SHIELDING METHODS DEVELOPMENT IN THE UNITED STATES*

F. R. Mynatt
Oak Ridge National Laboratory
Oak Ridge, Tennessee, U.S.A.

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ABSTRACT

A generalized shielding methodology has been developed in the U.S.A. that is adaptable to the shielding analyses of all reactor types. Thus far used primarily for liquid-metal fast breeder reactors, the methodology includes several component activities: (1) developing methods for calculating radiation transport through reactor-shield systems; (2) processing cross-section libraries; (3) performing design calculations for specific systems; (4) performing and analyzing pertinent integral experiments; (5) performing sensitivity studies on both the design calculations and the experimental analyses; and, finally, (6) calculating shield design parameters and their uncertainties. The criteria for the methodology are a 5 to 10% accuracy for responses at locations near the core and a factor of 2 accuracy for responses at distant locations. The methodology has been successfully adapted to most in-vessel and ex-vessel problems encountered in the shield analyses of the Fast Flux Test Facility and the Fast Flux Test Facility and the Clinch River Breeder Reactor; however, improved techniques are needed for calculating regions in which radiation streaming is dominant. Areas of the methodology in which significant progress has recently been made are those involving the development of cross-section libraries, sensitivity analysis methods, and transport codes.

A SYSTEMS VIEW OF SHIELDING METHODS

When considering the subject of shielding methods, one often directs attention to a computer code or to a particular method or algorithm for solving the transport equation. If, however, one considers shielding which is available or can be made available and the design decisions or analysis results that are required, it is clear that the tools available to the shielding analyst form a complex system. There are a number of alternative paths by which one can achieve the desired results. These alternative paths may differ in their inherent accuracy or in the ultimate accuracy that can be obtained, and they may differ in their ability to handle problems other than those for which they are immediately designed. All systems, however, share common features and their differences lie primarily in the way the various possible resources are weighted in putting together an approach toward desired answers. A complete and comprehensive study of a given shielding methodology would be extensive and is beyond the scope of this paper. Features of the U.S.A. shielding methodology will, however, be discussed here, and these may be at least indicative of the dimensions of a more complete consideration.

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An elementary block diagram of the primary shielding methodology employed in the United States is shown in Fig. 1. This methodology is adaptable to all neutron and gamma-ray transport and effects problems for all major power reactor types, although to date it has been used primarily for the loop-type plutonium-fueled liquid-metal fast breeder reactors (LMFBRs) pursued so vigorously for the past seven years. In particular, it has been applied to the Fast Flux Test Facility (FFTF) and the Clinch River Breeder Reactor (CRBR). The required accuracy i 5 to 10% for responses at locations distant from the core.

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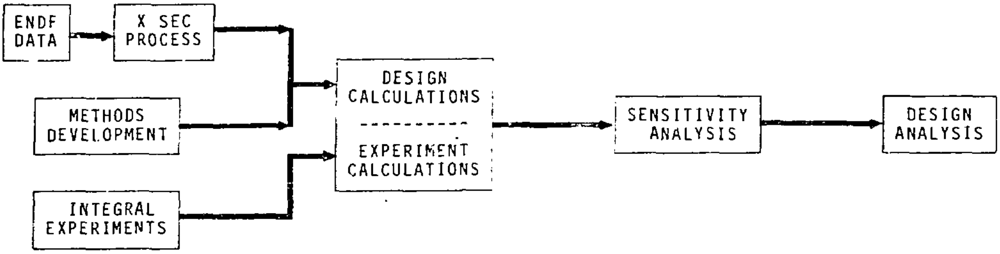

Fig. 1. Flow Diagram of U. S. Reactor Shielding Methodology

As shown in the figure, the methodology includes a method development activity and a cross-section processing activity which combine to provide the capability for performing design calculations or experimental analyses. The experimental analyses are tailored to provide data for testing the calculational methods and/or the cross-section data for particular design problems. The methodology also includes a sensitivity analysis capability which establishes the uncertainties on the calculated results that are due to uncertainties in the cross sections used in the calculations. Finally, design parameters with specified uncertainties are selected on the basis of all these results. Of course, in actual practice, the methodology includes numerous feedback paths between the various activities.

The computer codes provided by the methods development activity are general-purpose one- and two-dimensional discrete ordinates transport codes and three-dimensional Monte Carlo transport codes in the multigroup formalism. This primary calculational capability is backed up by "point" transport codes used as research tools to verify the adequacy of the energy group structure and other features of the multigroup codes as they are adapted to specific design problems. Insofar as possible, the latest ENDF data are used for processing into the multigroup (or point) cross-section sets.

The capability for sensitivity calculations is provided by special versions of the discrete ordinates codes utilizing generalized perturbation theory. The processed cross sections contain uncertainties and correlations, and these are used by the sensitivity analysis codes to estimate the *a priori* uncertainty in the results obtained in the design calculations.

In its simplest form, an analysis of an integral experiment provides a bias factor that can be applied as appropriate to the design calculation, the bias factor being derived from the ratio of the experimental result to that obtained in the analysis. However, much more information can be obtained from integral experiments. For example, a sensitivity analysis which shows whether the experimental analysis and the design calculation are sensitive to the same cross sections will indicate the relevance of a bias factor. Also, with the sensitivity analysis capabilities available, a mathematically rigorous estimate of design parameter uncertainties can be made in whoch the effects of discrepancies between pertinent experiments and their analyses are considered directly. In this procedure, the cross sections (and their uncertainties) used in the final design calculation have been adjusted with sensitivity analysis (usually within the bounds of their uncertainties) to agree with or approach the cross sections required for an experimental analysis. This approach yields cross sections that agree with measured quantities after the measured values have themselves been adjusted within the bounds of its uncertainty. The cross sections emphasized are those to which the experimental analysis (and the design calculation) are most sensitive The net effect is that the uncertainties on the resulting design parameter are reduced. This procedure is, in fact, a much more sophisticated method for biasing design-related results and determining the associated uncertainties than simply applying a bias factor, although it still is problem-dependent to some degree.

Obviously the various components of this shielding methodology are interrelated and the effect of each component on the ultimate precision of a design parameter is difficult to isolate because of compensating or overriding effects of other components. However, the methodology is sufficiently general that by shifting the emphasis in data files and integral experiments, it can be extrapolated or adapted to all reactor types. Thus for a large diversity of problems, such as are currently arising in the U.S., it should be more efficient that a methodology solely for a specific reactor system.

REVIEW OF LMFBR SHIELDING CAPABILITY

At this point it is useful to review the progress that has been made in the U.S. shielding methodology as it has been used for LMFBRs. With respect to in-vessel shielding problems, it can be stated that an adequate methodology exists for calculating six major problem areas: (1) neutron transport in deep sodium, (2) neutron transport in thick iron and stainless steel, (3) the shielding effectiveness of the upper axial and head shielding, (4) the shielding effectiveness of the radial blanket and shield, (5) neutron penetration through the lower axial shield and its sodium coolant channels, and (6) stored fuel effects. Here the term adequate methodology implies that while there are aspects of these problems that still cannot be calculated accurately, design calculations for the importance parameters of interest are accurate to within the range of the projected accuracy needs.

The capability for calculating neutron transport through thick sodium is crucial to LMFBRs, and a recent sensitivity study[1] of a series of Tower Shielding Facility experiments[2] has greatly advanced our understanding of this problem. This work led to remeasurements and reevaluations of some of the sodium cross sections, and it also showed that the important radiation transport phenomena in sodium change dramatically with the incident neutron spectrum, the exit response function, and the media preceding and following the sodium zones.

The problem of neutron transport in iron and stainless steel is somewhat less complex. It has been determined that dominant iron minima must be represented explicitly in the cross-section set but that the number of such minima are relatively few. New multigroup cross-section sets explicitly contain these minima (as well as the extremely important 300-keV minimum in sodium). Even so, the neutron spectrum above 1 MeV cannot be calculated accurately for transmission through stainless steel, and for the extremities of the radiation problems this may lead to bias factors as large as a factor of two.

The upper axial and head shielding problem is primarily that of deep neutron penetration in sodium preceded and followed by stainless steel. Within the limits mentioned above, experiments have confirmed that this type of problem can be calculated with acceptable accuracy.

For the problem of the radial blanket and shield, experiments have confirmed that the calculations are near the desired accuracy for the materials considered (stainless steel and inconel). However, for the larger LMFBR cores, lighter materials will be required to reduce the weight of the radial shield, and additional experiments will be necessary to verify the methods for these materials.

The lower axial shield problem is very similar to the radial shield problem since the sodium channel diameters required to obtain acceptable pressure drops are still sufficiently small to minimize streaming effects.

The production of fission neutrons in the fuel stored inside the reactor vessel was a particularly dominant effect in the FFTF design and to a lesser extent in the CRBR design. The complications caused in the shielding analyses were quite severe, and we are fortunate that in-vessel fuel storage may not be a design choice of the future.

With respect to LMFBR ex-vessel shielding problems, at least five problems areas, all of them involving radiation streaming, still require an improved methodology in order to meet the factor of two criterion. The first four involve neutron streaming, the regions of concern being (1) the primary pipe chaseways, (2) heating and ventilation ducts, (3) the reactor cavity, and (4) the gaps around the rotating plugs in the reactor head. The fifth involves gamma-ray streaming through plant shield penetrations.

The problem of neutrons streaming in primary pipe chaseways and activating the sodium coolant in the secondary sodium loop exceeds the capabilities of 3-D Monte Carlo codes with respect to statistical accuracy and overall reliability, and the geometry precludes the use of any existing or potential discrete ordinates codes with any degree of confidence. Fortunately, the primary piping in the FFTF and CRBR contain enough bends within the reactor cavities to essentially eliminate the problem. However, the large LMFBR designs tend to have less space in the cavity, with no bends at all allowed in some cases. Thus streaming will occur and the capability for calculating it must be developed. The problem of neutrons streaming through heating and ventilation ducts is very similar, except that in this case the concern is usually the contribution to some neutron response, such as dose rate.

Neutron streaming in the reactor cavity was a severe problem for both the FFTF and the CRBR and will be for virtually all similar reactors. In order to meet the criterion for locations outside the cavity, parametric experimentation will be required to verify the methods used.

The problem of neutron streaming through the gaps around the rotating plugs in the reactor head, identified early in the FFTF shielding program, will be particularly important for designs in which the reactor cavity streaming effect is successfully sovled or for pot-type LMFBR designs that inherently have minimum cavity streaming effects. Again, parametric experiments will be needed to verify the accuracy of the methods.

The problem of gamma-ray streaming in plant shield penetrations is compounded in an LMFBR because of the very high gamma-ray activity levels in the primary system piping, the very hot pipes which require thick insulation in the pipe penetrations, and the desire to minimize overall plant dimensions. There are several ways to reduce gamma-ray streaming from cell to cell in the plant, but when several bends are involved in the penetrations, the analysis methods are inadequate and experimental verification is insufficient.

RADIATION STREAMING METHODS

The preceding discussion points out the need for improved methods for calculating radiation streaming problems. Basically, a streaming problem can be defined as one in which a significant contribution to the radiation effect of concern is dominated by geometric attenuation introduced by the shielding materials. Thus, the geometrical feastures of a calculational method, as opposed to the deep-penetration features, are more significantly involved than in other types of shielding calculations.

In some cases, the geometry and materials used are such that a streaming problem can be calculated by a combination of collisionless flights and albedo reflections, or with semiempirical approaches. However, when significant contributions from material attenuation and diffusion must also be considered, the problems become quite difficult and the current methods are inadequate. The development of empirical methods is also difficult

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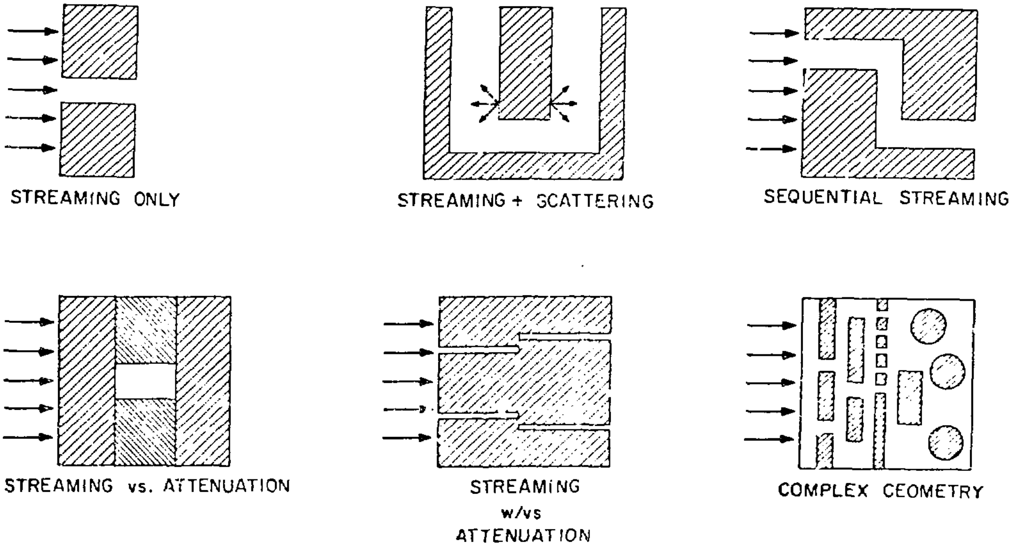
STREAMING ONLY

STREAMING + SCATTERING

SEQUENTIAL STREAMING

STREAMING vs. ATTENUATION

STREAMING w/vs ATTENUATION

COMPLEX GEOMETRY

Fig. 2. Typical Streaming Configuration

since the problems tend to occur outside the reactor vessel and any integral experiments mocking up these large regions would be quite elaborate and expensive.

Schematics representing several different types of streaming problems are shown in Fig. 2. First is the <u>streaming only</u> case which may be a simple situation that a trivial hand calculation can resolve. Next is the <u>streaming</u> + <u>scattering</u> case which also is not too difficult, even though the scattering sources must be identified. The case of <u>sequential streaming</u> requires both that the location of the scattering points be determined and that the spectral changes involved in the scattering be accurately calculated, but even this problem is relatively easy unless the streaming passageways are filled with materials that introduce additional scattering and attenuation, as happens when a pipe and insulation are inserted in a pipe chase.

Streaming in competition with or in sequence with attenuation, as shown by the two lower left sketches in Fig. 2, is a difficult calculation, but for relatively simple geometries it can be performed with the discrete

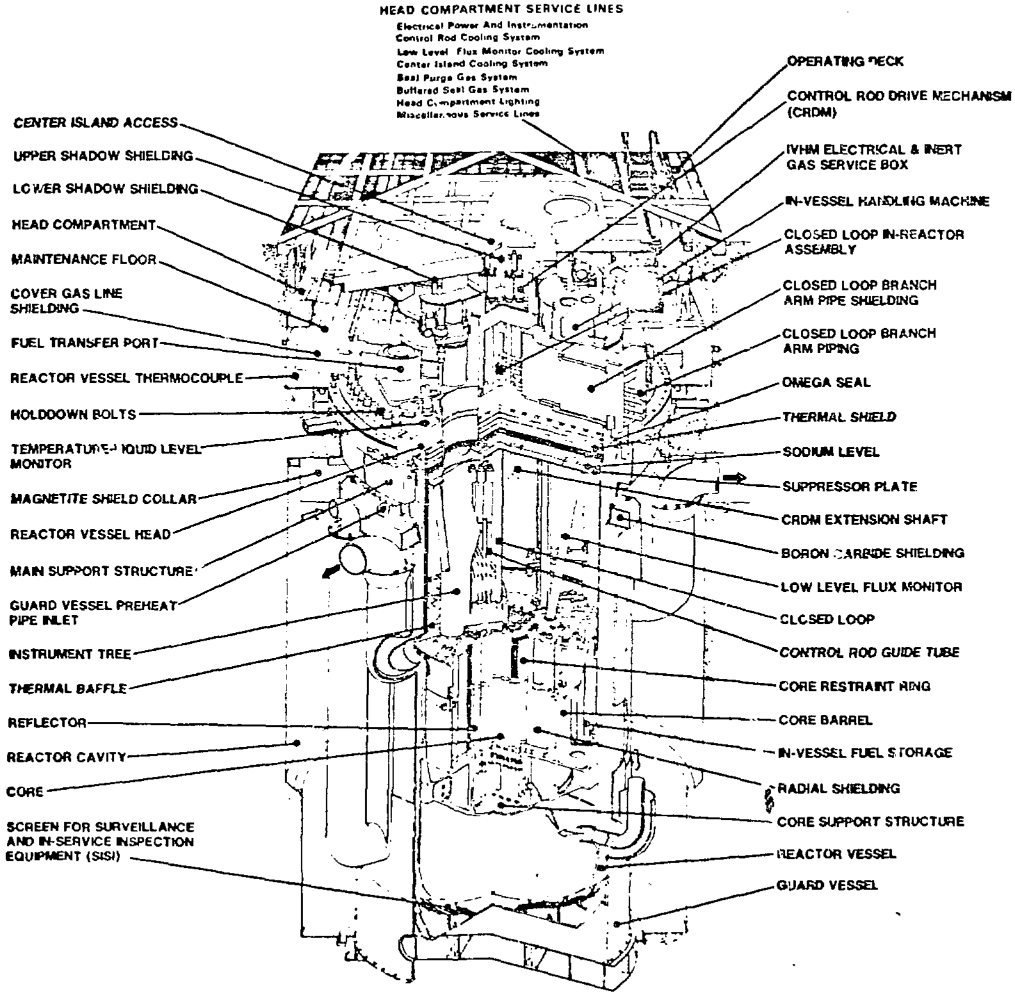

Fig. 3. Sketch of FFTF Head Compartment and Reactor Cavity Systems

ordinates method. This capability is due to two major features of the method: it can accurately calculate deep-penetration problems with anisotropic scattering in two dimenions, and in R-Z geometries and with biased quadratures, it can accurately calculate streaming.

The complex geometry sketch in Fig. 2 represents any system with a complex collection of components, pipes, wires, etc. in which neiter a clear and continuous streaming path nor a large zone of bulk material can readily be identified. Only with appropriate calculations can one determine whether a geometry of this type can be represented by a homogeneous material distribution or if certain "streaming" or "streaming + scattering" paths exist.

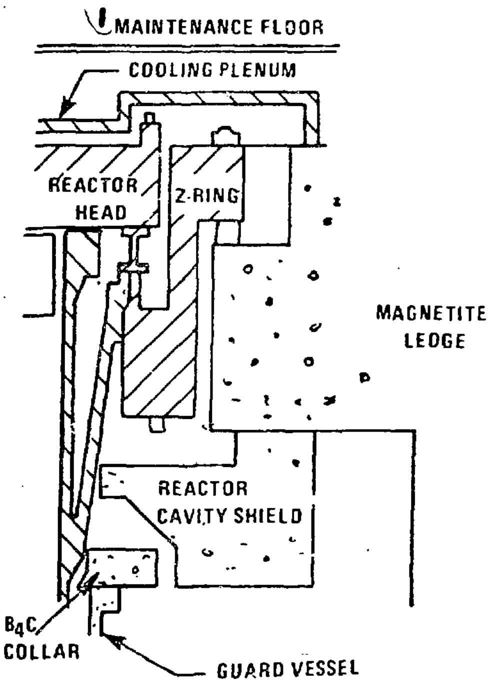

Fig. 4. Simplified Sketch of FFTF Vessel Support Ring and Reactor Cavity Shield.

A number of complex geometry streaming problems were encountered in the shielding analyses of the FFTF, particularly in the reactor cavity shield surrounding in the reactor vessel (see Fig. 3).[3] Consider, for example the region including the main support structure and the magnetite shield collar (also called the reactor cavity shield). A simplified sketch of this region is shown in Fig. 4. On an *a priori* basis, a dominant upward streaming path could not be determined, no could the scattering sources be located so that a simple void-streaming calculation would be possible. Detailed analyses of this problem were performed utilizing both discrete ordinates and Monte Carlo methods, and a technique had to be devised which would yield changes in the dose rates on the maintenance floor with changes in the dimensions or compositions of the reactor cavity shield and surrounding components. Thus the accuracy of the relative calculations had to be much greater than the accuracy of the absolute calculation. This requirement was met by using the discrete ordinates method when the geometry permitted its application and using the Monte Carlo method for very careful calculations in which the statistical error was less than the accuracy required for the nominal dose rate. Even so, some cases existed in which a design change did not result in a statistically significant change in the answer, and the only conclusion that could be made was that the answer was not sensitive to the design change.

The FFTF is not atypical of future reactor designs, and our shielding methodology must be prepared to address such problems with sufficient efficiency and accuracy. At least five areas in which additional development is needed are identifiable. First, improved codes for coupling discrete ordinates and Monte Carlo calculations are needed. Currently a surface coupling code is being used, but such methods need to be generalized and a volume coupling code based on the adjoint difference formulation[4] should be developed. Second, improved Monte Carlo biasing techniques are needed, and they should be relatively understandable by shield analysts who are not necessarily experts in Monte Carlo methods. Third, a 3-D discrete ordinates code (θ-R-Z) is needed for occasional use in problems that cannot be modeled in 2-D geometry and cannot be calculated with 3-D Monte Carlo methods because deep penetration is involved. Fourth, improved semiempirical techniques are needed, especially for multileg ducts such as those associated with primary loop piping or with plant shield penetrations. A technique for this problem could probably be based on a Monte Carlo method with albedo scatter, with the empiricism included as a built-in biasing scheme. This would be somewhat more acceptable than a purely empirical approach and should yield comparable computational efficiency and accuracy. Greater accuracy would be obtained if aspects of scattering and penetration are added to the multileg duct problem. Finally, a fifth need is for parametric verification experiments for the various problem areas that have been identified.

RECENT PROGRESS IN U.S. SHIELD METHODOLOGY

Several important improvements have been made in the U.S. shielding methodology in recent years, some of them resulting from efforts that are only peripheral to the shielding program. Three areas in which progress has been particularly noteworthy are (1) cross-section processing and multigroup library production, (2) sensitivity analysis methods, and (3) radiation transport methods.

At ORNL the techniques for processing cross sections into multigroup libraries have been modified more than once. In early work essentially all cross sections were processed in the GAM-II energy group structure,[5] which is relatively constant in energy. Later it was discovered that biased group structures in which specific groups are dedicated to dominant windows would solve deep-penetration problems, and the trend was to develop problem-dependent structures directly from point data. This approach, which was embodied in the AMPX processing system,[6] was subsequently found to be impractical. The current approach is a compromise in which a relatively detailed group structure is tailored or biased to include the important cross sections for several types of problems. This structure is then collapsed to broader groups for a specific type of problem, the major difficulty being the selection of the few-group structure. The cross sections are processed into the group structures by AMPX which uses the Bondarenko method (as embodied in the MINX[7] and SPHINX[8] codes) to include resonance self-shielding at the multigroup level. With this approach, a cross-section library consisting of 171 neutron groups and 36 gamma-ray groups was developed with the expectation that it would be collapsed for use in fusion reactor shielding problems in LMFBR core physics and shielding problem.[9] A few cross-section subsets have already been developed from this library and it now appears that a 45-16 subset will be used not only for fusion and LMFBR problems but also for light-water reactor shielding problems. Another feature of this library is that it will soon include delayed gamma-ray files and activation files.

As mentioned earlier, a limited capability is maintained for transport calculations using point cross sections rather than multigroup cross sections. The processing for such libraries has also been improved, and point calculations (with the discrete ordinates ANISN code[10]) have been performed for comparison with multigroup calculations using the 171-36 cross-section library. The library appears to be a good compromise, although many concerns remain; in particular, the treatment of high-energy neutrons in stainless steel may require a different group structure or even careful point calculations.

In the area of sensitivity analysis, several significant advances have been made, all of which are given in more detail elsewhere.[1] A modular code system called the FORSS system has been developed at ORNL for determining the sensitivity of calculated parameters of reactor and/or shield systems to the cross sections used in the calculations and assigning uncertainties to the calculated parameters based on uncertainties associated with the cross sections. The system requires that the evaluated ENDF cross-section data include uncertainty files, and some preliminary files have been completed for important shield materials. Also, as mentioned earlier in this paper, FORSS can adjust the cross sections for a given application on the basis of information received from pertinent integral experiments.

In a related effort, "channel theory" has been developed which can be used to determine the pathways in space followed by particles that successfully travel from their source to a location of interest. This technique, which is especially useful when shields contain streaming paths, is described in another paper.[11] Recently, work has been performed to extend this technique to channels in energy, which would contribute significantly to an understanding of the transport mechanisms in deep-penetration problems. The plans are that eventually channel theory will be included in the FORSS system.

In the third area of recent improvements, that of transport codes, the major new code at ORNL is Version IV of the 2-D discrete ordinates code DOT, which is discussed in a separate paper.[12] This code has two important new features: The first is that arbitrarily large problems can be solved with a reasonably small amount of computer memory, thereby reducing memory requirements for a given problem and eliminating the necessity for performing several overlapping calculations in order to solve a large problem. The second is that both the spatial mesh and the angular quadrature can be modified or changed by zone, which will allow a great saving in mesh points, and, more importantly, will allow biased quadratures to be used only in the regions where streaming is the dominant phenomenon. Test problems have shown that savings in computational time by factors of 2 to 10 are realized. In addition, DOT-IV has been completely reprogrammed and uses the latest and best acceleration techniques and numerical methods for both reactor physics and shielding problems.

Significant improvements have also been made in ORNL's Monte Carlo code MORSE. The latest version of this code, called MORSE-SGC,[13] employs a recent version of combinatorial geometry. It also has been completely reprogrammed so that it can operate with a small amount of computer memory, thus making it suitable for virtaully all present major computing machines and hopefully for future machines.

At Los Alamos Scientific Laboratory, efforts in transport methods in recent years have stressed the method of finite elements. Two of the codes, TRIPLET[14] and TRIDENT,[15] both available from the code centers, employ triangular mesh with linear discontinuous functions and an iterative weighted residual solution technique. The use of the triangular mesh allows greater freedom in allocating the mesh boundaries to follow curved or complex geometries in planar or cylindrical geometry; however, this new capability must be carefully tested before it can be applied in ongoing design efforts.

CONCLUSIONS

To summarize, it appears evident that the U.S. shielding methodology system is functioning in a reasonably balanced fashion. The understanding of the system aspects is sufficient that evaluation of improvements or accuracy in any major feature of the system can be performed in terms of overall system performance. In view of the U.S. reactor program redirection, it appears that the shielding methods system that has been chosen is appropriate and perhaps even fortunate in that it can be extended to other reactor types and design with no substantial loss of previous capabilities. In the review of the LMFBR shielding technology, it has been observed that most in-vessel and deep-penetration problems can be reasonably well handled within the current methodology. However, several ex-vessel problem areas are considered to be inadequate with respect to the current methodology and all of these areas are dominated by or heavily influenced by streaming phenomena. In further consideration of streaming problems, several needed improvements have been identified, including additional work in the Monte Carlo method and in a 3-D discrete ordinate codes in Θ-R-Z geometry. Also, a somewhat unique recommendation is made that a semiempirical technique be developed for ducts that is based on the Monte Carlo method with albedo scattering, using built-in albedos and an appropriate biasing scheme. Although the pace of methods development for shielding has slowed in recent years owing to the increasing capabilities of the accumulated methods, continued progress has been made in the areas of cross-section processing and libraries, sensitivity methods, and transport codes. These new capabilities will continue to improve the overall shielding methodology and will further improve our ability to redirect this methodology to alternative reactor fuels and types.

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Q(A.2) Certain codes are using reflection surfaces in order to cut down calculational running time. Did you ever look into the validity of such assumption? George G. Biro - Gibbs & Hall, Inc.

A(A.2) Yes. We have checked validity but not to the extent needed. The albedo approach is to be a time-saver for complex problems. As such, it is difficult to verify all the possibilities. We proceed therefore with somewhat incomplete verification and a large degree of faith.

Q(A.2) In connection with the use of albedo Monte Carlo method, what is your opinion on the present availability of albedo data? Carlo Ponti - EURATOM

A(A.2) This is an area of considerable concern. Albedo data should be generated and distributed like cross section libraries. We have recently generated albedo data for water, concrete, and steel over concrete. In addition to the problem of generating and preparing albedo data, there is a major problem associated with collapsing the data to reasonable quantities for design calculations.

Q(A.2) In your opinion, how far have we come since the late sixties in the improvement of techniques to handle streaming problems. It seems to me that the many analytical techniques available in the late sixties seemed quite adequate (if somewhat conservative) to handle complex streaming situations. H. E. Hungerford - Purdue University

A(A.2) Certainly, many problems could be handled adequately well by empirical or semi-empirical methods as used by experienced analysts. The Monte Carlo and discrete ordinates methods which also were developed in the late sixties offer in combination several advantages. One is the ability to analyze truly complex problems with multiple paths for streaming and attenuation which may defy the separation of effects required by empirical methods. Another advantage is the ability to more fully understand transport mechanisms; and by piecewise verification, these rigorous methods may be qualified for analysis of a wide range of problems not previously encountered in design experience. The rigorous methods do not reduce the need for competent analysts who also apply empirical approaches. They do extend the analysts capability into more complex problems and higher precision, and they add the force of more clearly defined verification and estimation of uncertainties.

Q(A.2) Will you repeat your comments on the ray effect and tell us whether this has been a problem in FFTF shield design? Warren F. Miller, Jr. LASL

A(A.2) The "ray effect" is a phenomenon with good and bad effects. It arises because of the tight mesh coupling in the r-z gometry between the space mesh and angle mesh. The r-θ geometry has loose coupling and a discrete ray will be obscured regardless of the mesh detail. For a diffusion problem with a point source, the few angles in a quadrature may produce unwanted flux structure. In an r-z geometry with a slot, the tight coupling allows use of a fine-mesh biased quadrature to accurately calculate both streaming and bulk transport. In one ORNL developed finite element code with discrete angles, the triangular mesh interacted with the angle mesh to diffuse the transport. This mesh diffusion even in a void prevented accurate calculation of streaming in a slot shield penetration.

Q(A.2) Comment on need for improved biasing techniques in Monte Carlo. Many techniques are available and known to researchers in the field but have not been implemented and are not being used. Our experience has shown that apparently difficult streaming problems can be treated accurately, with these techniques, and with only a moderate computer cost. Herbert Steinberg - MAGI

A(A.2) I certainly agree with your point. It is difficult to convey even the existing biasing expertise to those performing the bulk of design calculations. To the extent possible, we are trying to prepare user information on biasing which is easier to understand and apply.

Q(A.2) Please comment on calculations with multigroup versus pointwise cross sections.

A(A.2) This is a point which is quite debatable and has been debated extensively. To some extent multigroup libraries are needed to inject a degree of consistency among calculations. Of course, we can be consistently wrong if a data feature is obscured by the group structure. However, we have shown extensively that biased group structures allow multigroup calculations of equal precision with the best point methods. Also only a few cross sections minima need be represented in a given problem. Point codes vary in accuracy, some with energy grids at execution which are crude relative to good multigroup data and some which can treat data detail well. In point calculations, the user is responsible for decisions regarding the energy grid.