDWARE REQUIREMENTS

The code is operable on a 32K IBM 7090 computer with standard I-O equipment.

## 9. COMPUTER SOFTWARE REQUIREMENTS

The code may be compiled and executed on the IBM FORTRAN IV IJOB Monitor within the IBM IBSYS Operating System.

## 10. REFERENCES

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", WL-TDR-64-71, Volume I and II. (August 1964).

John A. Barton and G. L. Keister, "Symposium on Space Radiation Environment", D2-90684-1 (April 1965).

## 11. CONTENTS OF CODE PACKAGE

The package contains the following items:

- a. the pertinent referenced documents, and
- b. a reel of magnetic tape on which is written in 3 files:  
the BCD source card deck, the binary card deck, BCD input for a sample problem and a 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 code package should send a reel of magnetic tape to the above address.

## 13. DATE OF ABSTRACT

January 1968.

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## RSIC CODE PACKAGE CCC-78

## 1. NAME AND TITLE OF CODE

BED: Electron Penetration Code for Space Vehicles.

## 2. CONTRIBUTORS

NASA Langley Research Center, Hampton, Virginia.

Air Force Weapons Laboratory, Kirtland Air Force Base, New Mexico.

The Boeing Company, Seattle, Washington.

## 3. CODING LANGUAGE AND COMPUTER

FORTRAN IV; IBM 7090 and 7094.

## 4. NATURE OF PROBLEM SOLVED

BED computes the space radiation dose produced by electrons being transmitted through a radiation shield.

Required inputs for the code are:

1. electron flux-to-dose conversion table,
2. energy spectrum being used,
3. atomic number of shield material, and
4. layer thickness for solid angle for each sector.

The code will handle only one type of material at a time. Dose may be calculated at any point inside a vehicle.

## 5. METHOD OF SOLUTION

From an empirical study, it is found that electrons from radiation belts encircling the earth will produce a radiation dose ( $D_i$ ) equal to:

$$D_i = \Omega \int_{E_0}^E \left[ T(E) \varphi(E) \frac{dE}{dx}(E) \right] dE$$

where the following is true:

1.  $T(E)$  is the transmission of one electron through a shield of thickness  $x$  and atomic number  $z$ ,
2.  $\varphi(E)$  is the energy spectrum from a particular radiation belt, and
3.  $\frac{dE}{dx}(E)$  is the ionization loss for an electron of energy  $E(\text{MeV})$  of the material for which the dose is calculated.

For each thickness  $x$ , the product of  $T(E)$ ,  $\varphi(E)$ , and  $\frac{dE}{dx}(E)$  is integrated over the range of possible values of  $E$  to obtain the dose. If the dose is desired through a particular solid angle of shield, say  $\Omega$ , then the integral is multiplied by  $\Omega$ . The total dose through the shield is then taken to be the sum of the individual doses through particular thicknesses and at various solid angles.

#### 6. RESTRICTIONS OR LIMITATIONS

The mathematical formula for transmission of electrons is derived from theoretical electron Monte Carlo data which is lacking in experimental verification.

The analysis is for one material only, and a shield of more than one kind of material must be treated as though every layer consists of the same material.

#### 7. TYPICAL RUNNING TIME

No study has been made by RSIC as to typical running time. Estimated running time of sample problem: 0.03 hour.

#### 8. COMPUTER HARDWARE REQUIREMENTS

A 32K computer is required. Code is operable on the IBM 7090 and 7094 with standard I-O equipment.

#### 9. COMPUTER SOFTWARE REQUIREMENTS

The code is operable on the IBM FORTRAN IV IJOB Monitor within the IBSYS Operating System. Only systems, input, and output tape assignments are made.

## 10. REFERENCES

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", WL-TDR-64-71, Volume I and II. (August 1964).

John A. Barton and G. L. Keister, "Symposium on Space Radiation Environment", D2-90684-1 (April 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 files:  
the BCD source card decks, the binary card decks, BCD input for a sample problem and a BCD output listing from 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  
P. O. Box X  
Oak Ridge, Tennessee 37830

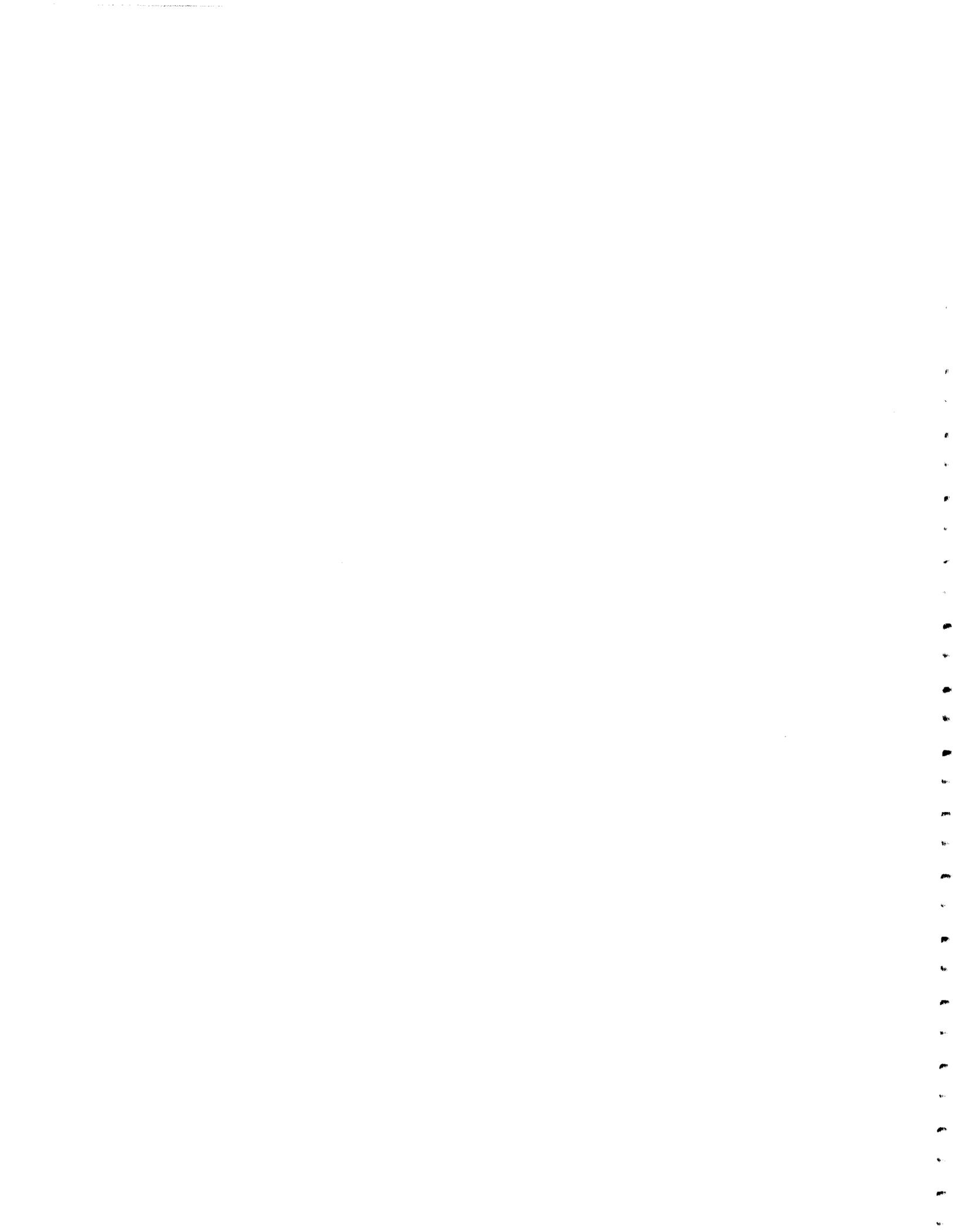
or telephoned to

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.

## 13. DATE OF ABSTRACT

January 1968.



## RSIC CODE PACKAGE CCC-79

## 1. NAME AND TITLE OF CODE

ISOSHL D: Kernel Integration Code - General Purpose Isotope Shielding Analysis.

Two versions are packaged: ISOSHL D I and II. RIBD is used in both versions as a subroutine to calculate fission product inventories. The CCC-31/BREMRAD code package can be used to calculate the bremsstrahlung spectrum mesh, but must be requested separately.

## 2. CONTRIBUTOR

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

## 3. CODING LANGUAGE AND COMPUTER

(A) ISOSHL D I: FORTRAN IV; IBM 7090.

(B) ISOSHL D II: FORTRAN IV; IBM 360.

## 4. NATURE OF PROBLEM SOLVED

ISOSHL D calculates the decay gamma-ray and bremsstrahlung dose at the exterior of a shielded radiation source. The source may be one of a number of common geometric shapes. If the radiation source originated as one or a group of fission products produced under known irradiation conditions, then the strength of the source is also calculated. The code calculates shield region mass attenuation coefficients, buildup factors, and other basic data necessary to solve the specific problem.

## 5. METHOD OF SOLUTION

ISOSHL D performs kernel integration for common geometric shapes. The "standard" point attenuation kernel (buildup factor x exponential attenuation + geometry factor) is numerically integrated over the source volume for 25 source energy groups. Buildup is considered characteristic of the last shield region (or a different specified region) but dependent on the total number of mean free paths from

source to dose point, and is obtained by interpolation on effective atomic number from a table of point isotopic buildup factor data. Mixed mass attenuation coefficients are obtained from a library of basic data using code input material density specifications. The source strength may be specified 1) as the emissions from a selection of fission products irradiated under specific conditions, 2) the curies of particular fission and/or activation products, or 3) a number of photons per second of energy E specified by input. An exponential source distribution may be specified for those source geometries which are applicable. If the source originates in a combination of fission products and their daughters, these are calculated by a fission product inventory procedure which runs through transmutation (decay chain) calculations for each fission product and daughter. The latest modification (ISOSHL D  $\rightarrow$  ISOSHL D II) adds the capability for calculating shielded dose rates from bremsstrahlung sources. This addition consists of routines for calculating the bremsstrahlung source spectra from the beta decay properties of the isotope(s) of interest. Bremsstrahlung photons per group for 25 energy groups (9 groups below 0.1 MeV have been added) are obtained by interpolation from tables of resolved spectra. This spectral mesh, for internal and external bremsstrahlung, is tabulated as a function of the following parameters: beta-emitting and stopping nuclides with atomic numbers of 10, 30, 50, 70, and 90; ratios of photon energy to beta end point energy for 25 intervals from 0.00375 to 1.0; beta and point energies at the intervals 0.1, 0.2, 0.5, 1, 2, and 4 MeV. Buildup factors for photon energies less than 0.1 MeV are interpolated from a table which contains data for 5 values of initial photon energy in the range 0.01 to 0.2 MeV, seven values of shield thickness in the range 1 to 20 mfp, and 6 atomic numbers in the range 13 to 92.

The entire shielding problem is solved for most types of isotope shielding applications without reference to shielding handbooks for basic data.

## 6. RESTRICTIONS OR LIMITATIONS

These limits apply: 5 source cooling times, 500 radioactive isotopes, 5 shield regions including source region, 25 energy groups, 20 materials in each shield region, choice of 11 source geometries.

## 7. TYPICAL RUNNING TIME

Dose from cylindrical volume source - 20 integration increments in each direction, fission product inventory calculations with 5 decay times, 25 energy groups, 4 shield layers, 5 materials homogenized into each shield layer and the source volume --- 6 minutes UNIVAC 1107. (Most other source geometries require less computation time.)

## 8. COMPUTER HARDWARE REQUIREMENTS

The codes were originally designed for the 65K UNIVAC 1107. They have been modified by RSIC to run on the IBM 7090 (I) and the 360 (II).

## 9. COMPUTER SOFTWARE REQUIREMENTS

The codes were originally designed for the UNIVAC 1107 EXEC II Monitor System.

ISOSHL D I is available as an overlay job on the IBM FORTRAN IV IJOB Monitor within the IBSYS Operating System. The ALTIO package is used.

ISOSHL D II is available for the IBM 360 computer and has been run on the Model 50 on the Level H compiler.

A library of data is packaged for each version.

## 10. REFERENCES

R. L. Engle, J. Greenborg, and M. M. Hendrickson, "ISOSHL D - A Computer Code for General Purpose Isotope Shielding Analysis", BNWL-236 (June 1966).

R. O. Gumprecht, "RIBD-Radioisotope Buildup and Decay", Unpublished data.

H. H. Van Tuyl, "BREM RAD - A Computer Code for External and

Internal Bremsstrahlung Calculations, HW-83784 (September 1964)  
(Packaged in CCC-31 only).

G. L. Simmons, J. J. Regimbal, J. Greenborg, E. L. Kelly, Jr.,  
and H. H. Van Tuyl, "ISOSHLD-II: Code Revision to Include Calcula-  
tions of Dose Rate from Shielded Bremsstrahlung Sources",  
BNWL-236SUP1 (March 1967).

11. CONTENTS OF CODE PACKAGE

The package contains the following items:

- a. the referenced documents, with exception noted, and
- b. for each code version a reel of magnetic tape has been written which contains the BCD source card decks, the binary card decks, a BCD library of data, BCD input for a sample problem, and a BCD output listing for the problem.

12. HOW TO OBTAIN PACKAGE

Inquiries or requests for the code package should 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, specifying the version desired.

13. DATE OF ABSTRACT

January 1968.

## RSIC CODE PACKAGE CCC-80

## 1. NAME AND TITLE OF CODE

GASS: Monte Carlo Calculation of Self Shielding by Encapsulated Gamma-Ray Sources.

## 2. CONTRIBUTOR

University of Illinois, Civil Engineering and Nuclear Engineering Program, Urbana, Illinois.

## 3. CODING LANGUAGE AND COMPUTER

FORTRAN IV and MAP; IBM 7090 and 7094.

## 4. NATURE OF PROBLEM SOLVED

GASS was designed to calculate the energy and angular distributions of photons emitted from a cylindrical or spherical encapsulated source.

## 5. METHOD OF SOLUTION

GASS represents an approximate solution by Monte Carlo methods to the steady state transport equation.

A time-saving procedure utilizing a "do-nothing" cross section allows ray tracing to proceed without many cross section look-ups and partial path length calculations. A choice is then made where the collision occurs between scattering, absorption, or the straightahead "do-nothing" reaction.

## 6. RESTRICTIONS OR LIMITATIONS

There are no known restrictions implied by storage allocation.

## 7. TYPICAL RUNNING TIME

Processing of 20,000 histories in the sample problem required 3 minutes on the IBM 7090 and 1.3 minutes on the 7094.

## 8. COMPUTER HARDWARE REQUIREMENTS

The code was designed for the 32K IBM 709<sup>4</sup> computer with 3 tape units.

## 9. COMPUTER SOFTWARE REQUIREMENTS

The code may be compiled and executed on the IBM FORTRAN IV IJOB Monitor and other compatible systems. Standard I-O and systems tape assignments are made.

## 10. REFERENCE

K. Preiss, "Monte Carlo Calculation of Self Shielding by Encapsulated Gamma Ray Sources", UI -NRSS -3 (August 1966).

## 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 3 files:  
the BCD source card deck, the binary card deck set up with BCD input for a sample problem and a 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.

## 13. DATE OF ABSTRACT

January 1968.

## RSIC CODE PACKAGE CCC-81A

## 1. NAME AND TITLE OF CODE

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

The UNC-SAM system represents such extensive modification and development of the CCC-13/ADONIS code system that a name change and separate packaging was deemed advisable. The system is a chained series of independent programs which processes cross sections and geometry data, does the transport problem, and edits the results. These codes are listed as auxiliary routines.

## AUXILIARY ROUTINES

GENDA: Cross Section Data Generator.

GENPRO: Processes GENDA Data to Determine Tables of  
Probabilities.

DATORG: Problem Data Generator.

EZGEOM: Geometry Data Expander.

VANGEN: Volume Anisotropic Source Generator.

GASP: Gamma Source Particle Generator.

INPUTD: Inputs Data for ADONIS.

ADONIS: Transport Code.

STATC: Edits ADONIS Output.

SAM: Program Sequencer.

## 2. CONTRIBUTORS

United Nuclear Corporation, White Plains, New York.

U. S. Army Ballistics Research Laboratory, Aberdeen Proving  
Ground, Maryland.

## 3. CODING LANGUAGE AND COMPUTER

FORTRAN and CODAP-2; CDC-1604.

## 4. NATURE OF PROBLEM SOLVED

UNC-SAM calculates the solution to the Boltzmann transport equation in complex three dimensional geometry. The code will

calculate fluxes, flux dependent functionals such as doses, and their standard deviations in geometry comprised of rectangular parallelepipeds, which in turn may contain spheres, cylinders, or parallelepipeds. UNC-SAM has been used in the following problems: transmission of neutrons through ducted shields, determining the production of gamma rays in shielding materials by high energy neutrons, transmission of neutrons and gamma rays through air (taking account of air and ground scattering), and weapon vulnerability studies.

## 5. METHOD OF SOLUTION

The ADONIS part of UNC-SAM is a Monte Carlo code that tracks either neutrons or gamma rays through shields composed of rectangular parallelepipeds of differing compositions. Particle splitting is employed to improve the efficiency of the calculation by assigning importance weights to each of the regions. A special "flux at a point" routine was designed to reduce problem running time when fluxes are required at improbable regions of phase space.

The code considers the following neutron interactions: isotropic and anisotropic elastic scattering, discrete and continuum inelastic scattering, scattering by hydrogen, the  $(n,2n)$  reaction in beryllium, and absorption.

The gamma portion of the code considers the Compton scattering of photons. The photoelectric effect and pair production are regarded as absorption processes.

Particle histories in ADONIS are terminated when: an absorption occurs, the particle degrades below an arbitrary cutoff energy, a kill occurs due to its low weight, or the particle escapes.

Cross-section data are obtained from a Master Element Data Tape for either neutrons or gamma rays prepared from GENDA and GENPRO. DATORG generates the specific cross section data needed for a given problem.

A source tape containing the position coordinates, direction cosines, and the initial energy of the neutron or gamma ray is generated by VANGEN or GASP as required. VANGEN will handle a

point, uniform plane, or uniform volume source which may be isotropic, anisotropic according to a given distribution, mono-directional, or isotropic in the half plane. The energy variation may be described as a fission spectrum, as monoenergetic, or as an arbitrary distributed energy source with equal probability steps prescribed by input. GASP is used to compute gamma-ray production from a primary neutron problem to serve as a source for a secondary gamma-ray problem.

EZGEOM takes a simplified geometrical description of the physical system and produces the complex set of data needed by ADONIS.

#### 6. RESTRICTIONS OR LIMITATIONS

A maximum of 80 regions may be used, where a region is defined as a rectangular parallelepiped of either finite or infinite dimensions. The totality of all such regions covers x, y, and z space. A maximum of 80 complex surfaces is permitted. A complex surface is defined as a side of a parallelepiped adjacent to more than one region.

#### 7. TYPICAL RUNNING TIME

No study has been made by RSIC as to typical running time. Estimated running time of sample problem through all routines: problem 1, 5 minutes; problem 2, 15 minutes, 20 seconds.

#### 8. COMPUTER HARDWARE REQUIREMENTS

UNC-SAM was designed for the 32K CDC 1604 computer with 6 or more tape units.

#### 9. COMPUTER SOFTWARE REQUIREMENTS

The codes may be compiled and executed on the FORTRAN 63 COOP Monitor System.

#### 10. REFERENCE

B. Eisenman and F. R. Nakache, "UNC-SAM: A Fortran Monte Carlo System for the Evaluation of Neutron or Gamma-Ray Transport

in Three-Dimensional Geometry", UNC-5093 (August 1964).

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 15 files:  
the BCD source card decks, the binary card deck, BCD  
input for a sample problem and 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  
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

January 1968.

## RSIC CODE PACKAGE CCC-81B

## 1. NAME AND TITLE OF CODE

UNC-SAM-2: Monte Carlo Time Dependent Three-Dimensional  
Complex Geometry Shielding Code System.

UNC-SAM-2 is a modification and extension of UNC-SAM. This abstract assumes a working knowledge of UNC-SAM and denotes the difference reflected in UNC-SAM-2.

## AUXILIARY ROUTINES

GENDA: Cross Section Data Generator.  
 GENPRO: Processes GENDA Data to Determine Tables of  
Probabilities.  
 TUNC: Program Sequencer.  
 BAND: Organized Data Tape (ODT) Generator.  
 BEDIT: ODT Editor.  
 GEOM: Geometry Data Expander.  
 INPUTD: Problem Specification Input.  
 MONTE: Transport Code  
 PEDIT: Edits MONTE Output.  
 TRANSMIT: Transmitted Particles Source Generator.  
 GASP: Gamma Source Particle Generator.  
 RESPONSE: Response Function Generator.

## 2. CONTRIBUTORS

United Nuclear Corporation, White Plains, New York.  
 U. S. Army Ballistics Research Laboratory, Aberdeen Proving  
Ground, Maryland.

## 3. CODING LANGUAGE AND COMPUTER

FORTRAN and CODAP-2; CDC-1604.

## 4. NATURE OF PROBLEM SOLVED

UNC-SAM-2 calculates the solution to the Boltzmann transport equation in complex three dimensional geometry. The code will

calculate fluxes, flux dependent functionals such as doses, and their standard deviations in geometry comprised of rectangular parallelepipeds, which in turn may contain spheres, cylinders, or parallelepipeds. Time dependence of the transport of neutrons or photons through matter is evaluated.

#### 5. METHOD OF SOLUTION

UNC-SAM-2 differs from UNC-SAM in the following:

1. It is time dependent.
2. It allows the energy range to be split into bands which are treated one at a time so that a linear energy structure may be treated. It also has a supergroup option.
3. It uses a more sophisticated tracking technique which should result in lower variance per particle and faster running times for a given accuracy.
4. It allows angular and energy dependent weighting.
5. Sources are specified as integrals.

The geometry treatment in the two codes is identical. Within an energy band cross section data is stored the same way. Both codes will accept GASP tapes as source functions.

#### 6. RESTRICTIONS OR LIMITATIONS

A maximum of 80 regions may be used, where a region is defined as a rectangular parallelepiped of either finite or infinite dimensions. The totality of all such regions covers x, y, and z space. A maximum of 80 complex surfaces is permitted. A complex surface is defined as a side of a parallelepiped adjacent to more than one region.

#### 7. TYPICAL RUNNING TIME

Estimated running time for sample problem: 5 minutes.

#### 8. COMPUTER HARDWARE REQUIREMENTS

UNC-SAM-2 was designed for the 32K CDC 1604 computer with 6 or more tape units.

## 9. COMPUTER SOFTWARE REQUIREMENTS

The codes may be compiled and executed on the 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).

B. Eisenman and F. R. Nakache, "UNC-SAM: A FORTRAN Monte Carlo System for the Evaluation of Neutron or Gamma-Ray Transport in Three-Dimensional Geometry", UNC-5093 (August 1964).

## 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 separate files: the BCD source card decks, the binary card deck, BCD input for a sample problem, and 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  
Post Office Box X  
Oak Ridge, Tennessee 37830

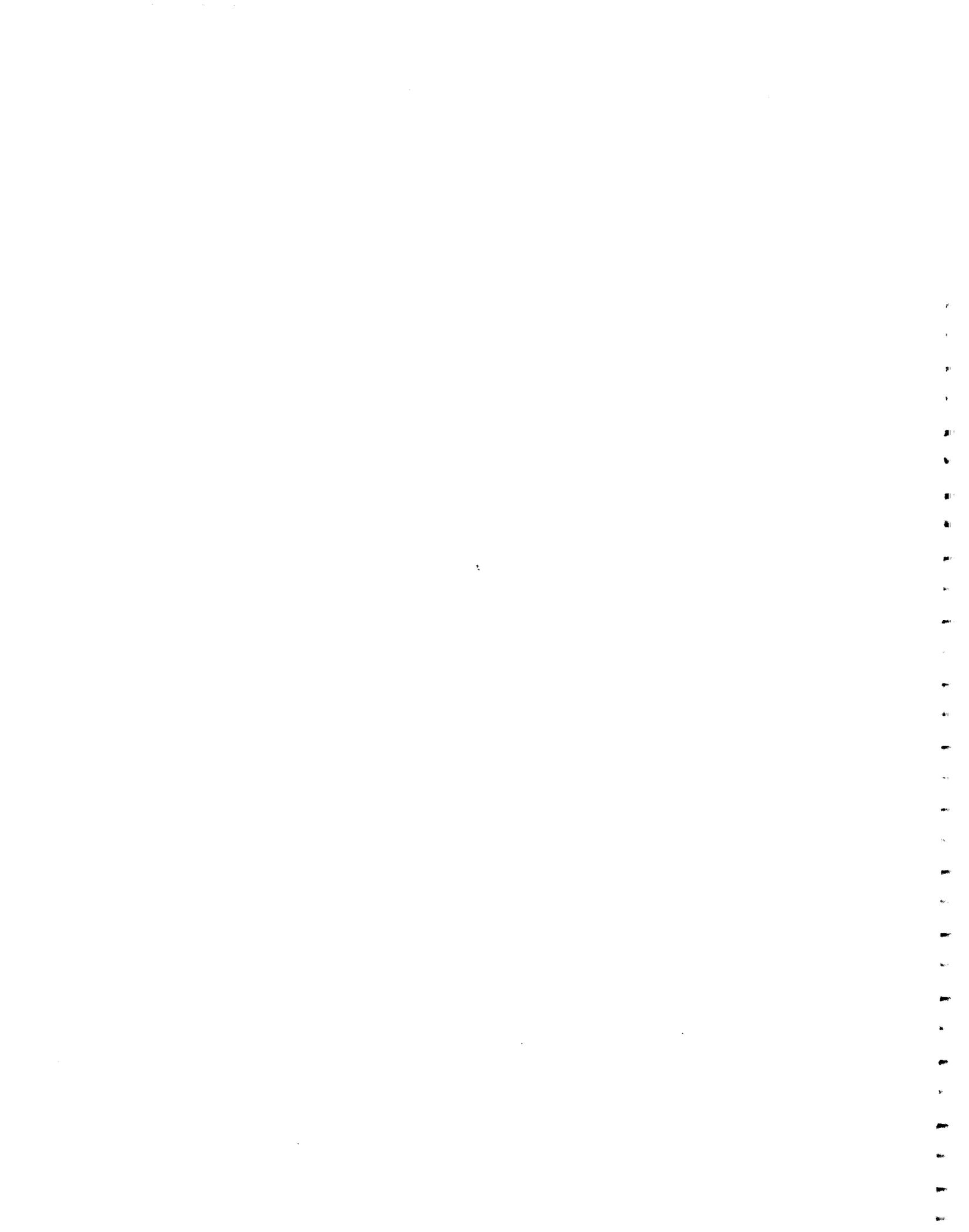
or telephoned to

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

January 1968.



## RSIC CODE PACKAGE CCC-82

## 1. NAME AND TITLE OF CODE

ANISN: Multigroup One Dimensional Discrete Ordinates  
Transport Code with Anisotropic Scattering.

## AUXILIARY ROUTINE

Library Generator.

ANISN supercedes DTF-II (NAA-SR-10951, March 25, 1966) which followed a series of developmental efforts over a period of years. An early version, DSN, was developed in the FLOCO language by Bengt Carlson of the Los Alamos Scientific Laboratory. A revision of DSN, called DTK, was written to incorporate improved convergence technique and ease of operation. DTF was a FORTRAN version of DTK written by UNC and LASL personnel. DTF-II evolved from DTF at Atomics International and in turn evolved into ANISN.

## 2. CONTRIBUTORS

Computing Technology Center and Oak Ridge National Laboratory, Union Carbide Corporation, Nuclear Division, Oak Ridge, Tennessee.

## 3. CODING LANGUAGE AND COMPUTER

FORTRAN IV; IBM 7090 and 7094.  
FORTRAN IV (H); IBM 360.

## 4. NATURE OF PROBLEM SOLVED

ANISN solves the one dimensional Boltzmann transport equation for neutrons or gamma rays in slab, sphere, or cylinder geometry. The source may be fixed, fission, or a subcritical combination of the two. Criticality search may be performed on any one of several parameters. Cross sections may be weighted using the space and energy dependent flux generated in solving the transport equation.

## 5. METHOD OF SOLUTION

The solution technique is an advanced discrete ordinates method which represents a generalization of the method originated by G. C. Wick and greatly developed and extended to curvilinear geometry by B. G. Carlson at Los Alamos Scientific Laboratory.

ANISN was designed to solve deep-penetration problems in which angle-dependent spectra are calculated in detail. The principal feature that makes ANISN suitable for such problems is the use of a programming technique with optional data-storage configurations which allows execution of small, intermediate, and extremely large problems. ANISN also includes a technique for handling general anisotropic scattering, pointwise convergence criteria, and alternate step function difference equations that effectively remove the oscillating flux distributions sometimes found in discrete ordinates solutions.

## 6. RESTRICTIONS OR LIMITATIONS

Problem size is limited only by machine size.

## 7. TYPICAL RUNNING TIME

ANISN can compute  $\sim 1200$  angular fluxes per second (maximum) on the IBM 7090. Depending on problem size, type, and convergence, running time has varied from less than one minute to several hours.

Estimated running time of the 10 sample problems on the IBM 7090: 35 minutes.

## 8. COMPUTER HARDWARE REQUIREMENTS

- a. IBM 7090, 7094, or 360; minimum 32K words.
- b. As few as one tape excluding standard I/O.
- c. Not channel dependent.
- d. Punch, printer.
- e. IBM 7090/7094-User may supply clock routine. Clock is sampled on IBM 360.

## 9. COMPUTER SOFTWARE REQUIREMENTS

- a. IBM 7090/7094 FORTRAN IV, version 13 Monitor System with ALTIO.

IBM 360 FORTRAN IV (H) Compiler, dated November 1966 or later.

- b. Standard operation, no intervention required.
- c. Is an overlay job. Punched card output optional, controlled by input.

10. REFERENCES

W. W. Engle, Jr., "ANISN, A One Dimensional Discrete Ordinates Transport Code with Anisotropic Scattering", K-1693 (March 1967).

W. W. Engle, M. A. Boling, and B. W. Colston, "DTF-II, A One Dimensional, Multigroup Neutron Transport Program", NAA-SR-10951 (March 1966).

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 7 files:  
the BCD source card decks, the binary card decks, BCD input for 10 sample problems, and a BCD output listing from 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  
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.

13. DATE OF ABSTRACT

January 1968.



## RSIC CODE PACKAGE CCC-83

## 1. NAME AND TITLE OF CODE

RAID: Monte Carlo Multibend Duct Shielding Code.

RAID is a translation and extensive modification of the CCC-9/L-05 code package. The changes were drastic enough to warrant separate packaging under a new CCC number.

## 2. CONTRIBUTOR

USAF Nuclear Aerospace Research Facility (NARF), General Dynamics, Fort Worth, Texas.

## 3. CODING LANGUAGE AND COMPUTER

FORTRAN IV; IBM 7090 and 7094.

## 4. NATURE OF PROBLEM SOLVED

RAID evaluates the energy, angular distribution, and intensity of either the scattered neutron or gamma-ray flux that penetrates a straight cylindrical or a multibend duct. A neutron or gamma-ray source may be described with a set of from 1 to 30 point sources. The energy and angular distribution and the intensity of the scattered flux are calculated for each of a set of from 1 to 30 detector points.

The code has been used in the analysis of the gamma radiation environment above a fallout field, where the effect of the air-ground interface is taken into account.

The geometry routine is sufficiently flexible for the description of many of the shield configurations encountered in reactor, weapons, and space shielding.

## 5. METHOD OF SOLUTION

The Monte Carlo method represents a Neumann series solution of the integral transport equation. Included in the sampling techniques are splitting, Russian Roulette, statistical estimation, and a method of importance sampling in the source angular

distributions.

The code provides for the calculation of the energy and angular distribution of the scattered neutron or gamma-ray flux, the total scattered neutron or gamma-ray flux and dose rates, and the unscattered neutron or gamma-ray flux and dose rate.

A library of cross sections which can be easily changed is packaged with the code.

#### 6. RESTRICTIONS OR LIMITATIONS

There are no known restrictions implied by storage allocation or argument range due to approximations.

#### 7. TYPICAL RUNNING TIME

No study of typical running time has been made by RSIC. Estimated running time for the two packaged sample problems on the IBM 7090: 0.11 hour.

#### 8. COMPUTER HARDWARE REQUIREMENTS

The code was designed to run on the 32K IBM 7090 and 7094, and on the CDC 6600 computers. 12 tape units are used on the IBM 7090.

#### 9. COMPUTER SOFTWARE REQUIREMENTS

The code has been compiled and problems executed on the IBM FORTRAN IV IJOB Monitor in the IBSYS Operating System. Standard I-O, overlay and systems, and 8 additional tape assignments were made. It was designed to be compatible with FORTRAN IV for the CDC 6600 computer.

#### 10. REFERENCE

J. A. Moore, J. B. Eggen, and F. O. Leopard, "Monte Carlo Procedure for Analysis of Radiation in Ducts (RAID)", NARF-DC-Memo 1.115 (December 1966).

#### 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 3 separate files: the BCD source card decks, the binary card decks, a BCD cross section library, BCD input for two sample problems, and a BCD output listing 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

January 1968.



## RSIC CODE PACKAGE CCC-84

## 1. NAME AND TITLE OF CODE

SHADRAC: Kernel Integration Code - Shield Heating and Dose Rate Calculation in Complex Geometry.

The code is a complete rewrite of machine language computer codes C-17 and L-63, packaged respectively as CCC-5 and CCC-6. SHADRAC represents such a considerable improvement over the old code versions as to warrant a name change and separate packaging. It is also known by the GD Digital Computing Laboratory's internal accounting designation, G-30.

## 2. CONTRIBUTOR

USAF Nuclear Aerospace Research Facility, General Dynamics, Fort Worth, Texas.

## 3. CODING LANGUAGE AND COMPUTER

FORTRAN IV; IBM 7090 and 7094.

## 4. NATURE OF PROBLEM SOLVED

SHADRAC calculates the neutron and/or gamma-ray spectra, heat generation rate, and/or dose rate at each of a group of point detectors, due to each of a group of point sources. The sources may be divided into sets, with each set having a unique source spectra. The spectrum, heating rate, and/or dose rate for each detector, summed over each source-point set and over the entire source group, may also be computed. Complex geometry may be treated.

## 5. METHOD OF SOLUTION

Point-to-point kernels, based upon the differential energy spectra for a point isotropic source in an infinite medium, are integrated over various sources. The data used is based on the moments-method solution of the fast-neutron or gamma-ray transport equation. The stepping-point method is used to solve for the path lengths from source to detector in each region.

The gamma-ray absorption coefficients are based on interpolations of the photoelectric and pair production cross sections so that the coefficients may be computed for all media of the system. The effective atomic number is interpolated from a table of atomic numbers versus the absorption coefficient per electron.

The mode of distributing the source points is chosen (either equal interval or according to Gaussian quadrature abscissa) which locates the coordinate planes that are perpendicular to the coordinate axes. The intersections of these planes are source point locations.

Modifications in SHADRAC resulting in improvement over earlier models are: direct computation of unscattered flux, removal of all neutron energy modes except the first, greater capabilities in the use of the source spectra, library tape storage of material data, coding of gamma-ray data into the program, and improvement of the output format.

#### 6. RESTRICTIONS OR LIMITATIONS

Enough physical and source description capability is provided by the program so that there should be little uncertainty except that associated with the point-to-point kernels as applied to specific geometries. Inhomogeneities generally increase the error since it is necessary to use arbitrary prescriptions for combining homogeneous media data.

#### 7. TYPICAL RUNNING TIME

No study has been made at this time to determine typical running time. Estimated running time of sample problem on the IBM 7090: 0.05 hour.

#### 8. COMPUTER HARDWARE REQUIREMENTS

IBM 7090 and 7094 32K computer with 7 tape units.

## 9. COMPUTER SOFTWARE REQUIREMENTS

The code was designed for and is operable in the IBM FORTRAN IV Operating System, using the ALTIO package. Four pool tapes, in addition to standard input, output, and systems tapes, are assigned. Allowance has been made for a library to be read in from tape. It is currently being read as standard input data.

## 10. REFERENCE

J. A. Moore, J. B. Eggen, C. W. Austin, D. H. Huckaby, and R. A. Miller, "Shield Heating and Dose Rate Attenuation Calculation (SHADRAC)", NARF-DC-MEMO #1.097 (March 1966).

## 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 3 files: the BCD source card decks, the binary card decks set up with a BCD library of cross sections and input for a sample problem, and a BCD output listing from 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

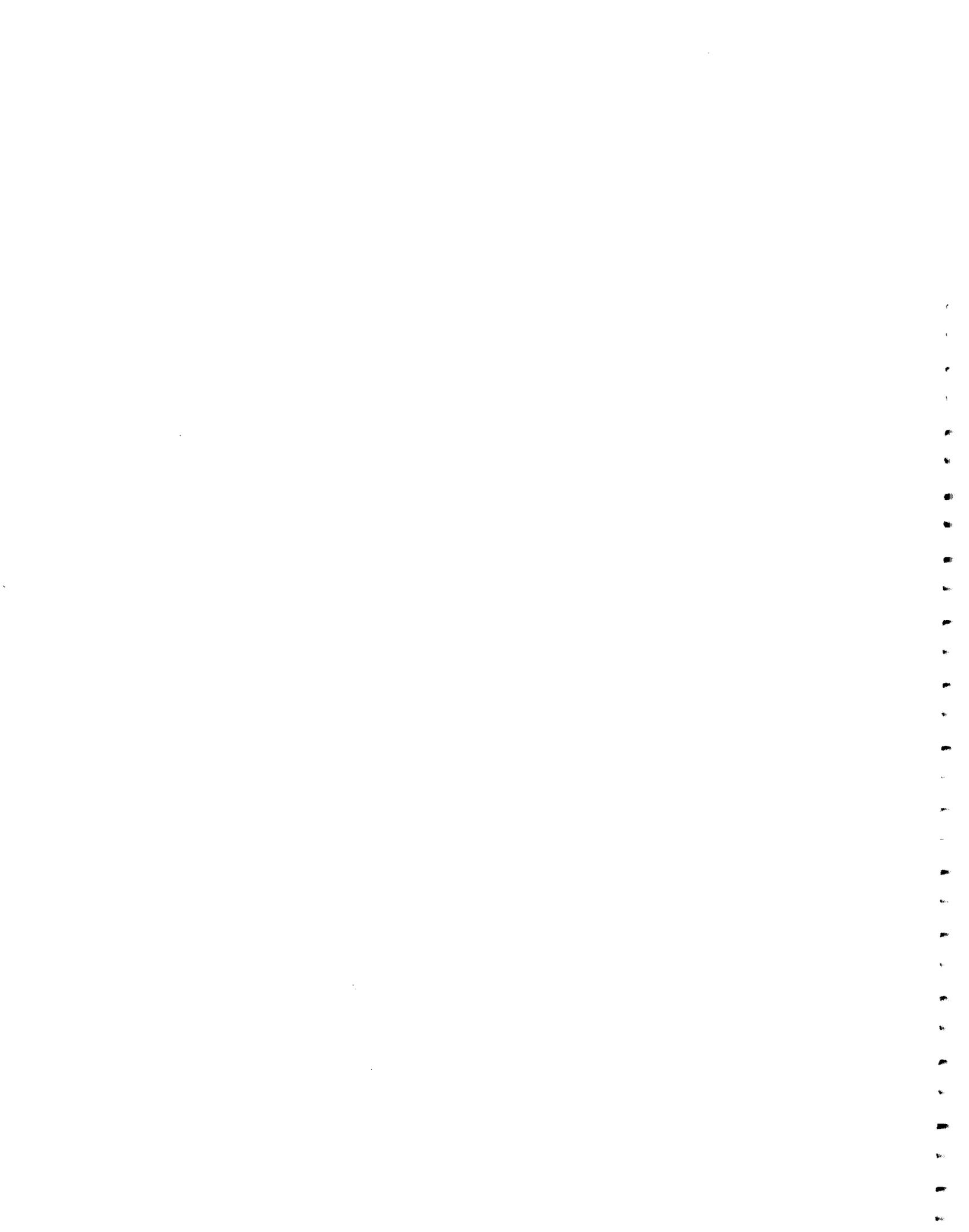
or telephoned to

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

January 1968.



## RSIC CODE PACKAGE CCC-85

## 1. NAME AND TITLE OF CODE

MOMGEN-MOMDIS: Moments Method Reconstruction of Scattered  
Gamma-Ray Distributions.

## AUXILIARY ROUTINES

MOMGEN: Moments Generator.

MOMDIS: Differential Angular-energy Number Flux Distribution  
Calculator.

## 2. CONTRIBUTORS

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.

## 3. CODING LANGUAGE AND COMPUTER

FORTRAN II, IV; IBM 704 and 7090.

## 4. NATURE OF PROBLEM SOLVED

MOMGEN-MOMDIS represent a theoretical procedure to determine  
the distribution of radiation which has been scattered by a barrier.  
The calculation is made for gamma rays which are scattered in an  
infinite homogeneous medium from a plane perpendicular mono-  
energetic beam of gamma-rays. Specifically, the differential  
angular-energy distributions of gamma-ray fluxes scattered from  
media of various thicknesses are calculated.

## 5. METHOD OF SOLUTION

There are two phases in the moments method procedure. Phase  
one consists of generating moments (for the distributions) and  
phase two consists of constructing the distributions.

In phase one (MOMGEN), three sets of moments are computed.  
They are moments for total, first and second order scattering.  
The moments used in the next phase, referred to as third and  
higher order scattering moments, are obtained by adding moments

for first and second scattering and subtracting their sum from total scattering moments.

In phase two (MOMDIS), a biorthogonal polynomial expansion is used to "fit" the moments. The resulting distribution is for third and higher order scattering. Distributions for first and second order scattering are determined from analytical expressions. Finally, a composite spectrum is formed by adding the three distributions.

#### 6. RESTRICTIONS OR LIMITATIONS

The calculation is limited to photons scattered in a forward direction. Distributions for no more than 6 thicknesses, 10 angles of scatter, and 105 wavelengths may be calculated in each run.

#### 7. TYPICAL RUNNING TIME

The combined running time for MOMGEN and MOMDIS is approximately 25 minutes per one hundred wavelengths on the IBM 704.

Estimated running time of the sample problem on the IBM 7090, in hours: MOMGEN, 0.09; MOMDIS, 0.03.

#### 8. COMPUTER HARDWARE REQUIREMENTS

The codes are designed for a 32K core storage memory, two magnetic tape drives and a printer. They are operable on the IBM 704 and the IBM 7090 computers.

#### 9. COMPUTER SOFTWARE REQUIREMENTS

The codes were designed for the IBM 704 FORTRAN II Monitor System. They were run by RSIC on the IBM FORTRAN IV IBJOB Monitor within the IBSYS Operating System. Tape assignments were made for I-O, systems, and one scratch tape.

A library of data for use in MOMGEN is packaged.

#### 10. REFERENCES

C. V. Smith, "A Moments Method Computer Code for Reconstructing Scattered Gamma Ray Distributions", USNRDL-TR-67-9 (January 1967).

H. Goldstein, J. E. Wilkins, Jr., "Calculations of the Penetration of Gamma Rays", NYO-3075 (1954).

11. CONTENTS OF CODE PACKAGE

The package contains the following items:

- a. the first referenced document, and
- b. a reel of magnetic tape on which is written in 6 files:  
the BCD source card decks, the binary card decks, a library of data, BCD input for a sample problem and a BCD output listing from 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  
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.

13. DATE OF ABSTRACT

January 1968.

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## RSIC CODE PACKAGE CCC- 86

## 1. NAME AND TITLE OF CODE

HANGER: Monte Carlo, Cylindrically Symmetric Shield, Neutron Transport and Heating Code.

## AUXILIARY ROUTINES

DATORG: Organized Data Tape Generator.

SOTGEN: Source Data Generator.

DOSE: Dose Calculator.

BEDIT: Transmitted Particle Sorter.

CAPTS: Propellant Capture Calculator.

The code was originally written by UNC in FAP and FORTRAN. It was rewritten in FORTRAN IV by NASA Lewis Research Center personnel.

## 2. CONTRIBUTOR

United Nuclear Corporation, Development Division - NDA, White Plains, New York.

NASA Lewis Research Center, Cleveland, Ohio.

## 3. CODING LANGUAGE AND COMPUTER

FORTRAN IV; IBM 7090.

## 4. NATURE OF PROBLEM SOLVED

HANGER was designed to calculate neutron transport in cylindrically symmetric NERVA configurations with particular emphasis on the heating produced by neutron interactions in liquid hydrogen. A homogenized reactor provides the radiation source, and a shield exists between the fuel tank and the reactor. The assemblage is assumed to be in a vacuum. The code is designed to calculate, for a given power distribution in the reactor of a nuclear rocket engine:

1. the neutron energy flux spectrum throughout the reactor, shield, and hydrogen propellant,

2. the heating from neutron interactions throughout the hydrogen propellant,
3. the energy spectrum and spatial and angular distribution of the neutron current leakage out of the reactor, shield, and propellant boundaries,
4. the secondary gamma sources arising from neutron capture and inelastic scattering.

Particular attention was devoted to low energy neutron interactions in hydrogen since such interactions lead to important capture gamma-ray sources.

## 5. METHOD OF SOLUTION

A source consisting of the phase space coordinates for the ensemble of neutrons to be processed as Monte Carlo case histories must be supplied on magnetic tape. The source may either be supplied along the face of the hydrogen propellant (when the reactor and shield are absent), or a power pattern [whose spatial distribution is assumed to be of the form  $f(r)g(z)$ ] throughout the reactor may be given from which the required source tape will be generated. Step functions  $f(r)$  and  $g(z)$  are given as input. The direction cosines are picked from an isotropic distribution. The energies are chosen from a truncated fission spectrum whose extreme energies  $E_U$ ,  $E_L$  are input.

The executive routine reads 100 source particles at a time into memory. These particles, along with their generated latents, are tracked until they are: (a) absorbed, (b) escaped, (c) degraded below a cutoff energy, or (d) killed by Russian Roulette. When these particles and their latents have been processed, the next 100 source particles are read in and the process continues until the desired total of neutron histories has been executed or an edit has been forced.

The basic input to the code consists of the geometrical input defining the configuration to be treated and the concentrations of the materials present. The basic output of the code includes

the neutron fluxes and their standard deviations in all reactor and shield regions as well as in certain specified regions in the hydrogen propellant. The heat deposition in prescribed annular regions of the hydrogen propellant is edited. In addition, information concerning leakage and gamma-ray sources is recorded on magnetic tapes for future use.

Special edits have been prepared to evaluate from the results of the main program: fast neutron dose for near axis points, the spatial distribution of neutron captures in hydrogen, and the angle-energy distribution of particles passing through specified planes of the configuration.

#### 6. RESTRICTIONS OR LIMITATIONS

Number of physical regions  $\leq$  30

Number of elements in region  $\leq$  10

Number of energies  $\leq$  11 (HANGER),  $\leq$  100 (DOSE)

#### 7. TYPICAL RUNNING TIME

No study has been made to determine typical running time. Estimated running time for sample problem on the IBM 7090, in hours: HANGER, 0.05; DATORG, 0.05; SOTGEN, 0.04. No timing has been done by RSIC on DOSE, BEDIT, and CAPTS.

#### 8. COMPUTER HARDWARE REQUIREMENTS

IBM 32K 7090 or 7094 computers with 7 tape units.

#### 9. COMPUTER SOFTWARE REQUIREMENTS

Standard IJOB Monitor System may be used with systems, input, output, and 4 pool tapes assigned. The ALTIO package is used to conserve storage space. A library of cross sections has been packaged.

#### 10. REFERENCE

G. Rabinowitz, F. Malmberg, M. Shapiro, "An IBM-7090 Monte Carlo Program to Calculate Neutron Transport in Cylindrically Symmetric Shields with Particular Emphasis on Neutron Heating in

Liquid Hydrogen", UNC-5043 (November 1962).

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 19 files:  
the BCD source card deck for each code packaged, BCD  
input data for a sample problem, a BCD library of cross  
sections, binary card decks for each code, and output  
from 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

January 1968.

## RSIC CODE PACKAGE CCC-87

## 1. NAME AND TITLE OF CODE

LG-H: Ray Analysis Cylindrical Duct Kernel Code for Neutrons and Gamma Rays.

LG-H is also a part of the ENEA Computer Programme Library, listed as Abstract No. ENEA 124.

## 2. CONTRIBUTORS

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

Hitachi Central Research Laboratory, Hitachi Ltd.

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; IBM 7090 and the IBM 7040 and 7044.

## 4. NATURE OF PROBLEM SOLVED

LG-H was designed to solve problems connected with gamma rays and neutrons streaming through straight cylindrical ducts. Fluxes and doses are studied at the duct exit surface of the shield barrier from an infinite plane source on the other side of the shield.

## 5. METHOD OF SOLUTION

The infinite source region is divided into 4 parts and the contribution of each part is calculated separately by the ray analysis method. The total dose at the detector and the contribution to it of each part of the plane source are calculated, using an assumed kernel, by varying the parameters, including the source angular distribution, position of the detector, duct radius, duct length, energy of radiation and materials of the shield.

## 6. RESTRICTIONS OR LIMITATIONS

The following limits apply:

Duct radius  $\leq$  100 cm.

Distance  $\leq$  100 cm.

Duct length  $\leq$  100 cm.

Maximum angular dependency 100

Maximum buildup factor constant 100

## 7. TYPICAL RUNNING TIME

Typical running time is less than 2 seconds per case.

Estimated running time for sample problems: 0.01 hour  
on the IBM 7090.

## 8. COMPUTER HARDWARE REQUIREMENTS

A 32K computer with 3 tape units is required. The code was designed to be run on the IBM 7040, 7044, and 7090 computers.

## 9. COMPUTER SOFTWARE REQUIREMENTS

The code has been compiled and executed on the IBM FORTRAN IV IJOB Monitor in the IBSYS Operating System. Only I-O and systems tape assignments were made.

## 10. REFERENCE

Mitsuyuki Kitazume, Akira Tsuruo, and Mitsuo Shindo, "Gamma Rays and Neutrons Streaming About a Cylindrical Duct by the Ray Analysis Method", Informal Memo (1966).

## 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 3 files:  
the BCD source card decks, the binary card deck, BCD input for a sample problem, and a 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  
Post Office Box X  
Oak Ridge, Tennessee 37830

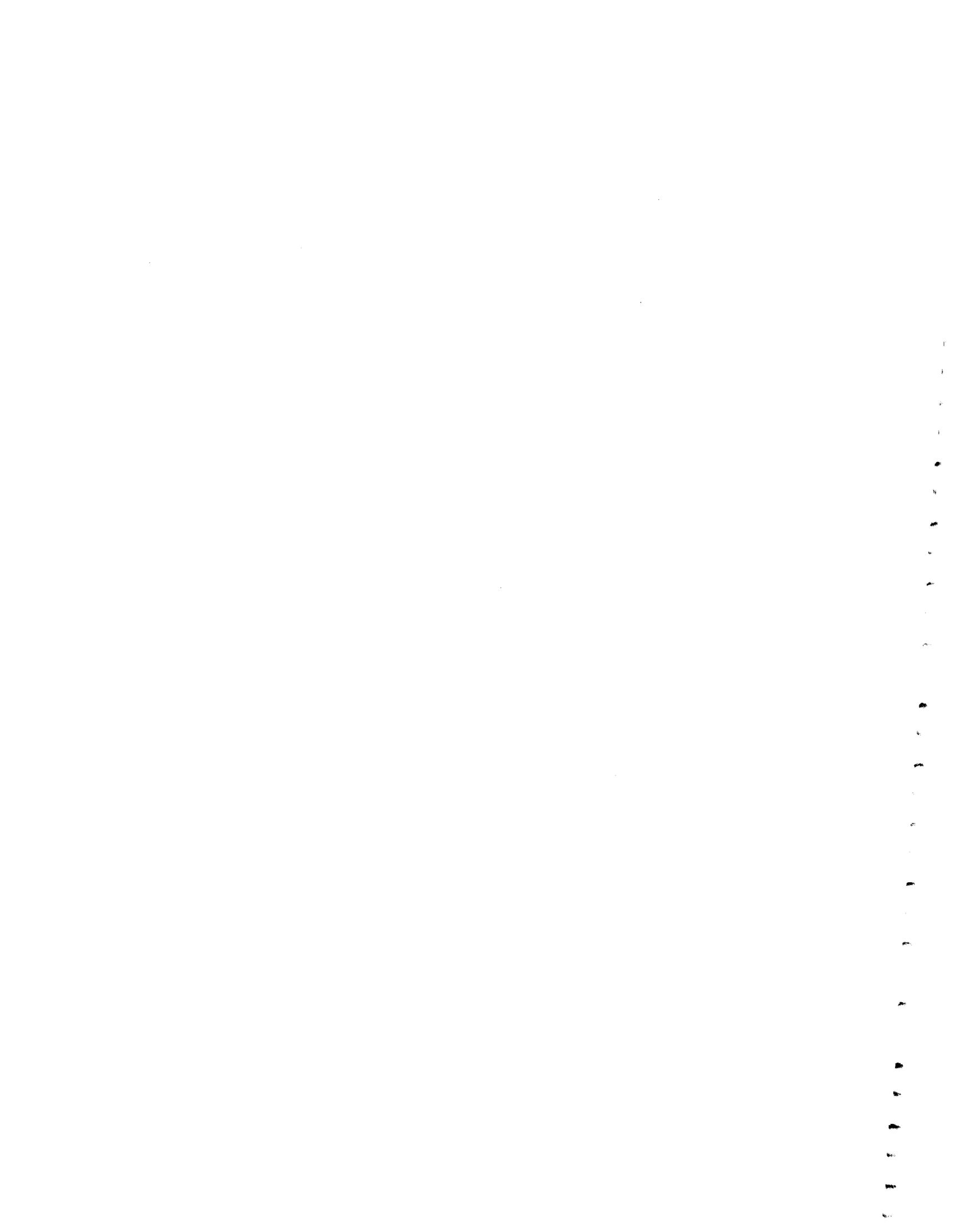
or telephoned to

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

January 1968.



## RSIC CODE PACKAGE CCC-88

## 1. NAME AND TITLE OF CODE

RADOS: Gamma-Ray Dose Estimation from Cloud of Radioactive Gases by Kernel Integration.

## 2. CONTRIBUTOR

Savannah River Laboratory, E. I. du Pont de Nemours and Company, Aiken, South Carolina.

## 3. CODING LANGUAGE AND COMPUTER

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

## 4. NATURE OF PROBLEM SOLVED

RADOS represents the finite spatial distribution of airborne source material as an infinite number of line sources. The code provides a means of rapidly calculating whole body gamma-ray dose from a finite cloud of radioactive material.

## 5. METHOD OF SOLUTION

The calculational method is based on the assumption that there is no change in the size and shape of the radioactive cloud of unit thickness during the time the cloud will irradiate the receptor. For an instantaneous release, the entire dose may be considered as being accumulated in unit time from a cloud extending to infinity in both directions from the receptor. This assumption permits the total dose to be calculated as the sum of the doses from an infinite number of line sources.

## 6. RESTRICTIONS OR LIMITATIONS

The following limitations must be considered:

- a. The code permits dose calculations only on the axial center of the cloud path.
- b. The receptor should be  $\geq$  600 meters downwind from the release point.

- c. The gamma-ray energy group structure is limited to the range 0.5 MeV to 2.0 MeV.

#### 7. TYPICAL RUNNING TIME

The program required less than 2 seconds on the IBM 360/65 for a four energy group, 12 isotope, 1 distance problem.

Estimated running time on the IBM 360/75 for the packaged sample problem: 1 minute, 18 seconds.

#### 8. COMPUTER HARDWARE REQUIREMENTS

The code was designed for the IBM 360/65, using standard I-O.

#### 9. COMPUTER SOFTWARE REQUIREMENTS

The code is operable on the IBM-360/75 Operating System using OS-360 Fortran H Compiler.

#### 10. REFERENCE

R. E. Cooper, "RADOS, A Code to Estimate Gamma Dose from a Cloud of Radioactive Gases," DP-1098 (June 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 three files: the source card deck (EBCDIC), the BCD input for a sample problem, and the output listing from 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  
P. O. Box X  
Oak Ridge, Tennessee 37830

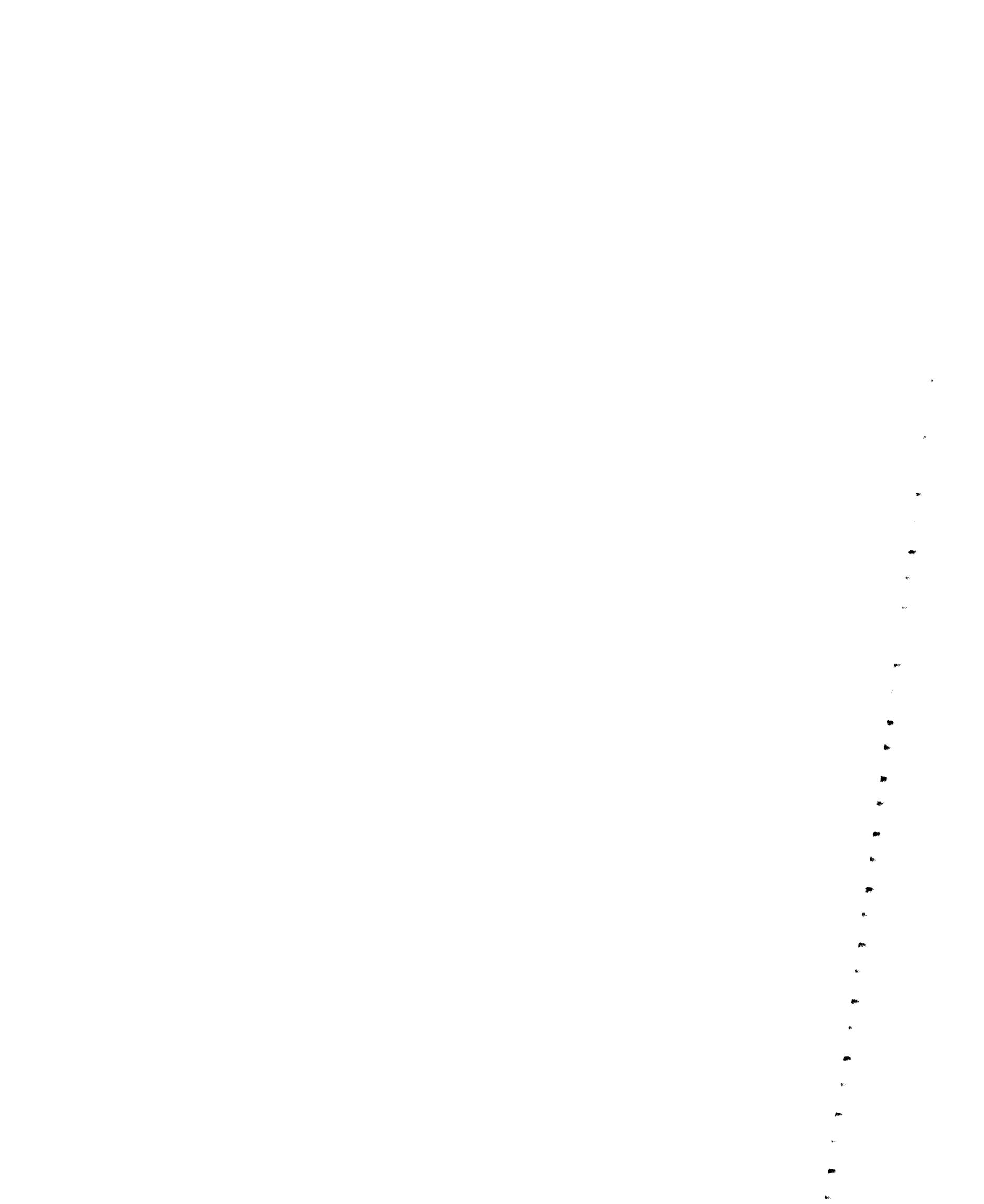
or telephoned to

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.

13. DATE OF ABSTRACT

May 1969.



## RSIC CODE PACKAGE CCC-89

## 1. NAME AND TITLE OF CODE

DOT: Multigroup Two-Dimensional Discrete Ordinates Transport Code with Anisotropic Scattering.

DOT was first available in 1966. The DOT-II version was made available in 1969.

## 2. CONTRIBUTORS

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.

## 3. CODING LANGUAGE AND COMPUTER

FORTRAN IV; IBM 7090, IBM 360/75, and CDC 6600.

## 4. NATURE OF PROBLEM SOLVED

DOT is a general purpose program which solves the linear, energy-dependent, Boltzmann transport equation for two-dimensional r-z, x-y, and r- $\theta$  geometries. The basic form of the solution is the quantity  $\bar{\phi}(r_i, z_j, E_g, \Omega_d) \Delta E_g = \phi_{ijgd}$ , the flux, averaged in the spatial interval surrounding  $r_i, z_j$ , integrated over the energy group g, and averaged in the solid segment surrounding  $\Omega_d$ .

## 5. METHOD OF SOLUTION

The gradient or convection term in the Boltzmann equation is approximated by a finite difference technique known as discrete ordinates - Carlson's  $S_n$  method. The inscatter integral is approximated by expanding the differential cross-section in a Legendre series which allows the integral to be computed by quadrature. DOT will solve forward or adjoint, homogeneous or inhomogeneous problems. The inhomogeneous problems may have a

volume distributed source or a specified angular flux at the right or top boundaries; fissions may be included for a subcritical system. Homogeneous (eigenvalued) problems will determine the following: (1) static multiplication factor, "k", (2) time absorption, "Rossia", (3) concentration for a specified k, (4) zone thickness for a specified k.

The primary differences between DDK and DOT are given in the following:

- a. general anisotropic scattering is allowed,
- b. boundary sources may be treated by specifying the angular flux at the right or top boundaries,
- c. angular fluxes may be printed or written on binary tape,
- d. the pass B scaling for problems with reflected right or top boundaries has been removed,
- e. if specified, a pointwise inner iteration flux convergence criteria is used instead of the integral test,
- f. the integral inner iteration convergence criteria specifies that the average absolute pointwise flux error be less than epsilon,
- g. input data is processed by the FIDO routine used in DTF-II and ANISN, and
- h. if the linear difference equations produce a negative flux, the flux is recalculated using the step function difference equations. This technique, which inhibits oscillation due to extrapolation, is optional.

## 6. RESTRICTIONS OR LIMITATIONS

Problem size is limited by machine size.

## 7. TYPICAL RUNNING TIME

No statistics have been accumulated by RSIC as to typical running time.

Estimated running time on the IBM 7090 for the two packaged sample problems: 16 minutes.

## 8. COMPUTER HARDWARE REQUIREMENTS

Standard equipment is used on either computer, and a maximum of 7 tapes or direct access devices in addition to I-O is utilized.

## 9. COMPUTER SOFTWARE REQUIREMENTS

CCC-89A: The code is operable on the IBM 7090/7094 IBSYS Operating System using IBJOB Processor.

CCC-89B: The code is operable on the IBM 360/75 Operating System using G Compiler.

CCC-89C The code is operable on the IBM 360/75 Operating System using OS-360 Fortran H Compiler.

CCC-89D: The code is operable on the CDC-6600.

Any non-standard library routines used are included in the affected code package.

## 10. REFERENCES

F. R. Mynatt, "A User's Manual for DOT," K-1694 (January 1967).

F. R. Mynatt, "Development of Two-Dimensional Discrete Ordinates Transport Theory for Radiation Shielding," to be published in Nucl. Sci. Eng. (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 separate files: the source card decks and BCD input and output listing from 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

May 1969.

## RSIC CODE PACKAGE CCC-90

## 1. NAME AND TITLE OF CODE

AMC: Monte Carlo Albedo Code for Neutron and Capture Gamma-Ray Distributions in Rectangular Concrete Ducts.

## 2. CONTRIBUTOR

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

## 3. CODING LANGUAGE AND COMPUTER

FORTRAN IV and MAP; IBM 360/75.

## 4. NATURE OF PROBLEM SOLVED

AMC calculates neutron and associated capture gamma-ray transmission through concrete-walled rectangular ducts of varying complexity to establish design criteria for entryway shielding against initial weapons radiation.

## 5. METHOD OF SOLUTION

AMC can be used to calculate the fluxes and dose rates inside large rectangular ducts constructed of concrete walls and arising from neutrons of all energies incident upon the duct mouth. An option in the code allows a simultaneous calculation of the secondary gamma-ray dose rates arising from wall capture as well. The code treats duct transmission by employing the albedo concept in conjunction with a Monte Carlo treatment of history generation.

In its present form the code can calculate the fast-neutron dose rates, epicalcium differential energy fluxes, thermal fluxes, and wall-capture secondary gamma-ray dose rates arising from an arbitrary incident neutron source for the following concrete-walled rectangular structures:

- a. a completely enclosed room,
- b. an open straight duct or a straight duct closed at one end,
- c. an open or closed two-legged duct with one right-angled bend,

- d. an open or closed three-legged duct with two right-angled bends,
- e. a three-legged duct open or closed at the mouth, but opening into a room at the end of the third leg, with two right-angled bends,
- f. any of geometries c through e above with the first leg sloping downward, the mouth and the remaining legs being horizontal.

In geometries a through e the mouth is vertical and the legs horizontal.

#### 6. RESTRICTIONS OR LIMITATIONS

The structure considered must have dimensions of a foot or greater, a limitation imposed by the albedo model.

The present description is limited to six distinct types of rectangular duct geometries but includes ducts having up to three legs with two right-angle bends.

#### 7. TYPICAL RUNNING TIME

Estimated running time for packaged sample problem: 10 minutes.

#### 8. COMPUTER HARDWARE REQUIREMENTS

The code was designed for the IBM 360/75 with standard I-O and one tape unit.

#### 9. COMPUTER SOFTWARE REQUIREMENTS

The code is written in FORTRAN IV with the exception of two function subprograms written in MAP. It is operable on the IBM 360/75 Operating System using OS-360 Fortran H Compiler.

A file of epicadmium data or fast-neutron data is input on logical 3 or 4 respectively, according to value of the mode-determining option.

## 10. REFERENCE

R. E. Maerker and V. R. Cain, "AMC: A Monte Carlo Code Utilizing the Albedo Approach for Calculating Neutron and Capture Gamma-Ray Distributions in Rectangular Concrete Ducts," ORNL-3964 (August 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 8 files:  
BCD library data, the source card decks, BCD input for two sample problems and BCD output list from 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

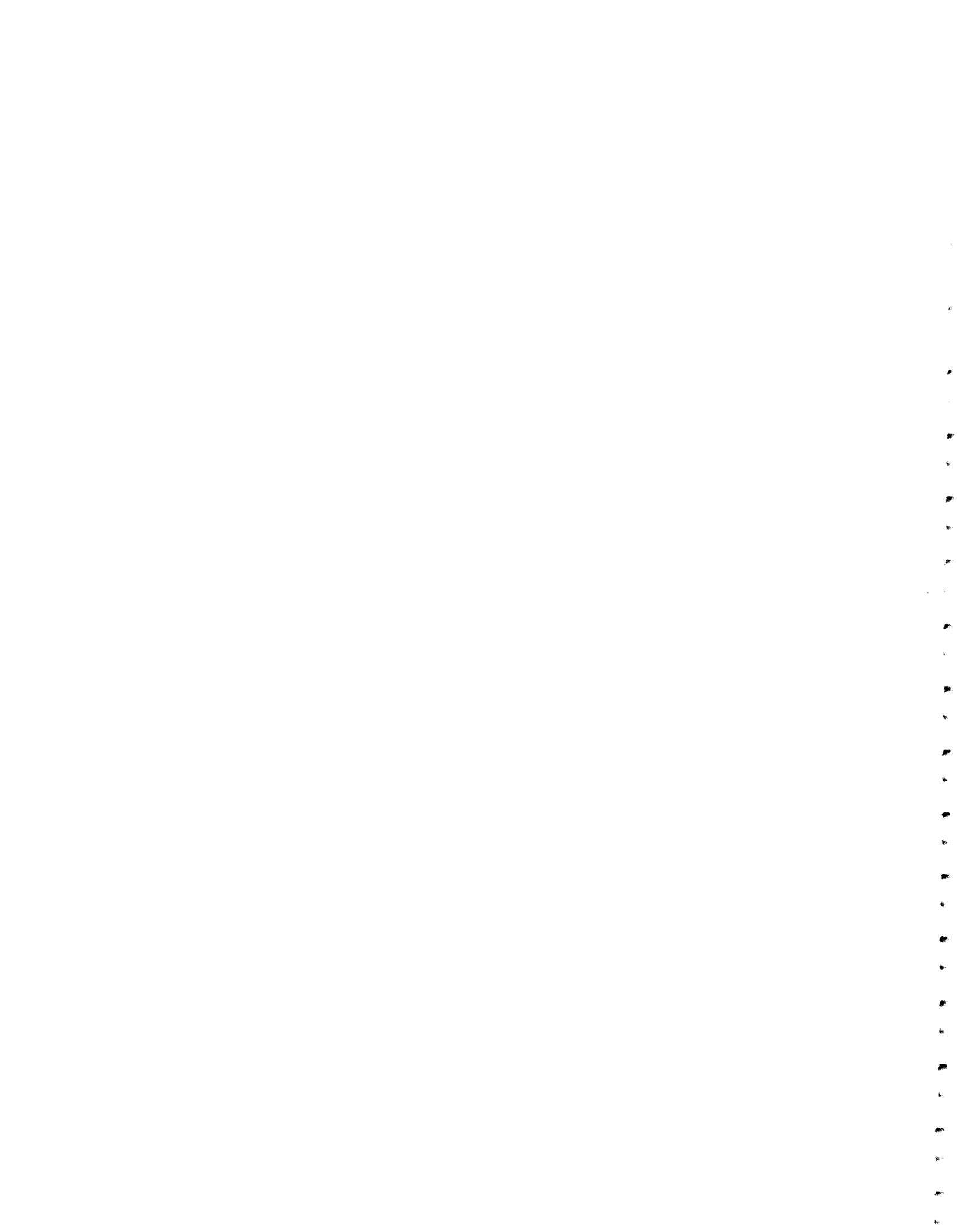
or telephoned to

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.

## 13. DATE OF ABSTRACT

May 1969.



## RSIC CODE PACKAGE CCC-91

## 1. NAME AND TITLE OF CODE

NEFIRS: Multigroup Spinney Method Removal-Diffusion Code  
for Neutrons.

## 2. CONTRIBUTOR

Gulf General Atomic, San Diego, California.

## 3. CODING LANGUAGE AND COMPUTER

FORTRAN IV and MAP; IBM 360/75.

## 4. NATURE OF PROBLEM SOLVED

Multigroup neutron fluxes and reaction rates are calculated for a specified shield system starting from flux levels to be specified for its inner interface. Problems are solved for 1 dimension in either plane, cylindrical, or spherical geometry.

## 5. METHOD OF SOLUTION

The neutron flux distribution is calculated using a multigroup age-diffusion technique. A modified uncollided (removal) flux and a related removal collision density of fission neutrons having energies greater than one MeV are computed to serve as a spatially varying source term for the first diffusion group (Spinney Method). The attenuation of these fast neutrons is computed analytically for 17 equally spaced energy groups. The removal cross sections are calculated from the total cross sections at the energies in question but with a correction that takes approximate account of the anisotropy of elastic scattering at high energies according to the prescription of Spinney.

The diffusion portion of NEFIRS is designed to handle up to eight energy groups the spacing of which is fixed if the neutronic data are taken from the code library. Down-scattering is treated by age-theory methods. The solution to the diffusion equation is approximated by a set of finite difference equations of the type

employed in the WANDA codes.

All neutronic constants required for a problem may be generated through the data routine of the code. However, the diffusion routine can also be executed using constants read in as input.

The code consists of several basic sections: a routine to determine the spatial variation of the uncollided flux in the shield resulting from fission sources within the core, a routine to compute from a library of basic data the neutronic constants necessary for an eight-group diffusion calculation through the various layers of structural or shielding materials surrounding the core, a routine to solve the one-dimensional diffusion equation for a multilayer shield arrangement in one of the three standard geometries, a library containing the basic data for up to twenty shield materials, a routine to evaluate the second-order exponential integral used in the analytic evaluation of the spatial distribution of the uncollided flux.

#### 6. RESTRICTIONS OR LIMITATIONS

The library is restricted to the data of twenty different materials. The number of meshpoints throughout the entire shield system must not exceed 1000. No more than eight diffusion groups can be handled.

#### 7. TYPICAL RUNNING TIME

A completed problem with about 260 meshpoints executed from tape on a UNIVAC 1108 required 15 seconds for computing and printing.

Estimated running time for the packaged sample problem on the IBM 360/75: 1 minute, 18 seconds.

#### 8. COMPUTER HARDWARE REQUIREMENTS

The code was designed for the UNIVAC 1108 computer and was converted by RSIC to run on the IBM 360/75. A basic storage capacity of about 40 K is required.

#### 9. COMPUTER SOFTWARE REQUIREMENTS

The packaged version is operable on the IBM 360/75 Operating

System using OS-360 Fortran H Compiler.

A special subroutine DIVID is included in this package.

10. REFERENCE

C. A. Goetzmann, "NEFIRS, A Computer Program for Exploratory Studies of Neutron Flux Distributions in Reactor Shields," GA-8069 (August 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 4 files:  
the BCD source card decks, the BCD input for a sample problem, and an output list 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  
Post Office Box X  
Oak Ridge, Tennessee 37830

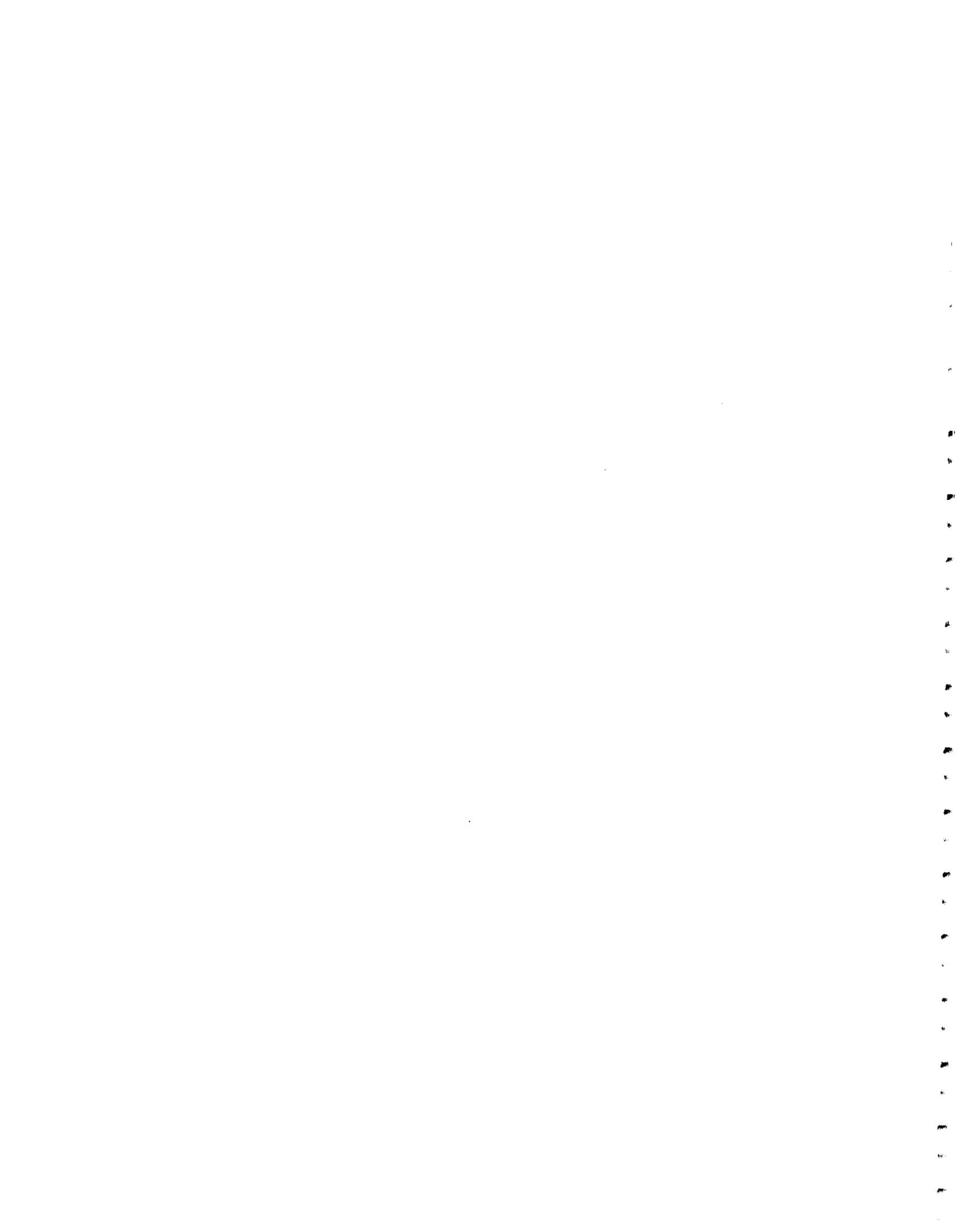
or telephoned to

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

May 1969.



## RSIC CODE PACKAGE CCC-92

## 1. NAME AND TITLE OF CODE

SAP - N and G: Neutron and Gamma-Ray Albedo Model Scatter  
Shield Analysis Code.

## 2. CONTRIBUTOR

Westinghouse Astronuclear Laboratory, Pittsburgh, Pennsylvania.

## 3. CODING LANGUAGE AND COMPUTER

FORTRAN IV; IBM 7090 and 7094.

## 4. NATURE OF PROBLEM SOLVED

Gamma-ray or neutron dose rates from radiation reflected from  
plane or cylindrical surfaces are calculated at receiver points.

## 5. METHOD OF SOLUTION

Formulas involving albedo data and incident dose rates are  
integrated over surfaces. The neutron albedo data are based on  
French and Wells and the gamma-ray data on Berger and Raso.  
"Opaque" finite planes may be included to account for shielding by  
"optically dense" material encountered by reflected radiation. These  
planes may be assigned an attenuation factor.

## 6. RESTRICTIONS OR LIMITATIONS

The following limitations are noted.

Maximum number of receiver points:	1000
Maximum number of opaque planes or shielding cylindrical surfaces:	100
Maximum number of energy groups:	100
Scattering cylindrical surface must be parallel to Z-axis.	

## 7. TYPICAL RUNNING TIME

Estimated running times for packaged sample problems in minutes:  
SAP-N, 0.01; SAP-G, 0.01.

## 8. COMPUTER HARDWARE REQUIREMENTS

The codes were designed for a 32-K IBM 7090 using standard I-O.

## 9. COMPUTER SOFTWARE REQUIREMENTS

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

## 10. REFERENCES

L. D. Stephenson, F. Planinsek, A. Killinger, and B. J. Sapovchak, "SAP - A Neutron and Gamma Ray Albedo Model Scatter Shield Analysis Program," WANL-TME-1273 (September 1965).

L. D. Stephenson and B. J. Sapovchak, "N-SAP and G-SAP - Neutron and Gamma Ray Albedo Model Scatter Shield Analysis Programs," WANL-TME-1273 Revision A (December 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 8 files:  
the source card decks, the binary card decks, BCD input for two sample problems, and a BCD output listing for each 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

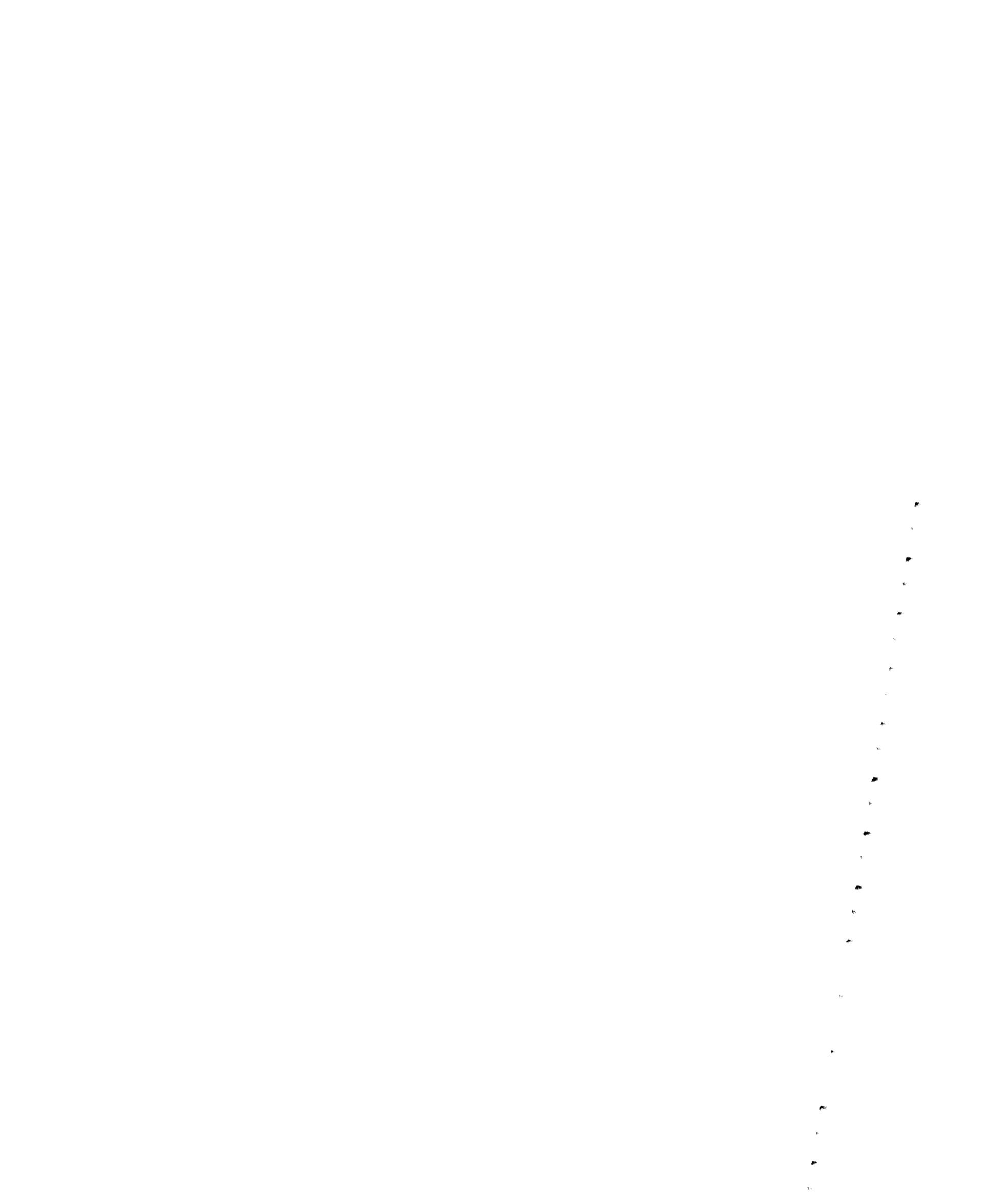
or telephoned to

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

May 1969.



## RSIC CODE PACKAGE CCC-93

## 1. NAME AND TITLE OF CODE

MCFLARE: Monte Carlo Code to Simulate Solar Flare Events and Estimate Probable Doses Encountered on Interplanetary Missions.

## 2. CONTRIBUTOR

NASA Lewis Research Center, Shielding Analysis Section, Nuclear Systems Division, Cleveland, Ohio.

## 3. CODING LANGUAGE AND COMPUTER

FORTRAN IV; IBM 7090 and 7094.

## 4. NATURE OF PROBLEM SOLVED

The program estimates probability of exceeding dose from solar flares encountered on an interplanetary mission for various (input) shields.

## 5. METHOD OF SOLUTION

MCFLARE uses Monte Carlo methods to simulate solar flare occurrences during an interplanetary space voyage. The total biological dose inside a shielded crew compartment due to the flares encountered is determined. MCFLARE evaluates the doses obtained on a large number of trips having identical trajectories. From these results, a dose  $D_p$  having a probability  $p$  of not being exceeded during the voyage can be determined as a function of  $p$  for any shield configuration.

The user of the code selects any number of solar flares considered to be representative of the ones that will occur during future solar active periods. The flares are assumed to occur randomly during these periods. The dose at a distance of 1 AU\* from the

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\*astronomical unit =  $1.496 \times 10^8$  km.

sun from each of these flares behind any shield configuration investigated is input to MCFLARE. MCFLARE accounts for the dependence of the dose received from a flare on the distance from the sun according to a  $(1/r)^\alpha$  variation, where  $r$  is the distance from the sun and the exponent  $\alpha$  is assigned any integral value including zero. From trajectory parameters, which are also input to the computer program, the distance from the sun as a function of time during the trip is calculated.

#### 6. RESTRICTIONS OR LIMITATIONS

Limitations on number of bins used for scoring are:

Number of different flare events considered	$\leq$	25
Number of different shields considered	$\leq$	7
Number of dose bins of unit width to be used	$\leq$	900

#### 7. TYPICAL RUNNING TIME

Sample problem described in reference took 2 minutes for 13 flare events, 40000 trips.

Estimated running time of packaged sample problem: 0.11 hours.

#### 8. COMPUTER HARDWARE REQUIREMENTS

The code was designed for a 32K IBM 7094-II using standard I-O.

#### 9. COMPUTER SOFTWARE REQUIREMENTS

The code is operable on the IBM 7090/7094 IBSYS Operating System using IBJOB Processor. A random number generator is included in this package.

#### 10. REFERENCE

Gerald P. Lahti, Irving M. Karp, and Burt M. Rosenbaum, "MCFLARE, A Monte Carlo Code to Simulate Solar Flare Events and Estimate Probable Doses Encountered on Interplanetary Missions," NASA TN D-4311 (February 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:  
the source card deck, the binary card deck, BCD input for  
a sample problem, and a BCD output listing for 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

May 1969.



## RSIC CODE PACKAGE CCC-94

## 1. NAME AND TITLE OF CODE

KAP-V: Kernel Integration Code in Complex Geometry.

KAP-V is part of a system of codes for shield design called "Synthesis of Computational Methods for the Design and Analysis of Radiation Shields for Nuclear Rocket Systems."\* The codes in this system are packaged by RSIC as CCC-94 through CCC-98. The programs in CCC-94, CCC-95, and CCC-96 are termed by the originators the "early" design method.

## 2. CONTRIBUTOR

Westinghouse Astronuclear Laboratory, Pittsburgh, Pennsylvania.  
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

Neutron and/or gamma radiation levels are calculated at detector points located within or outside a complex radiation source geometry describable by a combination of quadratic surfaces. A variety of options are available for describing cylindrical, spherical, disc, line, or point sources or source distributions in complex geometries. The output can be flux, dose, or heating rate.

## 5. METHOD OF SOLUTION

The point kernel method is used. The attenuation function, or kernel, for gamma rays employs exponential attenuation with a buildup factor. Three optional fast neutron attenuation functions are included: (1) a modified Albert-Welton function for calculating fast neutron dose rate using removal cross sections, (2) a bivariate polynomial expression for computing neutron spectra using infinite media moments data, and (3) a monovariant polynomial expression for

computing neutron spectra using infinite media moments data.

6. RESTRICTIONS OR LIMITATIONS

The following limitations are noted:

maximum number of boundary surfaces: 100  
maximum number of geometric regions or zones: 100  
maximum number of sets of response functions: 10

7. TYPICAL RUNNING TIME

Estimated running time of sample problem: 3 minutes.

8. COMPUTER HARDWARE REQUIREMENTS

The code was designed for a 32K IBM 7090 or 7094 using standard I-O and one pool tape.

9. COMPUTER SOFTWARE REQUIREMENTS

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

10. REFERENCES

R. K. Disney and M. A. Capo, "Synthesis of Computational Methods for the Design and Analysis of Radiation Shields for Nuclear Rocket Systems - KAP-V: The Point Kernel Attenuation Program," WANL-PR-(LL)-010, Volume 4 (1967).

M. A. Capo, R. K. Disney, T. M. Jordan, and H. C. Woodsum, \*"Synopsis of Methods and Results of Analyses," WANL-PR-(LL)-010, Volume 1 (1967).

M. A. Capo, \*"Sample Nuclear Rocket Reactor Problem for the KAP-V Code," WANL-PR-(LL)-014, Volume 7 (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 4 files:

the source card deck, the binary card 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  
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

May 1969.



## RSIC CODE PACKAGE CCC-95

## 1. NAME AND TITLE OF CODE

TAPAT: Multigroup One-Dimensional Discrete Ordinates Code.  
AUXILIARY ROUTINES

POINT: Data Generator and Cross Section Library.

TAPAT is part of a system of codes for shield design called "Synthesis of Computational Methods for the Design and Analysis of Radiation Shields for Nuclear Rocket Systems."\* The codes in this system are packaged by RSIC as CCC-94 through CCC-98. The programs in CCC-94, CCC-95, and CCC-96 are termed the "early" design methods by the originators.

## 2. CONTRIBUTOR

Westinghouse Astronuclear Laboratory, Pittsburgh, Pennsylvania.  
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

The POINT code prepares the bulk of the input data in the form required for the one-dimensional TAPAT system, the two-dimensional flux data processing code NAGS and the two-dimensional discrete ordinates code CCC-97/ODD-K.

The TAPAT system consists of 5 programs that solve the one-dimensional Boltzmann equation using  $S_n$  and/or diffusion theory, and perform data processing functions to provide a complete (neutron and photon radiation analysis) of a nuclear reactor or other radiation source.

There are five types of problems:

- (1) an eigenvalue problem with no fixed (distributed or boundary) sources.

- (2) a fixed (distributed or boundary) source problem with no fissions.
- (3) a fixed (distributed or boundary) source problem with fissions.
- (4) a concentration search problem where a specified concentration is varied until a desired eigenvalue is realized.
- (5) a region thickness search problem where a specified region thickness is varied until a desired eigenvalue is reached.

## 5. METHOD OF SOLUTION

The 5 programs of TAPAT are consistent versions of:

- (1) the ADDICT diffusion theory program which solves the multi-group diffusion equation in one-dimensional slab, cylindrical, or spherical geometries.
- (2) the MIST transport theory program which solves the multi-group discrete ordinates  $S_n$  equation in one-dimensional slab geometry.
- (3) the MISHT  $S_n$  program for one-dimensional spherical geometry.
- (4) the TOPIC  $S_n$  program for one-dimensional cylindrical geometry.
- (5) the FLUX EDIT data processing program which calculates reaction rates, distributed neutron and photon fixed sources, separable spatial neutron and photon energy source distributions, and other quantities of interest, such as heating rates.

The POINT program contains the basic microscopic cross section library, and input data processing routines to prepare input for the one-dimensional TAPAT program system, the two-dimensional transport program CCC-97/ODD-K, and the data processing code for ODD-K, NAGS.

The calculation of macroscopic cross sections readily usable in the transport programs, TAPAT and ODD-K, is achieved by employing a built-in library of region-independent neutron and gamma-ray microscopic cross sections. These group-averaged cross sections are applicable for the analysis of graphite-moderated nuclear reactor

rocket systems. The program also calculates response functions as punched card output for use in both the TAPAT and NAGS programs to calculate, for example, neutron and gamma ray energy deposition, fixed neutron sources, or prompt and secondary gamma ray sources.

#### 6. RESTRICTIONS OR LIMITATIONS

The following limitations apply:

- up to first order Legendre expansion of scattering function within a group,
- isotropic group-to-group scattering,
- number of mesh points  $M$ :  $3 \leq M < 100$ ,
- total number of regions  $J$ :  $1 \leq J \leq 30$ ,
- total number of energy groups  $N$ :  $1 \leq N \leq 20$ ,
- number of downscatter groups  $NDS$ :  $1 \leq NDS \leq 6$ ,
- number of upscatter groups  $NPS$ :  $1 \leq NPS \leq 5$ ,
- number of cross section sets (elements, isotopes, or materials)  $MMIX$ :  $0 < MMIX \leq 20$ ,
- order of  $S_n$  angular quadrature: 2 or 4, and
- order of Gauss quadrature in 0 (TOPIC):  $MIK \leq 7$ .

#### 7. TYPICAL RUNNING TIME

Estimated running times for sample problem in minutes: POINT, 4; TAPAT, 21.

#### 8. COMPUTER HARDWARE REQUIREMENTS

The codes were designed for a 32K IBM 7090 and 7094 and makes use of as many as 7 tapes in addition to I-0.

#### 9. COMPUTER SOFTWARE REQUIREMENTS

The codes are operable on the IBM 7090/7094 IBSYS Operating System using IBJOB Processor. TAPAT makes use of the overlay feature.

#### 10. REFERENCES

R. G. Soltesz and R. K. Disney, "The Point Program and Cross Section Library," WANL-PR-(LL)-010, Volume 2 (December 1967).

R. K. Disney, \*'"TAPAT, The FORTRAN IV Program System,"  
WANL-PR-(LL)-010, Volume 3 (December 1967).

M. A. Capo, \*'"Basic Geometry for the Nuclear Rocket Reactor  
Sample Problem," WANL-PR-(LL)-014, Volume 1 (December 1967).

R. G. Soltész, \*'"Sample Nuclear Rocket Reactor Problem for the  
POINT Program," WANL-PR-(LL)-014, Volume 2 (December 1967).

R. K. Disney and M. A. Capo, \*'"Sample Nuclear Rocket Reactor  
Problem for the TAPAT Program," WANL-PR-(LL)-014, Volume 3  
(December 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 9 files  
for each packaged code: a source card deck, a binary card  
deck, a BCD input for the sample problem, 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 code package should send a reel of magnetic  
tape to the above address.

#### 13. DATE OF ABSTRACT

May 1969.

## RSIC CODE PACKAGE CCC-96

## 1. NAME AND TITLE OF CODE

TIC-TOC-TOE: On-Axis Liquid Hydrogen Propellant Tank Heating  
Code

## AUXILIARY ROUTINES

POINT: Data Generator and Cross Section Library

TIC-TOC-TOE is part of a system of codes for shield design called "Synthesis of Computational Methods for the Design and Analysis of Radiation Shields for Nuclear Rocket Systems."\* The codes in this system are packaged by RSIC as CCC-94 through -98. The programs in CCC-94,95, and 96 are termed the "early" design method by the originators.

## 2. CONTRIBUTOR

Westinghouse Astronuclear Laboratory, Pittsburgh, Pennsylvania.

## 3. CODING LANGUAGE AND COMPUTER

FORTRAN IV; IBM 7090 and 7094.

## 4. NATURE OF PROBLEM SOLVED

Calculations are performed of radiation heating rate distribution in on-axis liquid hydrogen tanks. Neutron and photon sources are expressed as multigroup, angular equivalent point sources. The tank geometry is described by a series of axisymmetric truncated cones and/or cylinders.

Quantities computed at specified points in the tank are

- (1) gamma-ray heating by energy group,
- (2) neutron kinetic heating by fast energy group,
- (3) capture gamma-ray heating due to neutron captures in liquid hydrogen, and
- (4) capture gamma-ray heating due to neutron captures in tank wall.

These same quantities are also obtained for points at the corners of the volume elements employed in performing volume integrations over the tank. Quantities obtained from the volume integration include:

- (1) total gamma-ray and neutron heating rate as a function of propellant height,
- (2) radial and volume averaged heating rates as a function of propellant height,
- (3) radial averaged temperature rise as a function of time for a no-mix fluid model, and
- (4) temperature rise as a function of time for a complete mix fluid model.

#### 5. METHOD OF SOLUTION

Radiation levels from Monte Carlo calculations (propellant heating rates and thermal neutron capture rates as a function of depth and radius in a cylindrical tank for unit monoenergetic sources) have been empirically fitted as a function of  $r$  and  $z$ . These data are combined with equivalent point source strengths determined by a code such as KAP-V in a kernel technique to yield heating rates at specified points.

Propellant temperature is computed using the specific heat and integrating over time. Either a "no-mix" or "complete mix" model is assumed.

#### 6. RESTRICTIONS OR LIMITATIONS

Number of angles used to tabulate input fluxes:  $2 \leq \text{NAMAX} \leq 25$ .  
 Number of conical and cylindrical sections:  $1 \leq \text{NRMAX} \leq 24$ .  
 Number of discrete points in tank:  $0 \leq \text{NPOINT} \leq 100$ .

#### 7. TYPICAL RUNNING TIME

Timing of the sample problem on an IBM 7090 was as follows:  
 POINT, 4 minutes and TIC-TOC-TOE, 3 minutes.

#### 8. COMPUTER HARDWARE REQUIREMENTS

The code was designed for a 32K IBM 7090 and 7094.

## 9. COMPUTER SOFTWARE REQUIREMENTS

TIC-TOC-TOE and its auxiliary POINT may be compiled and executed on the IBM IJOB Monitor in the IBSYS operating system or on similar software.

## 10. REFERENCES

T. M. Jordan and H. C. Woodsum, \*"TIC-TOC-TOE, A Fortran Program for the Temperature in the Coolant Tank and Other Calculations and for the Thermal Neutron Originating Energy," WANL-PR-(LL)-010, Volume 5 (December 1967).

M. A. Capo, \*"Basic Geometry for the Nuclear Rocket Reactor Sample Problem," WANL-PR(LL)-014, Volume 1 (December 1967).

M. A. Capo and T. M. Jordan, \*"A Sample Nuclear Rocket Reactor Problem for the TIC-TOC-TOE Code," WANL-PR-(LL)-014, Volume 9 (December 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 9 files:  
5 files for POINT and 4 for TIC-TOC-TOE, the BCD source card decks, BCD input for a sample problem for each code and a BCD output listing 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  
May 1969.

## RSIC CODE PACKAGE CCC-97

## 1. NAME AND TITLE OF CODE

ODD-K: Multigroup Two-Dimensional Discrete Ordinates Code.

## AUXILIARY ROUTINES

POINT: Data Generator and Cross Section Library.

NAGS: Data Generator - Sources and Energy Deposition  
in Two-Dimensional Geometry.

DAFT: Angular Flux Code.

ODD-K is a WANL modification of the LASL DDK to improve the transport analysis. It is part of a system of codes for shield design called "Synthesis of Computational Methods for the Design and Analysis of Radiation Shields for Nuclear Rocket Systems."\* The codes in this system are packaged by RSIC as CCC-94 through CCC-98.

## 2. CONTRIBUTOR

Westinghouse Astronuclear Laboratory, Pittsburgh, Pennsylvania.

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

## 3. CODING LANGUAGE AND COMPUTER

A revised FLOCO (ODD-K), FORTRAN IV (NAGS), and FORTRAN IV and MAP (DAFT); IBM 7090 and 7094.

## 4. NATURE OF PROBLEM SOLVED

ODD-K solves the two-dimensional Boltzmann equation for  $(r,z)$ ,  $(r,\theta)$ , and  $(x,y)$  geometry. Neutron and photon spatial and energy distributions throughout the reactor system and spatial, energy, and angular distributions of leakage fluxes at the surface can be calculated.

The ODD-K code contains the following additional revisions to the LASL DDK code:

- (1) Initial problem normalization with distributed fixed sources was re-programmed to alleviate neutron (or photon energy) balance difficulty.

- (2) A difficulty in the multiplication operations in the mesh cell angular flux solution was eliminated. This decreased problem running times by at least a factor of two.
- (3) Distributed fixed source input was changed to allow this data input as a single logical record on a FORTRAN IV binary tape.
- (4) Scalar and angular flux data output was changed to generate FORTRAN IV binary tapes for use as input to the NAGS and DAFT programs.
- (5) Parameter and program storage allocation was changed to maximize core memory storage available for input data.
- (6) Modifications were made to allow calculation of various response functions for each mesh cell without lengthy input.

The DAFT program is the data processing routine which prepares angular, spatial, and energy distribution data which may be used as input to the CCC-98/FASTER Monte Carlo program from the surface angular leakage flux data of the two-dimensional transport program ODD-K.

The NAGS program is a series of routines which process multi-group neutron and photon energy fluxes for two-dimensional (R,Z or R, $\theta$ ) geometry models. Problems handled by the NAGS program are: (1) calculation of neutron and photon energy sources and distributions, (2) calculation of neutron and gamma ray dose rates, and (3) calculation of neutron, photon, and total energy deposition data.

## 5. METHOD OF SOLUTION

ODD-K solves the transport equation by the  $S_n$  discrete ordinates method.

The POINT program contains the basic microscopic cross section library and input data processing routines to prepare input for the one-dimensional CCC-95/TAPAT program system, the two-dimensional transport program ODD-K, and the data processing code for ODD-K, NAGS.

The calculation of macroscopic cross sections readily usable in the transport programs, TAPAT and ODD-K, is achieved by employing a

built-in library of region-independent neutron and gamma-ray microscopic cross sections. These group-averaged cross sections are applicable for the analysis of graphite-moderated nuclear reactor rocket systems. The program also calculates response functions as punched card output for use in both the TAPAT and NAGS programs to calculate, for example, neutron and gamma-ray energy deposition, fixed neutron sources, or prompt and secondary gamma-ray sources.

Fluxes input to the NAGS program are obtained from the ODD-K two dimensional transport program. Additional input data needed for the NAGS program is prepared automatically by the POINT program.

The four basic operations of the NAGS program are:

- (1) normalization of multigroup neutron and photon energy fluxes to units of per fission, per watt, or a selected power level.
- (2) redefinition of the mesh cell description for use in photon problems.
- (3) calculation of photon and neutron sources and distribution and heating rates for each region in the problem.
- (4) calculation of the total photon and neutron sources and fissions in the reactor.

The DAFT program reduces the multigroup angular fluxes at the surface (bottom, top, and lateral surfaces of an r,z reactor geometry) into separable, spatial, angular, and energy data for a limited number of surface areas. This program can coalesce the surface flux data from an ODD-K problem into as few as three surface areas (bottom, side, top) or into as many as the ODD-K problem had for the outer boundary mesh intervals, representing the bottom, top, and lateral surfaces of the reactor geometry. In addition, the multigroup representation can be coalesced into as few as one group of data or as many as the neutron (16) or photon (13) energy groups in the ODD-K problem.

## 6. RESTRICTIONS OR LIMITATIONS

Scattering approximation  $\leq P_1$ .

## 7. TYPICAL RUNNING TIME

Timing of the sample problem through each program is as follows:

POINT, 4 minutes,  
 NAGS, 6 1/2 minutes,  
 DAFT, not timed, and  
 ODD-K, 2 hours, 15 minutes.

## 8. COMPUTER HARDWARE REQUIREMENTS

A 32K IBM 7090 or 7094 is required. A maximum of 8 tape units is required.

## 9. COMPUTER SOFTWARE REQUIREMENTS

ODD-K is operable by indirect system control of the independent IBM FORTRAN II Monitor System using a revised FLOCO assembly program, FLOCOW-II.

POINT, NAGS, and DAFT are operable on the IBM 7090/7094 IBSYS Operating System using IBJOB Processor.

## 10. REFERENCES

R. G. Soltesz and R. K. Disney, \*"The POINT Program and Cross Section Library," WANL-PR-(LL)-010, Volume 2 (December 1967).

R. K. Disney, R. G. Soltesz, and S. L. Ziegler, \*"NAGS, A FORTRAN IV Data Processing Program for Calculation of Neutron and Gamma Ray Sources and Neutron and Gamma Ray Heating in Two Dimensional Geometries," WANL-PR-(LL)-010, Volume 7 (December 1967).

R. K. Disney and S. L. Zeigler, \*"DAFT, ODD-K Angular Flux Tapes Program," WANL-PR-(LL)-010, Volume 8 (December 1967).

R. K. Disney, \*"ODD-K, A Two-Dimensional Transport Code," WANL-PR-(LL)-010, Volume 6 (December 1967).

M. A. Capo, \*"Basic Geometry for the Nuclear Rocket Reactor Sample Problem," WANL-PR-(LL)-014, Volume 1 (December 1967).

R. G. Soltesz, \*"Sample Nuclear Rocket Reactor Problem for the Point Program," WANL-PR-(LL)-014, Volume 2 (December 1967).

R. K. Disney, \*"Sample Nuclear Rocket Reactor Problem for the ODD-K Two-Dimensional Transport Code," WANL-PR-(LL)-014, Volume 4

(December 1967).

R. K. Disney, \*"Sample Nuclear Rocket Reactor Problem for the NAGS Program," WANL-PR-(LL)-014, Volume 5 (December 1967).

R. K. Disney, \*"Sample Nuclear Rocket Reactor Problem for the DAFT Program," WANL-PR-(LL)-014, Volume 6 (December 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 14 files: all BCD source card decks, binary card decks, BCD input for sample problems, and BCD output listings for 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

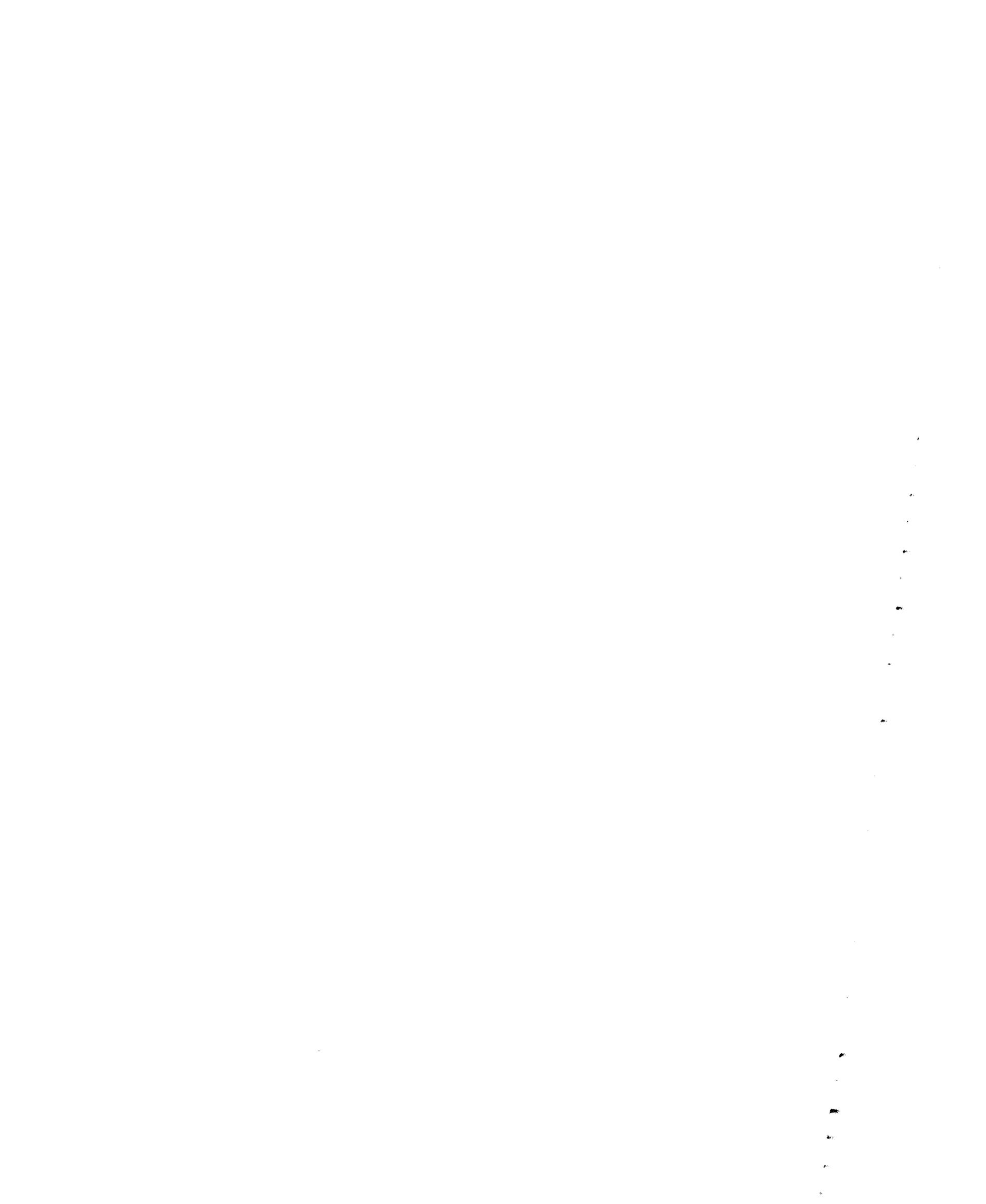
or telephoned to

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.

13. DATE OF ABSTRACT

May 1969.



## RSIC CODE PACKAGE CCC-98

## 1. NAME AND TITLE OF CODE

FASTER: Monte Carlo Transport Code in Complex Geometry.

FASTER is part of a system of codes for shield design called "Synthesis of Computational Methods for the Design and Analysis of Radiation Shields for Nuclear Rocket Systems,"\*. The codes in this system are packaged by RSIC as CCC-94 through CCC-98. CCC-97 and CCC-98 represent what the originators termed the "final" design method.

## 2. CONTRIBUTORS

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.

## 3. CODING LANGUAGE AND COMPUTER

FORTRAN IV

CCC-98A: IBM 7090 and 7094 (December 1967).

CCC-98B: IBM 360/75 (Boeing version, October 1968).

CCC-98C: IBM 360/75 (AGC modification - leakage tally added).

CCC-98D: UNIVAC 1108.

## 4. NATURE OF PROBLEM SOLVED

Energy dependent neutron or photon fluxes are calculated at points, surfaces, and regions of complex geometries. The general quadric surface equation describes the geometry. Common equations for planes, cones, elliptical cylinders and ellipsoids can also be used as input description of the surfaces. The source is described in rectangular, cylindrical, or spherical coordinates. The spatial, angular, and energy source description is assumed to be separable.

## 5. METHOD OF SOLUTION

FASTER is a Monte Carlo code which treats the entire spectrum of particle energies simultaneously which eliminates much of the usual repetitive geometric computations resulting in significant computer time savings. Sampling all possible scattering energies for each collision and extensive importance sampling reduces the variance. For neutron transport, using group cross sections results in additional time savings.

Distinctive features of the Monte Carlo method as employed in the FASTER program include:

- (1) application of random sampling to the spatial and angular integrations only,
- (2) consistent use of energy-averaged sampling functions,
- (3) approximation of importance functions by point kernel techniques,
- (4) analytic treatment of the energy variable over its entire range, and
- (5) zero variance energy integration of the scattered source equations.

## 6. RESTRICTIONS OR LIMITATIONS

None noted.

## 7. TYPICAL RUNNING TIME

Estimated running time of packaged sample problems:

- A: 24 minutes, IBM 7090.
- B: 7 minutes, IBM 360/75.
- C: 8 minutes, IBM 360/75.
- D: not tested by RSIC.

## 8. COMPUTER HARDWARE REQUIREMENTS

At least 32K memory is required and two tape units.

## 9. COMPUTER SOFTWARE REQUIREMENTS

Standard FORTRAN IV compilers will be sufficient for all versions.

## 10. REFERENCES

T. M. Jordan, \*"FASTER, A FORTRAN Analytic Solution of the Transport Equations by Random Sampling," WANL-PR-(LL)-010, Volume 9 (December 1967).

M. A. Capo, \*"Basic Geometry for the Nuclear Rocket Reactor Sample Problem," WANL-PR(LL)-014, Volume 1 (December 1967).

M. A. Capo and T. M. Jordan, \*"Sample Nuclear Rocket Reactor Problem for the FASTER Code," WANL-PR(LL)-014, Volume 8 (December 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 separate files: the BCD source card decks, BCD input for a sample problem, and a BCD output listing of the 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

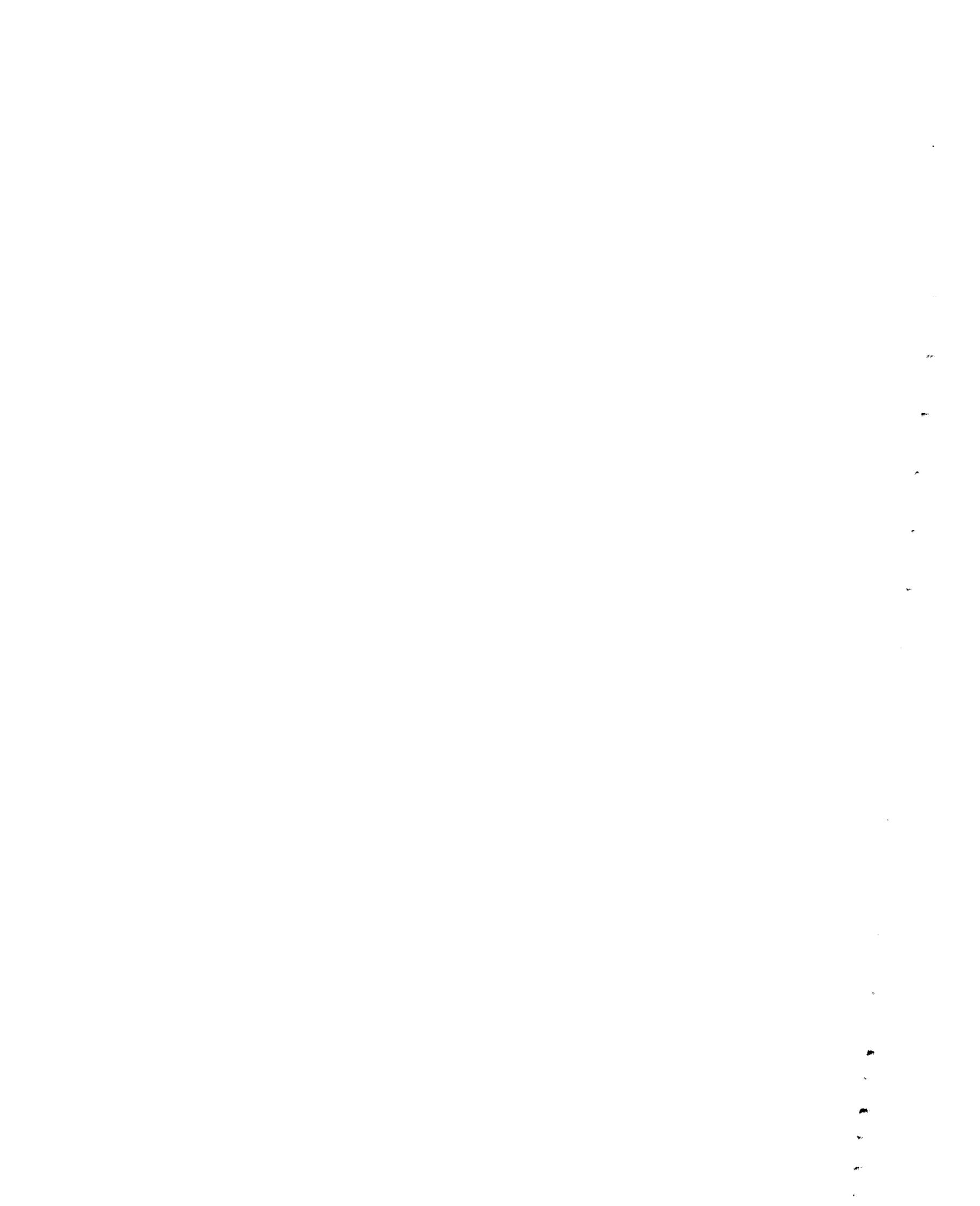
or telephoned to

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

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

## 13. DATE OF ABSTRACT

May 1969.



## RSIC CODE PACKAGE CCC-99

## 1. NAME AND TITLE OF CODE

PLUME: Gamma-Ray Dose Rate from a Radioactive Cloud -  
Kernel Integration Code.

## 2. CONTRIBUTOR

Operations Division, Oak Ridge National Laboratory, Oak Ridge,  
Tennessee.

## 3. CODING LANGUAGE AND COMPUTER

FORTRAN IV; IBM 360/75.

## 4. NATURE OF PROBLEM SOLVED

Estimates are obtained of certain internal and external radiation doses delivered to receptors who are exposed to the cloud of radioactive effluent for various periods of time and at various locations relative to the point of emission (the stack at site of a reactor accident). The internal dose considered is the thyroid dose due to the inhalation of iodine. The external doses are upper limits to the whole body doses to be expected from iodine and noble gases floating downward from the reactor site.

## 5. METHOD OF SOLUTION

It is assumed that a quantity of volatile fission products is released into a reactor containment volume via a reactor accident, is mixed instantaneously with the building air, and the homogeneous mixture is emitted to the atmosphere at a constant rate through a stack. The concentration of radioisotopes downwind is expressed in terms of the product of the "stack factor", which contains information on atmospheric dispersion, and a "source term" containing information on the rate of emission, decay, and dilution within the building. The atmospheric dispersion is estimated by the Gaussian Plume Formula corrected for ground reflection and the presence of a thermal inversion layer. Meteorological data is

built into the program which allows the plume description for various stability conditions and wind velocities. The source terms available in the code allow for treatment of iodine isotopes (neglecting iodine daughters), treatment of noble gases with solid daughters removed by filtering, and treatment of noble gas daughters of iodine parents removed by filtering.

The internal dose is taken as that due to inhalation of iodine parents only and involves the evaluation of an expression which is the time integral of the concentration multiplied by appropriate factors. The external dose requires an integration of the time-integrated concentrations over the plume volume, with appropriate gamma-ray buildup factors being used. The program uses a Gauss-Hermite quadrature scheme on the y-integration and a Gauss-Legendre quadrature scheme on the x- and y-integrations.

#### 6. RESTRICTIONS OR LIMITATIONS

The upper limit on the x-integration (direction of wind) should not exceed 20,000 meters because of the built-in meteorological data. This can be overcome by inserting additional data into the meteorological tables.

#### 7. TYPICAL RUNNING TIME

Estimated running time of the packaged sample problem: 17 minutes.

#### 8. COMPUTER HARDWARE REQUIREMENTS

The code was designed for the IBM 360/75 computer with standard I-O.

#### 9. COMPUTER SOFTWARE REQUIREMENTS

The code is operable on the IBM-360/75 Operating System using OS-360 Fortran H Compiler.

#### 10. REFERENCE

F. T. Binford, J. Barish, F. B. K. Kam, "Estimation of Radiation Doses Following a Reactor Accident," ORNL-4086 (February 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 3 files:  
the source card deck, input for a sample problem, and  
an output listing of 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

May 1969.



## RSIC CODE PACKAGE CCC-100

## 1. NAME AND TITLE OF CODE

K019 : Shield Thickness Calculation Program for Space Vehicles.

## 2. CONTRIBUTOR

NASA Manned Spacecraft Center, Houston, Texas.

## 3. CODING LANGUAGE AND COMPUTER

FORTRAN IV; IBM 360/75.

## 4. NATURE OF PROBLEM SOLVED

Shield thickness information is computed for use as input to codes which compute proton doses at points interior to an occupied spacecraft with complex shield geometry. For each detection point considered, the shield thicknesses computed are along rays emanating outward from the point, so that, in effect, all possible directions from which penetrating radiation could come are considered.

## 5. METHOD OF SOLUTION

Detection points within a complex structure are examined to find the thickness of shielding material present along rays directed from each point to the exterior of the structure. The ray directions for each point are such that they intersect the center equal areas on a unit sphere with origin at the point. For this purpose, the area of the unit sphere may be divided into 80, 320, 720, or 1280 smaller, non-overlapping areas of equal size. The complex structure may have shielding geometry described by combinations of hexahedrons, cylinders, spheres, hemispheres, and cones. Provision is also allowed for the presence of up to six phantom men in the structure, with arms and legs of each phantom in arbitrary positions. The detection points can be arbitrarily located within the structure, even inside a shield or phantom component. The thickness of shield material (in inches) along each ray is determined and expressed in terms of thickness (in  $\text{g}/\text{cm}^2$ ) of some standard material such as aluminum. The total

equivalent shield thickness ( $\text{g}/\text{cm}^2$ ) along each ray is the output of the program. This information is output for each detector point considered.

#### 6. RESTRICTIONS OR LIMITATIONS

Maximum number of materials	152
Maximum number of dose points	50
Maximum number of dose points in cylindrical or hemispherical components of a phantom	5
Maximum number of dose points in hexahedron components of a phantom	5

#### 7. TYPICAL RUNNING TIME

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

#### 8. COMPUTER HARDWARE REQUIREMENTS

The code was originally designed for the IBM 7090, then was re-programmed and has been run on the UNIVAC 1107 and the IBM 360/75. Packaged version is for the latter computer using standard I-O and 2 pool tapes or direct access devices.

#### 9. COMPUTER SOFTWARE REQUIREMENTS

The code is operable on the IBM-360/75 Operating System using OS-360 Fortran H Compiler.

#### 10. REFERENCE

C. Forrest Malone, "Program K019, Shield Thickness Calculation Program - Volume I, Computer Users Manual," NASA-MS-3066 (May 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 3 files:

the BCD source card deck, BCD input for a sample problem and the 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  
Post Office Box X  
Oak Ridge, Tennessee 37830

or telephoned to

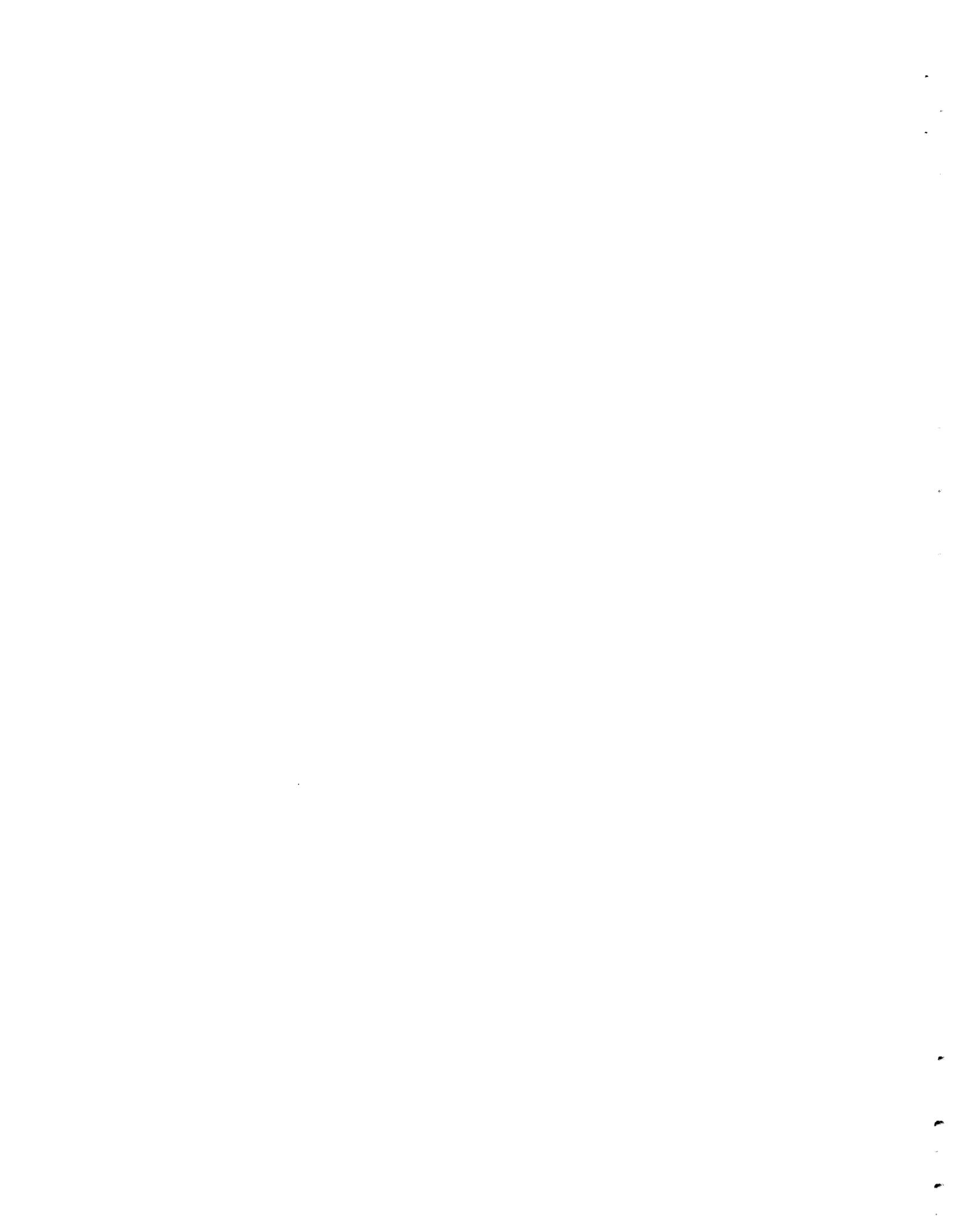
Area Code 615; 483-8611, extension 3-6944, or to

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

May 1969.



## RSIC CODE PACKAGE CCC-101

## 1. NAME AND TITLE OF CODE

NAP: Multigroup Time-Dependent Neutron Activation Prediction Code.

## 2. CONTRIBUTORS

IIT Research Institute, Chicago, Illinois.

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

Neutron induced gamma-ray activities are computed for use in transport codes. Activation calculations can be made for structural materials, reactor coolants, or any material exposed to a neutron flux. Results are given in terms of neutron flux, decay chain atom densities, photon emission rates, and energy. The code may also be used in the interpretation of activation data such as neutron spectra measurements or isotopic analysis.

## 5. METHOD OF SOLUTION

The neutron flux is evaluated by an optional one-dimensional multigroup discrete ordinates calculation which should be performed for problems of significant spatial variation. Isotropic scattering and scattering to the next lower group only are assumed. The incident neutron flux as a function of group is a function of time through a time-dependent power level.

Decay chains for  $(n,\gamma)$ ,  $(n,p)$ ,  $(n,\alpha)$ , and  $(n,2n)$  reactions are determined in succession. The NAP Gamma Radiation Library supplies the data of the radioisotope decay chains and modes of decay. Data for more than 800 isotopes are present.

Resonance self-shielding is accounted for by computation of effective resonance integrals using the NR or NR1A approximations.

Cross sections are obtained from the NAP Library, supplied by the user or calculated, if desired, by known systematics for the thermal region, by statistical resonance theory for the epithermal region, and by compound nucleus formation theory for (n,p), (n, $\alpha$ ), and (n,2n) reactions.

#### 6. RESTRICTIONS OR LIMITATIONS

Time steps of power	$\leq$	50
Time steps of activities	$\leq$	200
Decay chain steps, maximum of	$\leq$	4 (or until stable)
Energy groups	$\leq$	43
Spatial regions	$\leq$	20
Quadrature	$\leq$	10
Mesh points	$\leq$	100
Number of resolved resonances treated	$\leq$	9

#### 7. TYPICAL RUNNING TIME

Estimated running time of packaged sample problem: 1.5 minutes each.

#### 8. COMPUTER HARDWARE REQUIREMENTS

The code was designed for the 32K IBM 7094 with standard I-O and 4 tape units.

#### 9. COMPUTER SOFTWARE REQUIREMENTS

The code is operable on the IBM 7090/7094 IBSYS Operating System using IJOB Processor. The cross sections and gamma radiation libraries are input as binary files.

## 10. REFERENCES

David A. Klopp, "Prediction of Neutron Induced Activation - Volume I - NAP Code Manual," IITRI-A 6088-21 (January 1966).

David A. Klopp, "Prediction of Neutron Induced Activation - Volume II - NAP: Physical Models and Experimental Validation," IITRI-A6088-22.

## 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 8 files: the BCD source card deck, BCD and binary cross section libraries, BCD input for two sample problems, and BCD output listings from 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.

## 13. DATE OF ABSTRACT

May 1969.



## RSIC CODE PACKAGE CCC-102

## 1. NAME AND TITLE OF CODE

SURF: Conical and Plane Surface Single Scattering Code.

## 2. CONTRIBUTOR

Nuclear Materials and Propulsion Operation, General Electric Company, Cincinnati, Ohio.

## 3. CODING LANGUAGE AND COMPUTER

FORTRAN IV; GE-625 and IBM 360/75.

## 4. NATURE OF PROBLEM SOLVED

The code calculates dose rates and flux spectra at unshielded detection points. Radiation at the detection points is composed of direct and once-scattered contributions. The latter component is that scattered from a homogeneous conical shell or slab thin enough so that attenuation and higher order scattering can be neglected. The radiation source can be neutrons or gamma rays. For neutrons, elastic and inelastic scattering is treated. For gamma rays, Klein-Nishina scattering is treated.

Detection points can be located arbitrarily with respect to both the scattering structure and the radiation source. The radiation source can be represented as that escaping from a reactor or shield with an arbitrary energy spectrum. In addition, the source energy range can be divided into three subranges, each governed by a different angular distribution, which the user can arbitrarily specify.

## 5. METHOD OF SOLUTION

The source distribution is represented in terms of intensities in discrete energy groups and angle intervals. The scattering structure is divided into discrete volume elements. Each volume element contributes to the flux in a certain energy group at a detection point by means of scattering radiation incident on the volume element from a source point. The flux spectrum of scattered

radiation at each detection point is then calculated by summing scattering contributions from all volume elements and from all source points. The direct contributions, computed in a straightforward manner, are also added. The dose rate is then computed by multiplying the flux in each energy group by an appropriate conversion factor and summing the results. The user specifies, as input, source point coordinates and intensities for the source energy and angle intervals. Also, the structure configuration and volume element division instructions must be provided.

Neutron elastic scattering is treated as being isotropic in the center-of-mass system, and inelastic scattering is treated as being isotropic in the lab system. The user provides appropriate cross sections. In addition, energy transfer probabilities for inelastic scattering must be provided.

Photon scattering is automatically handled by means of the Compton and Klein-Nishina equations.

#### 6. RESTRICTIONS OR LIMITATIONS

Maximum number of point sources	50
Maximum number of energy groups	30
Maximum number of materials	6
Maximum number of detection points	50
Maximum number of source polar angles for discrete intervals	12
Maximum number of source azimuthal angles for discrete intervals	12
Maximum number of scattering angles for discrete intervals	12

#### 7. TYPICAL RUNNING TIME

Estimated running time of packaged sample problem on the IBM 360, 2 minutes.

## 8. COMPUTER HARDWARE REQUIREMENTS

The code was designed for a 32K GE-625 Computer and was made operable on the IBM 360/75 by RSIC.

## 9. COMPUTER SOFTWARE REQUIREMENTS

The code is operable on the IBM 360/75 Operating System using OS-360 Fortran H Compiler.

## 10. REFERENCE

J. E. MacDonald, "Conical and Plane Surface Scattering Program - SURF," GEMP 582 (February 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 3 files: the source card deck, input for a sample problem, and an 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  
Post Office Box X  
Oak Ridge, Tennessee 37830

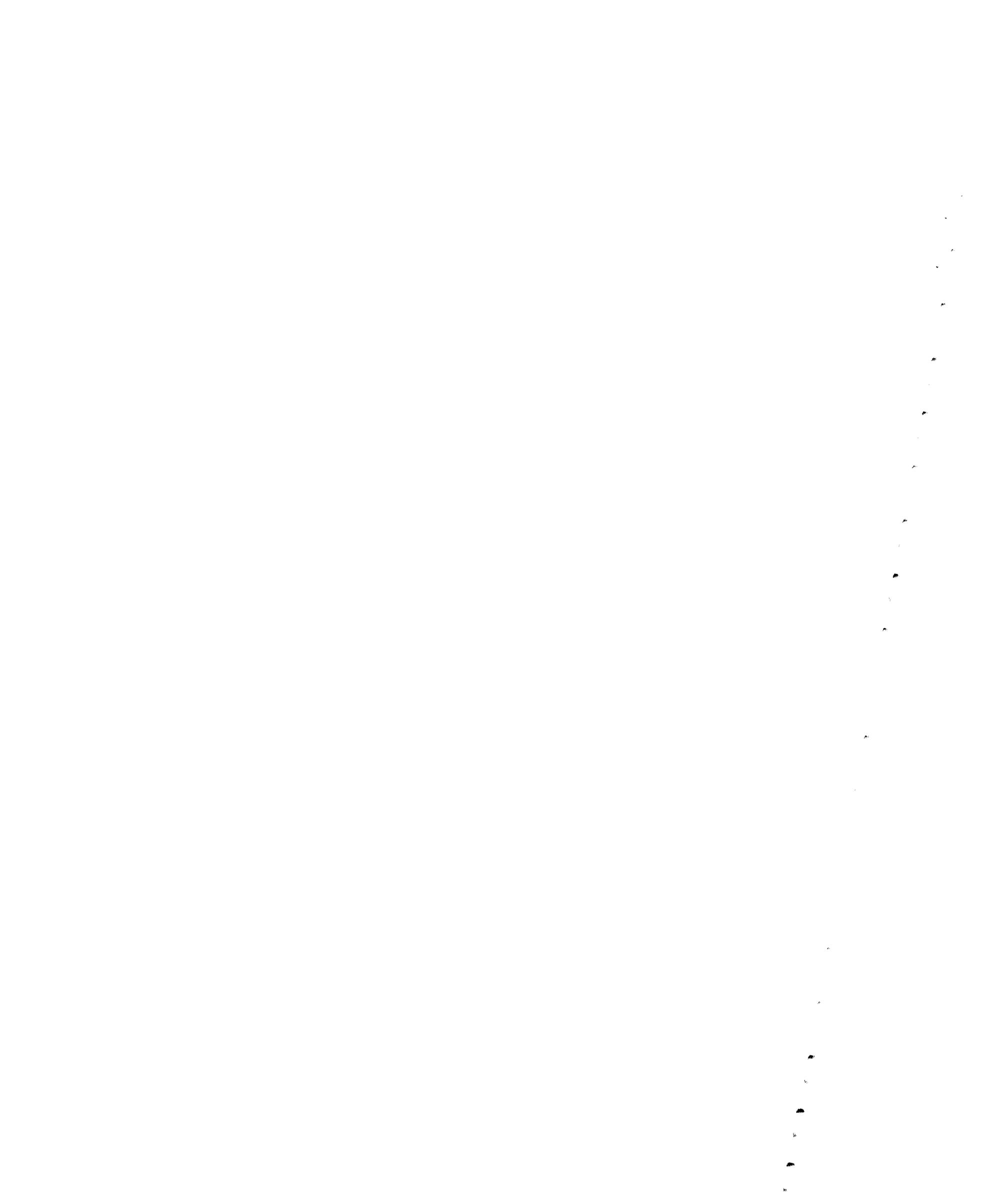
or telephoned to

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

May 1969.



## RSIC CODE PACKAGE CCC-103

## 1. NAME AND TITLE OF CODE

OPEX-II: Radiation Shield Optimization Code.

## 2. CONTRIBUTOR

National Aeronautics and Space Administration, Lewis Research Center, Cleveland, Ohio.

## 3. CODING LANGUAGE AND COMPUTER

FORTRAN IV; IBM 7090/7094.

## 4. NATURE OF PROBLEM SOLVED

OPEX-II will determine the minimum weight configuration for spherical laminated shields subject to the constraint of a constant total dose rate at a selected detector point. Once an optimized base case is obtained, effects of nominal changes in reactor size, power level, and dose constraint on shield weight may be estimated. Cost optimization may also be performed by specifying cost per unit volume rather than density for each region.

## 5. METHOD OF SOLUTION

The code is an extension of CCC-63/OPEX. The method of steepest descent has been maintained but the code has been completely rewritten to simplify data input, incorporate spherical geometry, expand the output, alter the dose-thickness relation, and effect other improvements. The dose rate is approximated by a sum of exponentials representing each dose component obtained from experiments or analytical calculations.

## 6. RESTRICTIONS OR LIMITATIONS

Number of regions:  $\leq 25$   
Number of terms in expression for dose rate:  $\leq 25$

## 7. TYPICAL RUNNING TIME

Estimated running time of the packaged sample problem on the IBM-7090, approximately 1 minute.

## 8. COMPUTER HARDWARE REQUIREMENTS

The code was designed for the IBM 7094 and uses standard I-0.

## 9. COMPUTER SOFTWARE REQUIREMENTS

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

## 10. REFERENCE

G. P. Lahti, "OPEX-II, A Radiation Shield Optimization Code," NASA TM X-1769 (March 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 several files: source card deck, input for a sample problem, 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.

## 13. DATE OF ABSTRACT

May 1969.

## RSIC CODE PACKAGE CCC-104

## 1. NAME AND TITLE OF CODE

EDNA: Electron Dose and Number Analysis Code by Kernel Integration.

## 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

The electron number flux and energy flux deposition rates are calculated as a function of depth in an aluminum slab. The impinging electron source can be normally incident or have either an isotropic or cosine angular distribution with respect to current. The energy spectrum of the incident electrons is arbitrary but is assumed to vary as  $Ae^{-BE}$  between input data values.

## 5. METHOD OF SOLUTION

The electron energy flux deposition rate at depth  $z$  in an aluminum slab is computed by integrating, over energy, the product of the incident energy flux spectrum and the probability density for energy flux deposition at that depth. The integration is performed by evaluating the integrand at the midpoint of each integration step, multiplying by the step width, and summing the result. For this evaluation, the incident spectrum is assumed to vary as  $Ae^{-BE}$  between input data points. The probability densities for energy flux depositions which are used in EDNA are those given in Ref. 1. These were derived from analyses of Monte Carlo results by Berger and Seltzer, for electron transmission and reflection factors. They are in the form of analytic expressions. The coefficients for the ex-

pressions are given in Ref. 1, and the user must provide appropriate values for these as input to EDNA.

An analogous treatment is used for computing the electron number flux deposition rate.

The energy range may be divided into several intervals, each with a different number of integration steps. The energy spectrum of incident particles is specified in input by a set of input energy and flux values.

#### 6. RESTRICTIONS OR LIMITATIONS

Some dimensional limitations are as follows:

Maximum number of values describing incident spectrum:	40
Maximum number of energy intervals for which a different number of integration steps is provided:	10

#### 7. TYPICAL RUNNING TIME

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

#### 8. COMPUTER HARDWARE REQUIREMENTS

The code was designed for the IBM 1130. It is also operable on the IBM 7090 with standard I-O units.

#### 9. COMPUTER SOFTWARE REQUIREMENTS

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

#### 10. REFERENCES

Charles E. Wuller, Jr., "Electron Transport from a Cosine Law Source," IN-SSL-N-68-13, Internal Note at Marshall Space Flight Center, Huntsville, Alabama, (September 1968).

Martin O. Burrell, J. J. Wright, and John W. Watts, Jr., "The Calculation of Electron and Bremsstrahlung Dose Rates," Protection Against Space Radiation, NASA SP-169, p. 529 (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 on 4 files:  
the source card deck, BCD input for a sample problem, the  
binary card deck, 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

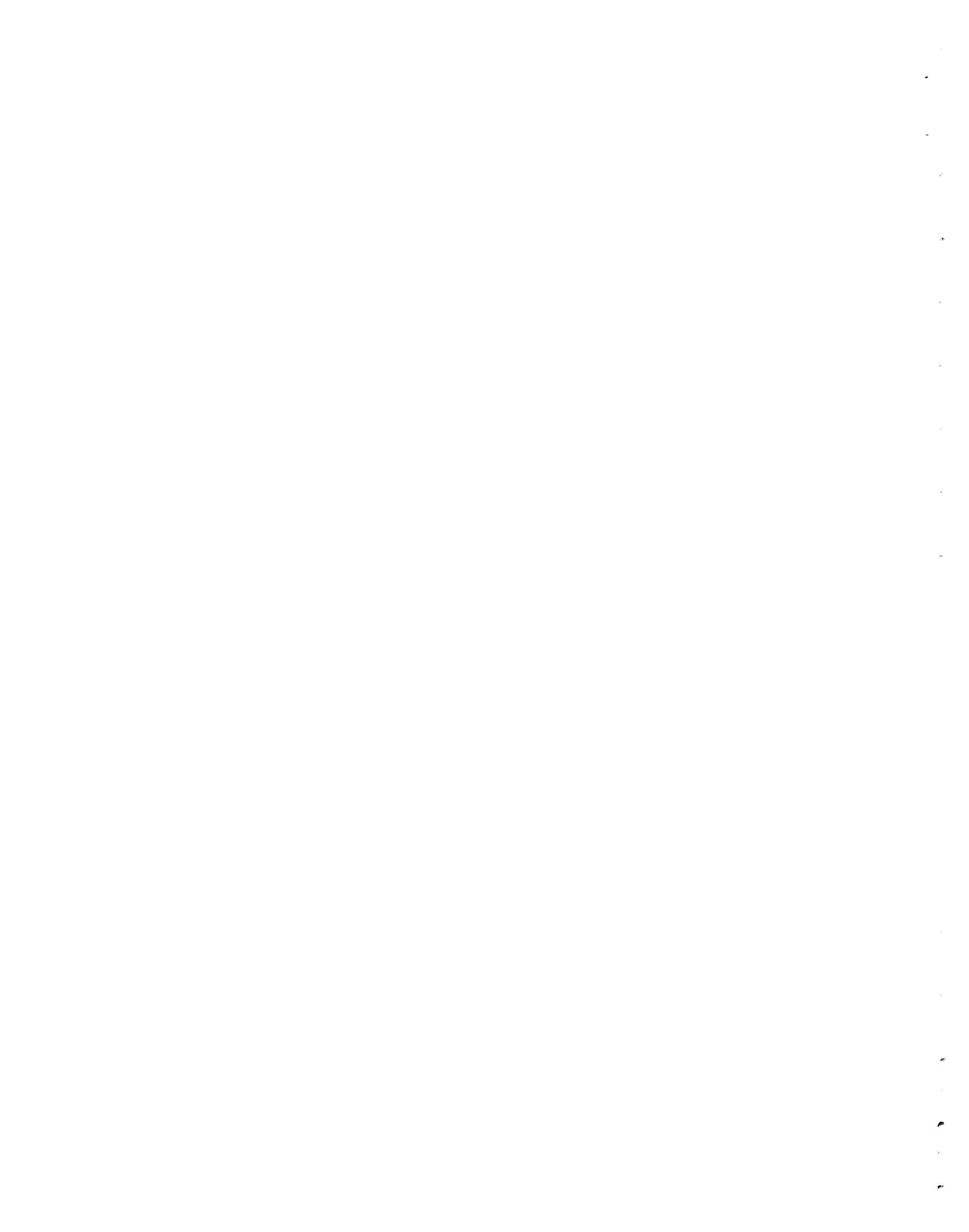
or telephoned to

Area Code 614; 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

May 1969.



## RSIC CODE PACKAGE CCC-105

## 1. NAME AND TITLE OF CODE

RDMM: Neutron Spectra Determination by Activation Detector Analysis.

## 2. CONTRIBUTOR

EURATOM Joint Nuclear Research Center, Ispra Establishment, Italy.

## 3. CODING LANGUAGE AND COMPUTER

FORTRAN IV; CCC-105A IBM 7090 and 7094, CCC-105B IBM 360/75.

## 4. NATURE OF PROBLEM SOLVED

Differential fast neutron spectra are calculated from experimental threshold foil activation detector data.

## 5. METHOD OF SOLUTION

By a least-squares technique, or Relative Deviation Minimization Method (RDMM), the square of the difference between the measured activation rate and energy integral of the calculated reaction rate (cross section times flux density) is minimized. The flux density is taken to be a weighting function multiplied by a sum of a system of functions which are polynomials in energy.

A procedure, based on the Monte Carlo method, is used to statistically study the effect of experimental input errors on the solution obtained.

## 6. RESTRICTIONS OR LIMITATIONS

N-point Newton-Cotes Integration  $2 \leq N \leq 7$

## 7. TYPICAL RUNNING TIME

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

## 8. COMPUTER HARDWARE REQUIREMENTS

At least a 32K memory is required and standard I-0.

## 9. COMPUTER SOFTWARE REQUIREMENTS

CCC-105A: The code is operable on the IBM 7090/7094 IBSYS Operating System using IBJOB Processor.

CCC-105B: The code is operable on the IBM 360/75 Operating System using OS-360 Fortran H Compiler.

A random number generator routine is included in each version.

## 10. REFERENCES

G. Di Cola and A. Rota, "RDMM - A Code for Fast Neutron Spectra Determination by Activation Analysis," EUR 2985.e (1966).

G. Di Cola and A. Rota, "Calculation of Differential Fast-Neutron Spectra from Threshold-Foil Activation Data by Least-Squares Series Expansion Methods," Nuc. Sci. Eng. 23, 344-353 (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 separate files: source deck, input for a sample problem, and the output listing from 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

May 1969.

## RSIC CODE PACKAGE CCC-106

## 1. NAME AND TITLE OF CODE

PF-COMP: Building Fallout Radiation Protection Factor Analysis.

## 2. CONTRIBUTORS

Research Triangle Institute, Raleigh, North Carolina.

U. S. Army Office of Civil Defense, Washington, D. C.

## 3. CODING LANGUAGE AND COMPUTER

FORTRAN 64; CDC 3600.

## 4. NATURE OF PROBLEM SOLVED

The PF-COMP program performs analysis of structures to determine the protection offered against fallout gamma radiation. It was designed for use, primarily, in such programs as the Military Overseas Shelter Survey, the Evaluation of Fallout Protection in Homes, and continuing National Fallout Shelter Surveys.

## 5. METHOD OF SOLUTION

A building protection factor (PF) calculational procedure is employed. The code is an implementation of the "Engineering Manual Method" of fallout analysis. The procedure considers contributions from the roof, setbacks, and limited planes of contamination (including areaways). The effects of apertures, interior partitions, floors, detector height above planes of contamination, mutual shielding, and building geometry are included.

This program is capable of determining the PF in the center of a building part, at eight additional predetermined locations, and at one additional arbitrary location. On the basis of the PF's at these points, the approximate area of the building part having a given protection factor is determined.

## 6. RESTRICTIONS OR LIMITATIONS

There are no apparent restrictions.

## 7. TYPICAL RUNNING TIME

No statistics have been accumulated by RSIC as to typical running time.

Estimated running time on the CDC 3600 for the 34 packaged sample problems: 48 minutes.

## 8. COMPUTER HARDWARE REQUIREMENTS

The code was designed for a CDC 3600 using standard I-O and a maximum of four tape units.

## 9. COMPUTER SOFTWARE REQUIREMENTS

The code is operable on the CDC 3600 scope system. A RTI table tape is required as input. MOSS data input by tape is optional.

## 10. REFERENCES

## a. Background material:

L. V. Spencer, "Structure Shielding Against Fallout Radiation from Nuclear Weapons," NBS Monograph 42 (1962).

## b. Code package material:

E. L. Hill, F. Johnson, and R. O. Syday, Jr., "Computer Program for Analysis of Building Protection Factors," Research Triangle Institute Research Memorandum RM-205-1, Part II (July 1965).

E. L. Hill, F. Johnson, and R. O. Syday, Jr., "Summary of a Computer Program for Analysis of Building Protection Factors," OCD Working Paper (March 1965).

M. D. Wright and R. O. Syday, Jr., "PF-COMP Computer Program Test Building," OCD Working Paper (January 1966).

OCD Publication, "Design and Review of Structures for Protection from Fallout Gamma Radiation," PM-100-1 (February 1965).

OCD Publication, "Military Overseas Shelter Survey, Phase I, Data Collection Form Instructions," Prepared by Research Triangle Institute (October 1965).

OCD Publication, "Military Overseas Shelter Survey, Phase 2, Data Collection Form Instructions," Prepared by Research Triangle Institute

(October 1965).

C. Eisenhauer, "An Engineering Method for Calculating Protection Afforded by Structures Against Fallout Radiation," PM-100-1 Supplement (January 1964).

11. CONTENTS OF CODE PACKAGE

The package contains the following items:

- a. the referenced documents, section b,
- b. a reel of magnetic tape on which is written in several files: the BCD source card deck, RTI library tables, input for 34 sample problems, Edited Input Data and Error Table, and output listings 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 code package should send a reel of magnetic tape to the above address.

13. DATE OF ABSTRACT

May 1969.



## RSIC CODE PACKAGE CCC-107

## 1. NAME AND TITLE OF CODE

ETRAN: Monte Carlo Code System for Electron and Photon Transport Through Slabs.

## AUXILIARY ROUTINES

DATAPAC 4: Data generation - single-scattering and multiple-scattering for a particular material from a master library.

## 2. CONTRIBUTOR

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

## 3. CODING LANGUAGE AND COMPUTER

FORTRAN IV & V; CCC-107A/ETRAN 15 - UNIVAC 1108.

FORTRAN IV; CCC-107B/ETRAN 16 - IBM 360/75, CCC-107C/ETRAN 16B - IBM 360/75.

## 4. NATURE OF PROBLEM SOLVED

ETRAN computes the transport of electrons and photons through plane-parallel slab targets that have a finite thickness in one dimension and are unbound in the other two dimensions.

The incident radiation can consist of a beam of either electrons or photons with specified spectral and directional distribution.

Options are available by which all orders of the electron-photon cascade can be included in the calculation. Thus electrons are allowed to give rise to secondary knock-on electrons, continuous bremsstrahlung and characteristic X-rays; and photons are allowed to produce photo-electrons, Compton electrons, and electron-positron pairs. Annihilation quanta, fluorescence radiation, and Auger electrons are also taken into account. If desired, the Monte Carlo histories of all generations of secondary radiations are followed.

The information produced by ETRAN includes the following items: (1) the reflection and transmission of electrons or photons,

differential in energy and direction; (2) the production of continuous bremsstrahlung and characteristic X-rays by electrons and the emergence of such radiations from the target (differential in photon energy and direction); (3) the spectrum of the amounts of energy left behind in a thick target by an incident electron beam; (4) the deposition of energy and charge by an electron beam as function of the depth in the target; (5) the flux of electrons, differential in energy, as function of the depth in the target.

#### 5. METHOD OF SOLUTION

A program called DATAPAC 4 takes data for a particular material from a library tape and further processes them. The function of DATAPAC 4 is to produce single-scattering and multiple-scattering data in the form of tabular arrays (again stored on magnetic tape) which facilitate the rapid sampling of electron and photon Monte Carlo histories in ETRAN.

The photon component of the electron-photon cascade is calculated by conventional random sampling that imitates the physical processes of Compton scattering, photon electric absorption, and pair production. In the calculation of the electron component, no attempt is made to follow successive individual interactions with atoms and atomic electrons because these are too numerous. Instead, a Monte Carlo model is used in which attention is focused on the effect of groups of successive collisions.

The electron tracks to be sampled are divided into a large number of short segments, and the energy loss and angular deflection in each segment are sampled from pertinent theoretical multiple scattering distributions. At the end of each short step, the direction of motion of the electron is changed by a multiple scattering angular deflection that is sampled from the Goudsmit-Saunderson distribution. This distribution is assumed to be the same for all short steps lying within a given step. The energy loss in a step, resulting from the cumulative effect of many inelastic collisions, is sampled from a distribution that is a convolution of a Landau distribution with a Gaussian. An option is also provided for using the continuous-

slowing-down approximation in which energy-loss fluctuations are disregarded and the energy loss by collisions is simply computed with the use of the stopping power formula.

The production of knock-on electrons is sampled in each short step with the use of a probability distribution derived from the Moller cross section for collisions between free electrons (binding effects are disregarded). Histories of these particles are then followed by procedures identical with those used for the primary electrons.

The production of continuous bremsstrahlung photons is sampled in each short step with the use of a probability distribution derived from the bremsstrahlung cross section (Bethe-Heitler theory with modifications taking into account the correct high-frequency limit, empirical corrections, etc.). The probability is usually quite small that more than one bremsstrahlung photon will be produced in a single short step. Allowance is made for such a contingency by sampling the frequency of bremsstrahlung production events from a Poisson distribution. The energy of the secondary bremsstrahlung photons is subtracted from the energy of the electrons producing them. Thus photon emission contributes to the energy-loss straggling of the electrons. The photons are started out at a random position in the short step in a direction relative to that of the primary electron specified by the sampled intrinsic bremsstrahlung emission angle. For problems in which the production of thick-target bremsstrahlung is of prime interest, there is an option to increase the rate of occurrence of bremsstrahlung events artificially by a specified factor.

The production of secondary characteristic X-rays in each short step is sampled with the use of the K-ionization cross sections of Arthurs and Moiseiwitsch and Kolbenstvedt.

The program is arranged so as to treat simultaneously many slab targets with different thicknesses.

Boundary crossings (transmission and reflection) usually occur in the middle of a short step. The energy with which the electron crosses the border is determined by subtracting from the energy at the beginning of the step an energy loss sampled from the Landau-

Blunck-Leisegang distribution for the fraction of the step taken to the boundary. The direction at the time of crossing is determined by changing the direction of motion at the beginning of the short step involved, using a deflection sampled from an exponential approximation to the Goudsmit-Saunderson distribution for the fraction of the short step to the boundary.

The target is subdivided into many thin sublayers of equal thickness, and the energy deposited in each sublayer is recorded for each sampled track. The energy allowed to be deposited is that dissipated by electrons in inelastic collisions resulting in the production of slow secondary electrons with energies below the chosen cut-off value. The energy given to fast secondary electrons with energies above the cut-off is not immediately scored, because the histories of these electrons are followed further so that the energy may eventually be deposited in a sublayer different from the one in which the electrons were produced. Bremsstrahlung losses are similarly not scored immediately. Photons are allowed first to penetrate further through the medium so that the energy of the electrons set in motion by them may eventually be deposited in a different sublayer.

The treatment of charge deposition is quite similar to that of energy deposition, involving the scoring of charge deposited in sublayers. A track is assumed to "end" when the residual range of the electron is so small compared to the size of the sublayers that escape to a different sublayer is no longer possible. When secondary electrons are produced, either as the result of a knock-on collision or as the result of Compton scattering or photon-electric absorption, a unit charge is withdrawn from the sublayer in which the electron is born. The charge is then allowed to be carried to a different sublayer. Electron-positron pairs are excluded from this scheme because on the average their production does not lead to a net transfer of charge.

The electron flux is computed in ETRAN as a quantity differential in energy but integrated over all directions.\* A Monte Carlo

estimate of the flux is obtained by dividing the target into many sublayers and scoring the tracklength of electrons with specified energies in each of the sublayers. The average tracklength per incident electron divided by the thickness of the sublayer provides an estimate of the average flux in the sublayer.

The flux calculation includes primary as well as secondary electrons with energies down to some cut-off value which is chosen so that the electron is effectively trapped in the sublayer in which it finds itself, because its residual range is smaller than the distance to the nearest sublayer boundaries.

\* In the IBM 360/75 version, the flux differential in angle is also computed, and a distinction is made between the flux in the forward and backward directions.

#### 6. RESTRICTIONS OR LIMITATIONS

No dimensional limitations are noted.

#### 7. TYPICAL RUNNING TIME

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

DATAPAC 4, 5 minutes

ETRAN 16B, 3 minutes

No available record for other hardware.

#### 8. COMPUTER HARDWARE REQUIREMENTS

The code requires the use of a large computer. It is operable on the UNIVAC 1108 and the IBM 360/75, with standard I-O and two tape units or direct access devices.

#### 9. COMPUTER SOFTWARE REQUIREMENTS

CCC-107A/ETRAN 15 (UNIVAC-1108 version) as distributed by RSIC requires the availability of a Fortran V compiler.

CCC-107B,C/ETRAN 16 and 16B are operable on the IBM 360/75 operating system using OS/360 Fortran H compiler.

## 10. REFERENCES

M. J. Berger and S. M. Seltzer, "Electron and Photon Transport Programs (Program DATAPAC 4)," NBS-9836 (June 1968).

M. J. Berger and S. M. Seltzer, "Electron and Photon Transport Programs (Program ETRAN 15)," NBS-9837 (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 separate files: three files of library data, a source deck for each code, input for a sample problem, 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.

## 13. DATE OF ABSTRACT

May 1969.

## RSIC CODE PACKAGE CCC-108

## 1. NAME AND TITLE OF CODE

SPECTRA: Determination of Neutron Spectra from Activation Measurements.

## 2. CONTRIBUTOR

Sandia Laboratories, Albuquerque, New Mexico.

## 3. CODING LANGUAGE AND COMPUTER

FORTRAN; CCC-108A: CDC 1604, CCC-108B: CDC 3600.

## 4. NATURE OF PROBLEM SOLVED

Neutron energy spectra are determined by an analysis of experimental activation detector data.

## 5. METHOD OF SOLUTION

A technique is used for solving the Fredholm integral equations describing neutron activation spectroscopy measurements. Through the representation of the spectrum and cross sections as piecewise linear functions of energy, the set of activation integral equations is reduced to a matrix equation. By introducing a constraint involving a "best" trial spectrum, it is shown that certain additional physical information usually available to an experimenter can be utilized. Based upon this constraint, a unique solution to the matrix equation can be obtained.

The user of the code has the option of obtaining the closed form exact solution and/or an approximate solution arbitrarily close to the exact solution through an iterative procedure.

The code is accompanied by a cross section library of 28 specific reactions and a complete data handling program which allows the removal, addition, and/or editing of the data on the cross section tape.

## 6. RESTRICTIONS OR LIMITATIONS

Number of energy points  $\leq$  50

## 7. TYPICAL RUNNING TIME

Estimated running time of the sample problem on the CDC 1604

Cross Section Handling: 5 minutes

SPECTRA: 4 minutes

## 8. COMPUTER HARDWARE REQUIREMENTS

At least a 32K memory is required. The code was designed for the CDC 3600. It was converted for use on the CDC 1604 and uses standard I-O and 2 tapes.

## 9. COMPUTER SOFTWARE REQUIREMENTS

CCC-108A: This version is operable on the CDC 1604 using the COOP operating system.

CCC-108B: Untested by RSIC.

## 10. REFERENCE

C. R. Greer, J. A. Halbleib, J. V. Walker, "A Technique for Unfolding Neutron Spectra from Activation Measurements," SC-RR-67-746 (December 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 BCD source card decks, BCD input for a sample problem, and a BCD output listing of 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

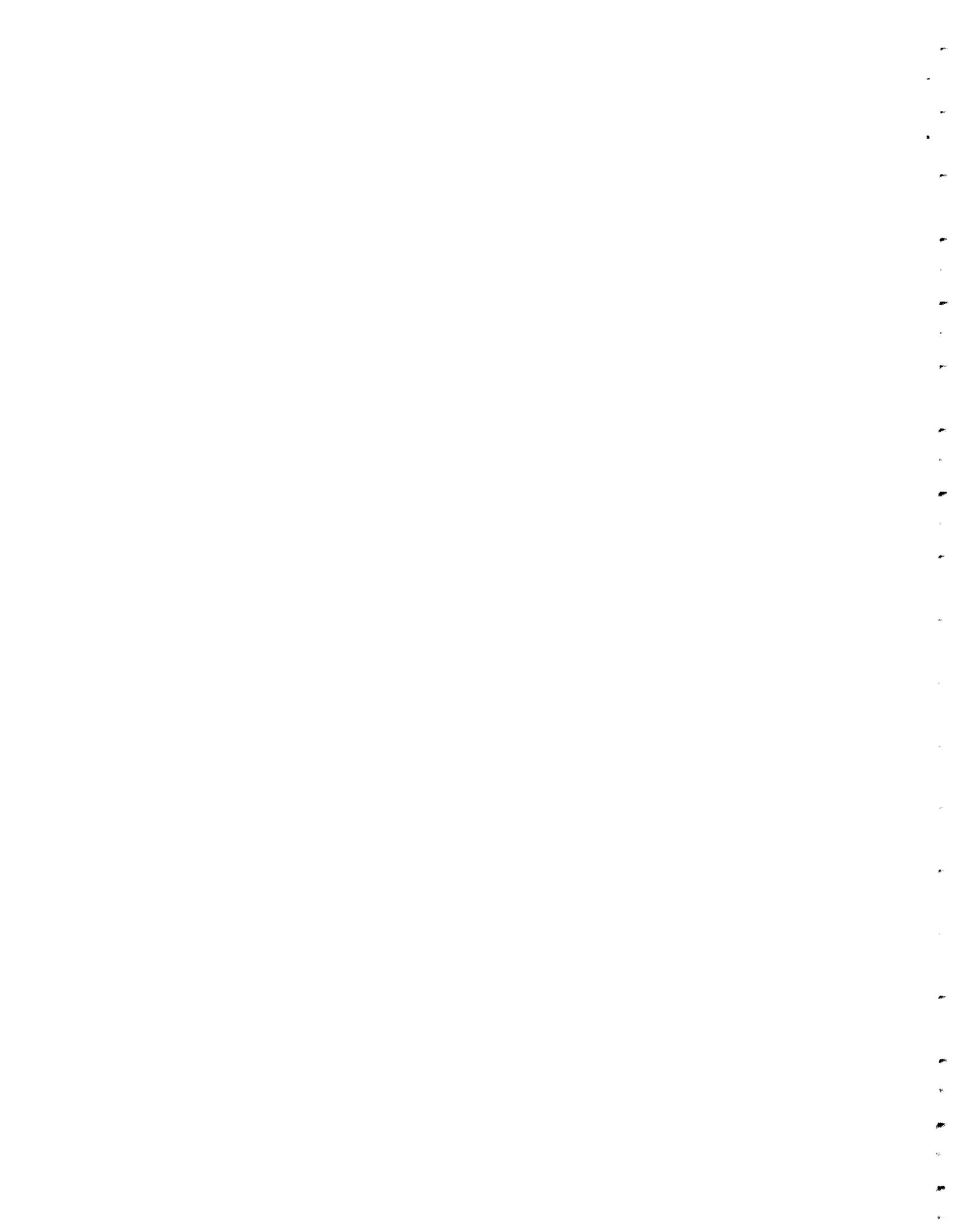
or telephoned to

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

May 1969.



## RSIC CODE PACKAGE CCC-109

1. NAME AND TITLE OF CODE  
SOSUM: Multigroup Beta and Gamma-Ray Energy Sources from Radioisotope Activities.
2. CONTRIBUTOR  
Atomics International, Canoga Park, California.
3. CODING LANGUAGE AND COMPUTER  
FORTRAN IV; IBM 360/75.
4. NATURE OF PROBLEM SOLVED  
Beta and gamma-ray source strengths, in a multigroup energy structure, are computed from isotope radioactivity levels given in terms of disintegration rates.
5. METHOD OF SOLUTION  
Data on the individual photon per disintegration for 181 isotopes are contained in the original library. By combining these data with activity data of a fission product inventory code such as CCC-69/CURIE, the decay energy per each of the specified energy groups is determined. The beta-ray energy is generally placed in one group on the basis of the assumed average energy, which is a function of the maximum energy and atomic number.
6. RESTRICTIONS OR LIMITATIONS  
None noted.
7. TYPICAL RUNNING TIME  
Estimated running time of the packaged sample problem on the IBM 360/75: 0.5 minutes.

8. COMPUTER HARDWARE REQUIREMENTS  
The code was designed for the IBM-360 using standard I-O.
9. COMPUTER SOFTWARE REQUIREMENTS  
The code is operable on the IBM 360/75 operating system using the OS/360 Fortran H compiler.
10. REFERENCE  
B. J. Dray, R. J. Thomson, "SOSUM, A Program to Compute Multi-group Beta and Gamma Energy Sources from Radioisotope Activities," AI-AEC-MEMO-12693 (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 several files: source deck (EBCDIC), library data, input for a sample problem, and output listing for 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  
May 1969.

## RSIC CODE PACKAGE CCC-110

## 1. NAME AND TITLE OF CODE

AIRTRANS: Monte Carlo Time and Energy-Dependent Three-Dimensional Radiation Transport Code.

## AUXILIARY ROUTINES

DATORG: Organized Data Tape Generator.

VANGEN: External Source Generator.

ASP: Anisotropic Secondary Particle Generator.

AIRSCA: Calculates Time-Energy Dependent Fluxes at Point in Atmosphere.

## 2. CONTRIBUTORS

United Nuclear Corporation, Elmsford, New York.

Lockheed Missile and Space Company, Sunnyvale, California.

## 3. CODING LANGUAGE AND COMPUTER

FORTRAN; CCC-110A: CDC-1604, CCC-110B: UNIVAC 1108.

## 4. NATURE OF PROBLEM SOLVED

The function of the AIRTRANS system is to calculate by Monte Carlo methods the radiation field produced by neutron and/or gamma-ray sources which are located in the atmosphere. The radiation field is expressed as the time- and energy-dependent flux at a maximum of 50 point detectors in the atmosphere. The system calculates uncollided fluxes analytically and collided fluxes by the "once-more collided" flux-at-a-point technique.

Energy-dependent response functions can be applied to the fluxes to obtain desired flux functionals, such as doses, at the detector point. AIRTRANS also can be employed to generate sources of secondary gamma radiation.

## 5. METHOD OF SOLUTION

Neutron interactions treated in the calculational scheme include elastic (isotropic and anisotropic) scattering, inelastic (discrete

level and continuum) scattering, and absorption. Charged particle reactions, e.g., (n,p) are treated as absorptions. A built-in kernel option can be employed to take neutrons from the 150 keV to thermal energy, thus eliminating the need for particle tracking in this energy range. Another option used in conjunction with the neutron transport problem creates an "interaction tape" which describes all the collision events that can lead to the production of secondary gamma rays. This interaction tape subsequently can be used to generate a source of secondary gamma rays.

The gamma-ray interactions considered include Compton scattering, pair production, and the photoelectric effect; the latter two processes are treated as absorption events.

Incorporated in the system is an option to use a simple importance sampling technique for detectors that are many mean free paths from the source. In essence, particles which fly far from the source are split into fragments, the degree of fragmentation being proportional to the penetration distance from the source. Each fragment is tracked separately, thus increasing the percentage of computer time spent following particles at the deep penetrations. Each fragment is assigned a "weight" which is inversely proportional to the degree of fragmentation suffered by the original source particle. All estimates of flux contributions by such a fragment are then multiplied by its assigned weight.

#### 6. RESTRICTIONS OR LIMITATIONS

$$N_D = \text{number of DETECTORS} \leq 50^*$$

$$N_E = \text{number of ENERGY BINS} \leq 50^*$$

$$N_T = \text{number of TIME BINS} \leq 50^*$$

\*Subject to  $3 N_D \times N_E \times N_T + 2 N_T \times N_E \leq 1500\text{-ITED}$ , where ITED = 1464 for neutrons and = 1054 for gammas.

#### 7. TYPICAL RUNNING TIME

Estimated running time of the sample problem on the CDC 1604: AIRSCA, 3 minutes; DATORG, 3 minutes. Sample problem does not use

VANGEN or ASP.

Typical running time on the UNIVAC 1108 varies from  $\frac{1}{4}$  to  $\frac{1}{2}$  sec/history/detector.

8. COMPUTER HARDWARE REQUIREMENTS

The code was originally designed for the CDC 1604 using standard I-O and a maximum of 4 tape units. It was converted for use on the UNIVAC 1108 by Lockheed Missile and Space Company.

9. COMPUTER SOFTWARE REQUIREMENTS

CCC-110A: This version is operable on the CDC 1604 using the COOP operating system.

CCC-110B: Untested by RSIC.

10. REFERENCES

G. L. Case, "Utilization Manual - AIRTRANS," LMSC-5234 (February 1968).

M. O. Cohen, "AIRTRANS - A Time-Dependent Monte Carlo System for Radiation Transport in a Variable Density Atmosphere and the Ground," UNC-5179 (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 for CCC-110A, 14 files and for CCC-110B, 6 files: BCD source card decks, BCD cross section library data, BCD input for sample problems and output listings from 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, specifying which code version is desired.

13. DATE OF ABSTRACT

May 1969.

## RSIC CODE PACKAGE CCC-111

## 1. NAME AND TITLE OF CODE

FLORA: Calculation of the Contribution of Fluorescence Radiation.

## 2. CONTRIBUTOR

Douglas Missile and Space Systems Division, Santa Monica, California.

## 3. CODING LANGUAGE AND COMPUTER

FORTRAN IV; UNIVAC 1108, IBM 7090.

## 4. NATURE OF PROBLEM SOLVED

FLORA is a program for calculating contributions to transmission and energy deposition from fluorescence radiation generated in media composed of single or layered slabs by incident photon fluxes. The code considers only the photoelectric effect and is therefore applicable to situations where that process is the dominating absorption mechanism. The incident spectrum can be arbitrarily represented by tabulated input values or can be chosen as a polynomial in energy or a function of the energy raised to an arbitrary power.

## 5. METHOD OF SOLUTION

The source strength of a given line of fluorescence radiation is given by the integral over incident energy of the product of removal rate (by photoelectric effect), fluorescence probability, and the relative importance of the line. The transmission of initial and fluorescence radiation is assumed to be described by exponential attenuation because only the photoelectric process is considered. The fluorescence spectrum across any plane in the slab can be calculated by integrating the source strength over space, while accounting for attenuation. In FLORA, energy integrations are done by

trapezoidal rule and spatial integrations by Simpson's parabolic rule.

The code calculates the uncollided spectrum and energy fluence at each generating plane (a position, specified by the user, at which various quantities are to be calculated). The fluorescence energy fluences, in both forward and backward directions, at each generating plane, are also computed. In addition, the forward tertiary fluence (fluence caused by L-lines forming from K-lines and by M-lines forming from K- and L-lines, etc.) is also computed.

The uncollided, forward fluorescence, backward fluorescence, and tertiary fluences are summed to obtain the total fluence at each generating plane. The uncollided spectrum and the forward and backward fluorescence spectra are summed to obtain the total spectrum at each generating plane. The net shell and total shell fluences are also calculated.

The user must supply as input information describing the shell cross sections, yields, energies, intensities, etc.

## 6. RESTRICTIONS OR LIMITATIONS

The following are dimensional limitations:

Maximum number of generating planes	: 10
Maximum number of shells	: 5
Maximum number of lines for each shell	: 15
Maximum number of incident energy values	: 30

## 7. TYPICAL RUNNING TIME

Estimated running of the packaged sample problem on the IBM 7090: 2 minutes.

## 8. COMPUTER HARDWARE REQUIREMENTS

The code was designed for the UNIVAC 1108 computer and was made operable on the IBM 7090 by RSIC. Standard I-O and 1 pool tape are needed.

## 9. COMPUTER SOFTWARE REQUIREMENTS

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

A special input subroutine (REED) is used with this code, and is included in the code package.

## 10. REFERENCE

B. Barnett, J. R. Hobart, J. A. Belcher, "NUCLEFF-16, FLORA: Program for the Calculation of the Contributions of Fluorescence Radiation," DAC-60654 (October 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: source card deck, input for a sample problem, output listing of the sample problem, and the binary deck.

NOTE: Although a number of subroutines for plotting on the SC-4020 are contained in this package, plotting capabilities are unavailable in this 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

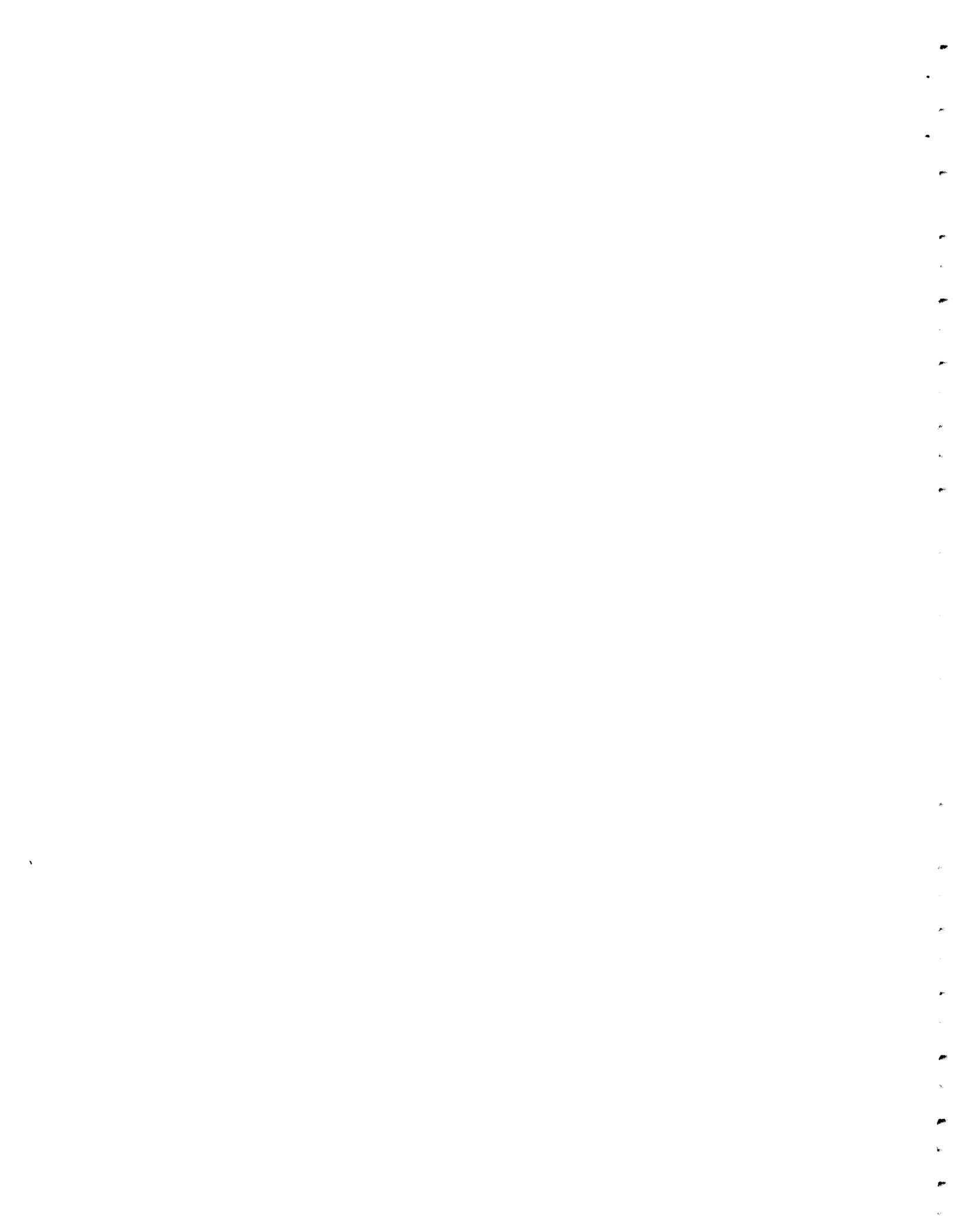
or telephoned to

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

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

## 13. DATE OF ABSTRACT

May 1969.



## RSIC CODE PACKAGE CCC-112

## 1. NAME AND TITLE OF CODE

SAND: Neutron Flux Spectra Determination by Multiple Foil Activation - Iterative Method.

Versions of the SAND program were received through the services of the General Electric Company, TEMPO, Data Center, Santa Barbara, California, and Computer Sciences, Richland, Washington.

## 2. CONTRIBUTORS

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.

## 3. CODING LANGUAGE AND COMPUTER

FORTRAN IV; CCC-112A: UNIVAC 1108, CCC-112B: IBM 360/75, CCC-112C: CDC 6600.

## 4. NATURE OF PROBLEM SOLVED

Neutron energy spectra are determined by an analysis of experimental activation detector data.

## 5. METHOD OF SOLUTION

An iterative perturbation method is used to obtain a "best fit" neutron flux spectrum for a given input set of infinitely dilute foil activities. The calculational procedure consists of the selection of a known flux spectrum form to serve as the initial approximation to the solution, and subsequent iteration to a form acceptable as an appropriate solution. The solution is specified either as time-integrated flux (fluence) for a pulsed environment or as a flux for a steady-state neutron environment.

A reaction cross-section library is provided with the code.

6. RESTRICTIONS OR LIMITATIONS

None noted.

7. TYPICAL RUNNING TIME

Estimated running time of sample problem: 2-3 minutes on the IBM 360/75; no available record for other hardware.

8. COMPUTER HARDWARE REQUIREMENTS

The code requires the use of a large computer; is operable on UNIVAC 1108, IBM 360/75 and CDC 6600.

9. COMPUTER SOFTWARE REQUIREMENTS

Standard FORTRAN IV compilers may be used.

10. REFERENCES

W. N. McElroy, S. Berg, T. Crockett, and R. G. Hawkins, "A Computer-Automated Iterative Method for Neutron Flux Spectra Determination by Foil Activation - A Study of the Iterative Method," AFWL-TR 67-41, Volume I (September 1967).

S. Berg and W. N. McElroy, "A Computer-Automated Iterative Method for Neutron Flux Spectra Determination by Foil Activation - SAND II and Associated Codes," AFWL-TR-67-41, Volume II (September 1967).

S. Berg, "Modification of SAND II," BNWL-855 (August 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 deck, input for a sample problem, and an output listing of the problem for each of the packaged code versions.

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, specifying which version is desired.

13. DATE OF ABSTRACT

May 1969.



## RSIC CODE PACKAGE CCC-113

## 1. NAME AND TITLE OF CODE

ATHENA: Monte Carlo Radiation Transport and Gamma-Ray Heating Code System in Complex Three-Dimensional Geometries.

The ATHENA system derives its basic structure from UNCSAM (CCC-81) and includes GENDA, GENPRO, VANGEN, EZGEOM, TESTG, DATORG, INPUTD, STATC, GASP, NATALE, and ATHENA.

## 2. CONTRIBUTORS

United Nuclear Corporation, Elmsford, New York.  
NASA Lewis Research Center, Cleveland, Ohio.

## 3. CODING LANGUAGE AND COMPUTER

FORTRAN IV; IBM 7090 and 7094.

## 4. NATURE OF PROBLEM SOLVED

The ATHENA system consists of a number of coordinated computer programs for the solution of radiation transport and heating problems in complex geometries, with special emphasis on the computation of gamma-ray heating in reactor cores.

## 5. METHOD OF SOLUTION

The system recognizes two geometry types, designated as 360-degree and 30-degree.

The 360-degree option accepts a description of an essentially homogenized cylindrical reactor core, which can be subdivided radially and axially to permit variation of the composition within the core and to provide several in-core flux scoring regions.

The 30-degree option, designed to exploit the 12-fold symmetry of a hexagonal-lattice reactor core, permits a much closer approximation to the actual configuration of such a core. Within the space between vertical planes bounding a 30-degree sector of space, the configuration of individual fuel and poison rods may be defined, as

well as that of the coaxial fuel cylinders within a fuel rod. In the Monte Carlo tracking of neutrons or gamma rays within this structure, particles striking the bounding planes are reflected optically, yielding, after many histories, the correct flux patterns in the representative 30-degree sector, without the need for describing explicitly (or tracking particles in) the other, congruent sectors of the reactor.

The programs are modeled after those in the UNC-SAM system; both systems use essentially the same nuclear-data library, interaction routines, and source-tape formats. However, ATHENA embodies many generalizations and new features which greatly increase its usefulness.

Among the new features of this system are:

1. A generalized and extremely versatile source-generator program, VANGEN;
2. New geometry-testing features, in program TESTG and in portions of the parameter-input routine, INPUTD;
3. The optional 30-degree-sector representation of a reactor;
4. Acceptability of "straddling" cylinders, each internal to several rectangular regions;
5. New high-efficiency cylinder and sphere tracking routines;
6. Sector-cylinder scoring;
7. Automatic generation of gamma-heating response functions;
8. New routines for primary- and secondary-gamma scoring at in-core and out-of-core point detectors;
9. Energy as well as spatial importance sampling in all pertinent programs;
10. A new GASP (secondary-gamma source production) program;
11. A neutron-absorption tally program, NATALE, for interpreting neutron interaction tapes.

The main tracking and scoring program in the system is called ATHENA. Input to this program are the processed geometry, cross-section power-pattern, and problem parameters as read from the dump tape, plus a neutron or gamma source tape produced by VANGEN or GASP.

ATHENA reads in source particles in sets of 100 from the source tape and processes each of these particles through its migrations and interactions until absorption, transmission, death by energy or spatial Russian roulette, or degradation below an energy cutoff occurs. (In neutron problems with a thermal-group option, particles below the cutoff energy are returned to a thermal energy, and tracking continues.) As the particle travels, a variety of information is recorded, including:

1. Fluxes in specified energy bins and regions, along with their standard deviations, in both gamma and neutron problems;
2. In gamma problems, total and volumetric heat deposition and standard deviation thereof for each region and for equi-angular sectors of specified cylinders;
3. A neutron-interaction tape for future calculation of secondary-gamma sources (via GASP) or other response-function calculations (via (NATALE));
4. A transmitted-particle tape for future processing;
5. Counts of particle births, deaths, absorptions, and degradations by region and of births and deaths by energy.

ATHENA permits the use of spatial and energy importance sampling through the specification of region weights (energy-independent) and energy weights, piece-wise constant over arbitrary energy-bins (one energy-weight mesh, region-independent). When a particle crosses a boundary between tracking regions, splitting or Russian roulette is done, based on the ratio of the pertinent region weights. Similar operations are performed after each collision, based on the energy weights before and after the collision. Heating and track-length scores are all computed taking into account the energy weight of the particle. In the final edit, answers pertinent to a given region are multiplied by the region weight.

Finally, there is the option of specifying a number of point detectors at which primary-gamma and secondary-gamma fluxes and heating can be computed.

ATHENA sends back its answers to the common dump tape.

## 6. RESTRICTIONS OR LIMITATIONS

Geometry: 200 regions, 127 special (non-parallelepiped) regions.  
Compositions: 32 distinct compositions; 15 nuclides/ composition for neutrons, 20 for gamma problems.  
Delayed-gamma tables: 80 instants, 15 energies.  
Reactor operating history: 50 instants.  
Variable dimensioning is used for most of the output array.

## 7. TYPICAL RUNNING TIME

Variable, depending on problem complexity. A large-scale problem run by Klann and Paulson attained a statistical accuracy of about 10% in several relatively small scoring volumes distributed in a 144-region, 30°-type geometry; it ran 42 minutes on the IBM 7094 computer for 80,000 primary-gamma histories, 51 minutes for 8000 fission neutrons, and 41 minutes for the secondary gammas generated in the 8000 neutron histories.

Estimated running time of the packaged sample problems on the IBM 7090:

VANGEN: 4 minutes  
MEZDA: 6 minutes  
ATHENA: 11 minutes  
GASP: 2 minutes  
STATC: 3 minutes

## 8. COMPUTER HARDWARE REQUIREMENTS

The code was designed for the 32K CDC 1604 and the IBM 7094 (packaged version) and requires up to 4 tape units.

## 9. COMPUTER SOFTWARE REQUIREMENTS

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

## 10. REFERENCE

D. Spielberg, "ATHENA: A System of FORTRAN Programs for Radiation Transport and Heating Calculations in Complex Reactor Geometries,"

UNC-5148 (NASA CR-54905), (March 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 26 files:  
the BCD source card decks, BCD input for 2 sample problems,  
and the BCD output listing of the problems.
- c. a reel of tape on which is written in 3 files: BCD cross  
section data libraries for use in the system.

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

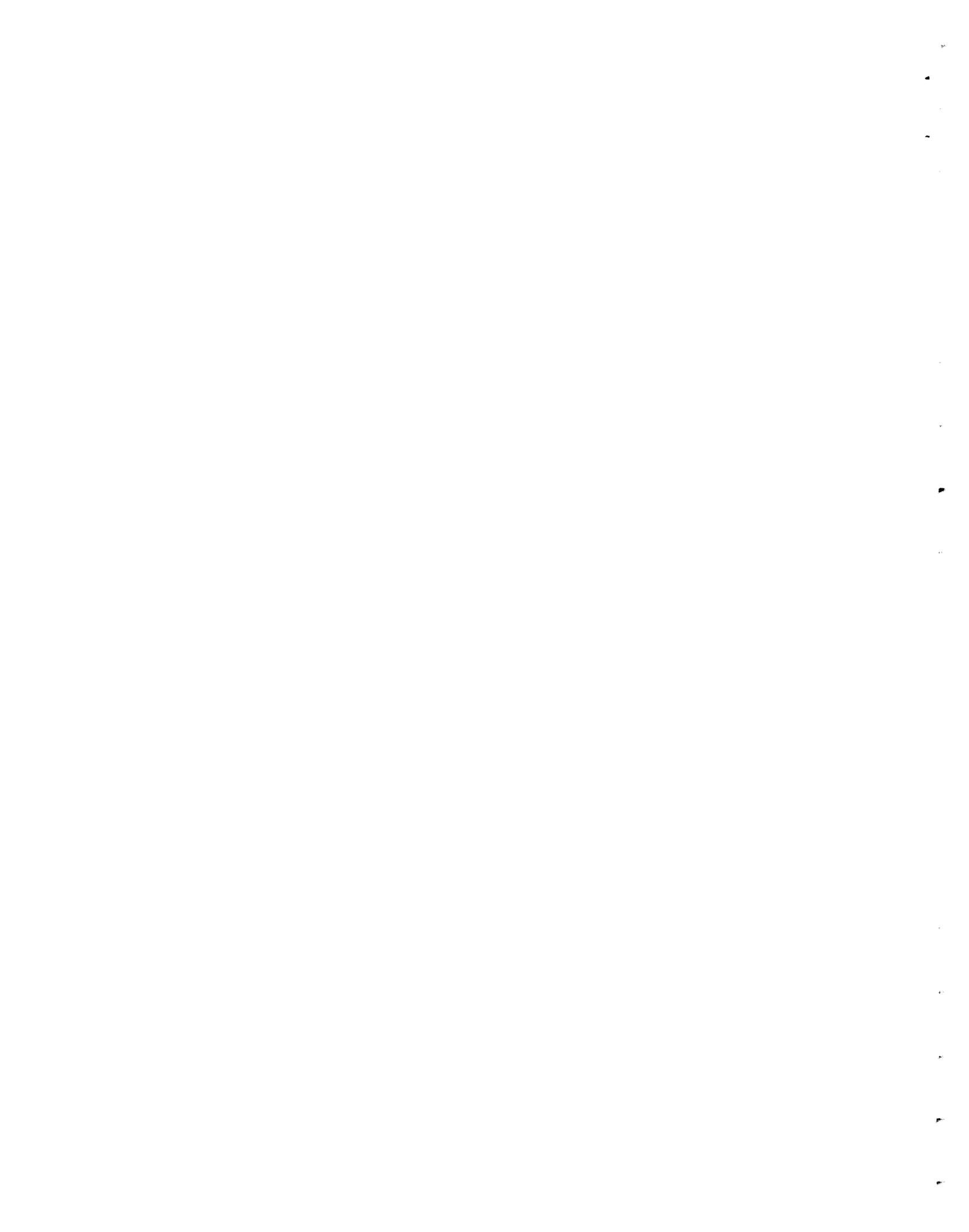
or telephoned to

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

Persons requesting the code package should send 2 reels of magnetic  
tape to the above address.

13. DATE OF ABSTRACT

May 1969.



## RSIC CODE PACKAGE CCC-114

## 1. NAME AND TITLE OF CODE

SAM-C: Monte Carlo Time Dependent Three-Dimensional Complex Geometry (Combinatorial) Shielding Code System.

## 2. CONTRIBUTOR

Mathematical Applications Group, Inc., White Plains, New York.  
U. S. Army Ballistic Research Laboratory, Aberdeen Proving Ground, Maryland.  
U. S. Army Nuclear Defense Laboratory, Edgewood Arsenal, Maryland.

## 3. CODING LANGUAGE AND COMPUTER

FORTRAN IV; CDC 6600.

## 4. NATURE OF PROBLEM SOLVED

SAM-C calculates the time-dependent transport of neutrons or gamma rays through complex three-dimensional geometrical configurations. The code will calculate the total flux in each region in a specified group of energy and time bins. The code can also compute fluxes at specified points within the geometry.

## 5. METHOD OF SOLUTION

SAM-C is based in large measure on the UNC-SAM-2 Monte Carlo program prepared by United Nuclear Corporation (with MAGI as subcontractor) for Aberdeen Proving Ground. The primary difference between UNC-SAM-2 and SAM-C is the geometrical description technique employed. SAM-C uses combinatorial geometry and is therefore capable of representing more complex assemblies. To use this geometry capability a number of modifications were made in the logic and storage requirements of UNC-SAM-2. In addition, a ray-tracing volume calculation routine was added since, for many of the shapes produced by the combinatorial geometry, it is impractical to determine the volume analytically. All the nuclear interaction routines of

of UNC-SAM-2 are unchanged.

Combinatorial geometry is a new and significant advance in the state-of-the-art of representing - in a computer - a complex three-dimensional structure. In effect, one represents a structure such as a tank in terms of sums, differences, and intersections of relatively simple bodies such as spheres, cylinders, etc. The input for such a description consists of the geometric location of the simple bodies and their dimensions, followed by a region definition table consisting of a series of equations defining each particular region of the structure in terms of the basic bodies. For example, if the total structure is a tank, then one region would be the gun barrel, which might be represented as the material located between two concentric cylinders.

The SAM-C program consists of two main programs and their associated subroutines.

The first main program is TUNC. TUNC reads and processes all input. The input is distributed among various common blocks and in an array labeled MASTER or ASTER. The two arrays are equivalenced and the different names are used to reference fixed point or floating point data. The MASTER-ASTER array contains the bulk of the input, i.e., cross sections, geometry, and Monte Carlo track length scores.

The second main program is called MONTE. MONTE controls the actual Monte Carlo calculation and the flux-at-a-point calculation. The routine controls the reading of cross-section bands and the writing on tape of the answer arrays.

## 6. RESTRICTIONS OR LIMITATIONS

Some dimensional limitations are as follows:

Maximum number of output flux energy bins:	200
Maximum number of points in the energy mesh for energy importance sampling:	50
Maximum number of points in the cosine mesh for angular importance sampling:	50
Maximum number of points in the time mesh for time-dependent problem:	50

Storage location for cross-section geometry, input and  
flux data: 30,000

7. TYPICAL RUNNING TIME  
No statistics have been accumulated by RSIC as to typical running time.
8. COMPUTER HARDWARE REQUIREMENTS  
The code was designed for the CDC 6600.
9. COMPUTER SOFTWARE REQUIREMENTS  
The code is operable on the Chippewa operating system.
10. REFERENCES
  - a. Background Material
    - (1) E. S. Troubetzkoy, "UNC-SAM-2: A FORTRAN Monte Carlo Program Treating Time-Dependent Neutron and Photon Transport Through Matter," UNC-5157 (September 1966).
  - b. Code Package Material
    - (2) W. Guber, R. Negel, R. Goldstein, P. S. Mittleman, and M. H. Kalos, "A Geometrical Description Technique Suitable for Computer Analysis of Both the Nuclear and Conventional Vulnerability of Armored Military Vehicles," MAGI-6701 (August 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: cross sections, source card deck, input data for a sample problem, and output listing for 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  
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.

13. DATE OF ABSTRACT

May 1969.

## RSIC CODE PACKAGE CCC-115

## 1. NAME AND TITLE OF CODE

GADJET: Monte Carlo Gamma-Ray Adjoint Energy Transport Code  
in Complex Three-Dimensional Geometry

## 2. CONTRIBUTORS

Radioptics, Inc., Plainview, New York.  
U. S. Naval Radiological Defense Laboratory, San Francisco,  
California.  
Office of Civil Defense, Washington, D.C.

## 3. CODING LANGUAGE AND COMPUTER

FORTRAN IV; CDC 6600.

## 4. NATURE OF PROBLEM SOLVED

The code is designed to calculate the effectiveness of structures (of arbitrary complexity) in shielding against fallout fields (finite or infinite), and to calculate the relative importance of each fallout region in contributing to the exposure rate in the structure. Complex realistic structures can be assumed, with windows, sills, black surfaces, etc., being allowed.

## 5. METHOD OF SOLUTION

The code solves the adjoint transport equation for the so-called "importance" function. A knowledge of the importance function, coupled with the source distribution, solves the problem of determining the response of an arbitrary gamma-ray detector located in a structure that is exposed to fallout gamma-ray sources.

GADJET handles the transport of photon energy through matter having a three-dimensional geometry composed of rectangular parallelepipeds which, in turn, may contain spheres, cylinders, parallelepipeds, or wedges. The geometry routines are based on those used in the UNC-SAM-1 code with only minor modifications.

GADJET consists of three independent programs: DATORG (cross-section processing and preparation); EZGEOM (geometry specifications); and ADJOINT, which calculates the solution of the transport problem with an edit of the results.

DATORG requires an Element Data File (EDF), a library of microscopic cross sections for elements which make up materials for a given problem. The microscopic cross sections are tabulated against energy at a prescribed energy mesh. The EDF must be supplied by the problem originator.

DATORG creates an Organized Data File (ODF) containing the total macroscopic cross sections and electron density based on the concentrations of materials on the EDF. The ODF is problem-dependent cross-section data tabulated against energy at a prescribed energy mesh as provided by the UNC-SAM data tape.

EZGEOM takes a simplified geometrical description of the physical system, as provided by the problem originator, and produces a rather complex set of data required by the transport program.

The problem originator must decompose space into a set of boxes known as "ordinary regions." Inside these boxes may be placed "non-ordinary regions" consisting of spheres, cylinders, wedges, or other boxes. Nonordinary regions may enclose other nonordinary regions. (See UNC-SAM report for further details.)

The output of EZGEOM and DATORG are combined and placed on the Organized Data File (ODF).

ADJOINT, using the appropriate Organized Data File (ODF) and parametric input defining a particular problem (including the structure) to be solved, selects a photon impinging on the specified detector position and attempts to reconstruct a history through the structure. During the reconstruction of the history, scores may be recorded where needed, collisions may occur and be noted, and latents may be created. Latents and photons coming out of collision (if any) are subject to treatment similar to photons originating at the detector. An estimated score is recorded for each photon emerging from collision if the photon path intersects a scoring surface. A scoring

surface at present is made up of portions of parallel planes perpendicular to a coordinate axis.

The calculation is done in statistical groups so that variances can be calculated. The results are displayed in bins corresponding to the manner in which the scoring surfaces have been sectorized. The total uncollided and collided scores for each sector of the scoring surfaces are displayed in the edit.

#### 6. RESTRICTIONS OR LIMITATIONS

Some dimensional limitations are as follows:

Maximum number of regions (ordinary and nonordinary):	200
Maximum number of nonordinary regions:	127
Maximum number of distinct nuclides:	33
Maximum number of compositions:	20
Maximum number of nuclides for any composition:	5
Maximum number of energies:	100

#### 7. TYPICAL RUNNING TIME

No studies of typical running time have been made by RSIC.

#### 8. COMPUTER HARDWARE REQUIREMENTS

The GADJET code was developed and used on the CDC-6600 at the Courant Institute, New York University. If the code is used on the machines with core systems less than 131K, some minor reprogramming is required to change array sizes in accomodating to the smaller core sizes. The 60-bit word size on the CDC 6600 is larger than that of other computers (usually 36 bits per word); this also involves some reprogramming if GADJET is used on other computers.

#### 9. COMPUTER SOFTWARE REQUIREMENTS

The code is operable on the Chippewa operating system.

## 10. REFERENCES

## a. Background Material

(1) B. Eisenman, F. Nakache, "UNC-SAM, A FORTRAN Monte Carlo System for Calculation of Neutron or Gamma-Ray Transport in Three-Dimensional Geometry," United Nuclear Corporation, Report UNC-5093 (1964).

## b. Code Package Material

(2) E. R. Friedman, M. H. Kalos, S. Presier, G. Rabinowitz, J. G. Beckerley, "The Numerical Solution of the Adjoint Transport Equation for Gamma Rays by the GADJET Code," NRDL-TRC-68-27 (1968).

(3) S. Presier, M. Kalos, A. Stathoplos, J. G. Beckerley, E.R. Friedman, G. Rabinowitz, H. Sadowski, L. A. Willis, "Calculation of Gamma Exposure Rates in an Open or Covered Basement in Fallout Fields Using the GADJET Code," NRDL-TRC-68-25 (1968).

## 11. CONTENTS OF CODE PACKAGE

The package contains the following items:

- a. the referenced documents, section b,
- b. a reel of magnetic tape on which is written in several files: BCD cross sections, the source card deck, input for a sample problem, 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

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

13. DATE OF ABSTRACT

May 1969.

