VIM - A CONTINUOUS ENERGY MONTE CARLO CODE AT ANL

R. N. Blomquist, R. M. Lell and E. M. Gelbard
Argonne National Laboratory
Argonne, Illinois, USA

ABSTRACT

The continuous energy Monte Carlo neutron transport code, VIM, and its auxiliaries, are briefly described. The ENDF/B cross section data processing procedure is summarized and its benchmarking agains MC^2-2 is reviewed. Several representative applications at ANL are described, including fast critical assembly benchmark calculations and STF and TREAT Upgrade benchmark calculations.

INTRODUCTION

The VIM code is a continuous energy Monte Carlo code designed primarily for fast reactor calculations, but also containing a thermal neutron scattering capability. The development of VIM and the associated fast reactor cross section processing codes was initiated at Atomics International[1] and has been continued at Argonne National Laboratory. VIM, now available through the National Energy Software Center, features a flexible geometrical capability, a neutron physics data base closely representing the ENDF/B data from which it has been derived, and a calculational output directed to the needs of the fast reactor analyst.

DESCRIPTION

The original VIM geometry package was designed to permit a simple description of plate-lattice critical experiments. All cells of identical characteristics, with plates, clad, and void defined by combination of rectangular parallelepipeds, need be specified only once; the full assembly is then described as a rectangular lattice constructed from the basic cells. The combinatorial geometry package developed for the code SAM-CE[2] has been implemented in VIM and extended to specific geometrical descriptions of particular interest in reactor analysis. The above two techniques have been combined in VIM to provide options for the description of repeating hexagonal and rectangular lattices with the in-cell geometrical definition employing the full combinatorial geometry capability. In addition, an infinite, homogeneous medium option is available to provide an efficient capability for data testing and cross section methods evaluation.

VIM produces three distinct estimates of the reactor eigenvalue. The analog, or last-event estimator scores $W(\nu\Sigma_f)^{isotope}/\Sigma_a^{total}$ whenever absorption by a fissile isotope occurs. Here, W is the neutron weight, and Σ is the macroscopic cross section. The collision estimator scores the fission production rate, $W(\nu\Sigma_f)^{total}/\Sigma_{tot}^{total}$, at each collision event. The track (or path) length estimator scores $W(\nu\Sigma_f^{tot})/\Sigma_a^{total}$ times

the track length for all tracks within each zone, including uncolliding tracks. The estimates of standard deviation in the eigenvalue estimates are obtained using the assumption (never strictly correct in an eigenvalue computation) that the contributions from all the history batches are statistically independent.[3]

VIM produces a statistical edit of various quantities after a user-specified number of batches. Both collision and track length estimation provide groupwise reaction rate estimates by region and by isotope, while track length estimation generates region-wise integrated group fluxes. Optionally, infinite dilution region-averaged microscopic reaction rate ratios may be obtained in a designated central region. Track length estimates of reaction rates and fluxes are used to provide estimates of broad-group microscopic and macroscopic cross sections over edit regions. All quantities are provided with standard deviation estimates which are based on the statistical independence of the batch data.

VIM may be used with a combination of several variance reduction techniques. Neutrons can be tracked by a combination of absorption and analog weighting which can be assigned by zone or by cell. For example one might use analog weighting in the core region for an eigenvalue calculation and absorption weighting in a blanket region to improve the statistics of low-energy effects. The user can select a cutoff energy below which all weighting is analog to reduce the effort spent on unimportant neutrons. Splitting and Russian roulette can be used to spatially modify the sample distribution, improving local statistics. Combined estimators[4] produce averages of the eigenvalue estimates in linear combinations determined by the statistical characteristics of the data, using the assumption of normality of the batch results. This is most effective as a variance reduction technique when two estimators are strongly negatively correlated. Simple averages of the estimates are also provided, with the estimated standard deviations including the effects of correlation between the estimates.

In addition to a startup source guess and a restart capability a number of initialization options are available. The user may provide a set of source sites from a similar previous calculation to avoid wasting the first several batches converging on a source shape. One can specify a fixed source of arbitrary spatial, angular and energy distributions by supplying a fixed source subroutine within the framework provided by VIM.

The physics data base for the VIM code consists of a library of binary files, with each file providing the physics data for one material. Each such "VIM material file" is the end product of a moderately complex computational path beginning with ENDF/B tape files. At each intermediate step, a code is used to process one or more intermediate data libraries and produce an output library of data available to a succeeding step. A flowchart of the generation system is shown in Fig. 1.

The initial step of cross section processing is the program VIMB, which reformats and reorders ENDF/B data and generates the energy grid. VIMB calculates the potential scattering cross section, threshold energies

for inelastic levels, and normalized cumulative secondary energy distributions for (n,2n), fission, and inelastic continuum scattering. A grid of energies for unresolved resonance parameters is generated if they are energy-dependent in ENDF/B, and the elastic scattering cross section energy grid is tightened to permit linear-linear interpolation meeting a user input accuracy criterion. A common energy grid is then constructed by merging the energy points for all reactions and inserting a background grid of 20 points per decade. VIMB also interpolates File 3 cross sections to the expanded energy grid and processes the anisotropic angular distribution data for elastic, (n,2n), and inelastic levels by (a) calculating a 200 point angular table at each energy from Legendre coefficients, (b) calculating a 200 point angular distribution from a tabulation by a cubic spline fit to the logarithm of the differential cross section vs. cosine of the angle, (c) calculating normalized cumulative distributions from the 200 point tables and collapsing to 20 equal-cosine intervals.

The UNIDOP code produces point cross section data in the resolved resonance region from S-wave and P-wave resonance parameters from VIMB output. For each isotope, an energy mesh is obtained by merging a 99-point distribution around each resonance energy with a background grid at equal lethargy intervals of at least 40 points per decade. Zero degree Kelvin resonance calculations are performed using either single-level or multi-level Breit-Wigner representations over the energy grid, and the point data are Doppler broadened to the first specified temperature (presumably 300 degrees K). Any File 3 background are added in, and the resulting data are then Doppler broadened to as many as four other user-specified temperatures. The cross section arrays are then thinned according to user specified accuracy of either interpolation accuracy on total cross section alone, absorption cross section alone, or total and absorption cross sections simultaneously. Resonance integrals are calculated before and after Doppler broadening, after any File 3 background are added in, and before and after thinning.

The AUROX code generates unresolved resonance data for single isotopes obtained from a VIMB output file into cross section probability tables using Monte Carlo methods. For each spin series of resonances, the Wigner distribution for resonance spacings is sampled independently to obtain a ladder of resonances, and the appropriate chi-square distributions are sampled for resonance width parameters. Pointwise cross sections for scattering, capture, and fission are then constructed on an arbitrary energy scale using energy dependent factors evaluated at the ENDF/B specified energy point with Doppler broadening to the desired temperatures. The average values of the cross sections between any two energy points on the grid are then binned by total cross section value with weight equal to the energy interval. Additional ladders are generated and the cross sections binned until either a user-specified number of ladders has been processed or until the standard deviation in the observed infinitely dilute average cross sections satisfies a user input criterion. The average cross section value in each bin is then calculated, the cumulative probability for sampling from a bin is obtained from the binned weights, and the resulting tables are normalized to the known infinite dilute average cross sections by applying an addi-

tive constant to each bin of a given reaction type, preserving the observed higher moments. The process is repeated for all the ENDF/B energy points on the VIMB output data set or for as many as have been specified by card input.

The REDUCE code processes cross section probability tables with a large number of probability bands into a library of tables with a small number of bands. Conventionally, an AUROX output data set with 99 point tables is processed to an output data set with 20 point tables. At user option, the probability bands are combined either by minimizing the absolute difference between input and output total cross sections for high atom densities or the mean square difference for very dilute concentrations. In addition, REDUCE calculates average self-shielded cross sections for an array of user-supplied values for additional equivalent potential scattering in barns/atom (the potential scattering for the material being processed is included in the probability table scattering cross section). REDUCE may be used solely to calculate effective cross sections from the original tables to be compared with reduced tables to examine the effect of the reducing algorithms and to compare with analytical calculations.

Since AUROX cannot process unresolved resonance data for a natural material which is a mix of isotopes, the probability tables must be generated for each isotope separately. MERGER is then used to prepare a single set of "material" tables from the isotopic tables and the corresponding isotopic abundances. In Fig. 1, MERGER would replace, precede, and/or follow REDUCE.

MERGER uses the principle that the unresolved resonances from different isotopes are uncorrelated. Consequently, the joint probability distribution for the cross sections of two isotopes is the product of the two individual distributions. In forming the joint distribution from tables of length N1 and N2, a distribution of length N1*N2 is formed. For each such probability interval, the combined cross section values are obtained from the weighted sums of the corresponding individual values, the weighting factors being the isotopic abundances. The resulting arrays are sorted in order of increasing combined total cross section and the cumulative probability distribution obtained. The algorithms of the REDUCE code are then used to reduce the N1*N2 band tables to the desired output table length. MERGER will not process multitemperature correlated tables, but tables at different temperatures may be processed independently.[5]

The code VIMTAP produces a VIM material file for free atoms by combining the output data sets of the VIMB, UNIDOP, and AUROX (or REDUCE or MERGER) codes. VIMTAP replaces the VIMB data in the resonance range with the UNIDOP and AUROX data, and finds threshold energies and indices for total inelastic scatter and fission cross sections. The angular data may be thinned, and the elastic scattering cross section is corrected at very low energies to account for thermal motion by using a free gas model. Angular distributions for (n,2n) and the inelastic continuum are not retained, and only one interpolation code is allowed per reaction type.

Only two subsections are allowed in the secondary energy distributions; if more are present, only the first two are retained. In addition, the number of probability tables for any one material is limited to 140.

For those materials with thermal scattering law data specified in File 7 of an ENDF/B thermal tape, the procedure becomes more complex. The File 7 data are processed by the FLANGEX code into a library of discrete-energy double-P1 $S(\alpha,\beta)$ scattering kernels and thermal inelastic scattering cross sections. The KERINTX code subsequently processes these into a library of thermal scattering probability tables and thermal inelastic cross sections. If the thermal scatterer is a solid, VIMB extracts the elastic scattering cross section and angular distributions in the thermal energy range from the ENDF/B thermal tape. For each such thermal scatterer, the corresponding free atom VIM material file is input to THTAPE, along with the KERINTX output library and, if required, the VIMB thermal data, to produce a VIM material file incorporating a full thermal treatment.

VIM itself employs several models to treat scattering phenomena relevant to thermal problems. For incident energies above 10eV, a full kinematic treatment of non-thermal free atom inelastic or elastic scattering is applied which ignores thermal motion of target atoms heavier than $^{1}_{1}H$. Below 10eV but above a user-supplied maximum thermal energy cutoff, scattering is treated as either scattering by a free gas or as isotropic center-of-mass non-moderating scattering, depending on the target mass. For solid materials, both thermal elastic and inelastic scattering is modeled, while for liquids, only thermal inelastic scattering is incorporated.

One present limitation of the VIM library generation system, and consequently of VIM itself, is the inability to treat all the possible inelastic processes described in the ENDF/B data. At the present time, elastic scattering (MT=2), (n, 2n) reaction (MT=16 only, or MT=24 in the absence of MT=16), fission (MT=18), discrete level inelastic scattering (MT=51 through MT=90), inelastic continuum scattering (MT=91), and "capture" (the sum of MT=102 through MT=114) are incorporated. The total cross section is then defined to be the sum of the cross sections for these reaction types.

The VIMB, UNIDOP, and VIMTAP codes in use at ANL are modifications of similarly-named codes developed by Atomics International.[6] The code AUROX was derived from the AI code U3R[7] after extensive development and modification. The bulk of the development and testing of cross section preparation methods for VIM which has been done at ANL has been directed toward the treatment of resolved and unresolved resonance data. The FLANGEX and KERINTX codes were developed from the FLANGE and KERINT codes of Honeck and Finch at Savannah River.[8]

Figure 2 illustrates the data flow for codes which access the library of VIM material files. The codes FILEONE, XSEDIT, and BANDIT are not really a part of the library generation system, but rather are

utility codes for VIM and are part of the VIM export package. The code XSEDIT provides a binary-to-BCD and BCD-to-binary conversion capability for the VIM data base to permit export of the VIM code to non-IBM installations without the library processing capability. It may also be used to produce an edited listing of library contents. The code FILEONE is used to scan the library of material files and prepare a data file containing variable dimensioning information. The code BANDIT partitions the data from up to 20 VIM material files into as many as 16 energy bands and produces corresponding cross section data subsets requiring about equal and minimum amounts of computer storage to be used during VIM calculations.

Two auxiliary codes provide the user with the capability to modify the structure of the output data and to reanalyze it. KEFCODE permits the user to obtain a statistical edit of the VIM eigenvalue estimators for a subset of neutron batches completed by VIM. For example, one might wish to ignore the first several batches representing the unconverged source, or to lump the batches into larger ones. Using RETALLY, the user can perform group collapse, region homogenization, skip early batches and process only some of the records, and produce a new modified edit of the VIM batch data.

QUALIFICATION OF VIM

VIM was brought to Argonne primarily as a tool for fast critical assembly experimental analysis and for analytical methods benchmarking, so the code has been thoroughly benchmarked on fast reactor problems. Since the criticals program has included the assessment of nuclear cross section data and processing methods, much of the validation of VIM has focused on this area.

Prael and Henryson[9,10] tested VIM's cross section data preparation and its solution of the slowing-down problem by comparison with the MC^2-2 code, which was tested at the same time. MC^2-2 solves the fundamental mode neutron slowing-down equations with high accuracy using multigroup, continuous slowing-down, and integral transport theory algorithms. Since both MC^2-2 and VIM were designed to model the slowing-down process in great detail, and since the methods of each are distinct, such comparison provides confidence in the accuracy of both codes. In MC^2-2, the resonance calculations used an ultrafine group structure ($\Delta u = 0.008$), except below 4 keV where a hyperfine structure ($\Delta u \sim 0.001$) was applied to the resolved resonance region.

To test the cross section preparation algorithms, several infinitely dilute, zero-dimensional slowing down problems were solved[9] using ENDF/B-III data. Comparison of the resolved resonance broad-group cross sections revealed several large, local discrepancies. The first resulted from insufficient energy point densities away from resonance peaks due to the application of a linear-linear interpolation scheme to data spaced for log-linear interpolation. The second discrepancy arose from incomplete summing of resonance contributions. The third difficulty was due to distortion of absorption cross sections between well-separated resonances because the thinning criterion was applied only to the total cross section. This was solved by thinning out only those points at which both the absorption and total cross sections meet the thinning criterion.

After correcting these discrepancies, two test problems with 10 MeV source were run to produce broad-group edits for 27 groups of lethargy width 0.5. The problems solved were an infinite homogeneous medium of ^{23}Na with an infinitely dilute admixture of heavy isotopes, and an infinite homogeneous medium of ^{12}C with an infinitely dilute admixture of structural material. The capture and fission cross sections for ^{238}U and ^{239}Pu generally agreed to within a few tenths of a percent. However, resolved resonance capture in ^{238}U was still in error by almost 5% in certain groups due to the linear interpolation method; unresolved resonance cross sections required improved numerical normalization of the resonance probability tables; and ^{238}U and ^{239}Pu unresolved resonance cross sections were in error by as much as 2% because VIM uses a linear energy interpolation of probability tables. In addition, errors in capture in structural materials necessitated extending the energy grid farther from resonances, and the fluctuation of structural material cross sections required a denser energy grid in the keV range to overcome use of linear probability tables. These difficulties were also eliminated by adjusting the energy grid algorithms where appropriate.

Once agreement in cross section processing between VIM and MC^2-2 was achieved, both codes were tested on a typical homogeneous zero-dimensional fast reactor slowing down problem with a composition representative of the benchmark critical ZPR 6 Assembly 7. ENDF/B-III data were used, and 24 group edits of flux, fission spectrum, isotopic reaction rates, and isotopic microscopic cross sections, were produced for comparison. Extremely close eigenvalue agreement was obtained, and group fluxes agreed to within 1% down to the resonance range where differences of several percent were observed. Isotopic capture and fission rates and broad group cross sections were within 1% except for capture in ^{238}U and fission in ^{239}Pu, which was traced to VIM's use of linear interpolation between probability tables for unresolved resonance cross sections. Other broad-group cross sections agreed to within a few tenths of 1%, and within 1% in the resonance ranges.

As a result of these comparisons, there is confidence that the VIM cross section data are accurately represented in the material libraries, and that the physical slowing-down process is properly treated. VIM has subsequently been applied in the analysis of a number of fast reactor critical experiments, providing the most stringent tests of VIM's ability to analyze actual reactor cores. Both basic integral parameters e.g., multiplication eigenvalue, and detailed information specifically relevant to the critical assemblies under study[11] have been used to compare VIM with physical systems.

REPRESENTATIVE APPLICATIONS

At ANL, VIM is applied to diverse sets of problems. Although it is not possible to include here detailed discussions of each such set, in this section we will briefly describe some of the more common applications.

The level of agreement attained in the comparisons of VIM with fast reactor cross section processing codes and the experience gained in using VIM for the analysis of criticals has led to a widespread use of VIM as the benchmark for fast reactor cross section processing methods and as an analysis tool for numerous aspects of critical experiments. The continuous energy cross section treatment and the essentially exact geometrical modelling used in VIM permit the analyst to focus on the source of errors in deterministic calculations, e.g., deterministic eigenvalue calculations for critical systems which are typically in error by approximately 1.5% with the ENDF-B/IV data base. Eigenvalue and integral reaction rate comparisons with experiment and with other calculations have improved the quality of experiment analysis. Specifically, most of the eigenvalue errors have been traced to the ENDF data, and the effects of multigroup cross section processing on experiment analysis have been quantified, within limits.

For the analysis of most criticals, the need for VIM extends considerably beyond testing the effects of cross section treatment. The representation of fine structural details is often very important in the analysis. For example, if one has a detector or foil located in the core, the structure near this detector or foil can affect the number of counts obtained. An accurate calculation for such a treatment requires a flexible geometry package beyond the scope of most deterministic codes. The breeding ratio measurement of ZPPR-4 is an example of the need for such detailed analysis.

Certain classes of criticals have geometries that are so complex or irregular that analysis by deterministic codes requires extensive geometric approximations. A recent series of safety-related criticals involved the study of damaged cores, including mockups with slumped fuel, large cavities, and other severe geometric distortions. Such abnormal configurations severely strain the usual xy, xyz, rθ, rθz, and triangular geometric representations in most diffusion and S_N codes. These models must therefore verified by accurate reference calculations using the exact geometric representation available in VIM. Even when core geometries are regular in other respects, certain aspects of the experiment may require the VIM capabilities, at least for benchmark calculations. Some criticals, e.g., those related to GCFR studies, are notable for large neutron leakage and streaming effects. Diffusion codes require special treatments, e.g., anisotropic diffusion coefficients[12] to account for such effects, and special treatment of unit cell streaming paths, while S_N codes may require high angular resolution for accurate modelling of axial leakage. VIM provides a reliable treatment of neutron streaming and leakage, whether in small regular channels or in large irregular cavities.

Because of the relatively recent incorporation of a thermal scattering law treatment in VIM[13], benchmarking of the thermal cross section library in VIM is much less extensive than that for the fast energy range. Nevertheless, such benchmarking has been done for materials of particular interest to individual programs and users, e.g., graphite for the SAREF program and light water reactor materials for various thermal reactor programs at Argonne. As in the case of fast reactor cross section

studies, the agreement found between VIM, other thermal reactor codes, and experiments, is such that VIM has come to be accepted by many users as a standard for comparison with other thermal reactor cross section preparation codes.

Several groups in the Applied Physics Division at ANL use VIM to benchmark cross section methods used in thermal reactor analysis. For most of these applications, corresponding unit cells are calculated with VIM and with the thermal reactor code of interest, e.g., EPRI-CELL. The analyst then compares integral parameters such as k and regional reaction rates, and microscopic quantities such as individual isotopic cross sections with corresponding quantities produced by the deterministic code of interest. The results of this comparison allow the user to identify possible problems in a multigroup cross section set and to attach a level of reliability to the multigroup cross section set. More elaborate types of comparisons are occasionally carried out. For example, one analyst desired to use EPRI-CELL for the analysis of a rather complex light water reactor fuel assembly design containing fuel and blanket pins of different sizes and water holes in an unusually tight lattice.[14] This design was so heterogeneous that a true unit cell did not exist within the fuel assembly, necessitating rather extensive geometric approximations in the application of EPRI-CELL to the system. As a check on the adequacy of the final EPRI-CELL model, the entire fuel assembly was modelled exactly in a VIM calculation. This application is typical of an entire class of problems where a geometrically exact VIM calculation is used to validate a model when circumstances force a code user to exceed its intended range of application.

VIM is also used with some frequency for more general types of reactor analysis. One program that relies heavily on VIM is the Safety Research Experiment Facilities (SAREF) program. The SAREF program was formerly directed toward the development of a conceptual design for the proposed Safety Test Facility (STF) and is presently concerned with the design of an upgraded core for the Transient Reactor Test Facility (TREAT) reactor. Because this program involves all aspects of core design for an actual reactor rather than general parametric or feasibility studies of reactor types and concepts, VIM usage in SAREF tends to be varied and complex in scope. Examples of this usage will be discussed in some detail.

VIM is used for cross section benchmarking in SAREF, but the procedure becomes more complicated than in the routine unit cell calculations cited above. Because SAREF is concerned with core design for an actual reactor rather than the study of a concept, the quality of multigroup cross sections used in SAREF reactor physics calculations is particularly important. Errors which might be acceptable for a feasibility study cannot be tolerated in a program dealing with modifications to an existing reactor. Ideally, multigroup cross section methods and reactor physics calculational techniques are validated by comparison with critical experiments, but for reasons of budget and schedule, this is not possible for the current TREAT Upgrade work. Consequently, more reliance must be placed on comparisons between VIM and multigroup deterministic

calculations. One makes the same type of comparisons here as in the simpler cell calculations discussed above, but in the SAREF applications one is dealing with many distinct regions of a complex coupled-core reactor rather than a simple unit cell.

Certain unusual features of the STF conceptual design and the TREAT Upgrade design and purposes make uncommon demands on analysis methods. Both STF and TREAT Upgrade are transient test reactors intended to provide a pulsed source of neutrons with specified characteristics to irradiate a cluster of target fuel pins in a test loop at the center of the reactor core. Consequently, the fission density distribution in these target pins and the relationship of this fission density to the fission density at specified locations in the reactor core are the crucial parameters of full-core physics calculations. The target pins and some of the other regions of interest represent very small fractions of the entire core volume, so a very careful application of splitting and Russian roulette techniques is required to reduce the relevant variances in these regions to an acceptable level. Some recent calculations with and without splitting have shown that true reaction rates in the smaller regions could never be separated from statistical noise were it not for splitting.

Calculations for the SAREF reactors are further complicated by the presence of large radial and azimuthal irregularities in these cores. Both STF and TREAT Upgrade have inner and outer core regions of markedly different compositions and properties. This leads to a strong radial dependence in the core fission density and flux spectrum. A more serious non-uniformity in these cores, however, is the presence of a large cavity caused by the removal of a row of fuel assemblies between the test loop and the boundary of the reactor. This void is introduced to allow experimenters to "view" fuel displacement in the central target pins during a transient irradiation experiment. This slot causes large azimuthal flux variations and strong neutron streaming effects which in turn lead to a marked azimuthal dependence of the core fission density. The streaming effect due to this slot is the most important reason for using VIM in SAREF core analysis, since the slot void invalidates diffusion calculations and S_N calculations would require a very fine angular mesh. The presence of control rods inserted to varying positions only complicates matters further.

Because it is not practical in terms of cost or calendar time to perform VIM calculations for every core configuration or design parameter of interest, reference configuration VIM calculations are used to correct important physics paramters from the less accurate diffusion and S_N calculations. Three examples of these correction factors will illustrate this. First, deterministic codes cannot geometrically represent either target pins or the test loop, so the crucial determinations of target pin energy deposition are seriously in error. Furthermore, the fission density distribution within the target pins is an important experimental parameter which cannot be adequately calculated deterministically. By applying VIM with full geometric detail to the test assembly, correction factors are generated which apply to a class of similar situations. Finally, it is very important that the location of the hot spots in the

core be known because they set the operational limits for the core. Because diffusion and S_N calculations are not generally able to accurately account for the effects of the slot void and control rods in a three-dimensional calculation, the VIM results are used to identify and study the core hot spots. By carefully selecting representative configurations to be studied with VIM, the designer can establish sets of correction factors to be applied to classes of similar core configurations.

At times, the required accuracy for fission density estimates has made full core Monte Carlo calculations impractical. But by imposing a fixed source of magnitude, initial energy, and initial direction important with respect to the test assembly, it is possible to generate a Green's function for the target pins and the neighboring fuel assemblies. A record of initial and terminal neutron sites is written by VIM, which is then processed to yield the desired Green's function. This function can be used to determine the response of some selected portion of the system, particularly the target pins, to a partial current imposed at some boundary of interest. This current is determined from diffusion or transport calculations which are capable of representing general core characteristics but not the fine structural details of the target region. One can economically determine the response of the test assembly fission density region to changes in core conditions by varying the partial currents at the target region boundary and applying the Green's function to them.

A related procedure requires the imposition of incoming partial currents on some boundary surrounding the region of interest. These partial currents can be obtained in the same manner as for the Green's function calculation outlined above. In this case, however, the exact magnitude of the partial current is used to determine the fixed source. The source neutrons and all of their daughter neutrons generated within the region of interest are followed to the point of termination either by absorption within the region or escape from it. The resulting VIM estimates for the central region closely approximate those from a full core calculation, but are much more economical because the focus of the Monte Carlo calculation is only on the target region.

Another Argonne program that relies heavily on VIM is the Reduced Enrichment Research and Test Reactor (RERTR) program, in which many of the reactors under study have cavities and/or beam ports for the irradiation of test materials. Because many of these reactors are used principally as irradiation facilities, their designs are oriented towards creation of a high neutron flux at particular locations in the reactor core. In addition, many of these reactors are rather small physically, with very important neutron leakage and streaming effects. These reactors often exhibit such strong heterogeneity that standard diffusion and S_N codes cannot be applied without significant approximations in the geometric representations of the reactor cores. Accurate treatment of heterogeneity can be important even in relatively large, low-leakage reactors: it is even more important in small reactors where leakage can be a major factor in the neutron economy.

An adequate treatment of heterogeneity can be very difficult with deterministic codes for several reasons. First, representation of fine structure in finite difference codes can require so many mesh points as to easily exceed available core storage. Second, theoretical considerations may limit the attainable accuracy with a deterministic code, e.g., the basic approximations inherent in diffusion theory. Third, spatial heterogeneity is often accompanied by various cross section effects that are difficult to deal with in multigroup treatments, e.g. self shielding.

Sequences of design calculations with VIM are no more practical in RERTR than in SAREF, for the same reasons of time and cost. One has little choice but to accept the geometric approximations in deterministic codes for normal calculations. However, important heterogeneous features can be examined by calculating a more-or-less exact representation of the reactor core with VIM. The VIM calculation provides the data for RERTR analysts to correct the deficiencies caused by geometrical approximations and, incidentally, to correct for errors introduced by multigroup cross section treatments.[15]

VIM has also been modified for special use in the design of the Intense Pulsed Neutron Source (IPNS) at Argonne, which employs a high energy proton beam from the zero Gradient Synchrotron to generate a high flux neutron beam for nuclear physics and materials research. The proton beam generates high energy neutrons in a heavy metal target. The emergent neutrons are then scattered into neutron beam tubes by moderating materials selected to produce neutron fluxes of specified characteristics, particularly energy. The highly localized and anisotropic nature of the proton beam and the resulting neutron fluxes necessitates an exact treatment of the angular variable, and the irregular arrangement of target, moderator, reflector, and beam tube regions in the shapes of cylinders and parallelapipeds requires a flexible geometric representation. HETC[16], a high energy nucleon-meson transport code is used to track incident protons and the few neutrons emergent from the proton target with energies above 15 MeV. The site coordinates and velocities of neutrons below 15 MeV are then saved as source sites for VIM. VIM tracks the neutrons in the usual way, except that at each collision, the probability of emerging as a beam tube particle is computed. After scoring this probability times the weight, the normal tracking process resumes. The HETC/VIM results are consistent with experiment within statistics (5-10% uncertainty).

SUMMARY

For a number of years, VIM has provided a reliable computational benchmark capability at Argonne because of the extensive benchmarking of VIM itself against other analytical tools and against numerous critical experiments. The code has been applied to analysis of a wide range of fast and thermal reactors as well as to other neutron transport calculations which require either flexible geometric representations or basic ENDF cross section data up to and including version V.

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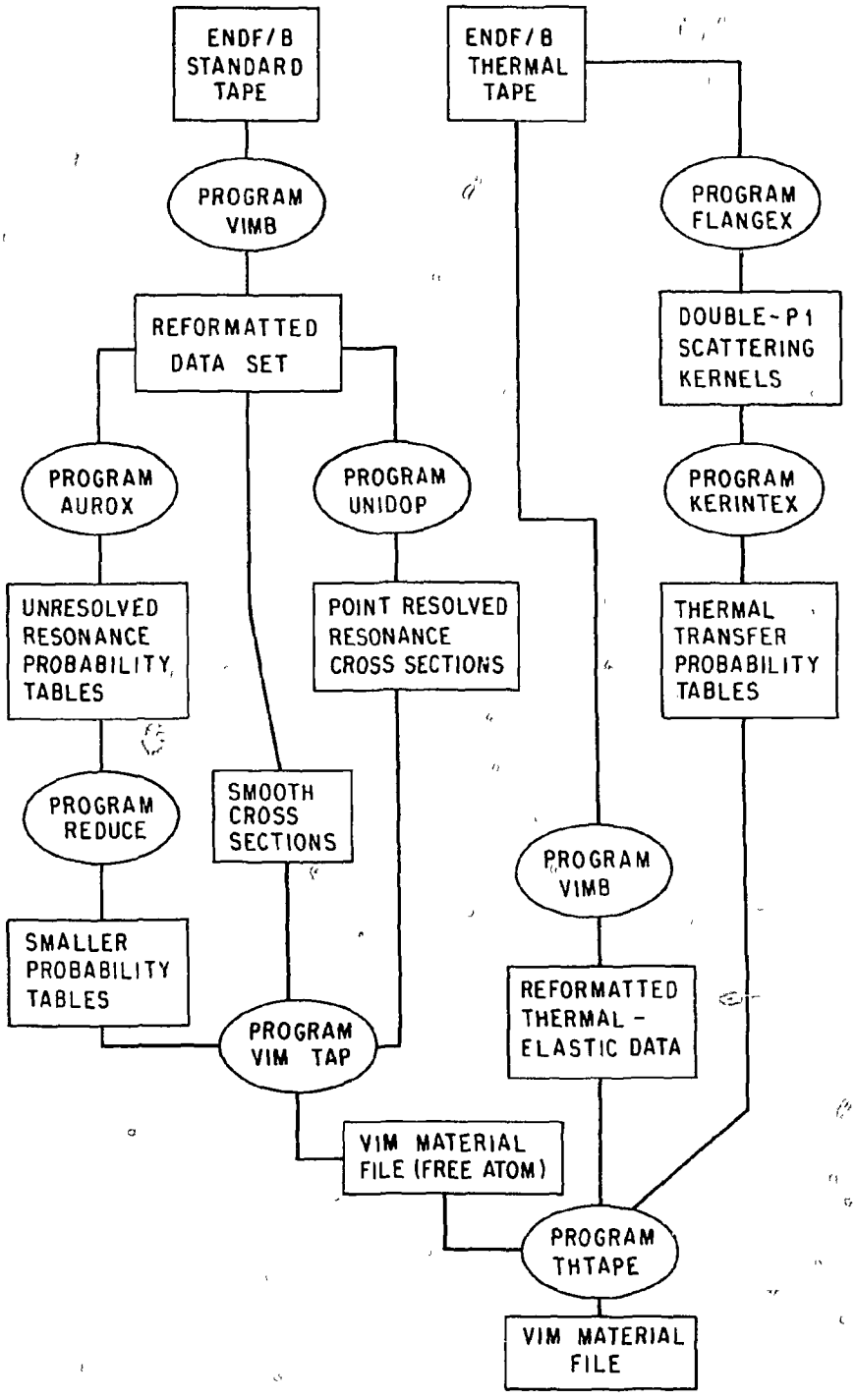

Fig. 1. VIM Cross Section Library Generation Procedure

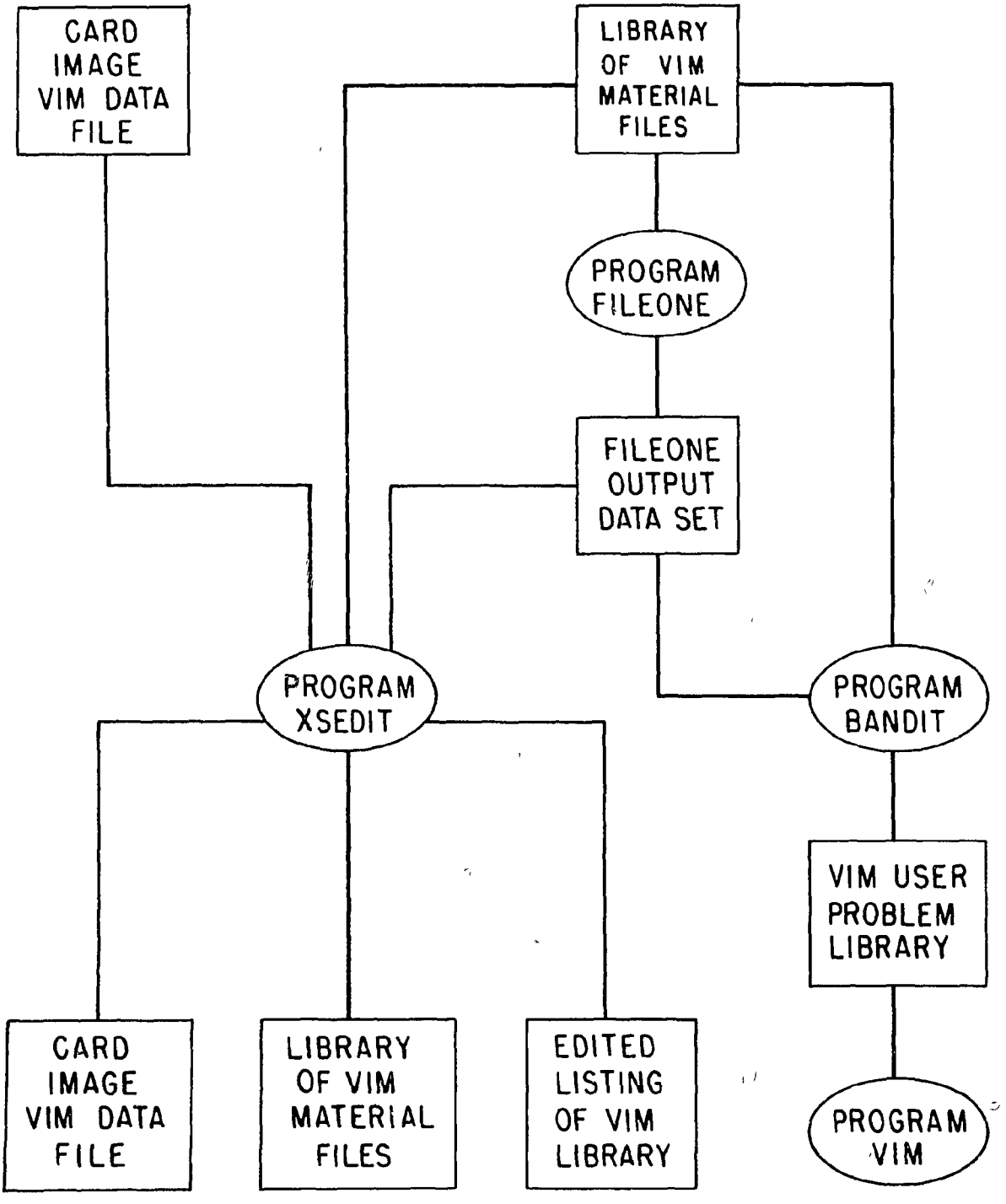

Fig. 2. VIM Cross Section Library Handling Utilities