THE SHIELDING FACTOR METHOD FOR PRODUCING EFFECTIVE CROSS SECTIONS: MINX/SPHINX AND THE CCCC INTERFACE SYSTEM

R. E. MacFarlane
Los Alamos Scientific Laboratory, University of California
Theoretical Division
P O Box 1663, Los Alamos, NM 87545

C. R. Weisbin
Oak Ridge National Laboratory
P O Box X, Oak Ridge, TN 37830

N. C. Paik
Westinghouse Advanced Reactor Division
P O Box 158, Madison, PA 15663

ABSTRACT

The Shielding Factor Method is an economical designer-oriented method for producing the coarse-group space and energy self-shielded cross sections needed for reactor-core analysis. Extensive experience with the ETOX/1DX and ENDRUN/TDOWN systems has made the SFM the method of choice for most US fast-reactor design activities. The MINX/SPHINX system was designed to expand upon the capabilities of the older SFM codes and to incorporate the new standard interfaces for fast reactor cross sections specified by the Committee for Computer Code Coordination. MINX is the cross-section processor. It generates multigroup cross sections, shielding factors, and group-to-group transfer matrices from ENDF/B-IV and writes them out as CCCC ISOTXS and BRKOXS files. It features detailed pointwise resonance reconstruction, accurate Doppler broadening, and an efficient treatment of anisotropic scattering. SPHINX is the space-and-energy shielding code. It uses specific mixture and geometry information together with equivalence principles to construct shielded macroscopic multigroup cross sections in as many as 240 groups. It then makes a flux calculation by diffusion or transport methods and collapses to an appropriate set of cell-averaged coarse-group effective cross sections. The integration of MINX and SPHINX with the CCCC interface system provides an efficient, accurate, and convenient system for producing effective cross sections for use in fast-reactor problems. The system has also proved useful in shielding and CTR applications.

INTRODUCTION

The complexity of a typical reactor core makes it impractical to solve the neutron transport problem with full space and energy detail. For this reason designers normally use effective cross sections averaged over relatively coarse energy groups and space regions. The Shielding Factor Method (SFM) is an economical method for producing these effective cross section that was originally developed in Russia.[1] Development of the SFM in the US has been chiefly for the fast-reactor program, and extensive experience has been accumulated with the ETOX[2]/1DX[3] and ENDRUN[4]/TDOWN[5] code systems. More recently the SFM has received increased attention for thermal power reactor analysis with the development of EPRI-CELL and EPRI-CPM for the electric utility industry.[6] SFM code systems are traditionally divided into two parts: the cross section processor (e.g., ETOX) and the space-energy collapse code (e.g., 1DX). The MINX/SPHINX system follows this pattern. It was designed to expand upon the capabilities of the older SFM codes and to incorporate the standard interface formats for fast reactor codes specified by the Committee for Computer Code Coordination (CCCC).[7] The MINX[8] cross section processor generates a library of multigroup cross sections, shielding factors, and group-to-group transfer matrices from ENDF/B-IV[9] evaluated nuclear data and writes it out as CCCC ISOTXS and BRKOXS files. The SPHINX[10] space-energy code uses specific mixture and geometry information together with equivalence principles and a diffusion or transport flux calculation to construct effective coarse-group cell-averaged macroscopic cross sections in CCCC format.

The MINX/SPHINX system is in routine use on both IBM and CDC equipment. Comparisons with the older SFM codes show generally good agreement. Comparisons with independent codes such as ETOE-2[11]/MC^2-2,[12] VIM,[13] and GGC-5[14] give confidence that the MINX/SPHINX system is suitable for the routine analysis of large fast-reactor cores.

THEORY OF THE SHIELDING FACTOR METHOD

The goal of the SFM is to define effective cross sections for some range of energy (E in group g) and some region of space ($\underline{r}$ in volume v) that preserve macroscopic overservables such as reaction rate. Clearly,

$$\sigma^i_{xgv} = \frac{\int_g dE \int_v d\underline{r}\ \sigma^i_x(E,\underline{r})\ \phi\ (E,\underline{r})}{\int_g dE \int_v d\underline{r}\ \phi\ (E,\underline{r})}\ , \qquad (1)$$

where σ^i_x is the cross section for isotope i and reaction x at E and $\underline{r}$, and ϕ is the neutron scalar flux at that energy and position. Similar expressions can be constructed to preserve the group-to-group scattering rates and the transport cross section.

Unfortunately, the flux needed for Eq. (1) is not known before the fact; in fact, it is one of the quantities being sought in the analysis. In addition it is very complex, being full of sharp dips and peaks caused by resonances in the cross sections. However, experience has shown that it is possible to separate the variation of the flux into a part that is relatively smooth with respect to energy group and spatial zone size and a remaining resonance part. The variations in the smooth part can be determined by a multigroup flux calculation, but the intra-group flux must be selected by model.

The class of codes represented by MC^2 and GGC-5 determines this model flux by making a detailed flux calculation for a simplified homogeneous system. This is an expensive procedure. The SFM codes, on the other hand, assume that the intragroup flux can be modeled as

$$\phi(E) = \frac{W(E)}{\Sigma_t(E)} \quad , \tag{2}$$

where W is a smooth function of energy reflecting the fission and scattering sources into E and Σ_t is the total macroscopic cross section. Formally, Eq. (2) gives the flux for an infinite homogeneous system satisfying the narrow resonance approximation. However, heterogeneous systems can be included using equivalence princples.[15] Extension to wide and intermediate-width resonances is also possible.[16]

Furthermore, in evaluating the numerator of Eq. (1), it is assumed that the important effect is the interaction between a resonance in σ_x and the dip in the flux caused by that resonance (self-shielding). The reaction rate becomes

$$\int_g \frac{\sigma_x^i(E)}{\sigma_0^i + \sigma_t^i(E)} W(E)dE \quad , \tag{3}$$

where

$$\sigma_0^i = \frac{1}{\rho_i} \sum_{j \neq i} \rho_j \sigma_t^j \quad , \tag{4}$$

and where ρ_i is the number density for isotope i in the homogeneous mixture. The simplification comes from assuming that σ_0 is constant in g. The result is a single parameter, the "background cross section per atom," which can be used to characterize self-shielding. The cross sections produced by MINX are computed using

$$\sigma^i_{xg} = \frac{\int_g \frac{\sigma^i_x}{\sigma_0 + \sigma^i_t} \, WdE}{\int_g \frac{1}{\sigma_0 + \sigma^i_t} \, WdE} \quad . \tag{5}$$

The results are tabulated as cross sections for $\sigma_0 = \infty$ and shielding factors

$$f^i_{xg} = \frac{\sigma^i_{xg}(\sigma_0)}{\sigma^i_{xg}(\infty)} \quad , \tag{6}$$

for several values of σ_0. SPHINX then computes σ^i_0 using Eq. (4). In heterogeneous systems an additional "escape cross section per atom" is added. The corresponding shielded cross section is then obtained by interpolating the f-factors for this σ_0. Temperature is handled in the same way.

This approach makes a composition-independent cross-section library possible. The economy of the SFM results from being able to use this library many times.

THE MINX PROCESSING CODE

MINX was designed to combine and improve upon the resonance capabilities of ETOX[2] and ENDRUN[4] and the anisotropic scattering capabilities of ETOG[17] and SUPERTOG.[18] It is a modular code that uses paging techniques and variable dimensioning to make it possible to process the complex evaluations found in ENDF/B-IV.[9] The normal flow through the code is pictured in Fig. 1.

First, detailed pointwise cross sections are generated from ENDF/B resonance parameters and cross sections using the method of RESEND.[19] The energy grid is suitable for linear interpolation to within a user specified accuracy. The results are written out as a "pointwise-ENDF" (PENDF) tape suitable for printing, plotting, or further processing.

These pointwise cross sections are then accurately Doppler broadened to any desired temperatures using the method of SIGMA-1.[20] This approach has the advantage of correctly broadening smooth cross sections, backgrounds, and multi-level representations. Since broadening is a smoothing process, the results are thinned to a user specified accuracy and written out as PENDF tapes. Examples of the number of points produced in reconstruction and Doppler thinning are given in Table 1. Although this highly accurate process is expensive, it only has to be done once for a particular evaluation. Many subsequent averaging runs with different parameters can be made using the one temperature-dependent PENDF

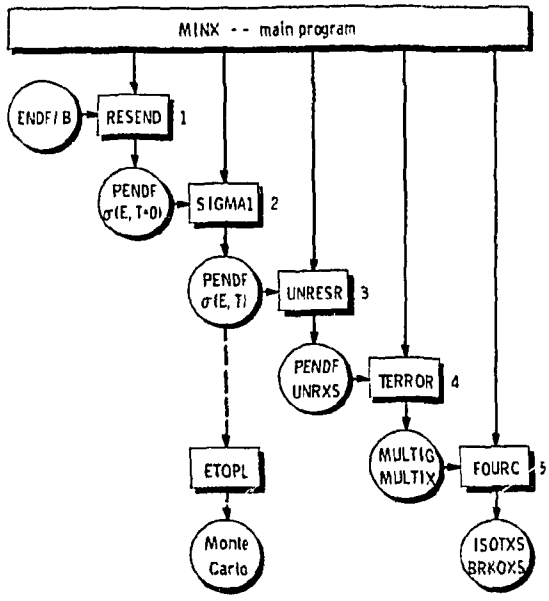

Fig. 1. Structure of MINX code illustrating functional blocks and data flow. ETOPL is not a part of MINX.

Table 1. Results of MINX Resonance Reconstruction and Doppler Broadening

Nuclide	Points at 0 K[a]	Points at 2100 K[b]	CP Seconds (CDC7600)
^{12}C	404	404	68.1
Fe	8798	5033	302.8
^{235}U	7209	2660	492.3
^{238}U	50372	6683	3483

[a] 0.5% reconstruction except 1.0% for ^{238}U.
[b] 0.1% thinning tolerance.

tape. The tape can also be reworked for use by continuous-energy Monte Carlo codes.[21]

This procedure will not work in the unresolved energy range where only statistical knowledge of the resonances is available. Effective pointwise cross sections vs T and σ_0 are produced by averaging over the ENDF/B distributions of resonance widths using methods based on ETOX.[2]

Multigroup cross sections are computed using Eq. (5) and appropriate generalizations. The group structure and smooth weight function are chosen by the user. The energy integrations are performed by adaptive quadrature starting from the union grid of the functions in the integrands. The nature of the PENDF cross section grid assures that all features are well-represented. Fission yields are averaged to preserve $\nu\sigma_f$ and slowing-down parameters are averaged to preserve $\bar{\mu}\sigma_e$ and $\xi\sigma_e$. The transport cross section is computed as $\sigma_t - \bar{\mu}\sigma_e$ where ^{e}current-weighting is used for the total cross section.

Elastic and discrete-inelastic scattering both obey two-body kinematics. MINX usually performs the resulting complex integrals over energy and angle with a semi-analytic method[22] based on an expansion in the laboratory system. The analytic integrals are obtained by a recursion relation, and the single energy integral is performed adaptively to a user specified tolerance. When this is not appropriate (e.g., light isotopes and near thresholds) MINX automatically changes to a direct numerical integration in the center-of-mass frame.

Group-to-group cross sections for continuum reactions are evaluated using analytic integrations over secondary energy and the standard adaptive quadrature for initial energy. Fission chi vectors by isotope are

produced by averaging the ENDF/B spectrum appropriate to a specified incident energy.

The final step is to format the results of the multigroup averaging module into the CCCC-III[7] ISOTXS (cross sections and matrices) and BRKOXS (shielding factors) files.

THE SPHINX SPACE-ENERGY CODE

SPHINX combines an extended version of the resonance treatment of 1DX[3] with the one-dimensional diffusion theory flux calculation of 1DX or the one-dimensional transport flux transport of ANISN.[23] It is modular in structure and uses the flexible POINTR system[24] of dynamic storage allocation. The entire code was assembled in accordance with the CCCC specifications for code compatibility.[7,25] The use of CCCC interface files makes communication with other CCCC-compatible codes such as TWOTRAN[26] and VENTURE[27] straightforward.

The basic structure of SPHINX is shown in Fig. 2. The various execution paths through the code are administered by the ZEUS CONTROL module using input data from the CCCC standard and code-dependent interfaces listed in Table 2. The fundamental cross-section data, intermediate results, and final answers are transmitted using the CCCC files described in Table 3.

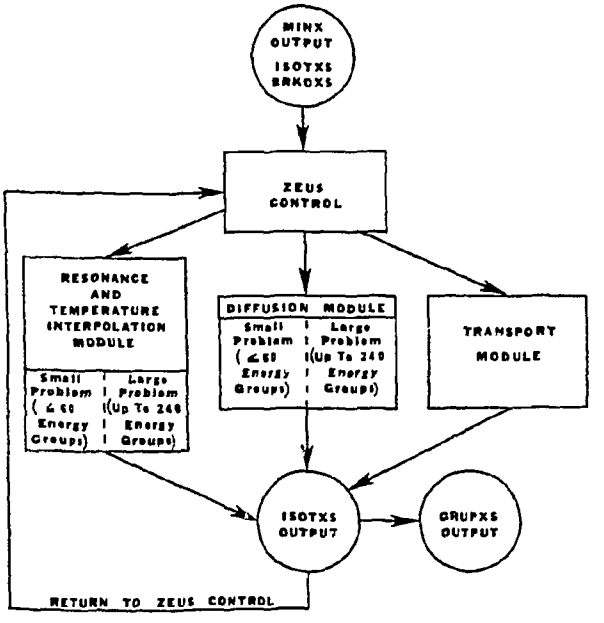

Fig. 2. Execution paths through SPHINX.

Table 2. SPHINX Control Files

Name	Standard	Description
ZEUS1A	no[a]	Modular control input
XSRINP	no	Resonance module input
SKODXI	no	Diffusion module input
ANISIN	no	Transport module input
ZNBTDN	no	Zone atomic densities
FPRINT	no	Print control
GEODST	yes[b]	Geometry description
NDXSRF	yes	Nuclide density and cross section parameters
ZNATON	yes	Zone nuclide atomic densities
SEARCH	yes	Criticality search parameters
SNCONS	yes	S_n constants
FIXSRC	yes	Volume and surface sources

[a] See Ref. 10 for detailed specifications.

[b] See Ref. 7 for detailed specifications.

Table 3. SPHINX Standard[a] Interface Files

Name	Description
ISOTXS	Nuclide-ordered cross section data
GRUPXS	Group-ordered cross section data
BRKOXS	Resonance self-shielding data
RTFLUX	Regular scalar flux
ATFLUX	Adjoint scalar flux
RCURNT	Regular current
ACURNT	Adjoint current
RAFLUX	Regular angular flux
AAFLUX	Adjoint angular flux
RZFLUX	Regular zone-averaged flux
PWDINT	Power densities

[a] See Ref. 7 for detailed specifications.

The first step in most problems is to use the resonance module to prepare effective self-shielded cross sections appropriate to the specified composition and geometry. The background cross section σ_0 is computed for each group and nuclide using Eq. (4). An additional escape term can be added for one of the seven options: (1) cylindrical cell using Sauer's approximation[28] for the Dancoff factor in a hexagonal lattice, (2) cylindrical cell using Sauer's approximation to the Dancoff factor in a square lattice, (3) symmetric slab cell, (4) asymmetric slab cell, (5) isolated rod, (6) cylindrical cell with the Bell approximation[29] to the Dancoff factor, and (7) symmetric slab cell with the Bell approximation. Self-shielding factors are then computed at σ_0 by Langranian interpolation. Effective cross sections are defined as in 1DX except that provision is made for an elastic group-to-group matrix. The results are written in ISOTXS format for communication with the flux modules.

On option, the code then branches to the diffusion module. The calculation is identical to that in 1DX except that input is in ISOTXS format and cross-section storage has been modified to allow for up to 240 groups and for several additional partial reaction types (i.e, n2n, nα, nd, ...). The cross sections are then collapsed to a subset group structure using the computed flux and written out in ISOTXS format.

The optional transport module uses the S_n method to obtain the flux. The method is identical to ANISN except for the ISOTXS interface capability. When the flux has been obtained, cross sections are collapsed to a subset group structure and zone-averaged using either volume or flux weighting. This provides a capability for cell homogenization.

The use of standard files provides many possible paths. For example, the flux from a diffusion calculation is easily available as an input guess for a subsequent transport calculation using already shielded cross sections.

LIBRARIES AND UTILITIES

SPHINX is normally used with one of the existing multigroup libraries generated by MINX. LIB-IV[30] is a 50-group 101-isotope library generated from ENDF/B-IV. The library includes all the general purpose evaluations from ENDF/B-IV plus the two copper isotopes and the nine lumped fission products from ENDF/B-III. All materials were run at 300, 900, and 2100 K using 4 to 6 σ_0 values with decade steps. Scattering matrices are given to P_3. The weight function consists of a 1.4 MeV fission spectrum joined at 820.8 keV to a 1/E shape which, in turn, joins to a 0.025 eV Maxwellian at 0.10 eV. The library also contains delayed neutron yields and spectra for seven isotopes generated in CCCC DLAYXS format using NJOY.[31]

VITAMIN-C[32] is a 171-group library with 36 isotopes chosen for importance in fusion and fast-reactor calculations. The specifications

for temperature, σ_0, Legendre order, and weight function are similar to LIB-IV, except that a velocity exponential fusion peak has been attached in the 14 MeV range.

These libraries require several utility codes in order to knit them into a system with MINX and SPHINX. First, BINX[33] is a code for converting back-and-forth between binary and BCD modes for transmission of ISOTXS, BRKOXS, and DLAYXS files between laboratories. LINX[33] is a code for adding new isotopes to an existing CCCC cross-section library. Finally, CINX[34] is a collapse code that can be used to generate a subset library tailored to a particular set of problems. As an example, CINX has been used to produce a 126-group subset of VITAMIN-C especially designed for LMFBR core and shield analysis.[35] Figure 3 illustrates how these codes and libraries combine to form a complete system.

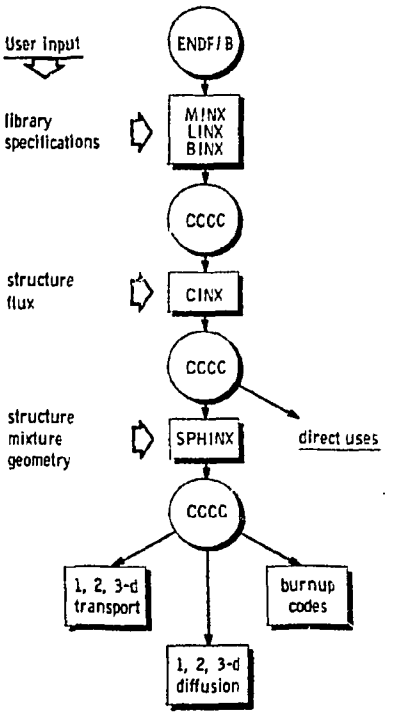

Fig. 3. Outline of CCCC interface system for generating multigroup constants for fast reactor analysis.

CODE VALIDATION

The MINX/SPHINX system has been tested in a variety of ways.[32,35] One ongoing project is a comparison of various processing codes being carried on by a committee of industrial and national laboratories (the DOE Code Evaluation Working Group). In order to minimize confusing complications, this group has analyzed a simple homogeneous composition typical of a large fast-breeder reactor core. The current results for some important parameters are given in Table 4. Larger differences exist between the fluxes and various cross sections. At the present time, for this type of problem, the chief causes of these differences seem to be: (1) group structure and weight function, (2) elastic removal treatment and (3) unresolved self-shielding. In any case, the numbers in Table 4 are less than the uncertainties associated with the basic evaluated data and with other design conservatisms. They imply that the MINX/SPHINX system is accurate for routine fast-reactor design.

Table 4. Comparison of Various Codes for a Buckled Homogeneous Fast Reactor Model

Parameter	ANL MC^2-2 Value	ARD 50g/SPHINX % diff	ORNL 126g/SPHINX % diff	LASL 50g/1DX % diff	LASL ETOX/1DX % diff	GE 50g/TDOWN % diff
k_{eff}	1.0040	0.10	0.31	0.19	0.13	0.17
C28/F49	0.1585	-0.06	-0.88	-0.26	-0.32	0.00
C28/F25	0.1447	-0.35	-0.76	-0.35	-0.48	0.28
F49/F25	0.9132	-0.33	0.12	-0.12	-0.23	0.22
F28/F25	0.0206	0.44	1.12	0.68	-0.44	4.27
F40/F25	0.1806	0.17	0.55	0.33	-0.44	1.55
F41/F25	1.294	0.29	0.36	0.33	-0.15	0.30

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IMPLEMENTATION OF THE RAPID CROSS SECTION ADJUSTMENT APPROACH AT GENERAL ELECTRIC

C. L. Cowan, E. Kujawski and R. Protsik
General Electric Co. - Fast Breeder Reactor Department
Sunnyvale, California

ABSTRACT

The General Electric rapid cross section adjustment approach was developed to use the shielding factor method for formulating multigroup cross sections. In this approach, spatial and composition dependent cross sections for a particular reactor or shield design are prepared from a generalized cross section library by the use of resonance self-shielding factors, and by the adjustment of elastic scattering cross sections for the local neutron flux spectra. The principal tool in the cross section adjustment package is the data processing code TDOWN. This code was specified to give the user a high degree of flexibility in the analysis of advanced reactor designs. Of particular interest in the analysis of critical experiments is the ability to carry out cell heterogeneity self-shielding calculations using a multiregion equivalence relationship, and the homogenization of the cross sections over the specified cell with the flux weighting obtained from transport theory calculations. Extensive testing of the rapid cross section adjustment approach, including comparisons with Monte Carlo methods, has indicated that this approach can be utilized with a high degree of confidence in the design analysis of complex fast reactor systems.

INTRODUCTION AND REVIEW

The generation of multigroup cross sections for the analysis of fast reactors at General Electric is based upon a rapid cross section adjustment approach. This approach was specified to utilize the shielding factor method for processing basic nuclear data, and has undergone extensive development since the middle 1960s. Many of the early underlying theories for this approach are described in the compilations of group constants by I.I. Bondarenko.(1)

The shielding factor method is basically a two-step process. In the first step, a generalized composition independent cross section library is created from the pointwise ENDF/B data. The generalized

library includes fine-group infinitely dilute cross sections, self-shielding factors and group-to-group transfer matrices for all materials of interest. This library serves as a preprocessed data base for generating the energy group constants for a specific fast reactor system. The self-shielding factors are peculiar to this approach and are given in tabular form in the generalized file for different values of the parameter σ_o, where for a given material, σ_o is designated as the ratio of the sum of the total cross sections for all other materials in a given composition to the atom density of the material in question.

The second step in the shielding factor approach involves the calculations of spatial and composition dependent cross sections for a particular reactor or shield design. The group constants are determined by interpolating on the self-shielding factors for the specific σ_o values which are computed for the given compositions. In addition, the elastic removal cross sections on the generalized file are corrected for the actual neutron flux distribution.

The early techniques for treating the shielding factor approach were incorporated into the ENDRUN - TDOWN code packages[2,3] at GE and the ETOX - 1DX code packages at HEDL.[4,5] The successful application of these techniques in the design analysis of fast reactor systems, has led to the development of a second generation of data processing codes, as characterized by the MINX[6] and SPHINX[7] systems. A significant development effort has also gone into the present version of TDOWN, and a number of these features are presented in the material which follows. TDOWN was specified to give the user maximum flexibility in using the rapid cross section adjustment approach for performing advanced design and critical experiments analysis. Problem descriptions can vary from a single composition with fluxes input, to a two-dimensional multiregion problem requiring several one-dimensional diffusion or transport theory flux solutions and resulting in a set of microscopic cross sections for each material in each region.

GENERAL FEATURES OF THE TDOWN CODE

The TDOWN code performs the following modifications to a generalized data library to generate composition and spatially dependent neutron cross sections for a specific fast reactor system:

1) Self-shielding factors (f-factors) are used to modify the infinitely-dilute cross sections for fission, capture, elastic-scattering, transport and total reactions. The f-factors are dependent upon the temperature of the material, the composition in which the material is found, and (optionally) on the cell configuration.

2) Eigenvalue and fixed source flux calculations may be carried out in fundamental mode, or in one-dimensional diffusion or

transport theory.

3) The elastic-removal cross sections are corrected for the energy distribution of the actual neutron flux. The neutron flux spectra may be input or taken as calculated in step (2) above.

4) The fission spectrum for a given composition is determined by weighting the flux averaged fission spectra for each isotope in accordance with its contribution to the neutron source.

5) The cross sections may be homogenized over all or a portion of specified cells, with flux-weighting obtained from transport theory calculations.

6) The cross sections may be condensed to one or more few group libraries, using any of the computed or input flux spectra.

A generalized flow diagram outlining the TDOWN procedures and operations is given in Figure 1. The computations for a given composition begin with the calculations of the self-shielding parameter for the i-th group and m-th material, $(\sigma_o)_{im}$. This parameter is calculated in an iterative process with the maximum number of iterations specified by the user. Following the determination of the $(\sigma_o)_{im}$ values, resonance self-shielding correction factors are computed by interpolating between the f-factors which are tabulated for discrete values of the temperature and σ_o. The f-factor interpolation on σ_o is carried out by means of a Taylor series expansion about neighboring values whereas the interpolation on the temperature is based upon the logarithmic proportionality between the tabulated values.

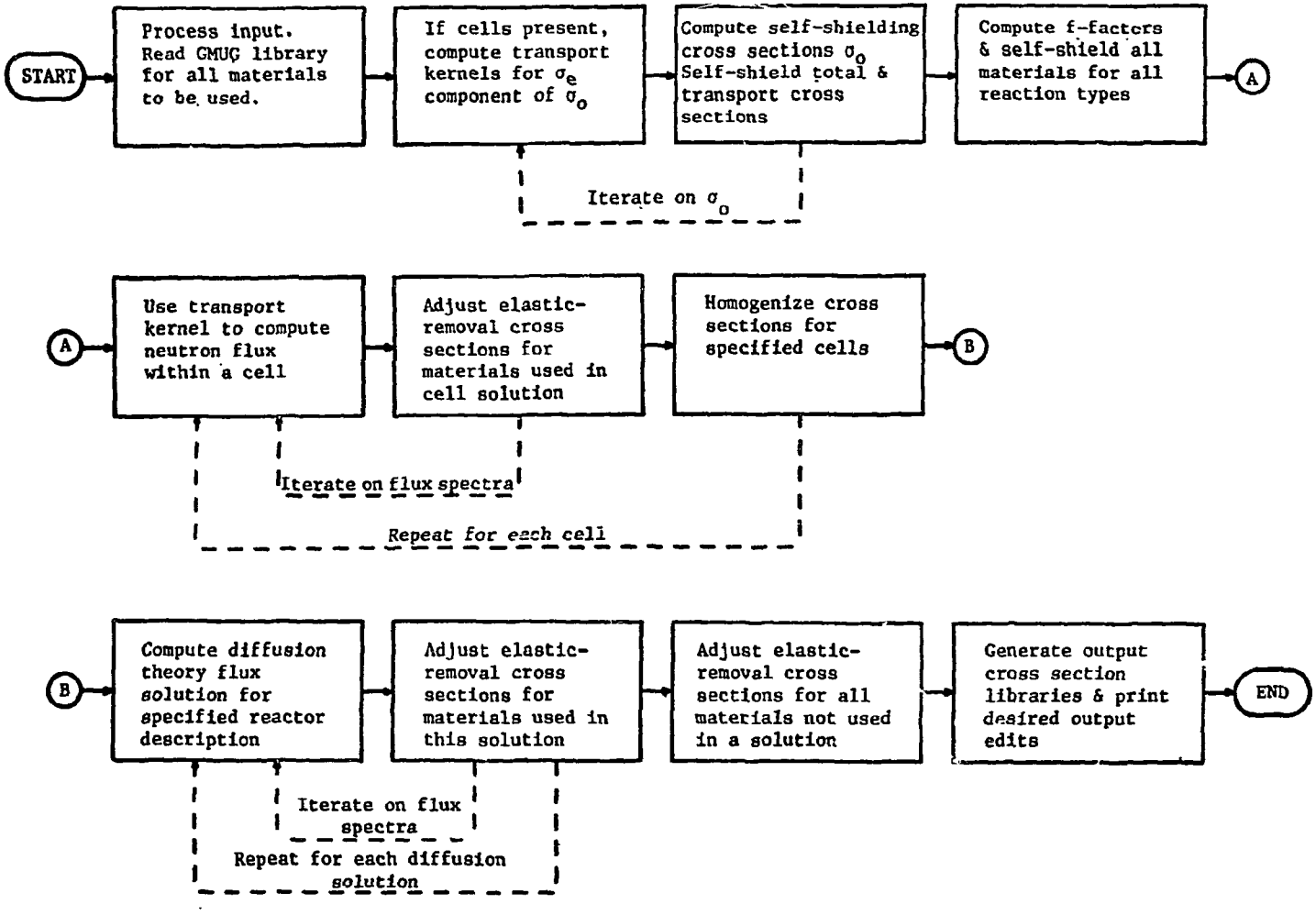
The technique for carrying out the spectral adjustment to the elastic removal cross section in TDOWN is based upon a Taylor series expansion of the ratio of the neutron fluxes for the ℓth and 0th flux spectrum iterations about some average lethargy $\bar{u}$. In this expansion, only the first two terms are retained to give,

$$R_{im}^{(\ell)}(u) \equiv \frac{\phi^{(\ell)}(u)}{\phi_{im}^{(0)}(u)} \simeq R_{im}^{(\ell)}(\bar{u}) + (u-\bar{u})\, R_{im}'^{(\ell)}(\bar{u}) \qquad (1)$$

Here $\phi_{im}^{(0)}$ is the intragroup flux weighting spectrum used to generate the group constants for material m in the generalized data library. Ideally, this term can be expressed as a function of lethargy as follows:

$$\phi_{im}^{(0)}(u) = \left[\frac{(\sigma_o)_{im} + (\sigma_t)_{im}}{(\sigma_o)_{im} + \sigma_{t,m}(u)}\right] \psi(u), \qquad (2)$$

FIGURE 1 GENERALIZED FLOW DIAGRAM OF TDOWN COMPUTATIONS



where (σ_o) is the standard self-shielding parameter, and $\psi(u)$ corresponds to the fission spectrum or a constant, depending on whether the lethargy is above or below some specified cutoff point.

Using Equation (1) it can be shown that the elastic removal cross section after the ℓth iteration is approximately given by

$$(\sigma_{er})_{im}^{(\ell)} \sim (\sigma_{er})_{im}^{(0)} \frac{R_{im}^{(\ell)}(\bar{u} + \delta\bar{u})}{R_{im}^{(\ell)}(\bar{u})} \tag{3a}$$

where

$$\bar{u} + \delta\bar{u} = u_i - \gamma_{im} \tag{3b}$$

$$\gamma_{im} = \frac{\int_{u_i - u_{i-1}}^{u_i} du(u_i - u)\, P_m(u \rightarrow > u_i)}{\int_{u_i - u_{i-1}}^{u_i} du\, P_m(u \rightarrow > u_i)} \tag{3c}$$

u_i = Upper lethargy for group i, and

$P_m(u \rightarrow > u_i)$ = Probability of scattering from lethargy u to a lethargy greater than u_i

It is noted that the probability term P_m includes an expression for anisotropic scattering.

Using Equation (2) the elastic removal adjustment factor can be rewritten in a form which readily lends itself to a physical interpretation:

$$\frac{R_{im}^{(\ell)}(\bar{u} + \delta\bar{u})}{R_{im}^{(\ell)}(\bar{u})} = \left[\frac{\Sigma_t(u_i - \gamma_{im})\ \phi^{(\ell)}(u_i - \gamma_{im})}{\Sigma_{t,i}\ \phi_i^{(\ell)}/\Delta u_i}\right]\left[\frac{\psi_i/\Delta u_i}{\psi(u_i - \gamma_{im})}\right], \tag{4}$$

where

$\Sigma_{t,i}$ and ϕ_i = ith group total macroscopic cross section and flux corresponding to $\Sigma_t(u)$ and $\phi^{\ell}(u)$, respectively

ψ_i = group flux corresponding to $\psi(u)$

Δu_i = width of group i

To evaluate the adjustment factor in Equation (4), the reaction rate and original flux spectrum are approximated by continuous linear functions which conserve the integrals of the reaction rates and $\psi(u)$ in the neighborhood of each group and over the entire lethargy span. It should be noted that this requires a break at the boundary between adjacent groups of unequal lethargy width. By rearranging Equation (4) and substituting into Equation (3a), the elastic removal correction in TDOWN is expressed as follows:

$$(\sigma_{er})_{im}^{(\ell)} = (\sigma_{er})_{im}^{(o)} \left[\frac{1 + \left(\frac{\Delta u_i - 2\gamma_{im}}{\Delta u_i + \Delta u_{i+1}} \right) \left(\frac{\Sigma_{t,i+1}\, \phi_{i+1}^{(\ell)}}{\Sigma_{t,i}\, \phi_i^{(\ell)}} - \frac{\Delta u_{i+1}}{\Delta u_i} \right)}{1 + \left(\frac{\Delta u_i - 2\gamma_{im}}{\Delta u_i + \Delta u_{i+1}} \right) \left(\frac{\psi_{i+1}}{\psi_i} - \frac{\Delta u_{i+1}}{\Delta u_i} \right)} \right], \quad (5)$$

Heterogeneity is treated in TDOWN by using an optional procedure for calculating the σ_o's based upon multiregion equivalence theory. The optional treatment is limited to one-dimensional slab, cylindrical or spherical cell geometries as supplied by the user. The multiregion equivalence relationship permits the computation of "self-shielded" cross sections in a complex multiregion cell. This formulation is especially important for the ZPPR plate cells, where the use of a "two-region equivalence relation" is at best ambiguous. The geometry is properly accounted for through the use of the transport coefficients or first flight collision probabilities $T_{zz'}$ between regions z and z'. In this approach, a generalized equivalence relationship is derived with the self-shielding parameter given as follows:

$$(\sigma_o)_m = (\sigma_o)_{mz} + (\sigma_e)_{mz} \quad (6)$$

Here $(\sigma_e)_{mz}$ is defined as an escape cross section which is material and region dependent (the group indicators have been omitted for convenience). The escape cross section is expressed in terms of the collision probabilities as,

$$(\sigma_e)_{mz} = \frac{(\Sigma_t)_z}{N_{mz}} \left(\frac{1 - T_{zz}}{T'_{zz}} \right), \quad (7)$$

where

$$T'_{zz} = T_{zz} + \sum_{z' \neq z} T_{zz'} \frac{N_{mz'}\, \sigma_{mz'}}{(\Sigma_t)_{z'}} \frac{(\Sigma_t)_z}{N_{mz}\, \sigma_{mz}} \quad (8)$$

The coefficients, $T_{zz'}$, may be conveniently computed using S_N-transport theory by considering unit sources. The problem of choosing an appropriate "modified Bell approximation" is thereby avoided. This transport kernel may then be used to generate a flux solution for use in homogenizing the cross sections in the cell.

The broad range of computations available in TDOWN provide a high degree of flexibility in specifying the cross sections for the analysis of a particular fast reactor design. Thus, the cross sections for each material on the output library from TDOWN are determined by 1) the source material on the generalized file, 2) the composition set and material temperature which were specified for carrying out the self-shielding calculations, 3) the spectral set which was utilized in the elastic-removal correction, 4) the heterogeneity description, and 5) the spectral set which was utilized in the condensation of the fine-group data library. A two-dimensional reactor description is input to aid the user in setting up several one-dimensional diffusion theory flux solutions through specified zones in the two-dimensional model. Specific cell descriptions may also be input for multizone self-shielding and for transport theory flux solutions. Composition sets may, or may not, correspond to the compositions in either the two-dimensional or cell models. Thus, a single TDOWN problem can be utilized to provide output data sets for the same material in the same composition at different temperatures, or even cross section sets for different neutronics problems. The flux spectra sets may be obtained from any of the zero or one-dimensional flux solutions which are carried out by TDOWN, or may be input.

TESTING THE RAPID CROSS SECTION ADJUSTMENT APPROACH

The shielding factor methods have undergone an extensive validation. Detailed comparisons with the Monte Carlo code VIM[8] and the direct data processing code MC^2-2[9] have been carried out as part of the activities of the Data Processing Subcommittee of the Code Evaluation Working Group. The differences between the results of calculations using the shielding factor approach and the more accurate methods were found to be small for the important integral parameters. The multiregion heterogeneity treatment incorporated in TDOWN has also been tested against the VIM computations for the ZPPR inner core normal cell. The results of the reaction rate calculations within the individual plates are summarized in Table I. The agreement between the results of the shielding factor heterogeneity calculations and the Monte Carlo calculations were found to be excellent. These studies have also indicated some limitations in the applicability of the two-region equivalence relationships. In particular, the shielding effects for the structural materials have been found to be significant for certain cell configurations.

Based upon the testing results described above and the successful application of the shielding factor method in the analysis of critical experiments it is concluded that the TDOWN code can be used with a high degree of confidence in the analysis of fast reactor systems.

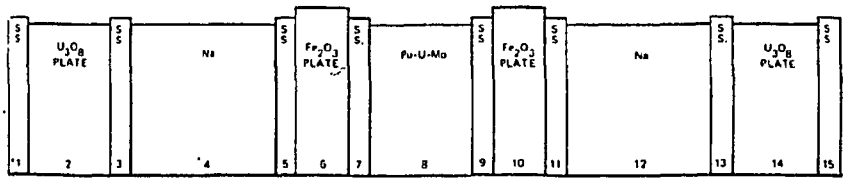
TABLE I

COMPARISON OF TDOWN AND VIM HETEROGENEITY FACTORS[a]
FOR PLATEWISE ONE-GROUP CROSS SECTIONS

One-Group Cross Section[b]	TDOWN Heterogeneity	TDOWN/VIM[c]
$^{238}U(n,\gamma)$ (U_3O_8 Plate)	1.0057	0.997 ± 0.28%
$^{238}U(n,\gamma)$ (Pu-U-Mo Plate)	0.9509	1.001 ± 0.27%
$^{238}U(n,f)$ (U_3O_8 Plate)	0.9293	0.986 ± 0.39%
$^{238}U(n,f)$ (Pu-U-Mo Plate)	1.1283	0.981 ± 0.34%
$^{235}U(n,f)$ (Pu-U-Mo Plate)	1.0022	1.002 ± 0.26%
$^{239}Pu(n,\gamma)$ (Pu-U-Mo Plate)	0.9527	1.004 ± 0.37%
$^{239}Pu(n,f)$ (Pu-U-Mo Plate)	0.9968	1.002 ± 0.20%
Fe(n,γ) (Can Zone 1)	0.9962	1.009 ± 1.02%
Fe(n,γ) (Can Zone 3)	1.0320	1.018 ± 1.31%
Fe(n,γ) (Na Plate)	1.0215	0.999 ± 1.06%
Fe(n,γ) (FeO_3 Plate)	0.9897	0.998 ± 0.98%
Fe(n,γ) (Can Zone 7)	0.9992	0.968 ± 1.61%

a) The heterogeneity factors are defined as the ratio of the one-group platewise cross sections for the normal inner core ZPPR cell divided by the corresponding one-group cross sections for the homogeneous cell.

b) The plate layout for the ZPPR cell is shown as follows:



c) The assigned uncertainty is simply the uncertainty in the individual isotopic reaction rate for the heterogeneous cell given by VIM using track length estimators after 200,000 histories.

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