NUCLEAR SCIENCE AND ENGINEERING: 26, 283-284 (1966)

Computer Code Abstracts

FCC

A Fundamental Mode Code for Fast Reactor Analysis*

1. Name of Code: FCC[1]
2. Computer for Which Code is Designed: UNIVAC 1107. Programming Language: FORTRAN IV.
3. Nature of Code: FCC is a multipurpose data manipulation code for use in fast reactor analysis. The code can be used to:

   a) compute resonance-shielded cross sections using data in the Russian format (shielding factors and infinite-dilution cross sections)[2]

   b) compute multigroup fundamental-mode flux and adjoint flux

   c) compute and punch group collapsed microscopic or macroscopic cross sections in DTF format

   d) compute fuel burnup.
4. Restrictions on Complexity: The code will handle up to 26 energy groups with 14 isotopes in a 64 K memory.
5. Running Time: About 15 sec/case on a UNIVAC 1107.
6. Unusual Features: An extensive cross check (for consistency) of the input data is performed. By these checks, numerous printing errors in the Russian data compilation[2] have been detected.
7. Status: In use.
8. Material Available: The FORTRAN source deck and the Russian data compilation are available from Battelle-Northwest.
9. *References:*

[1]W. W. Little, Jr. and R. W. Hardie, "FCC - A Fundamental Mode Code for Fast Reactor Analysis," BNWL-234, Battelle-Northwest Laboratory (March 1966).

[2]I. I. Bondarenko et al., *Group Constants for Nuclear Reactor Calculations*, Consultants Bureau, New York (1964).

W. W. Little, Jr.
R. W. Hardie

Battelle Memorial Institute
Pacific Northwest Laboratory
Richland, Washington

Received April 18, 1966

*Work performed under the auspices of the USAEC.

GAZED

1. Name of Code: GAZED[1]
2. Computer for Which Code is Designed: IBM 7040/7044 and IBM 7090/7094.
   Programming System: FORTRAN IV including MAP. Monitor: 7040/7044 IBSYS System, Version 9/Mod 3 and IBM 7090/7094 IBSYS Operating System, Version 8.
3. Nature of Problem Solved: Reactor depletion utilizing one-dimensional, multigroup, neutron diffusion theory and a semi-analytic method for the calculation of nuclear burnup.
4. Method of Solution: Neutron diffusion equations described by highly accurate difference equations having 4'th-order error at interior points and 3'rd-order error at material discontinuities. Source iterations using Chebyshev polynomials. Flux iterations based on splitting scatter-transfer matrix into down- and up-scatter parts. Provision made for accelerating the flux iterations via a polynomial method. Criticality searches based on first-order perturbation theory. The nonlinear reactor depletion equations are solved by a semi-analytic method based on the use of perturbation theory to solve nonlinear differential equations. There results a sequence of successive approximations (outer burnup iterations). The atom density iterates are determined analytically. However, the source terms are determined via numerical quadrature.
5. Restrictions on Complexity:
   Problem size limitations: There is a maximum size problem, but there is no simple set of relationships to describe it. There are no fixed dimensions of the FORTRAN type and each of the links of the program is optimized to maximize the available storage. The program will exit from the computer with an error comment, if a greater-than-maximum GAZED problem is attempted, and it can then be broken into a sequence of less-than-maximum problems by choosing a smaller number of time points or load periods.
   Machine requirements: 32K memory, 5-channel C tapes, and 4-channel B tapes.
6. Typical Running Times: 18 min plus 2.4 min for the edit for a fast gas-cooled reactor (FGCR) problem. This problem was run straight burnup (i.e., no criticality search) and only one outer burnup iteration was performed. Six main time points (i.e., flux calculations performed) were used, along with 5 subintervals per time step (i.e., flux renormalized to get desired power). Dimensions were 10 group (down-scatter only), 10