GAZE-II*

1. Name of code: GAZE-II
2. Computer for which code is designed: IBM 7090
   Programming system: FORTRAN including FAP
   Monitor: 709/7090 FORTRAN Monitor System
3. Nature of problem solved: One-space dimensional, multigroup neutron diffusion including criticality searches on position of any radius or on volume fractions of any combination of nuclides in any combination of materials.
4. Method of solution: Highly accurate difference equations with 4th order error at interior points and 3rd order error at material discontinuities. Outer iterations using Chebyshev extrapolation. Inner (flux) iterations based on splitting scatter-transfer matrix into down- and up-scatter parts. Criticality searches using first order perturbation theory.
5. Restrictions on the complexity of the problem:
   Problem size limitations:
   $$\begin{cases} L + 5I + 9K + LM + 2IM + 9IK + (M+1)I^2 \leqq 21998 \\ K \geqq 3 \end{cases}$$
   where
   L = number of nuclides
   I = number of groups
   K = number of space points, including both boundaries
   M = number of materials
   Machine requirements: 32K memory, 5 channel A tapes, and 2 channel B tapes.
6. Typical running times: 3.15 min for large system requiring 198 outer iterations to converge. This problem had 9 groups total with 6 thermal groups, 5 materials, 41 space points, and 4 nuclear species. The same problem took 2.95 min for 9ZOOM and 15.65 min for FN. An 18-group problem with no upscattering and requiring 11 outer iterations to converge required 1.19 min for GAZE. This problem had 2 materials, 59 space points, and 2 nuclear species. The same problem took 1.80 min for AIM-6.
7. Unusual features: All the features of 9ZOOM, including scatter transfers allowed between all groups. In addition, output fluxes from each problem are punched on cards and can be used as input starting fluxes for any problem. Capability of adjoint as well as direct flux solution. An extensive edit, including point values of fluxes, flux gradients, capture, fission, and sources densities. Many forms of peak-to-average flux are included as well as capture and fission rates for every nuclear species in every material. Macroscopic cross sections for every material are given along with transverse, radial, and total bucklings in every material and group. In addition, a complete breakdown of the neutron balance is given for each group in every material and in the entire system. Various criticality searches can be performed and are based on first order perturbation theory.
8. Present status: Production. Inquiries should be addressed to: G. D. Joanou, General Atomic Division/General Dynamics Corporation, P. O. Box 608, San Diego 12, California.
9. *References:* S. R. Lenihan, "GAZE—A One-Dimensional, Multigroup, Neutron Diffusion Theory Code for the IBM 7090." GA-3152 (May 4, 1962).

* Work supported under a joint program of General Atomic and Empire State Atomic Development Associates, Inc.

S. P. Stone, E. T. Collins, and S. R. Lenihan, "9-ZOOM—A One-Dimensional, Multigroup, Neutron Diffusion Theory Reactor Code for the IBM 709." UCRL-5682 (August 25, 1959).

P. G. Fischer, "Multigroup, Multiregion, One Space Dimension Neutron Diffusion Theory Calculation—Program FN." XDC 60-3-68.

H. P. Flatt and D. C. Baller, "NAA Program Description, The AIM-6 Code." (January 1961).

S. R. Lenihan
General Dynamics/
General Atomic Division
San Diego, California

SPMCN-1

1. Name of code: SPMCN-1
2. Computer for which code is designed: IBM 7090, 7094
   Programming system: Fortran II
3. Nature of problem solved: SPMCN-1 is a multiregion, multienergy group, time dependent Monte Carlo code for neutrons in a homogeneous medium where the neutron interactions considered are scattering and absorption.

   The neutron source is a point source at the spatial coordinate $(X, Y, Z) = (0, 0, 0)$. The original neutron energies are generated from an input energy spectrum. Initial neutron directions are picked from input angular distributions. There is an input angular distribution for each energy group. The maximum number of energy groups is 32 and the maximum number of entries in the angular distribution for each energy group is 50. The geometry of the system is described by specifying various spheres about the point $(X, Y, Z) = (0, 0, 0)$. The result is a series of spherical shells about the point (0, 0, 0). Input is limited to 150 such spheres and therefore limits the number of spherical shells (regions) to 150.

   One material—homogenous and of constant density—occupies all regions. This material may be composed of a maximum of 5 isotopes (or elements).

   All answers are obtained by analytic estimation with the exception of Monte Carlo tallies of particle and neutron leakage from the system. Answers consist of the following:
   (1) The flux and dose rate for each spherical shell (region) by energy group.
   (2) Neutron and particle leakage from the system.
   (3) The distribution of the neutron leakage from the boundary sphere of the system by energy group and for each energy group, the angular distribution of neutrons.
4. Restrictions on the complexity of the problem: Maximum number of energy groups—32; maximum number of regions—150; maximum number of isotopes—5.
5. Typical running times: This code runs 1 to 3 min depending on the complexity of the problem.
6. Unusual features of the code: Russian roulette is played with any particle whose weight becomes too small. Splitting in prescribed regions and analytic estimation are used to increase the validity of the results for the least amount of computing time.
7. Present status: In use.
8. *Reference:* M. J. Kniedler and T. Jordan, SPMCN-1,

Neutron Monte Carlo Code for Spherical Geometry. MND 4539 (January 19, 1963)

M. J. KNIEDLER
Martin Marietta Corp.
Baltimore 3, Maryland

MO322

1. Program name: MO322
2. Computer for which program is designed: Philco-2000 Programming system: FORTRAN II
3. Nature of the problem solved: MO322 is a one energy, Monte Carlo calculation of the fraction of neutrons absorbed in spheres randomly distributed in an otherwise infinite homogeneous medium. A finite number of spheres is loaded randomly into a three-dimensional box. This box is assumed to repeat in all directions to fill all space. Thus, each sphere which has its center in the box but which is cut by one or more sides of the box has from one to seven corresponding spheres outside the box which lean into the box and which are displaced from the original sphere along one or more coordinates by the width of the box. The neutron source is either flat in the region external to the spheres, flat in the spheres, or flat everywhere. Individual neutrons are followed in their wanderings through the box. When one leaves the box on one side, it re-enters with the same path orientation on the opposite side. Absorption and isotropic scattering are included in both compositions. The box can be reloaded with a different random distribution of spheres in the course of the calculation.
4. Restrictions on the complexity of the problem: 150 spheres in the box, 100 box loadings, 100 experiments per box loading, 500 histories per experiment.
5. Typical running times: With 100 spheres in the box making up 10% of the volume of the box the following times are representative:
   0.19 min to load the box
   5.7 min per 100 histories
   0.5–1.0 hr over-all, to get a reasonable answer.
6. Unusual features of the program: The variances in the track length estimates of the compositionwise absorptions have been reduced by dividing the average partial track lengths (in absorption mean free paths) by the average total track length. On the one hand, this makes the conventional statistical analysis, based on a normal distribution of the results of experiments, inapplicable. On the other hand it makes the program possible by reducing the running time to a reasonable value.
7. Present status: In use.
8. *Reference:* R. M. Cantwell, "MO322 and MO332—FORTRAN Programs for Calculating Neutron Absorption in Spheres Distributed Randomly." WAPD-TM-352 (October 1962).

R. M. CANTWELL
Westinghouse Electric Corporation
Bettis Atomic Power Laboratory
Pittsburgh, Pennsylvania

PLMCN-1

1. Name of code: PLMCN-1
2. Computer for which code is designed: IBM 7090, 7094 Programming system: Fortran II
3. Nature of problem solved: PLMCN-1 is a multiregion, multienergy group, time dependent Monte Carlo code for neutrons in a homogenous medium where the neutron interactions considered are scattering and absorption.

   The neutron source is an infinite plane perpendicular to the z-axis of the system at $Z = 0$. The original neutron energies are generated from an input energy spectrum. Initial neutron directions are picked from input angular distributions. There is an input angular distribution for each energy group. The maximum number of energy groups is 32 and the maximum number of entries in the angular distribution for each energy group is 50. The geometry of the system is described by specifying various planes parallel to the XY-plane. The result is a series of infinite slabs parallel to the source plane. Input is limited to 150 such planes and therefore limits the number of slabs (regions) to 150. (The plane at $Z = 0$ need not be inputed.)

   One material—homogenous and of constant density—occupies all regions. This material may be composed of a maximum of 5 isotopes (or elements).

   All answers are obtained by analytic estimation with the exception of Monte Carlo tallies of particle and neutron leakage from the system. Answers consist of the following:
   (1) The flux and dose rate for each slab (region) by energy group.
   (2) Neutron and particle leakage from the system.
   (3) The distribution of the neutron leakage from the boundary opposite the source plane by energy group and for each energy group, the angular distribution of neutrons.
4. Restrictions on the complexity of the problem: Maximum number of energy groups—32; maximum number of regions—150; maximum number of isotopes—5.
5. Typical running times: This code runs 1 to 3 min depending on the complexity of the problem.
6. Unusual features of the code: Russian roulette is played with any particle whose weight becomes too small. Splitting in prescribed regions and analytic estimation are used to increase the validity of the results for the least amount of computing time.
7. Present status: In use.
8. *Reference:* M. J. Kniedler and T. Jordan, PLMCN-1, Neutron Monte Carlo Code for Slab Geometry. MNI-C-2933 (January 19, 1963)

M. J. KNIEDLER
Martin Marietta Corp.
Baltimore 3, Maryland