Conf-731207--1

ETOE-2/MC^2-2/SDX Multigroup Neutron Cross-Section Processing

by H. Henryson, II, B. J. Toppel, and C. G. Stenberg

Presented at:
Seminar on Codes for Nuclear Data Processing,
Ispra, Varese, Italy, December 5-7, 1973

ARGONNE NATIONAL LABORATORY, ARGONNE, ILLINOIS

ETOE-2/MC^2-2/SDX Multigroup Neutron Cross-Section Processing*

H. Henryson, II, B. J. Toppel, and C. G. Stenberg
Argonne National Laboratory, Argonne, Illinois 60439, U.S.A.

The ETOE-2/MC^2-2/SDX code system provides a comprehensive neutron cross-section processing capability. The three components of the system include an ENDF/B library processor (ETOE-2), an ultra-fine-group fundamental mode calculation (MC^2-2), and an intermediate-group space-dependent capability (SDX). In this paper brief descriptions of the types of calculations are discussed. Selected results from typical problem executions are presented.

INTRODUCTION

The ETOE-2/MC^2-2/SDX multigroup cross-section processing codes have been designed to provide a comprehensive neutron cross-section generation capability for a wide range of applications including critical experiment analysis and core and shielding calculations. Fundamental nuclear data provide the primary input to the code system and the output consists of a user-specified multigroup cross-section data file. Great flexibility is provided to the user in specifying the rigor of a calculation, thus providing a unified cross-section processing system with a single data base which may be used for both survey and design calculations.

The principal program blocks of the code system include a library processor (ETOE-2), an ultra-fine-group fundamental mode calculation (MC^2-2) and an intermediate-group space-dependent capability (SDX). A block diagram indicating the general program flow is given in Fig. 1. Brief explanations of the physics methods and models incorporated in the ETOE-2/MC^2-2/SDX code system are presented below.

ETOE-2

The ETOE-2 program processes the fundamental nuclear data from an ENDF/B [1] data file and prepares eight binary library files for use by the computational modules of MC^2-2/SDX. Neither MC^2-2 nor SDX reads the ENDF/B data directly. Thus, one could prepare MC^2-2/SDX library files from an alternative input data base (e.g. UKNDL, KEDAK) by either conversion of the basic data to the ENDF/B formats or by the replacement of the ETOE-2 code with a new processing code which generates the MC^2-2/SDX libraries.

The ETOE-2 program performs five basic functions: (1) reformat data; (2) process "light" element ($A \leq 100$) resonance cross sections; (3) generate ultra-fine-group "floor" cross sections; (4) calculate function tables: and (5) convert all ENDF/B formats to laws which are allowed by MC^2-2/SDX.

The ENDF/B data files provide all data for a given material, whereas the MC^2-2/SDX computational modules require that the data be ordered by reaction type (e.g. resolved resonance parameters, scattering coefficients, etc.). The MC^2-2/SDX library structure has been designed to permit the efficient access to data by the computational modules. The eight data files in the MC^2-2/SDX library include resolved resonance parameters,

*Work performed under the auspices of the U. S. Atomic Energy Commission.

unresolved resonance parameters, ultra-fine-group nonresonance cross sections, inelastic and (n,2n) scattering data, fission spectrum parameters, and elastic scattering distributions, as well as function tables and an administrative file.

At user option the ETOE-2 code calculates resonance cross sections from ENDF/B resonance parameters for all materials of mass less than an input value. Generally a mass of 100 is used. These "light" element resonance cross sections are then combined with the ENDF/B "floor" cross sections and integrated over ultra-fine-group energy boundaries ($\Delta u \leq 0.008$) to provide the group cross sections required by MC^2-2/SDX. It is assumed that "light" element resonance cross sections are composition-independent on the ultra-fine-group level.

The ENDF/B formats permit a large number of options in describing the fundamental data. Many of these options have not been used in the three releases of ENDF/B to date and only a subset of the allowed ENDF/B laws are processed by MC^2-2/SDX. It is, therefore, necessary for the ETOE-2 code to process data given by any of the other laws and prepare these data in a format permitted by MC^2-2/SDX. Some examples may serve to illustrate the types of processing required by ETOE-2. The MC^2-2/SDX resolved resonance algorithms assume either a single level Breit-Wigner or a multilevel Adler-Adler description whereas ENDF/B also permits R-Matrix (Reich-Moore) and multilevel Breit-Wigner parameters. It is well known that equivalent multilevel Adler-Adler parameters may be derived from these models [2] and these equivalent parameters are calculated by ETOE-2. Similarly, ENDF/B permits six secondary energy distribution laws for inelastic and (n,2n) scattering, whereas MC^2-2 permits only three. The ETOE-2 code generates an arbitrary tabulated function if data are provided for any of the three laws not processed by MC^2-2. Similar examples may be cited in the processing of elastic scattering distributions and fission spectrum data. In general, the format conversions performed by ETOE-2 do not alter the basic data input on the ENDF/B files.

The ETOE-2 program thus provides an automated capability for the generation of MC^2-2/SDX library files from ENDF/B data. It performs the same functions for MC^2-2/SDX as the ETOE [3] and MERMC2 [4] codes perform for the MC^2 [5] cross-section preparation program. Since the library files generated by ETOE-2 are not composition dependent, the program need be executed only when new fundamental data become available (e.g. each release of ENDF/B). A limited capability is available to modify the data in the MC^2-2/SDX libraries, thus obviating the need to rerun ETOE-2 in order to study the sensitivity of multigroup cross sections to changes in fundamental data. A more extensive library modification capability is planned.

MC^2-2

Since its introduction in 1967, the MC^2 code has been used extensively for the preparation of multigroup cross sections. However, a number of recognized limitations and inconsistencies in the code pointed out the need for a new capability which could serve as a standard for fast reactor calculations. The MC^2-2 code has been developed to satisfy this need. Recent advances in neutron slowing-down theory, resonance theory, and numerical methods have been incorporated into the ultra-fine-group fundamental MC^2-2 calculation. The code development has generally proceeded according to the

implementation strategy outlined in Ref. 6 although a number of improvements and extensions have been included in the capability during the implementation and verification phases of the project. A block diagram indicating the flow through the computational modules of MC²-2 is given in Fig. 2.

The MC²-2 code solves the neutron slowing-down equations in the P_1, B_1, consistent P_1, and consistent B_1 approximations. The equations have the form [7]

$$\Sigma\phi_1 + \Sigma\phi_0 = S + \int \Sigma_s^0(u' \to u)\phi_0(u')\,du' \tag{1}$$

$$\frac{\Sigma}{3}\phi_0 + A_1\phi_1 = \int \Sigma_s^1(u' \to u)\phi_1(u')\,du' \tag{2}$$

$$\phi_L = -\frac{L}{2L+1}\frac{\Sigma\phi_{L-1}}{A_L}, \qquad L = 2 \ldots N, \tag{3}$$

where the extended transport coefficients A_L, which depend on the order and type of approximation, account for high-order transport and anisotropic scattering effects. If the elastic slowing-down density is defined as

$$q_L(u) = \int_{u-\epsilon}^{u} du'\,\Sigma_s^L(u')\phi_L(u')g^L(u,u'), \tag{4}$$

and the collision density is assumed to be slowly varying

$$\Sigma\phi_L(u') \cong \Sigma\phi_L(u) + (u' - u)\frac{d}{du}\left[\Sigma\phi_L(u)\right],$$

then Eqs. (1) and (2) may be written [8]

$$\frac{1}{N(u)}\left[\left(\Sigma - \Sigma_s^0\right)\left(A_1 - \Sigma_s^1 + \epsilon_1\xi_1\Sigma\right) + \frac{B^2}{3}\right]\left(\epsilon_0 q_0 + S\right)$$

$$+ \frac{B}{N(u)}\epsilon_0\xi_0\epsilon_1\xi_1 q_1 + \frac{dq_0}{du} = S \tag{5}$$

$$\frac{B}{3N(u)}\epsilon_1\xi_1\left(\epsilon_0 q_0 + S\right) + \frac{\epsilon_1 q_1}{N(u)}\left[\left(A_1 - \Sigma_s^1\right)\left(\Sigma - \Sigma_s^0 + \epsilon_0\xi_0\Sigma\right) + \frac{B^2}{3}\right] + \frac{dq_1}{du} = 0 \tag{6}$$

$$N(u) = \frac{B^2}{3} + \left(A_1 - \Sigma_s^1 + \epsilon_1\xi_1\Sigma\right)\left(\Sigma - \Sigma_s^0 + \epsilon_0\xi_0\Sigma\right). \tag{7}$$

The multigroup equations, (1) and (2), and the continuous slowing-down equations, (5) and (6), are solved on an ultra-fine-group lethargy mesh as depicted in Fig. 3.

The energy boundary between the multigroup and continuous slowing-down formulations is user-specified but must lie above the top of the resolved resonance energy region. This is a consequence of the resonance treatment discussed below. The moderating parameters, ξ_i and ϵ_i, in the continuous slowing-down formulation may be calculated using either Greuling-Goertzel [9] or Improved Greuling-Goertzel [7] algorithms.

The source term, S, in Eqs. (1), (5), and (6), represents contributions from inhomogeneous, fission, inelastic, and (n,2n) ultra-fine-group sources. Only elastic scattering is treated continuously in the continuous slowing-down formulation.

The resolved and unresolved resonance calculations of MC^2-2 are modeled after the work of Hwang [10] and represent a marked improvement in both accuracy and efficiency over the methods incorporated in the MC^2 code. The resonance algorithms make use of a generalized J*-integral formulation based on the narrow resonance approximation including overlap effects. The characteristics of these algorithms may be summarized briefly as follows:

(1) The integration procedure was optimized by utilizing the asymptotic properties of the integrands and the general characteristics of the Gauss-Jacobi quadrature. This was achieved by introducing a rational transformation of the variable of integration.

(2) For the relatively weak resonances which represent a significant portion in practical calculations, the J*-integral is evaluated analytically.

(3) For the resolved energy range, the new algorithms allow the use of the multilevel formalism in Adler-Adler form [11].

(4) For the unresolved energy range, the new algorithms provide an accurate estimate of the in-sequence overlap effect which accounts for the long-range correlation of levels described by Dyson [12], and include the influence of interference scattering.

As shown in Ref. 13, the J*-integral method provides an efficient means of accounting for resonance effects in the continuous slowing-down formulation. In particular, the continuous slowing-down equations, (5) and (6), may be solved for the "asymptotic" neutron slowing-down density ignoring narrow resonances. Then the resonance reaction rates are computed using the flux resulting from the asymptotic slowing down density attenuated by absorption in higher energy resonances. It is not difficult to show that

$$q(u_r + \cdot) = q_{as}(u_r) Q_r (1 - P_r) , \tag{8}$$

where

$$Q_r = \prod_{r'} (1 - P_{r'}) , \qquad u_{r'} < u_r \tag{9}$$

and

$$P_r = \left[\frac{\Gamma_{a,r}}{E_{0,r}} J^*_a - \frac{\Sigma_a}{\Sigma_t} \frac{\Gamma_{t,r}}{E_{0,r}} J^*_t \right] \frac{\Sigma_t \epsilon_{as}}{q_{as}} . \tag{10}$$

The ultra-fine-group flux derived from the attenuated slowing-down density may be used in the generation of broad-group cross sections by standard group-collapsing methods.

As noted above, the entire MC²-2 calculation is performed on an ultra-fine-group mesh. Inelastic and (n,2n) secondary energy distributions may be described by discrete levels, evaporation spectra and/or tabulated functions according to the ENDF/B specifications. Detailed angular distributions are used in calculating ultra-fine-group P_0 and P_1 elastic transfer matrices for all materials. For light elements, an analytic integration over the sink group is combined with a detailed numerical integration over the source group [14]. Heavy element transfer matrices are calculated semi-analytically according to the methods described in Ref. 15.

Options available to the user of MC²-2 include inhomogeneous group-dependent sources, group-dependent buckling, buckling search to critical, and isotope-dependent fission spectrum distributions. The user-specified cross-section file generated by MC²-2 is appropriate for neutronics calculations (up to 50 groups) or for use in intermediate group (50-300 groups) spectrum calculations. In particular, the intermediate group cross-section file may be used in the SDX capability described below. High-order Legendre moments of elastic transfer matrices may be calculated using a weighting spectrum generated recursively according to Eq. (3) and the numerical methods described in Ref. 16.

A hyper-fine-group ($\Delta u \approx 0.0002$) integral transport capability is also available at user option. The integral transport calculation is modeled after the RABID [17] and RABBLE [18] codes although significant modifications have been made to (1) eliminate precision difficulties; (2) make use of the MC²-2/SDX library data; (3) link the hyper-fine-group calculation to the ultra-fine-group calculation through the slowing-down source; and (4) increase the overall efficiency of the codes. This MC²-2 option is intended for use in the low-energy region (≤300 eV) where the narrow resonance approximation assumed in the J^*-integral formulation is known to break down. The upper energy for the integral transport calculation is user specified.

In order to compare the MC² and MC²-2 programs, a sample problem based on a typical ZPPR core configuration has been run on the IBM-350/195 at Argonne National Laboratory. The MC²-2 user options have been chosen to permit a direct comparison of the two codes. Problem specifications are given in Table I. The MC² code uses both a fine- and ultra-fine-group energy grid in the spectrum calculation. The fine-group structure is used to treat heavy element elastic scattering and inelastic and (n,2n) transfer as well as other slowly varying data. Studies have shown that MC² results are sensitive to the user-specified, fine-group lethargy width [19]. In running the sample problem, a fine-group width of 0.1 was therefore used instead of the more common one-fourth lethargy.

The ultra-fine-group spectra from MC²-2 and MC² are given in Figs. 4 and 5 respectively. It can be noted that the overall spectrum agreement is quite good. In the low energy region, below 4 keV, one observes considerably more structure in the MC²-2 spectrum. This is a consequence of both the different resolved resonance treatments and the detailed treatment of elastic scattering in MC²-2. Between 4 keV and 1 MeV where the scattering resonances of intermediate mass isotopes dominate the spectrum, the agreement is excellent. Above 1 MeV, the MC²-2 spectrum again shows greater detail

as a result of the ultra-fine-group treatment of inelastic and (n,2n) scattering. Fig. 6 shows this part of the spectrum in greater detail. The MC²-2 eigenvalue for this problem (0.9979) falls between the MC² ultra-fine-group (0.9964) and broad-group (0.9990) values. The two MC² eigenvalues differ because of the special resonance cross-section collapsing algorithm.

Timing comparisons for the two problems are shown in Table II. The MC²-2 calculation requires less than one-third the total computing time, one-sixth the CPU time, and one-half the core storage of the MC² calculation. Most of the time savings may be attributed to the speed of the resolved resonance calculation. The increased time required for the ultra-fine-group spectrum calculation is due to the inelastic and (n,2n) source calculation and other timing differences are due to increased programming efficiency and more rigorous numerical and physics models as discussed above.

SDX

A space-dependent cross-section generation capability is provided by the SDX option of the ETOE-2/MC²-2/SDX code system. The SDX option may be characterized by three salient features:

(1) the use of an intermediate-group microscopic cross-section library for all cross sections except those representing heavy element resonances;
(2) run time computation of intermediate-group heavy element resonance cross sections appropriate to the composition, cell structure, and temperature of the problem; and
(3) explicit treatment of all heterogeneity and multiregion spatial effects in one dimension.

The SDX option has been designed to provide the user great latitude in the rigor, complexity, and computational effort associated with a given problem. For example, it is possible, for each region in a multiregion problem, to obtain the intermediate-group cross sections, perform the resonance cross section and integral transport calculation for a heterogeneous cell model, and homogenize the intermediate-group cross sections. The homogenized cross sections for each region would then be used in a multiregion, intermediate-group diffusion theory calculation, and the resulting flux used to calculate broad-group spatially averaged cross sections on a cell-averaged and plate (pin)-wise basis. The rigor, and computational effort, of such a calculation could be relaxed by using a single set of intermediate-group cross sections for all regions, but still generating the heavy element resonance cross sections for each different region. The integral transport calculation could be omitted and volume-averaging used in the cell homogenization with or without the use of equivalence theory to account for heterogeneous effects in the calculation of heavy element resonance cross section. A block diagram indicating the major program modules of SDX is given in Fig. 7.

At its most rigorous the SDX calculation is more rigorous, yet significantly more economical, than the most rigorous previously existing fast neutron multigroup cross-section preparation capability. On the other end of the spectrum, the simplest SDX calculation provides in less than 2 min a broad-group cross-section set which should be

adequate for many analyses. In particular, the latter option should be an improvement upon the self-shielding factor schemes because of the more accurate resonance cross-section treatment.

Broad-group microscopic cross sections are composition-dependent because of the composition-dependence of the neutron flux (and current) weighting spectrum. Elastic removal and heavy element resonance cross sections are generally the most sensitive to composition due to intermediate element scattering resonances and heavy element resonances. In the SDX option the heavy element resonance cross sections are calculated on an intermediate-group level for each plate or pin type (using equivalence theory) or homogeneous mixture in each region of a multiregion problem. As Fig. 1 indicates, the resonance calculation in SDX uses the same program modules as MC^2-2. Intermediate-group resonance cross sections are calculated assuming a constant collision density per unit lethargy in SDX rather than by use of the attenuation treatment described earlier in MC^2-2. Thus the resonance algorithms employed in the SDX calculation combine a high degree of accuracy, when the narrow resonance approximation is valid, with modest computational time. It is assumed in the SDX option that all the remaining cross sections are composition-independent on the intermediate-group level and the intermediate-group spectrum will adequately reflect the composition-dependence for the purpose of obtaining broad-group cross sections. For current applications, intermediate-group cross-section libraries on the order of 150 to 200 groups have been generated. These libraries adequately "trace out" the scattering resonances in lighter elements such as oxygen and sodium and the lower energy resonances of iron but are generally inadequate to "trace out" the higher energy scattering resonances in intermediate mass nuclei. The intermediate-group cross-section libraries may be constructed from MC^2-2 ultra-fine group calculations as indicated in Fig. 1 or any other code which creates a cross-section file in the proper format.

Three options exist in SDX with respect to unit cell homogenization:

(1) A homogeneous mixture may be specified in which case resonance cross sections are computed for a homogeneous mixture and simply combined with the intermediate-group library data (i.e. no unit cell homogenization).

(2) If a heterogeneous unit cell is specified, heterogeneous resonance cross sections are computed for selected isotopes in the specified plate/pin types using equivalence theory. These resonance cross sections are combined with the intermediate-group library data and an infinite slab or cylinder integral transport calculation is performed for the unit cell. Spatial self-shielding factors and cell-averaged intermediate-group cross sections are calculated. The integral transport calculation is based on a modified version of the code CALHET [20].

(3) The integral transport calculation described in Item (2) may be omitted and volume averaging used to obtain all cross sections.

The intermediate-group library data and cell-averaged resonance cross sections are input to a one-dimensional diffusion theory calculation. The space-dependent calculation employs the space-energy factorization [21] approximation optionally as a final

solution or as a means for accelerating the direct intermediate-group solution, and employs power iteration with Chebyshev acceleration. A fundamental mode option is available for space-independent solutions. A modified version of the SEFD code [22] is used for the space-dependent calculation. Broad-group microscopic cross sections are averaged over the intermediate-group spectrum and over user-specified spatial regions in multiregion problems. Cell-averaged or homogenized cross sections are generated with cross sections appropriate to individual plates/pins available on option.

The computing time required for an SDX calculation is dependent upon the user options as discussed above. Three sample problems have been run on the IBM-360/195 at Argonne National Laboratory to provide an indication of time requirements. All problems were run with a single 156-group cross-section library. Both the unit cell calculation and the diffusion theory spectrum calculation depend upon the number of intermediate groups. In Table III timing results are presented for a single-region problem in both a homogeneous and heterogeneous (12-plate) representation. The computation times for the resolved and unresolved resonance calculations increase in the heterogeneous problem since the calculation must be performed for the homogeneous mixture as well as for each resonance material in each plate. In Table IV timings are presented for a multiregion space-dependent calculation in which each region was treated homogeneously. The simplest SDX calculation is represented by the homogeneous problem of Table III. The most detailed calculation would include multiregion effects as in Table IV along with a heterogeneous cell model as in Table III.

PROGRAMMING METHODS

The ETOE-2/MC^2-2/SDX code system has been developed at Argonne National Laboratory on an IBM-360/195. All programming has been done in FORTRAN and has adhered to standards established by the Committee on Computer Code Coordination. In addition, considerable effort has been expended to produce a code which may be implemented as simply as possible on other computers. In order to test the coding procedures, a preliminary version of MC^2-2 was converted to CDC compatible FORTRAN and sent to the Los Alamos Scientific Laboratory for execution on a CDC-6600 and CDC-7600. In the conversion, use was made of the IBM to CDC system codes described in Ref. 23. Only minor problems were encountered in making the code operational on CDC equipment.

The entire code is variably dimensioned through use of the subprogram package BPOINTER [24]. All binary I/O is performed by subroutine call so that great flexibility is provided in specifying installation-dependent data management strategies. Standard routines have been used for installation-dependent functions such as timing. Extensive use has been made of comment cards in order to make the codes as self-documenting as possible (approximately one-third of the more than 50,000 source cards of MC^2-2 are comments).

As pointed out above, the input to the code system is an ENDF/B data file. Broad-group cross-section files are output by the code system in either the ARC System XS.ISO format [24] or the ISOTXS format [25] specified by the Committee on Computer Code Coordination. BCD input is prepared according to standard ARC System conventions [24].

A detailed coding and algorithm verification program is currently in progress. It is anticipated that Version 1 of the code system will be released to the Argonne Code Center by the end of 1974. The ETOE-2/MC^2-2/SDX code system can, however, be expected to continue to evolve as appropriate development, testing, and assessment of feasibility and desirability are undertaken.

REFERENCES

[1] DRAKE, M.K., Ed., Data Formats and Procedures for the ENDF Neutron Cross Section Library, National Neutron Cross Section Center, USAEC Rep. BNL-50274 (ENDF-102) (1970).

[2] DE SAUSSURE, G., PEREZ, R.B., POLLA, A FORTRAN Program to Convert R-Matrix-Type Multilevel Resonance Parameters for Fissile Nuclei into Equivalent Kapur-Peierls-Type Parameters, USAEC Rep. ORNL-TM-2599 (1969).

[3] GREEN, D.M., PITTERLE, T.A., ETOE, A Program for ENDF/B to MC^2 Data Conversion, USAEC Rep. APDA-219 (ENDF-120) (1968).

[4] PENNINGTON, et al, Service Routines for the Multigroup Cross Section Code, MC^2, USAEC Rep. ANL-7654 (1970).

[5] TOPPEL, B.J., RAGO, A.L., O'SHEA, D.M., MC^2 - A Code to Calculate Multigroup Cross Sections, USAEC Rep. ANL-7318 (1967).

[6] TOPPEL, B.J., "The new multigroup cross section code, MC^2-2," Proc. Conf. New Developments in Reactor Mathematics and Applications, March 29-31, 1971, Idaho Falls, Idaho, USAEC Rep. CONF-710302, Vol. 2 (1971) 966.

[7] STACEY, W.M., JR., Continuous slowing down theory for anisotropic elastic neutron moderation in the P_N and B_N representations, Nucl. Sci. Engng. 41 (1970) 381.

[8] STACEY, W.M., JR., Continuous slowing down theory applied to fast reactor assemblies, Nucl. Sci. Engng. 41 (1970) 381.

[9] GOERTZEL, G., GREULING, E., An approximate method for treating neutron slowing down, Nucl. Sci. Engng. 7 (1960) 69.

[10] HWANG, R.N., Efficient methods for the treatment of resonance cross sections, Nucl. Sci. Engng. 52 (1973) 157.

[11] ADLER, D.B., ADLER, F.T., "Neutron cross sections in fissile elements," Proc. Conf. on Breeding, Economics and Safety in Large Fast Reactors, USAEC Rep. ANL-6792 (1963) 695.

[12] DYSON, F.J. Statistical theory levels of complex systems, III, J. Math. Phys. 3 (1962) 166.

[13] STACEY, W. M., JR., Resolved narrow resonance reaction rates in fast reactor mixtures, Nucl. Sci. Engng. 41 (1970) 455.

[14] HENRYSON, H., II, STENBERG, C.G., TOPPEL, B.J., Calculation of Elastic Scattering Matrices, USAEC Rep. ANL-7710 (1971) 395.

[15] HENRYSON, H., II, Multigroup elastic scattering cross sections for heavy elements, Nucl. Sci. Engng. 43 (1971) 235.

[16] GELBARD, E., HENRYSON, H., II, An efficient calculation of Legendre moments of multigroup elastic scattering transfer matrices, Trans. Am. nucl. Soc. 17 (1973) 263.

[17] OLSON, A.P., RABID: An Integral Transport-Theory Code for Neutron Slowing Down in Slab Cells, USAEC Rep. ANL-7645 (1970).

[18] KIER, P.H., RABBA, A.A., RABBLE, A Program for Computation of Resonance Absorption in Multiregion Reactor Cells, USAEC Rep. ANL-7326 (1967).

[19] STACEY, W.M., JR., et al, "Studies of methods for fast neutron multigroup cross section generation and their effect upon the neutronics properties of LMFBR critical assemblies," National Topical Meeting on New Developments in Reactor Physics and Shielding, Sept. 12-15, 1972, Kiamesha Lake, New York, USAEC Rep. CONF-720901, Book 2 (1972) 221.

[20] FILLMORE, F.L., The CALHET-2 Heterogeneous Perturbation Theory Code and Application to ZPR-3-48, USAEC Rep. AI-69-13 (1969).

[21] STACEY, W.M., JR., HENRYSON, H., II, "Applications of space-energy factorization to the solution of static fast reactor neutronics problems," Proc. Conf. New Developments in Reactor Mathematics and Applications, Idaho Falls, Idaho, USAEC Rep. CONF-710302, Vol. 2 (1971) 953.

[22] STACEY, W.M., JR., HENRYSON, H., II, Application of space-energy factorization to LMFBR diffusion theory calculations, Trans. Am. nucl. Soc. 14 (1971) 205.

[23] WOODRUFF, W.L., LEE, R.H., "An IBM-360 to CDC-6000/7000 code conversion package," Proc. Conf. Mathematical Models and Computational Techniques for Analysis of Nuclear Systems, April 9-11, 1973, Ann Arbor, Michigan, USAEC Rep. CONF-730414-P2 (1973) IV-75.

[24] JUST, L.C., et al, The System Aspects and Interface Data Sets of the Argonne Reactor Computation (ARC) System, USAEC Rep. ANL-7711 (1971).

[25] CARMICHAEL, B.M., MENELEY, D.A., VONDY, D., Report of the Subcommittee on Standard Interface Files, USAEC Rep. LA-5324-MS (1973).

TABLE I. MC^2-2 Sample Problem Specifications

Mixture:	^{238}Pu, ^{239}Pu, ^{240}Pu, ^{241}Pu, ^{242}Pu, ^{235}U, ^{238}U, ^{241}Am, ^{12}C*, ^{16}O,* ^{23}Na*, ^{27}Al*, Fe*, Cr*, Ni*, Mn, Cu*, Mo*, Si*
Options:	Δu^{ufg} = 1/120, 1860 ultra-fine groups; inconsistent P_1 homogeneous spectrum; fixed B^2; 27 broad groups; single fission spectrum
MC^2 Options:	Δu^{fg} = 1/10; Legendre elastic scattering materials: *
MC^2-2 Options:	CSD below ~4 keV; improved Goertzel-Greuling moderating parameters; unresolved resonance self-overlap; four nearest resolved resonances.

TABLE II. Sample Problem Timing

Code Area	MC^2-2 CPU (sec)	MC^2-2 CPU + PP (sec)		MC^2 CPU + PP (sec)	
Input	0.4	3	(0.6%)	5	(0.3%)
Unresolved resonance σ	18	28	(5.9%)	104	(6.8%)
Resolved resonance σ	12	16	(3.4%)	1006	(65.8%)
Interaction	2	7	(1.5%)	----	-----
	32	51	(10.8%)	1110	(72.5%)
Tabulated σ	2	10	(2.1%)	100	(6.5%)
Elastic matrices	39	172[a]	(36.4%)	114	(7.4%)
Moderating parameter	3	7	(1.5%)	----	-----
	44	189	(40.0%)	214	(14.0%)
Ultra-fine-group spectrum	54	81	(17.1%)	1	(0.1%)
Broad-group σ management	67	125	(26.5%)	193	(12.6%)
Inelastic σ management	3	22	(4.7%)	----	-----
	124	228	(48.3%)	194	(12.7%)
Broad-group spectrum	0.3	1	(0.2%)	6	(0.4%)
Total	201	472		1530	
Total CPU/total	0.4			0.8	
Core required	570 K bytes			1282 K bytes	

[a] 41 sec for all CSD.
86 sec for materials which get legendre treatment in MC^2.

TABLE III. Fundamental Mode SDX, 12 Isotopes (Time in sec)

Code Area	Homogeneous		Heterogeneous[a]	
	CPU	PP	CPU	PP
Input processor	<1	2	<1	2
Unresolved resonances	18	14	28	32
Resolved resonances	13	4	22	6
Interaction and cross-section preparation	2	9	3	13
Unit cell	--	--	36	9
Spectrum	8	7	8	7
Total	43	50	98	81

[a]12 plates of which 7 contained resonance isotopes.

TABLE IV. Homogeneous, Space-Dependent SDX - 16 Isotopes, 4 Regions (Time in sec)

SEF1D		Remainder		Total	
CPU	PP	CPU	PP	CPU	PP
73	23	92	126	165	149

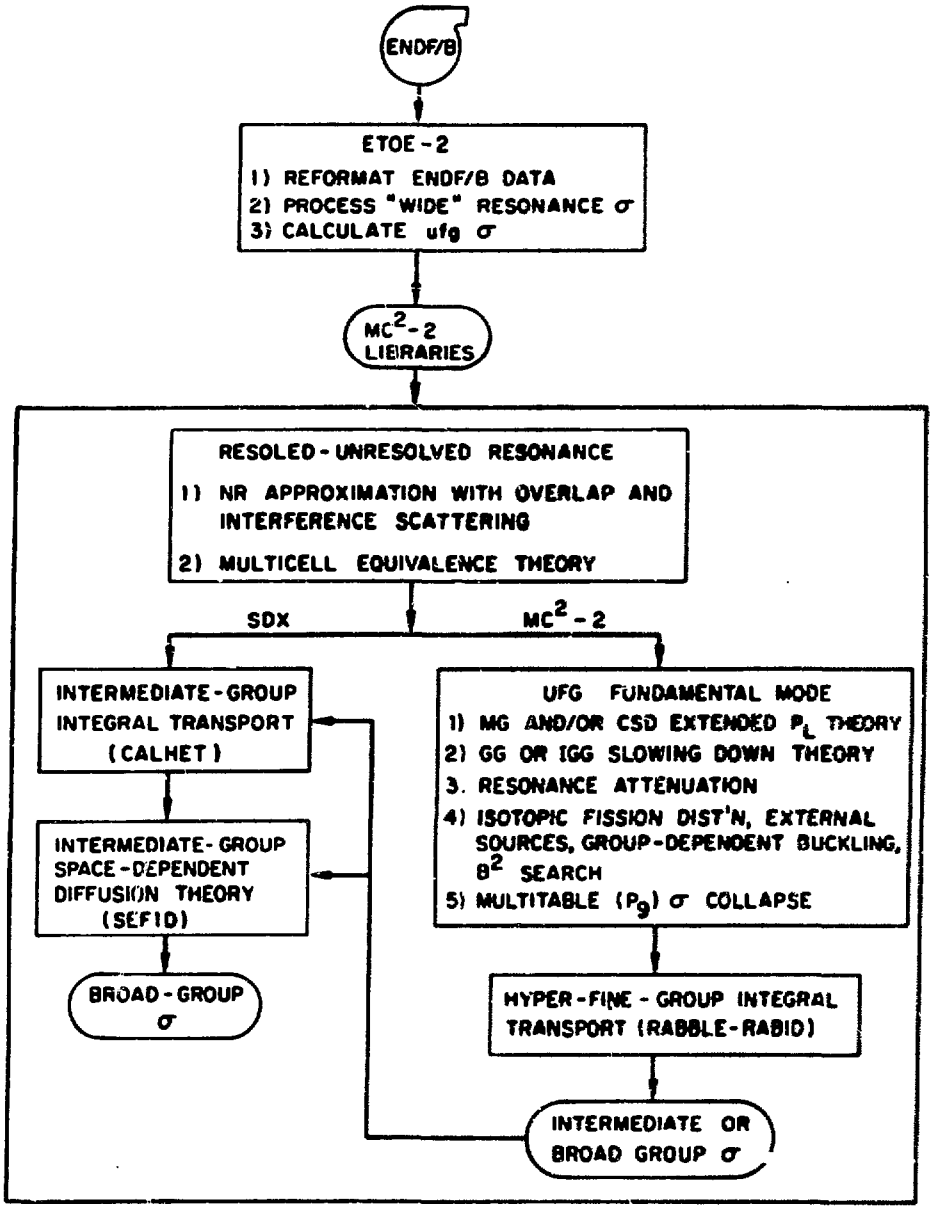

Fig. 1. ETOE-2/MC²-2/SDX.
(ANL Neg. No. 116-2120)

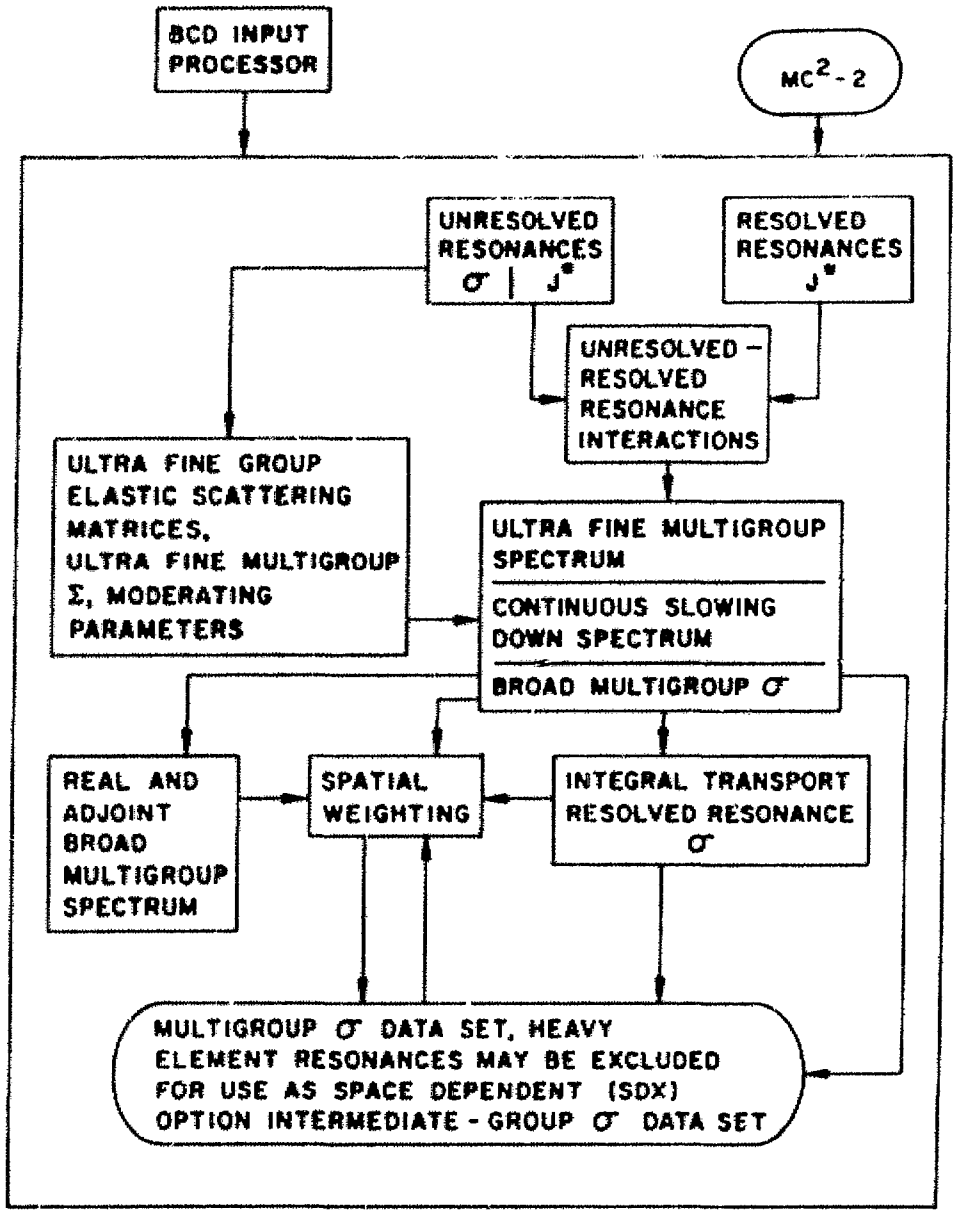

Fig. 2. MC^2 program flow.
(ANL Neg. No. 116-1773-I)

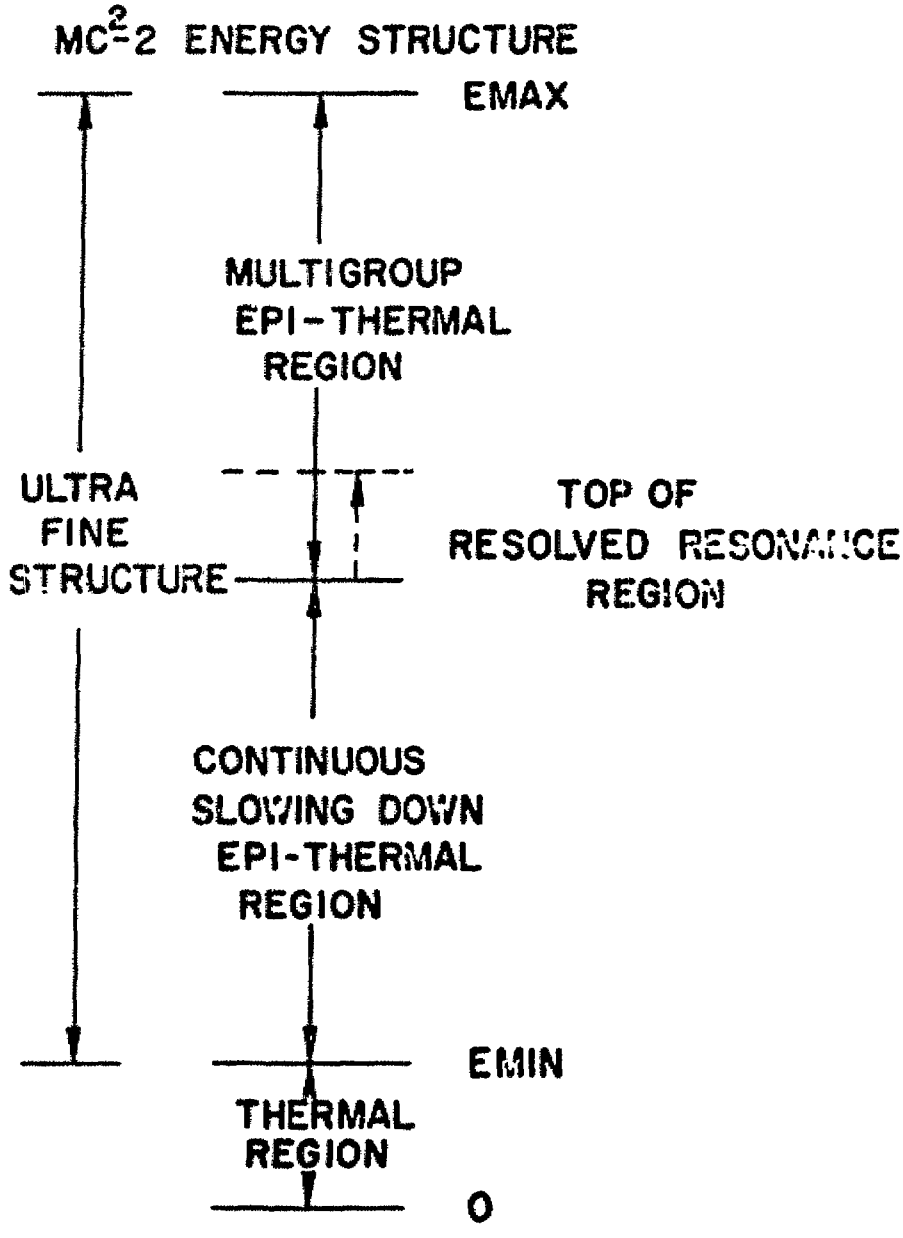

Fig. 3. MC²-2 energy structure.
(ANL Neg. No. 116-631)

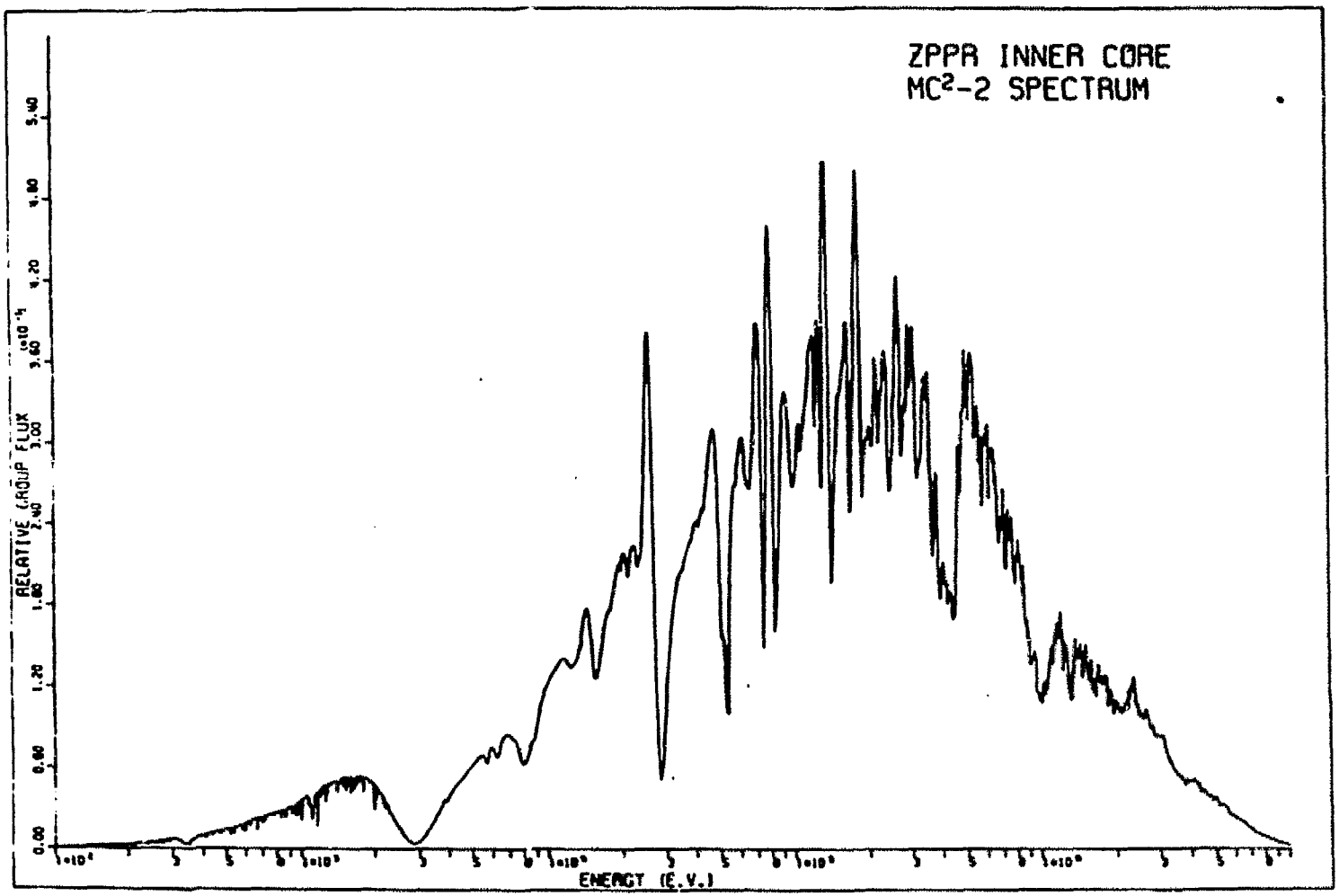

Fig. 4. MC²-2 spectrum, 0.0001-10 MeV.
(ANL Neg. No. 116-2104)

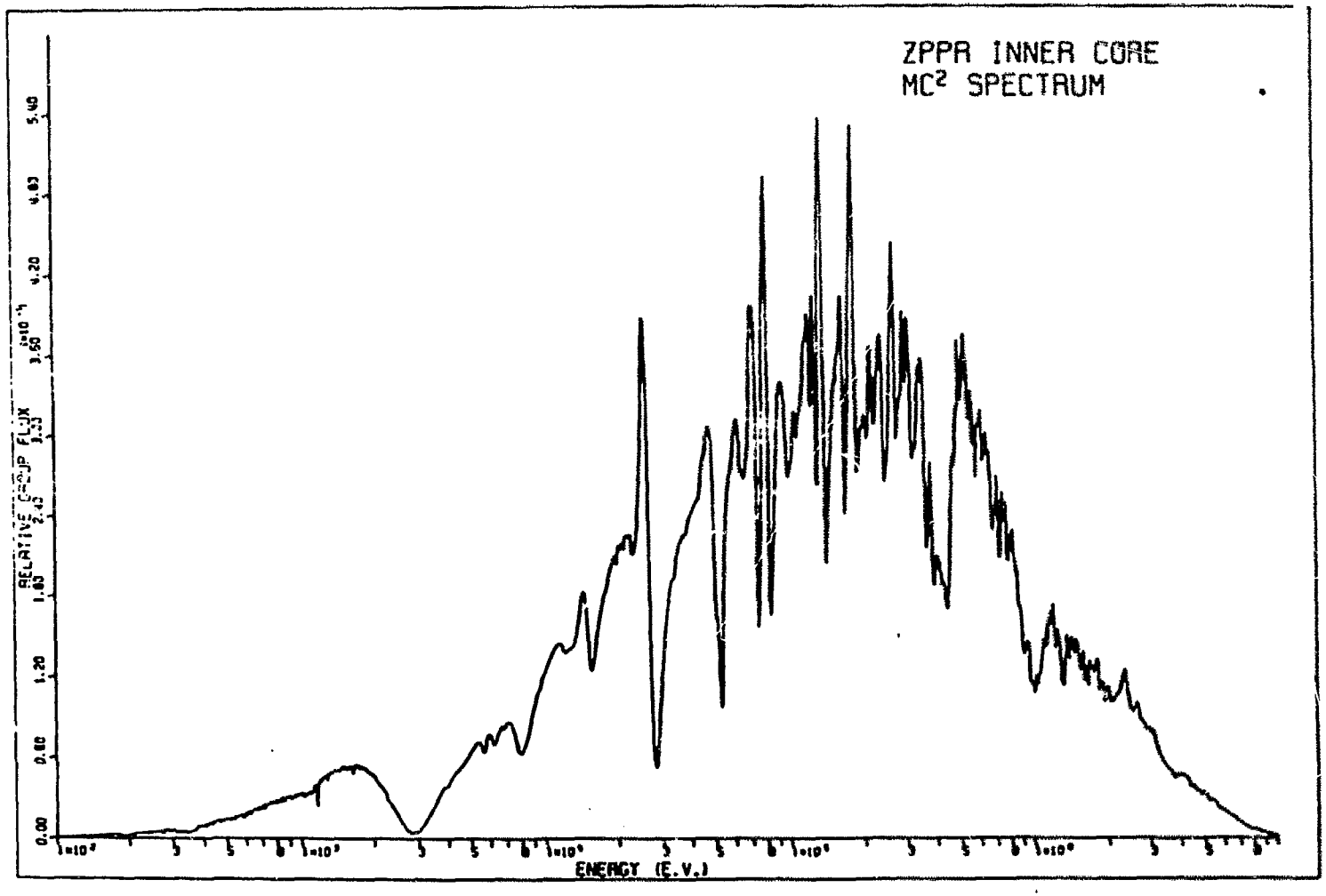

Fig. 5. MC² spectrum, 0.0001-10 MeV.
(ANL Neg. No. 116-2103)

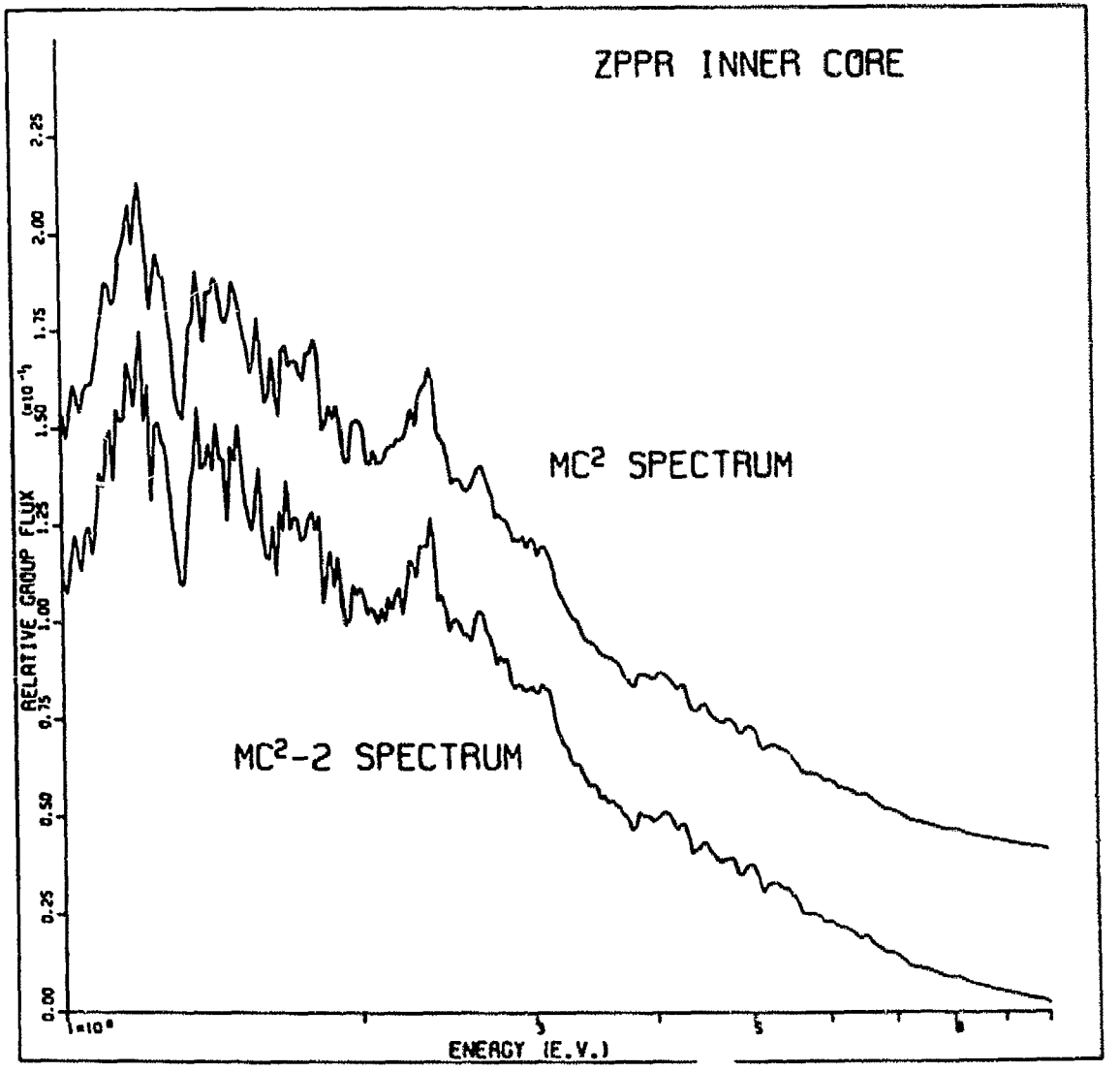

Fig. 6. MC²-2 and MC² spectrum, 1-10 MeV.
(ANL Neg. No. 116-2105)

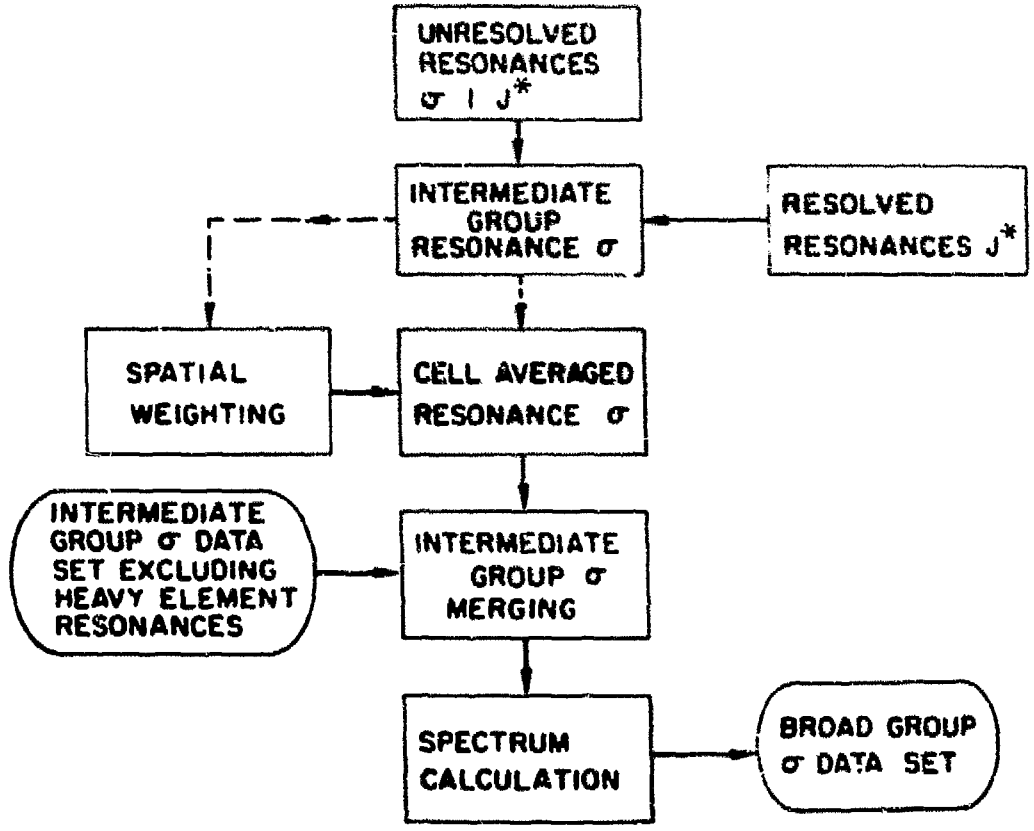

Fig. 7. SDX program flow.
(ANL Neg. No. 116-1771-I)