MOD-5

A Discrete Markov Slowing Down Code

1. Name of Program: MOD-5.
2. Computer for Which Program is Designed: IBM 360.
3. Nature of Physical Problem Solved: This code calculates the time- and energy-dependent evolution of the neutron density in homogenous media following initiation of (a) a monoenergetic source distributed over a finite time interval, or (b) a source of arbitrary spectrum with a delta-function distribution in time. Effectively, the code produces Green's Function solutions to the slowing down equation in discrete numerical form. Leakage is treated in the diffusion approximation. The program (a) calculates spectra and energy moments at selected times following the burst of source neutrons, (b) evaluates the time-dependent neutron density and slowing down density at selected energies and computes moments of these densities, (c) calculates time-dependent distributions of capture, leakage, and first fission, and moments of these distributions, (d) calculates steady-state central core neutron flux and leakage flux in detail and in group averaged form, and (e) calculates miscellaneous parameters such as k_{eff}.
4. Method of Solution: The code is based on a discrete Markov slowing down model developed by the author.[1,2] The energy range of interest in the slowing down region is divided into an arbitrary number of "states" and a Markov matrix is constructed which defines the probabilities for transition from one state to another in some finite time interval. The neutron density (defined as a vector over the state structure) is evolved in time by repetitive multiplication of the density vector into the transition matrix. Cross sections are currently derived from the 26 group Russian set.[3] Capture, fission, and leakage events are accounted for in three absorbing states.
5. Restrictions on Complexity of the Problem: Number of isotopes ≤ 5; number of real energy states ≤ 71; number of virtual energy states (see Ref. 2 or 4) ≤ 201; number of broad groups for input cross sections ≤ 26; number of time steps ≤ 400; number of time moments ≤ 21; number of groups from which inelastic scattering can take place ≤ 10. The program can be modified to handle up to 701 virtual energy states and 201 real states.
6. Typical Running Time: On the IBM 360 the running time in seconds is given approximately by the formula

$$t = 0.004\ N\ N_v + 10 \quad ,$$

where N is the number of real states and N_v the number of virtual states.

7. Unusual Features of the Program:
   a. Elastic scattering transition probabilities are calculated with a new stochastic algorithm.[1]
   b. The state structure can be calculated by the code to provide optimal (most accurate) treatment of elastic scattering.
   c. Calculation times may be greatly reduced by allowing the transition matrix to follow the neutron distribution. This "traveling array" technique[1,2] may also allow a much finer state structure for a given core capacity.
   d. A unique transition matrix generating technique provides transition matrices that are consistent, accurate, and stable regardless of energy range or time step width.
8. Related and Auxiliary Programs: SLOAD arranges the cross-section library in proper format and loads it on the appropriate direct access device.
9. Status: In use.
10. Machine Requirements: 175,000 bytes memory, normal input, output, program, and punch units, 6 cylinders on IBM 2314 or equivalent.
11. Programming Language Used: FORTRAN-IV.
12. Operating Systems Under Which Program is Executed: The program can operate under both CP/CMS and OS.
13. Material Available: The following material may be obtained from the Argonne Code Center, Argonne National Laboratory: MOD-5 source deck (2583 cards), SLOAD source deck and cross section, library (2084 cards), sample problems (22 cards). Reference report, NPS-61WN71061A.
14. *Acknowledgment*: This program and accompanying documentation are based in part on work supported by the Foundation Research Program at the Naval Postgraduate School. The Foundation Research Program is funded by the Office of Naval Research.
15. *References*:

[1]T. J. WILLIAMSON and R. W. ALBRECHT, *Nucl. Sci. Eng.*, **37**, 41 (1969).

[2]T. J. WILLIAMSON and R. W. ALBRECHT, *Nucl. Sci. Eng.*, **42**, 89 (1970).

[3]L. P. ABAGYAN, N. O. BAZAZYANTS, I. I. BONDARENKO, and M. N. NIKILEAV, *Group Constants for Nuclear Reactor Calculations*, Consultants Bureau Enterprises, Inc., New York (1964).

[4]T. J. WILLIAMSON, "MOD-5: A Computer Code for Calculations of Neutron Time-Energy Distributions in the Slowing Down Region," NPS-61WN71061A, Naval Postgraduate School, Monterey, California (1971).

T. J. Williamson

Physics Department, Code 61Wn
Naval Postgraduate School
Monterey, California 93940

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QX1

A One-Dimensional Multigroup Kinetics Program

1. Name of Program: QX1.
2. Computer for Which Program is Designed: CDC 3600, IBM 360.
3. Nature of Physical Problem Solved: QX1 solves the multigroup, one-dimensional time-dependent diffusion equations. Problem geometry may be plane, cylindrical, or spherical. Steady-state initial conditions may be established either for a source-free system or for a system with an external neutron source. The reactor may be perturbed by changing material volume fractions and/or temperatures or by changing the neutron source level. A first-collision time-dependent source distribution may be specified. Resonance absorption feedback is calculated by groupwise interpolation in a

cross section versus temperature table. A highly simplified fuel temperature model is included.

4. Method of Solution: The improved quasi-static method as described in Ref. 1 is used to solve the time-dependent problem. The method consists of factoring the total flux into the product of a space-energy-time-dependent shape function and a purely time-dependent amplitude function, normalized so that the most rapidly varying part of the total flux is included in the amplitude function. The coupled set of equations which results is solved iteratively. The factorization method was developed specifically for fast reactor safety analysis. The advantages of factorization are greatest for such systems, though the code has been shown to perform adequately on thermal reactor problems.

5. Restrictions on the Complexity of the Problem: 30 energy groups, 15 downscatter groups, 6 delayed neutron families, 20 spatial regions, 16 material mixtures per region, 150 mesh points.

6. Typical Running Time: A 29-group, 15 downscatter, 51 mesh point rod-drop problem run to 30 reactor sec executed in 29 min on the CDC 3600 and 10 min on the IBM 360/75. A 10-group, 6 downscatter, 53 point pulsed-reactor problem run to 1-msec reactor time executed in 10 min on the IBM 360/75.

7. Unusual Features of the Program: The running time can be reduced greatly for problems requiring relatively low accuracy, but the code has been shown to reproduce the results of direct finite-difference codes when convergence is tightened. An automatic time-step selector provides the capability of using small time steps in those portions of the transient where necessary to maintain accuracy, and larger time steps where possible to reduce running time. A true point kinetics problem can be run using only the initial shape function. A compact problem edit is given in terms of the familiar integral quantities of reactivity, effective delayed-neutron fraction, generation time, etc. Very general problem driving functions and time step controls may be used. A group-collapsing system is built into the problem preparation module of the code.

8. Status: Extensive testing and comparison with other kinetics codes has been completed. The code is in use at Argonne National Laboratory.

9. Machine Requirements: The CDC-3600 version requires a 64K memory and a maximum of eight tapes plus input, output, and punch tapes. The IBM-360 version requires 140K short words of memory, a maximum of eight disk datasets plus input, output, and punch datasets.

10. Programming Language Used: FORTRAN IV. Each version is independent of the peculiarities of local language to the maximum extent possible. Variances are documented on comment cards within the code. The CDC-3600 version has been adapted to the CDC-6000 series language through interaction with work carried out on a CDC-6500 machine.

11. Operating System: CDC-3600—SCOPE 6.2. IBM 360/75—OS/360 (Release 17).

12. Material Available: The source deck, cross-section library, sample problems, and the reference document[2] which describes the code are available from the Argonne Code Center. The requestor should specify either the CDC-3600 version or the IBM-360 version as required. The CDC-3600 version is recommended for adaptation to a CDC-6000 series machine.

13. *Acknowledgment*: This work was performed under the auspices of the U. S. Atomic Energy Commission.

14. *References*:

[1]K. O. OTT and D. A. MENELEY, *Nucl. Sci. Eng.*, **36**, 402 (1969).

[2]D. A. MENELEY, K. O. OTT, and E. S. WIENER, "Fast Reactor Kinetics—The QX1 Code," ANL-7769, Argonne National Laboratory (March 1971).

Daniel A. Meneley
E. S. Wiener

Argonne National Laboratory
9700 S. Cass Avenue
Argonne, Illinois 60439

Karl O. Ott

Purdue University
Nuclear Engineering Department
West Lafayette,
Indiana 47906

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SN35

1. Name of Code: SN35.

2. Computer for Which Program is Designed: ICL-1905.

3. Nature of Physical Problem Solved: SN35 solves the time-independent neutron transport approximation for specially dependent neutron flux distribution in spherical geometry. The code provides the pay-off matrices of a two-person nonzero sum game for given strategies and given quantities of fertile material in a fast critical system.

4. Method of Solution: It is a new approach to solve the neutron transport equation. The parameter which is modified at the end of each outer iteration is not critical radius but ratio of critical quantity of fissile material and given quantity of fertile material.

5. Restrictions on the Complexity of the Problem: 41 spatial points, 16 energy groups, 5 angular directions, 5 nuclides, 3 zones of heterogeneity.

6. Typical Running Time: 60 min.

7. Unusual Features of the Program: Number of spatial points used in every zone is proportional to its thickness which is variable with outer iteration.

8. Related and Auxiliary Programs: The problem solved is a nonlinear one; therefore, the algorithm is analogous but not identical with standard SN.

9. Status: In use.

10. Machine Requirements: 28K words core; no peripheral storage devices are required.

11. Programming Language Used: FORTRAN-1900.

12. Operating System on Monitor Under Which Program is Executed: Supervisor and, in particular, George.