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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).

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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

materials, 43 space points, and 32 nuclides. For the high-temperature gas-cooled reactor (HTGR), a typical problem took 12.4 min plus 2.4 min for the edit. This problem was run straight burnup and 3 outer burnup iterations were performed. Three main time points were used along with one constant coefficient subinterval per time step and 4 variable coefficient subintervals. Dimensions were 4 groups (down-scatter only), 17 materials, 75 space points, and 25 nuclides.

7. Unusual Features: All the features of GAZE[2], including scatter transfers between all groups, criticality searches, and an extensive edit. A semi-analytic eigenvalue method is used to calculate the nuclear burnup. The result of this method is that generally a much smaller number of time points is required than for the usual numerical (nonanalytic) solution of the nuclear burnup equations. Only enough time points are required to describe certain slowly varying source terms and, if any, control-rod volume-fraction variations. Depletion is calculated for each material composition (a material may fill only one space interval, if desired). Other unusual features are: time-dependent transverse bucklings (if desired); variation by nuclide of the fission spectrum; $(n,2n)$ transfers allowed between all energy groups; a table of disadvantage factors for each nulide (if desired); (n,γ), (n,α), (n,p), and fission cross sections are treated as district cross sections; multiple cross-section sets may be given for each nuclide to describe spacial variation due to temperature, etc.; time-dependent reactor power (if desired); diffusion calculations may be search only, search-hot-cold, or hot only; multiple load periods (if desired), each consecutive pair of load periods separated by a reload interval, during which interval control rods, fuel elements, etc., may be relocated, removed, or introduced; depletion scheme described in very general terms allowing predecessors for β-decay, α-decay, two (n,γ) processes, (n,α), (n,p), and $(n,2n)$; a very flexible procedure is allowed whereby a previously run problem may be restarted with changes in the original input data.

8. Present Status: Production. Requests should be submitted to: Dr. G. C. Pomraning, General Atomic Division, General Dynamics Corporation, P. O. Box 608, San Diego, California 92112.

9. *References:*

[1]S. R. Lenihan and R. M. Wagner, "GAZED - A Reactor Burnup Code for the IBM 7040/7044 and the IBM 7090/7094 Which Utilizes One-Dimensional, Multigroup, Neutron Diffusion Theory and an Eigenvalue Method for the Calculation of Nuclear Burnup," GA-7052, General Atomic Division, General Dynamics Corp. (March 25, 1966).

[2]S. R. Lenihan, "GAZE-2, A One-Dimensional, Multigroup, Neutron Diffusion Theory Code for the IBM-7090," GA-3152, General Atomic Division, General Dynamics Corp. (August 3, 1962).

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