LMFBR RADIATION TRANSPORT METHODS DEVELOPMENT*

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Abstract

The elements of a shielding program based on a fundamental approach are discussed with a brief evaluation of the present status of the technology. A procedure is presented for utilizing the best features of Monte Carlo, discrete ordinates, and diffusion theory in coupled calculations providing good accuracy at minimum expense. As an example, results are presented from a four-step coupled two-dimensional discrete ordinates calculation for determination of the neutron and secondary gamma-ray dose rates in the FTR head compartment. An assessment of computational methods development plans is presented.

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Introduction

In the 1962 Reactor Handbook, the comparison method of shield design is described as a simple empirical procedure by which performance data for a known reactor and shield are used to predict the results for a new configuration.[1] At the 1970 ENEA/IAEA Specialists Meeting on the Physics Problems of Reactor Shielding, the UKAEA paper title poses the question, "The Science of Reactor Shielding - An End to Empiricism in the 1970's?" In this paper[2] a case was presented for the use of the semi-empirical removal-diffusion method with multigroup constants systematically adjusted to elicit agreement with a variety of integral experiments. This approach does not differ fundamentally from the old comparison method, and from this point of view, shielding calculations appear to be only complex interpolations tied to integral experiments with a short umbilical cord. An opposing point of view of the fundamental approach is that point cross sections are problem-independent quantities describing reaction phenomena, and that shielding radiation transport calculations employ mathematically sound approximations to obtain solutions of the exact transport equation. In this view, integral experiments serve to check calculational approximations and deficiencies in basic cross sections, but are not used directly to make adjustments in the mathematical model.

Both the empirical and the fundamental approaches offer the capability for solving shielding problems. While in the past the physical complexity required an empirical approach, linear radiation transport problems are presently solvable on the most modern computers with sufficiently good approximations and geometric realism to encompass most shielding problems. In taking advantage of this capability, exploitation of the fundamental approach reduces the need for prototype experiments and greatly increases the accuracy and range of applicability of shielding radiation transport calculations. It is important that one programmatically adopt either the empirical or fundamental approaches to shielding and conduct research and application to support the chosen approach rather than trying to contribute to both since this would require duplication of effort and a tendency to switch back and forth, thereby reducing the rate of progress of both.

In this paper, a shielding program consistent with the fundamental approach will be discussed. An approach to optimally utilize the present and immediately foreseeable capability in order to obtain accurate calculations at the lowest cost is presented. As an example of present capability, the use of sequential overlapped coupled DOT-III[3] calculations of an important FTR shielding problem is discussed. Proposed improvements to the present calculational techniques are outlined.

Elements of a Shielding Method

Advanced shielding methodology originated largely as a result of demanding shielding requirements in the aircraft nuclear propulsion program (ANP) and the space nuclear program (SNAP) and is now actively pursued by the military Defense Nuclear Agency (DNA) and AEC space nuclear systems (SNS) and liquid metal fast breeder reactor (DRDT-LMFBR) programs. This methodology has developed rapidly since the mid 1960's when the Monte Carlo and discrete ordinates methods began to utilize the new super-computers. The basic characteristics of the advanced method are the flexible ability to solve widely different problems with the same data and technique and the ability to achieve an exact solution for any problem within the limits of the cross-section data or to economize for a particular problem type by appropriate adjustments to the method so as to not overcalculate the problem.

By far the most important assumption is that almost any practical problem can be solved exactly, and, hence, that observed errors can be eventually traced to deficiencies in the cross sections or the transport calculational method, and that, when corrected, the improved cross sections or method is applicable to any problem of similar requirements.

The advanced shielding method must include five basic technologies:

- (1) cross-section measurements
- (2) cross-section evaluations
- (3) cross-section processing
- (4) integral experiments
- (5) radiation transport calculations

Although the area of transport calculations receives the most detailed discussion in this paper, the status of work in the other four categories is very briefly summarized below since the actual shielding "method" must utilize all five interdependent technologies.

(1) Cross-Section Measurements. For years the shielding community has needed improved measurements of the total cross sections, especially at the interference minima, and of secondary gamma-ray production cross sections for all neutron energies. Some feel that it is impossible to measure cross sections with sufficient accuracy for shielding.[2] This opinion largely derives from experiencing long delays in obtaining cross-section measurements adequate for shielding. Improved communication has led to significant progress as is demonstrated by improved total cross-section measurements for iron[4] and an efficient procedure for secondary gamma-ray production cross-section measurements.[5] While all cross sections cannot be measured to high precision in the available time, sensitivity analyses and integral experiments can be used to determine priorities and requirements for measurement of the most important cross sections.

(2) Cross-Section Evaluations. The evaluator must provide the calculator with a complete point cross-section set whether the appropriate data exists or not. He is the communication link between measurer and calculator, and he receives and participates in the conclusions of integral experiment analysis. It is important that the evaluator be continuously available in this role. As sensitivity analysis of integral experiments indicate possible cross-section errors, it is necessary for the evaluator to act on this information. Otherwise, an impasse is reached which can only be resolved by empirical adjustments or the proliferation of separate evaluations.

As an example of this process, within the last year analysis of an iron benchmark experiment[6] led to the remeasurements of the iron total cross section between 20 and 300 keV revealing factors of 2 errors in the interference minima[4] and within the last few months reanalysis of the iron broomstick experiment[7,8] has called for new total cross-section measurements between 1 and 4 MeV where 10% errors in the minima are suspected.

(3) Cross-Section Processing. The ENDF system format now includes sufficient flexibility for most shielding problems and the official version-III evaluations for many materials are expected to be useful for both shielding and core physics. In order to solve widely different problems, multigroup shielding calculations require the utmost in flexibility for group energy widths, scattering angular distribution expansions, and weighting functions. For several years, the most flexible neutron multi-

group processing code was SUPERTOG[9] which had combined the virtues of CSP[10] and ETOG.[11] Secondary gamma-ray production cross sections were processed by a variety of codes such as POPOP4[12] since early ENDF evaluations had no data. These processing functions have now been inherited by the modular code system AMPX[13] which operates entirely from ENDF libraries and prepares neutron, gamma-ray, or coupled neutron-gamma-ray multigroup libraries. While the major effect of the new code is to provide greatly increased flexibility and quick response to processing requests, there is also opportunity for improved standardization through record keeping which identifies multigroup sets with regard to processing options as well as the mat and mod numbers of the point data file.

(4) Integral Experiments. In an empirical approach to shielding calculations, integral experiments serve the purpose of providing data for adjusting the computational algorithm. As such the experiments must closely resemble the design configuration and revised experiments are needed as the design changes even if essentially the same materials and sources are involved. In addition to the high cost of a continuous stream of experiments, experience has shown that shield designs change too quickly compared to the time needed for experiments, and one quickly is in the position of comparing apples and oranges.

In the advanced approach for shielding one can logically separate experiments for data testing and experiments for verifying basic methods or which serve as an expedient means of checking a complex calculational scheme having basic methods which are well known. A clear deficiency in this approach has been the need to determine if the data testing integral experiments are sensitive to the data in a way similar to the requirements of the design problem. Intuitive judgment and experience are most likely inadequate in this area and a sensitivity analysis approach[14] is being developed to fill the need.

In this approach, there are significant time delays in obtaining new cross-section measurements or in improving the calculational methods. When design decisions are required before the improvements are realized the present policy is to estimate the design problem uncertainty from the knowledge of the integral experiment.

The CSEWG shielding subcommittee has acknowledged several data testing experiments requiring only simple transport calculations[8,15,16] and one requiring a complex calculation.[17] Recent deep-penetration data testing experiments for iron[7] and sodium[18] are being fully documented. In all cases there is a clear need for sensitivity analyses when these experiments are used to evaluate uncertainties in a given design problem.

Integral experiments for methods verification ideally would involve only materials for which the cross-section data are adequately known. However, the lack of sufficient cross-section data and a fully developed sensitivity technology has led to a continuation of prototypical experiments for methods verification invariably involving both methods and data testing. In the analysis of such experiments the most exact methods are used and checked through intercomparison and data deficiencies are identified. When these are corrected through measurements and evaluations, the analysis should agree with experiment within experimental accuracy. Complex integral experiments are therefore regarded as data testing experiments where the appropriate sensitivity is incorporated by a prototypical configuration. In such experiments, prototypicality must include the source and detector response as well as material configuration. A recent development is the technique for

incorporating spectrum modifiers in such experiments to obtain prototype source spectra.[19,20] Methods verification is best performed through calculational intercomparisons, but this approach sometimes requires more time although less expense than experimental verification.

The CSEWG shielding subcommittee has acknowledge two methods testing experiments[21,22] while several others are being documented.[20,23-25] A peculiar example is the FTR lower axial shield prototype[20] where the streaming path was so complex and the need so urgent that an empirical streaming factor was sought. Although the conclusions obtained from evaluation of the experimental results were of significant value to the FTR design, the exercise demonstrated that the ambiguity in the empirical adjustment factor was large compared with the desired uncertainty.[19]

(5) Radiation Transport Calculations. As stated in the introduction, the key ingredient of the advanced shielding method is the ability to perform essentially exact numerical solutions of the linear Boltzmann transport equation for complex systems of practical interest. Although all facets of the shielding technology are required and are interdependent, this calculational capability is the crucial ingredient. Time and space do not allow a presentation here to substantiate this claim for the proof consists of a growing body of successful calculational comparisons and integral experiment analyses. A major aspect in achieving this area of confidence is the fact that the majority of practical problems are amenable to solution both by Monte Carlo and discrete ordinates methods in which the approximations are adjustable with ultimate accuracy limited only by computer capabilities. The utilization of general transport codes which do not have problem- or project-dependent limitations provide the methodology with flexibility. Similarly, for specific problem types time-saving approximations can be compared directly with precise solutions providing economy. This frees one from evaluating approximations by comparisons with integral experiments which invariably differ from the design problem.

At ORNL the most frequently used transport codes for reactor shield analysis are the multigroup MORSE[26] Monte Carlo and ANISN[27] and DOT-III[3] discrete ordinates codes. When non-multigroup "point" cross-section calculations are required, the O6R Monte Carlo code has been widely used in the past while for simple one-dimensional and two-dimensional problems point cross-section calculations with ANISN and DOT are being increasingly used.

There are four application areas which employ the transport codes with differing emphasis: (1) design support calculations, (2) integral experiment analysis, (3) cross-section sensitivity studies, and (4) automatic shield optimization. For the LMFBR program, an example of work in area (1) is given in this paper, areas (2) and (3) in companion papers,[14,18,24] and area (4) has been limited to space reactor shielding.[28]

A Plan for Economical Design Support Calculations

Overall economy in radiation transport calculations for LMFBR shield design can be achieved through a systematic approach utilizing the best features of each technique of the computational methods. An example of the possibilities is a procedure utilizing spacial coupling between discrete ordinates calculations and discrete ordinates and Monte Carlo calculations.

While limited to one- and two-dimensional geometries, multigroup discrete ordinates is the tool with greatest flexibility in the degree of approximation as one can vary the number of groups, angles, space points, and

scattering expansion greatly reducing the cost in many cases while preserving the same accuracy. In addition to discrete ordinates, ANISN and several versions of DOT offer diffusion theory as an option by group and one can use diffusion theory mixed with discrete ordinates in the coupled calculations. Spacial coupling of DOT and MORSE calculations with the DOMINO[29] program offers a capability that makes solution of three-dimensional problems possible in a time frame compatible with shield design. This occurs when the deep penetration parts of the problem can be well approximated in two dimensions and the three-dimensional Monte Carlo part of the problem can be solved without extensive biasing.

Several discrete ordinates calculations can be advantageously coupled along spacial surfaces to solve a single problem. This procedure follows from the fact that one can take from a problem the angular flux at an interior surface, and, using this as an external boundary condition, delete the part of the problem on one side of the boundary and obtain identically the same result on the remaining side. The advantage is that, while the angular flux at the surface should include all scatterings on both sides of the surface, the entire problem often need not be included. This allows one to bootstrap several small problem solutions to obtain the solution for a large problem.[30] Additional flexibility is gained by using an adjoint calculation for part of the problem and by the option to obtain an external surface angular flux from the first problem and use it as an internal boundary condition in the second problem.

The overlapped spacial coupling is used to advantage in three instances: (1) when a problem is too large for existing codes and computers and must be divided into subproblems, (2) when with existing codes one needs to change spacial mesh, group structure, or angular quadrature within a problem, and (3) when one can more efficiently perform parametric studies by isolating unchanging parts of the problem. In case (1) the total computer time used is greater than that which would be used to solve the entire problem in one calculation, but in cases (2) and (3) significant computer time can be saved.

The capabilities of forward and adjoint coupled calculations can be used to provide for inexpensive shield design studies. The procedure is to periodically determine baseline radiation environments by performing a forward two-dimensional discrete ordinates transport or transport diffusion calculation for the entire plant design to the extent where direct neutron and secondary gamma-ray radiation are important. At the same time, similar adjoint calculations are performed for each major criteria detector point (radiation damage response at core support and vessel points, various reaction rates in the radial cavity, and dose or activation responses in the head compartment and heat exchanger vault). Although these baseline calculations are expensive they would likely be performed only four times a year or so. At any surface in the system the product of the forward angular flux and one of the adjoint angular fluxes gives the contribution per unit surface area to the detector represented by the adjoint. Integration over a surface effectively enclosing the core or the detector point gives the entire detector response.

These periodically updated forward and adjoint angular fluxes should be available for any purpose in core or shield design. Individual shield design studies then require calculations covering only the zone being designed. For example, the effects of materials or geometry changes in the radial shield would use the forward angular fluxes to define the reflector-shield or blanket-shield boundary condition and the adjoint fluxes at the

outer surface as response functions. The radial shield design calculations could use discrete ordinates, diffusion, Monte Carlo, or any other method in any approximation. The calculation would be limited to the shield zone and the approximation could be checked against the existing discrete ordinates calculation for the unmodified design. The shield zone calculation could be one-, two-, or three-dimensional and effects of improvements to the baseline radial shield calculation could also be checked. One possible arrangement in the LMFBR community would be for the national laboratory shielding group to perform the large forward and adjoint calculations and provide computer codes, and for the industrial design groups to utilize these results to perform the inexpensive parametric study calculations. With this combination, the designer has the full power of the large calculation results to use in obtaining quick design change evaluations.

In addition to the procedure described above, the volume integration of the equivalent source[31] or adjoint difference[32] scheme can be used. In the adjoint difference approach the change in response at the i th detector is given by

$$\delta R_i = \int_{VOL} \phi_P^* [\delta S + \delta \Sigma_S \phi - \delta \Sigma_T \phi]$$

where the symbols are from the operator form of the transport equation,

$$L\phi + \Sigma_T \phi = \Sigma_S \phi + S,$$

ϕ_P^* represents the perturbed system adjoint flux for the i th detector and $\delta \Sigma_S = \Sigma_S^P - \Sigma_S$, $\delta \Sigma_T = \Sigma_T^P - \Sigma_T$, and $\delta R_i = R_i^P - R_i$. The integration is only over the volume where the changes occur and if the unperturbed adjoint flux is used, the approximation is that of linear perturbation theory. Therefore, from the baseline forward and adjoint calculations, a linear perturbation approximation to the effects of radial shield composition changes on all the important detectors could be obtained directly by simple integration. These baseline forward and adjoint calculations also provide all the information for cross-section sensitivity analysis.

The procedure proposed here is less expensive than the present shield analysis for FTR. The baseline calculations are expensive but their frequency of occurrence is reduced. More importantly the angular fluxes are to be available projectwide and redundant calculations are limited to checking purposes. Of course the prime feature is the ability to minimize the calculational expenditures for design studies by utilizing the baseline calculations to their fullest extent.

The DOT-III[3] transport code and FACT[33] spacial coupling code utilize the spacial coupling procedure. Both forward-forward and forward-adjoint coupling have been performed in previous studies.[34,35] Some of the more recent calculations use many of the features of the procedure discussed above and the example presented here demonstrates the capability of the method.

An Example of Present Capability Analysis of the FTR Head Compartment Shielding Problem

Figure 1 shows a somewhat obsolete drawing of the FTR with many details, such as cavity shielding, omitted. After many studies it was determined

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Fig. 1. A Pictorial Drawing of the FFTF Reactor.

that the dominant neutron transport path contributing toward neutron and secondary gamma-ray dose in the head compartment was transmission through the radial shield, induced fission in the stored fuel arrays, leakage into the radial cavity, and streaming through the complex vessel support system. In order to reduce the dose levels in the head compartment, the addition of a magnetite shield deck and B_4C shield ring in the radial cavity, and a borated polyethylene and steel shield floor in the head compartment are being studied. The shield design is performed by Westinghouse ARD with supporting radiation transport calculations performed by ORNL.

Because of spectra differences the neutrons from fissions in the stored fuel dominate the dose rate above the cover by two orders of magnitude as compared with neutrons originating in the core. The fission distribution in the core was obtained from a diffusion mode DOT calculation and the total fission power in the stored fuel was determined from a three-dimensional coupled DOT-DOMINO-MORSE calculation.[36] The first-round analysis of the shielding problem utilized a four-step sequence of overlapped DOT-III cylindrical r - z calculations with fixed source distributions in the core and stored fuel. The stored fuel source was distributed uniformly over a homogenized annulus normalized to the total stored fuel power obtained from the three-dimensional calculation.

The first step in this sequence was a 50-neutron group S_6 P_1 13328 point DOT calculation for the reactor from the midpoint of the core extending out to a radius of over 550 cm and axially up to a height of 730 cm. This calculation could be used to define the source incident to the vessel support system; however, it is assumed that the S_6 quadrature is inadequate for accurate determination of streaming in the cavity shield gaps, the geometric representation of the gap between the vessel and B_4C shield is poor, and gamma rays are not included.

A vertical boundary angular flux tape was obtained from step 1 at a radius of 295 cm which is at the outer surface of the baffle. The FACT code was then used to convert this boundary angular flux to an appropriate external left boundary flux for a 21-neutron group - 18 gamma-ray group S -166 P_1 2115 point calculation of the cavity. The purpose of this step 2 of the calculation is to obtain an adequate definition of streaming and include secondary gamma-ray transport in the cavity. The mesh in this step was different from that used in step 1, and separation of the cavity problem and shield deck problem was desired because additional shield deck configurations were to be studied. Step 2 produced a boundary flux tape for a horizontal boundary at an axial height of 500 cm.

Step 3 was a 21-neutron group - 18 gamma-ray group S -166 P_1 2880 point calculation and included the B_4C ring and magnetite deck in a zone extending radially from 295 to 555 cm and axially from 500 to 660 cm. The FACT code was used to prepare a bottom boundary flux condition from step 2 and a left boundary flux condition from step 1. Therefore, the calculation of this isolated zone incorporates the results of transport in zones below and to the left through the boundary conditions. A horizontal internal boundary angular flux tape was obtained at an axial height of 630 cm. The step 3 calculation extended to a height of approximately 660 cm, and it was assumed that the 30-cm overlap was sufficient to include the reflections from the materials above 630 cm.

The FACT code was again used to prepare a bottom boundary source for step 4 which was a 21-neutron group - 18 gamma-ray group S -166 P_1 4160 point calculation of the head region from the centerline to a radius of

555 cm and from an axial height of 630 cm to 886 cm. In this case the bottom boundary source from $r = 0.0$ to $r = 295$ cm was taken from step 1 results and the source from 295 cm to 555 cm was from step 3 results. This includes the thermal shield, cover, upper vessel flange, support arms, Z ring, support ledge, and the proposed head compartment shield floor consisting of 3 in. of borated polyethylene and 1 in. of steel.

Figures 2-5 show computer plots of the geometries of steps 1-4 and the neutron isodose contours determined by the DOT calculations. Figure 6 shows the secondary gamma-ray isodose contours for step 4. The portion of the step 4 calculation from the centerline to a radius of 295 cm was omitted from the plot to improve the enlargement of the complex section to the right of 295 cm.

Figure 2 shows the shape of dose contours in the sodium pool and reactor cavity, while Fig. 3 shows that the dose contours in the S-166 cavity case are not greatly different from those in Fig. 2. Figure 4 indicates that streaming is important in the inverted "L" gap between the B_4C shield and the magnetite shield and in the dogleg gap between the vessel flange and the B_4C shield. However, the design is reasonably well balanced inasmuch as additional shielding in this area would be circumvented by leakage outward from the vessel. The mission of the cavity shield is therefore to reduce the radiation incident on the head from cavity scattering to a level comparable with that from the sodium pool. The dimensions in this calculation are at operating temperatures since considerable movement takes place during startup.

The neutron dose contours in Fig. 5 show the importance of streaming in the gap between the vessel and the support arms. Dominant mechanical considerations excluded shielding from this area resulting in an imbalance in the shielding effectiveness. This calculation indicates that the maximum neutron dose at the top of the shield floor is 0.3 mrem/hr.

The gamma-ray dose contours in Fig. 6 show the effects of the large gap although in this case the gap between the Z ring and the support ledge is also an important streaming path. The calculation indicates that the maximum gamma-ray dose at the top of the shield floor is 0.43 mrem/hr.

In the four-step calculation described in the preceding, the coupling procedure was used between steps 1 and 2 to change group structure (thereby incorporating gamma-ray groups), to change angular quadrature to allow use of a 166-angle set biased so as to provide a reasonably good calculation of streaming in annular slits, and to revise the mesh allocation reducing the running time and memory requirements for the S-166 cavity calculation. The group structure and angular quadrature were common to steps 2, 3, and 4. The separation of these steps was primarily used to reduce problem size and to reallocate the mesh in order to more accurately define the geometry.

A variation of the above procedure can be used to illustrate the capability for parametric design calculations. An adjoint calculation of step 4 may be performed to obtain the importance distribution (adjoint angular flux) at the 630-cm height surface. Given this tape and the top boundary angular flux tape from step 3, the FACT code performs the necessary integration giving the dose rate above the head compartment shield floor. A study of the effects of various clearance gaps in the B_4C and magnetite shield system then only involves the repetition of step 3 with its input boundary source tapes from steps 1 and 2, and the FACT code folding with the tape from adjoint step 4 to give the resulting changes in dose in the head compartment. The cost of the parametric study is reduced to a minimum

Fig. 2. Geometry and Neutron Dose Contours For Step 1 of the Calculation - An S_6 , P_1 , 13328 Point R-Z Model of the Reactor.

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Fig. 3. Geometry and Neutron Dose Contours For Step 2 of the Calculation - A More Detailed S-166, P₁, 2115 Point R-Z Model of the Radial Cavity.

Fig. 5. Geometry and Neutron Dose Contours For Step 4 of the Calculation - An S-166, P₁, 4160 Point R-Z Model of the Cover, Vessel Support System, and Head Compartment Shield.

Fig. 6. Geometry and Gamma-Ray Dose Contours For Step 4 of the Calculation - An S-166, P₁, 4160 Point R-Z Model of the Cover, Vessel Support System, and Head Compartment Shield.

by calculating only over the section being changed. Additionally, DOT need not be used in the step 3 calculations, but point kernel, Monte Carlo, and, if it were applicable, even diffusion theory could be used through appropriate coupling with the boundary conditions.

The three-dimensional aspects of this problem remain to be studied as the shield design progresses. The major concerns are penetrations of the shield floor in the head compartment and penetrations of the cavity shields by piping and other equipment. Coupling of intermediate results from these DOT calculations with MORSE is being used to investigate these problems.

Excluding these three-dimensional aspects of the problem, the uncertainty due to cross sections, multigroup approximations, P_0 truncation, and approximations inherent in the discrete ordinates method is estimated to be a factor of five. Since the sensitivity analysis tool is not yet in routine use to correlate integral experiments and design problems, this uncertainty estimate contains a large portion of intuitive judgment. Studies which contribute toward the verification of this calculation include integral experiments for neutron transport in iron,[6] stainless steel,[37] sodium,[18] neutron scattering in a concrete walled cavity,[25] and neutron streaming in offset annular slits[23] and coolant pipe chase-ways.[24] Although the iron secondary gamma-ray production cross sections have been experimentally verified,[15,16,38] the neutron-to-gamma-dose ratio at the top of the cover is anomalously high, and a suitable integral experiment should be performed to verify the gamma-ray transport problem.

Desired Transport Method Developments

The present capability in diffusion theory, discrete ordinates transport theory, and Monte Carlo is extensive. The capability, however, is useless if the elapsed time for problem solving (not computer time) greatly exceeds the time required for effective design input. The continuing goal is to reduce the time for problem definition to debugged problem solution. The following is a brief list of development items which would improve the present capability:

(1) An improved two-dimensional discrete ordinates code having the capabilities of DOT-III plus zone-dependent spacial mesh and quadrature. This removes the inefficient overlap used in the present spacial coupling.

(2) The development of a general geometry capability with input similar to that of the combinatorial Monte Carlo geometry.[39] This geometric module should be used for both Monte Carlo and point kernel three-dimensional calculations and as the basis for two-dimensional models.

(3) Improved general-purpose programs for coupling discrete ordinates calculations. This would include a more general version of FACT,[33] and a general program for volume coupling with the adjoint difference method.

(4) An internal ray-tracing scheme similar to that of Vossbrecker [40] incorporated for use in diffusion theory and low-order discrete ordinates calculations.

(5) Increased capability for three-dimensional calculations to augment the present Monte Carlo and diffusion theory codes.

These improvements are needed but a higher priority is devoted to efforts for improving the means for utilizing the present capability in the most efficient way.

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