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Computer Code Abstracts*

ABSTRACT NO. 9

1. Name of code: FIRE
2. Computer programming system: Code is written for a 32K IBM-704 with at least three tape units. A more restricted 8K version is available. The code is in long-hand and was assembled by the standard Los Alamos assembly program (871).
3. Problem solved: FIRE solves the one-dimensional age-diffusion equations for slab (symmetric and asymmetric), cylinder, and sphere geometries.
4. Method for solution: Multigroup equations are solved by iterating on the fission source. Finite difference equations are used in multiregion problems, while a core radius may be specified for equivalent bare reactors problems with only the energy variable. Detailed thermal flux spectra may be computed using "up-scattering" and "down-scattering" cross sections. In computing ith iterate of flux, the $(i-1)$st iterate is used to compute fission source and up-scattering, and the ith iterate is used to compute down-scattering.
5. Basic physics approximations: Macauley-Coveyou approximation for hydrogen and age-diffusion theory for other elements. Neutrons are transferred either up or down by inelastic scattering matrix.
6. Restrictions on complexity of problem: Number of regions $R \leq 12$, the number of groups $I \leq 39$; $R \leq 24$, $I \leq 19$. Maximum number of space points is 180.
7. Typical running times: One half to two minutes for 18 group case with no up-scattering (7 to 14 iterations). A complicated case with a difficult thermal flux and a convergence criterion less than 10^{-6} may run an hour or more. Running times may be minimized through use of flux dumps to restart similar problems.
8. Unusual features: Inclusion of "up-scattering;" compatibility of cross-section input required by FIRE and DSN codes.
9. *References:* LA-2161: A One Dimensional Intermediate Reactor Computing Program. C. B. Mills and F. W. Brinkley, March 27, 1959. Deals with the theory behind code. LA-2161 Supplement: A One Dimensional Intermediate Reactor Computing Program. F. W. Brinkley, March 17, 1960. Deals with problem setup. Macauley-Coveyou approximation: communicated by B. T. Macauley, 1952.
10. Availability: Persons wishing to use code should contact F. W. Brinkley.

F. W. BRINKLEY AND C. B. MILLS
Los Alamos Scientific Laboratory
Los Alamos, New Mexico

ABSTRACT NO. 10

1. Name of code: EQUIPOISE-2
2. Computer for which code is designed and programming system: IBM-7090; FORTRAN II; to be run under the IBM-7090 FORTRAN Monitor System.
3. Nature of problem solved: Two-group, two-dimensional, neutron diffusion equations in cylindrical geometry. EQUIPOISE-2 is a 7090 version of EQUIPOISE-1.
4. Restrictions on the complexity of the problem:
   Maximum number of groups—2
   Maximum number of increments in r or z direction—37
   Machine requirements: 32K 7090 with 7 tape units.
5. Typical running times: 1–3 min for a 1000 point problem.
6. Unusual features of the code: Same as EQUIPOISE-1.
7. Present status: In use. Interested persons should contact authors.
8. *References:*
   T. B. Fowler and Melvin Tobias, EQUIPOISE-2—A two-dimensional, two-group, neutron diffusion code for the IBM-7090 computer, ORNL-CF-60-11-67 (November, 1960).
   Melvin Tobias and T. B. Fowler, EQUIPOISE, Reactor Code Abstracts, *Nuclear Sci. and Eng.* **9**, 101 (1961).

MELVIN TOBIAS AND T. B. FOWLER
Oak Ridge National Laboratory
Oak Ridge, Tennessee

ABSTRACT NO. 11

1. Code Designation: FF-MOCCA
2. Computer, programming system: IBM-704. Program which prepares input data (i.e., cross sections, angular distributions, level excitations, etc.) is in FORTRAN II. Remainder in SAP.
3. The FF-MOCCA Code computes the Fast Fission Ratio (δ) and the Fast Fission Factor (ϵ) by the Monte Carlo Method. The physical effects which can be included are: absorption, fast fission, $(n, 2n)$ reactions, isotropic and anisotropic elastic scattering (in cm system), and inelastic scattering by excitation of an energy level or in accordance with the statistical model. Absorption processes are calculated by the use of a weight factor, and fission processes by the method of fractional generated neutrons. The energies for primary and secondary generated neutrons are picked from Watt type fission spectra. The variation of ν (number of neutrons per fission) with energy is included. An arbitrary distribution of the primary fission neutrons in fuel rods can be used.

   The various cross sections in the program are group averaged. Up to 50 energy groups may be used. The unit cells in our calculations can be subdivided into a maxi-

* Computer codes for this section should be submitted directly to the *Code Abstract Editor*, Ely M. Gelbard, Bettis Atomic Power Laboratory, Westinghouse Electric Corporation, P. O. Box 1468, Pittsburgh 30, Pennsylvania.

mum of 15 regions which can be filled with different isotope mixtures. Each mixture can contain up to 10 isotopes.

The geometry routines can treat infinite slabs, staggered rectangular lattices, unstaggered rectangular lattices, hexagonal cells and rods bundles. They were originally written for the N.Y.U. Resonance Escape Probability Code. The calculations can be done for multiregion cells in an infinite lattice array or for a single cell only.

Specifically, the code computes the following quantities:

(a) The fast neutron ratio (γ), defined as the ratio of secondary neutrons to primary neutrons, for each fissionable isotope.
(b) The fast fission factor (ϵ) for the entire system.
(c) The fast fission factor ($\epsilon(1)$) for neutrons scattered only in the region where they were born.
(d) The fast fission factor ($\epsilon(N)$) for neutrons not scattered in the moderator. (The difference $\epsilon - \epsilon(N)$ is the so-called "back scattering effect.")
(e) The fractional absorption as a function of the geometric region and the energy group.
(f) The over-all absorption for equal volume elements in each region. This gives the spatial neutron distribution.
(g) The number of collisions necessary to leave the fast fission region as a function of the energy group in which the neutron started.
(h) The neutron escape probability into a specified region (for single cell calculations).
(i) The fast fission ratio (δ), defined as the ratio of fast fissions to thermal fissions. This ratio δ can easily be computed by the use of the fractional absorptions specified in (e).

4. Typical running times: The running times depend strongly on the number of collisions/history necessary to leave the fast fission energy region. A typical problem might use 2000 histories and require 6–9 min/10,000 collisions.
5. *References:*

H. Rief, An IBM-704 Monte Carlo Code to calculate the fast effects in homogeneous and heterogeneous systems, BNL Report (1960).

R. D. Richtmyer, R. Van Norton, and A. Wolfe, The Monte Carlo Calculation of Resonance Capture in Reactor Lattices, *Proc. 2nd Intern. Conf. Peaceful Uses Atomic Energy, Geneva* **16,** 180 (1958).

6. Availability: Those interested in using MOCCA should contact H. Honeck at Brookhaven National Laboratory.

H. Rief
Brookhaven National Laboratory
Upton, New York

Note

The authors of the paper "SM-2 Flexible Critical Experiments" [*Nuclear Sci. and Eng.* **9,** 41 (1961)] wish to acknowledge the many significant contributions made to that paper by Mr. John W. Noaks, former supervisor of the Alco Products, Inc. Critical Facility.