Computer Code Abstract

PHENIX

A Two-Dimensional Diffusion-Burnup-Refueling Code

1. Name of Code: PHENIX, A Two-Dimensional Diffusion-Burnup-Refueling Code.[1]

2. Computer for Which Code is Designed: CDC 6600. Programming Language: FORTRAN IV

3. Nature of Physical Problem Solved: PHENIX is a two-dimensional multigroup diffusion-burnup-refueling code for use with fast reactors. The code is designed primarily for fuel-cycle analysis of fast reactors and can be used to calculate the detailed burnup and refueling history of fast-breeder-reactor concepts having any generalized fractional batch reloading scheme. Either ordinary k_{eff} calculations or searches on material concentration or region dimensions can be performed at any time during the burnup history. The complete fuel-cycle history, i.e., the entire series of burnup intervals, can be calculated in one run, or each burnup interval can be analyzed separately. The refueling option of the code accounts for the spatial flux shifts over the reactor lifetime in the calculation of fuel discharge. Effects of this flux shift were presented recently,[2] and are discussed in detail in Ref. 3. For each burnup interval in the fuel-cycle history, the code performs the following operations:

   a. calculates, for each fuel isotope in each zone, both the atom density of the fuel fraction to be discharged and the atom density following the refueling

   b. calculates, for each fuel isotope, the total charge and discharge (in kg) of the entire reactor

   c. collapses the atom-density data for any number of regions, e.g., all the radial or axial blanket regions, into a single total charge and discharge

   d. punches on cards the charge-discharge mass balances for input to a reactor economics code (if desired).

   All of these calculations reflect the spectral and spatial-flux shifts from one burnup interval to another.

4. Method of Solution: Eigenvalues are computed by standard source-iteration techniques, with group-rebalancing, successive line over-relaxation, and fission source over-relaxation used to accelerate convergence. These methods are used in the two-dimensional diffusion theory code 2DB[4] and are incorporated in PHENIX. However, several basic differences exist between the 2DB methods and those used in PHENIX. In PHENIX, a sinusoidal initial flux guess can be used in which the code generates the appropriate values for the flux at each mesh point for any combination of reflective and vacuum boundary conditions. Additionally, in PHENIX, the line inversion can be performed by rows (radial), columns (axial), or by alternating the direction from one mesh sweep to the next. Based on experiments with different core geometries and different combinations of boundary conditions, the code will determine the "best" direction by considering the boundary conditions together with the average axial and radial mesh spacing. The concentration search calculation has also been changed to include the simultaneous addition or removal of any combination of materials in any combination of reactor zones. The performance of convergence tests and calculation of new eigenvalues in search problems are based on techniques used in the Los Alamos S_n codes DTF-IV[5] and 2DF.[6]

   Burnup is performed by PHENIX using zone-averaged total fluxes and zone- and group-averaged cross sections as in 2DB. Each input burnup time step is arbitrarily divided into ten smaller time steps, and the burnup equation is then solved as a march-out problem using the smaller time steps. A constant total power constraint is used to adjust the magnitude of the fluxes at the end of each subdivided time step.

   With the fractional batch refueling scheme used in PHENIX, the fuel fraction with the greatest burnup is discharged. This discharge is calculated by actually burning initially clean fuel over its period of residence in the reactor using the appropriate zone-averaged total fluxes and zone-group-averaged cross sections from previous burnup intervals. Refueling is then accomplished by subtracting the discharge-atom density from the homogenized-region atom density and adding the appropriate clean-fuel atom density. The principal advantage of this refueling technique is the requirement to explicitly tag each fuel isotope only once per region.

5. Restrictions on Complexity: Variable dimensioning is used to make maximum use of the computer fast memory. Nearly all subscripted variables are stored in a single array limited to $30\,000_{10}$ words. The present version of the code is also limited to 50 energy groups. Because the fluxes for each energy group are calculated successively, storage requirements in the $30\,000_{10}$ array mentioned above are relatively insensitive to the total number of energy groups.

6. Representative Running Times: A straight k_{eff} calculation in R-Z geometry with eight energy groups and 900 mesh points requires 40 sec running time on the CDC 6600. If the same problem is carried out over the complete cycle of the code, e.g., Initial Search →

Burnup → Final k_{eff} → Refuel, for the first burnup interval (clean reactor, sinusoidal flux guess), the running time is ~75 sec. Each subsequent burnup interval requires 60 to 65 sec for the same calculational sequence. The number of burnup intervals required to reach equilibrium is a direct function of the particular fractional batch refueling scheme. Thus, if the reactor requires five burnup intervals to reach equilibrium from the clean configuration, the total running time is between five and six minutes.

7. Related and Auxiliary Programs: Mesh-point and atom-density data can be generated by DPC,[7] a two-dimensional data preparation code, and input directly to PHENIX. This is particularly useful in the case of complex problems. Cross sections may be input via either cards or tape, and the format is compatible with MC^2 output[8] as well as that of other cross-section preparation codes.

8. Status: In use at LASL and at Air Force Weapons Laboratory (Kirtland AFB). Complete code package will be available in the Argonne Code Center by January 1, 1970.

9. Machine Requirements: A 65K memory, one random-access storage device (disk), and two magnetic-tape units. If cross sections are input from cards, only one tape unit is required. The other tape unit is used for flux guesses or dumps (if desired), and to store burnup data needed for the refueling portion of the program.

10. Operating System: The program is constructed within the SCOPE 3.2 Operating System of the CDC 6600.

11. Material Available: A source deck and listing and sample problem are presently available from the authors. The LA report on the code (Ref. 1, in press) will be ready by January 1970.

12. *References*

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[2] THOMAS J. HIRONS and R. DOUGLAS O'DELL, *Trans. Am. Nucl. Soc.*, **12**, 38 (1969).

[3] THOMAS J. HIRONS and R. DOUGLAS O'DELL, "Calculational Modeling Effects on Fast Breeder Fuel-Cycle Analysis," LA-4187, Los Alamos Scientific Laboratory (1969).

[4] W. W. LITTLE, Jr. and R. W. HARDIE, "2DB User's Manual," BNWL-831 Rev. 1, Battelle Northwest Laboratory (1969).

[5] K. D. LATHROP, "DTF-IV, A FORTRAN-IV Program for Solving the Multigroup Transport Equation with Anisotropic Scattering," LA-3373, Los Alamos Scientific Laboratory (1965).

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[7] W. H. HANNUM and B. M. CARMICHAEL, "DPC, A Two-Dimensional Data Preparation Code," LA-3427-MS, Los Alamos Scientific Laboratory (1965).

[8] B. J. TOPPEL, A. L. RAGO, and D. M. O'SHEA, "MC^2, A Code to Calculate Multigroup Cross Sections," ANL-7318, Argonne National Laboratory (1967).

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