

Computer Code Abstracts

M0176

1. Name of code: M0176
2. Computer for which code is designed: Philco-2000
Programming system: The subset of ALTAC which corresponds approximately to FORTRAN II. A 32K memory is required. No tapes are used except for input/output.
3. Nature of the problem solved: This program solves several P -approximations to the few group neutron transport equation in slab geometry. Fast groups are $P-1$ or $P-3$, with $P-0$, $P-1$, and $P-2$ transfer matrices, while thermal groups are $P-1$ or double $P-1$, with $P-0$ and $P-1$ transfer. A $P-1$ variational thermal treatment with two trial functions is included as an option. Boundary conditions are general. Eigenvalue, slowing down, or purely thermal problems are done. Strategy of solution is quite variable.

A $P-0$ source, either a fission neutron source or a source to thermal, can be input pointwise or as regionwise constant, and, for a purely thermal problem, a $P-1$ source to thermal can be input pointwise. Edits available include the $P-0$, $P-1$, $P-2$, and $P-3$ components of the flux at every mesh point in every group, the total thermal $P-0$ flux, the converged fission source (if an eigenvalue calculation has been done), fast and thermal pointwise activations of two arbitrary sets of compositionwise cross sections, and various region integrals.

Each group is solved iteratively, as in the FLIP code, with the requestor having the choice of iterating that group to convergence or leaving it after a specified number of iterations. Fission source iterations are done as in WANDA. Since, for a thin slab, the convergence of the few thermal groups is slow relative to the convergence of the fission source iterations, provision is made for iterating the thermal groups to convergence or a specified maximum number of times in each fission source iteration. The variational thermal groups are solved as ordinary thermal groups with current transfer.

4. Restrictions on the complexity of the problem: 250 points, 15 regions, 6 compositions, 6 fast groups, 6 thermal groups, 10 groups total.
5. Typical running times: A thin slab (fast convergence of fission source iterations) with a water reflector (slow convergence of thermal iterations) has been analyzed with 6 fast $P-3$ group and 4 thermal double $P-1$ groups using 90 points. The running time was 6 min.
6. Availability: in production at Bettis Atomic Power Laboratory. Copies of this program may be obtained from Mr. Robert A. Cohen, Manager, Customer Services, Philco Corporation, Government and Industrial Group, Computer Division, 3900 Welsh Road, Willow Grove, Pennsylvania.

7. References: 1. B. L. Anderson, J. A. Davis, E. M. Gelbard, and P. H. Jarvis, FLIP—An IBM-704 code to solve the P_L and double P_L equations in slab geometry. WAPD-TM-134 (March 1959).
2. O. J. Marlowe and M. C. Suggs, WANDA 5—A one-dimensional neutron diffusion equation program for the Philco-2000 computer. WAPD-TM-241 (November 1960).
3. R. M. Cantwell, M0176—A Fortran program to solve several P -approximations to the few group neutron transport equation in slab geometry. WAPD-TM-320 (April 1962).

R. M. CANTWELL
Bettis Atomic Power Laboratory
Pittsburgh, Pennsylvania

M0150

1. Name of code: M0150
2. Computer for which code is designed: Philco-2000.
Programming system: The subset of ALTAC which corresponds approximately to Fortran II. A 32K memory is required. No tapes are used except for input/output.
3. Nature of the problem solved: The double $P-3$ approximation to the one-velocity one-dimensional transport equation is solved in slab geometry. This is a FORTRAN version of the double $P-3$ option of the IBM-704 FLIP code. It accepts a pointwise isotropic source and anisotropic scattering up to the $P-2$ component. The second order differential equations are written as difference equations, which are solved repeatedly, one equation at a time, until convergence is obtained. No acceleration is used in the iterative solution since the second order differential equations have been transformed in such a way as to make the iteration process rapidly convergent.
4. Restrictions on the complexity of the problem: 50 regions, 10 compositions, 500 points.
5. Typical runningtimes: Less than 1 min for most problems.
6. Availability: in production at Bettis Atomic Power Laboratory. Copies of this program may be obtained from Mr. Robert A. Cohen, Manager, Customer Services, Philco Corporation, Government and Industrial Group, Computer Division, 3900 Welsh Road Willow Grove, Pennsylvania.
7. References: B. L. Anderson, J. A. Davis, E. M. Gelbard, and P. H. Jarvis, FLIP—An IBM-704 code to solve the P_L and double- P_L equations

in slab geometry. WAPD-TM-134 (March 1959).

R. M. Cantwell, M0150—A Fortran program to solve the double P -3 equations in slab geometry. WAPD-TM-315 (April 1962).

R. M. CANTWELL
*Bettis Atomic Power Laboratory
 Pittsburgh, Pennsylvania*

EQUIPOISE-3-A

1. Name of code: EQUIPOISE-3-A
2. Computer for which code is designed: IBM-7090
 Programming system: FORTRAN
3. Nature of problem solved: EQUIPOISE-3-A is a slightly revised version of EQUIPOISE-3
 (See abstract, *Nuclear Sci. and Eng.* **13**, 63 (1962)).
 The differences between EQUIPOISE-3 and 3-A are:
 - a. In addition to the standard EQUIPOISE-3 output, a picture is printed of the arrangement of materials in the reactor. This picture is similar to that produced by PDQ with the additional feature that should the material arrangement be in error, points in the

mesh which are not covered by a material are indicated in the picture by an asterisk.

- b. If the adjoint option is used, EQUIPOISE-3-A provides as additional output the prompt neutron lifetime and a list giving the reactivity per unit change in each group constant in each region of the reactor.
4. Restrictions: Same as for EQUIPOISE-3 with the exception that if the number of different materials in the reactor exceeds 35 no picture will be printed.
5. Typical running time: Same as for EQUIPOISE-3.
6. Present status: In use, available.
7. *References*: 1. T. B. Fowler and M. L. Tobias, EQUIPOISE-3: A two-dimensional, two-group neutron diffusion code for the IBM-7090 computer. ORNL-3199 (December 1959).
 2. C. W. Nestor, Jr., EQUIPOISE-3-A, ORNL-3199, Addendum (June 1962).
 C. W. NESTOR, JR., T. B. FOWLER,
 AND M. L. TOBIAS
*Oak Ridge National Laboratory**
Oak Ridge, Tennessee

* Operated by Union Carbide Corp. for the U.S.A.E.C.

Note

Volume **14**, No. 2, in the Letter to the Editor entitled "Two Regimes of Burnout (DNB) Correlated with Steam Energy Flow for Uniformly-Heated Channels," by William J. Levedahl, pp. 201-203, the work outlined was performed at the General Nuclear Engineering Corporation, Dunedin, Florida, under contract with the East Central Nuclear Group.