Computer Code Abstract

FOCUS

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1. Program Name and Title: FOCUS, a versatile nonmultigroup adjoint Monte Carlo neutron transport code.

2. Problem Solved: The FOCUS program[1] enables the calculation of any quantity related to neutron transport in reactor or shielding problems, but was especially designed to calculate differential quantities, such as point values at one or more of the space, energy, direction, and time variables of quantities like neutron flux, detector response, reaction rate, etc. or averages of such quantities over a small volume of the phase space.

   Different types of problems can be treated: systems with a fixed neutron source that may be a monodirectional source located outside the system, and eigenfunction problems in which the neutron source distribution is given by the (unknown) fundamental mode eigenfunction distribution. Using Monte Carlo methods, complex three-dimensional geometries and detailed cross-section information can be treated. Cross-section data are derived from ENDF/B, with anisotropic scattering and discrete or continuous inelastic scattering taken into account. Energy is treated as a continuous variable and time dependence may also be included.

3. Method of Solution: A transformed form of the adjoint Boltzmann equation in integral representation is solved for the space, energy, direction, and time variables by Monte Carlo methods. Adjoint particles are defined with properties in some respects contrary to those of neutrons. Adjoint particle histories are constructed from which estimates are obtained of the desired quantity. Adjoint cross sections are defined with which the nuclide and reaction type are selected in a collision. The energy after a collision is selected from adjoint energy distributions calculated together with the adjoint cross sections in advance of the actual Monte Carlo calculation. For multiplying systems, successive generations of adjoint particles are obtained. These generations die out for subcritical systems with a fixed neutron source and are kept approximately stationary for eigenfunction problems.

   Completely arbitrary problems can be handled by defining a neutron source and/or neutron detector in simple user-written subroutines. Importance sampling devices, such as splitting, Russian roulette, and path length stretching, which depend on energy and space region, are available.

4. Related Programs: ADX is a code[2] that calculates adjoint cross sections and energy distributions from ENDF/B. The ETOF program[3] composes a system data file for FOCUS with all cross-section data needed to treat a given system.

5. Unusual Features of the Program: An unusual feature of FOCUS as an adjoint Monte Carlo code is its ability to treat eigenfunction problems. An equivalent treatment of a one-velocity thermal group is introduced. Due to a strong control of the sequence of random numbers per particle history, differences in estimated quantities from two systems due to (small) differences in geometry or cross section can be calculated with relatively small standard deviation.

6. Restrictions on the Complexity of the Problem: The number of different cross-section media in a system is limited to 16. Each medium can contain at most ten different nuclides. The total number of different nuclides in the system is limited to 100. At most, nine fissionable nuclides are allowed in the system. No limits apply to the cross-section data or geometry description.

7. Computer: IBM 370.

8. Typical Running Time: Strongly dependent on the complexity of the problem, the particular quantity to be calculated, and the statistical accuracy desired. Running time may vary from ~0.5 min to several hours.

9. Programming Language: FORTRAN IV and assembler language.

10. Operating System: IBM 370 OS/VS2.

11. Machine Requirements: Core storage dependent on the complexity of the problem. In general, 256 to 320 kbytes will be sufficient. Simultaneous access to up to four files on disk or tape, depending on the options selected.

12. Availability: Source decks of FOCUS and all auxiliary programs with input and output of sample problems are available on magnetic tape from the Nuclear Energy Agency Data Bank, B.P. 9, 91190 Gif-sur-Yvette, France, together with reference reports.

13. *References:*

[1]J. E. HOOGENBOOM, "FOCUS–A Versatile Non-Multigroup Adjoint Monte Carlo Neutron Transport Code," IRI-131-77-06, Interuniversity Reactor Institute, Delft (1979).

[2]J. E. HOOGENBOOM and P. F. A. de LEEGE, "ADX–A Code to Calculate Adjoint Cross Sections from the ENDF/B File," IRI-131-77-04, Interuniversity Reactor Institute, Delft (1979).

[3]J. E. HOOGENBOOM, "ETOF–A Program to Prepare a Cross Section Data Tape from the ENDF/B File for the Adjoint Monte Carlo Code FOCUS," IRI-131-77-05, Interuniversity Reactor Institute, Delft (1979).