CONF-781117--3

Methods for U. S. Shielding Calculations - Applications to FFTF and CRBR Designs*

W. W. Engle, Jr. and F. R. Mynatt
Oak Ridge National Laboratory
Oak Ridge, Tennessee, USA
and
R. K. Disney
Westinghouse, Advanced Reactors Division
Madison, Pennsylvania, USA

ABSTRACT

The primary components of the U. S. reactor shielding methodology consist of: (1) powerful computer code systems based on discrete ordinates or Monte Carlo radiation transport calculational methods; (2) a data base of neutron and gamma-ray interaction and gamma-ray-production cross sections used as input in the codes; (3) a capability for processing the cross sections into multigroup or point energy formats as required by the codes; (4) large-scale integral shielding experiments designed to test cross-section data or techniques utilized in the calculations; and (5) a "sensitivity" analysis capability that can identify the most important interactions in a transport calculation and assign uncertainties to the calculated result that are based on uncertainties in all of the input data. The required accuracy for the methodology is to within 5 to 10% for responses at locations near the core to within a factor of 2 for responses at distant locations. Under these criteria, the methodology has proved to be adequate for in-vessel LMFBR calculations of neutron transport through deep sodium and thick iron and stainless steel shields, of neutron streaming through lower axial coolant channels and primary pipe chaseways, and of the effects of fuel stored within the reactor vessel. For ex-vessel LMFBR problems, the methodology requires considerable improvement, the areas of concern including neutron streaming through heating and ventilation ducts, through the cavity surrounding the reactor vessel, and through gaps around rotating plugs in the reactor head, as well as gamma-ray streaming through plant shield penetrations. In these types of streaming problems, complex geometrical features of the system play an important role and necessitate the development of special calculational techniques. Several such problems encountered in design calculations for the Fast Flux Tests Facility revealed generic areas requiring improved techniques. In the meantime, work on the primary components of the methodology continues, with significant progress having been made in recent years in the particular areas of cross-section processing and multigroup library production, in sensitivity analysis methods, and in the execution of the radiation transport code systems.

(Paper to be presented at FBR Shielding Seminar, Obninsk, USSR, Nov. 13-17, 1978.)

*Research sponsored by the U. S. Department of Energy under Contract W-7405-eng-26 with the Union Carbide Corporation.

1. Introduction

The shielding methods currently in use in the United States have been developed and refined over a fifteen-year period. The original development of present codes for shielding applications was supported by the U. S. space program. As that program was phased out, additional development was supported by defensive military programs and in recent years by the Fast Breeder Reactor programs. A complete and comprehensive study of a given shielding methodology would be extensive and is beyond the scope of this paper; however, the features of the U.S.A. shielding methodology discussed here will indicate the manner in which special problems are approached.

2. Overview of Shielding Methods

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) and gas-cooled fast-breeder reactors (GCFBRs). In particular, it has been applied to the Fast Flux Test Facility (FFTF) and the Clinch River Breeder Reactor (CRBR). The required accuracy is 5 to 10% for responses at locations near the core.

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 which 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 their uncertainties. 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 than a methodology solely for a specific reactor system.

3. 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 than an adequate methodology exists for calculating seven 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, (6) stored fuel effects, and (7) neutron streaming through the primary-pipe chaseways. Here the term adequate methodology implies that while there are aspects of these problems that still cannot be calculated accurately, design calculations for the important 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.

The problem of neutrons streaming in primary pipe chaseways and activating the sodium coolant in the secondary sodium loop exceeds the capabilities of standard techniques in 3-D Monte Carlo codes with respect to statistical accuracy and overall reliability, and the geometry (multiple 90° bends) precludes the use of any existing discrete ordinates codes and normal coupling techniques with any degree of confidence. However, the recent development of a right-angle coupling technique for R-Z geometry discrete ordinates calculations and its use in the analysis of a prototypic pipe chaseway experiment[3] at the Tower Shielding Facility have resulted in increased confidence in the analysis area. Additional development in albedo Monte Carlo techniques is expected to make significant contributions in this area also. Fortunately, the primary piping in the FFTF and CRBR contains 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 available.

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

The problem of neutrons streaming through heating and ventilation ducts is similar to the pipe chaseway streaming mentioned above except that more bends are usually required and the analysis techniques have not yet been applied to current designs. Also 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 solved 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 shielding 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.

4. 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 features 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 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 streaming only case which may be a simple situation that a trivial hand calculation can resolve. Next is the streaming + scattering case which also is not too difficult, even though the scattering sources must be identified. The case of sequential streaming 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-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 dimensions, and in R-Z geometries 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 neither 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.

A number of complex geometry streaming problems were encountered in the shielding analyses of the FFTF, particularly in the reactor cavity shield surrounding the reactor vessel.[4] Consider, for example, the region including the main support structure and the concrete shield collar (also called the reactor cavity shield). A simplified sketch of this region is shown in Fig. 3. On an *a priori* basis, a dominant upward streaming path could not be determined, nor 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 indicative 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[5] 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.

5. 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,[6] which has relatively constant group widths in lethargy. 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,[7] 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 emboded in the MINX[8] and SPHINX[9] codes) to include resonance self-shielding at the multigroup level. With this approach, a cross-section library consisting of 171 neutron groups was developed with the expectation that it would be collapsed for use in fusion reactor shielding problems and in LMFBR core physics and shielding problems.[10] 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[11]) 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[12] 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 calcualted 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"[13] 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 illustrated in Fig. 4 which is a channel theory plot of the geometry shown in Fig. 3. The peak or ridges in the figure define the dominant streaming paths through the complex geometry. 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.[14] 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,[15] 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 virtually all present major computing machines and hopefully for future machines. New plotting techniques which provide three-dimensional views of the complex geometries are valuable aids in correcting geometry input descriptions and in visualizing and evaluating any compromises in geometry descriptions.

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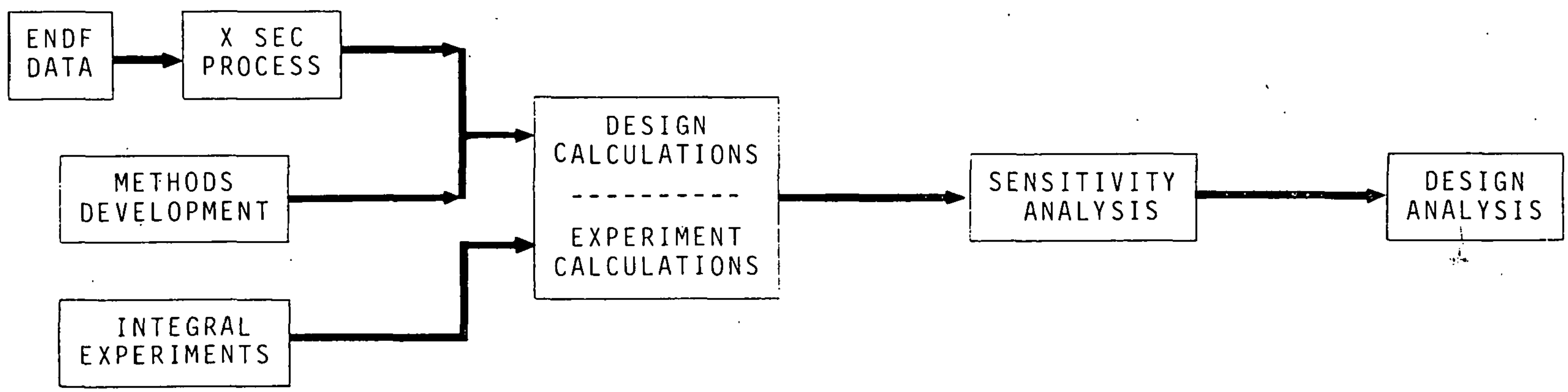

Fig. 1. Flow Diagram of U. S. Reactor Shielding Methodology.

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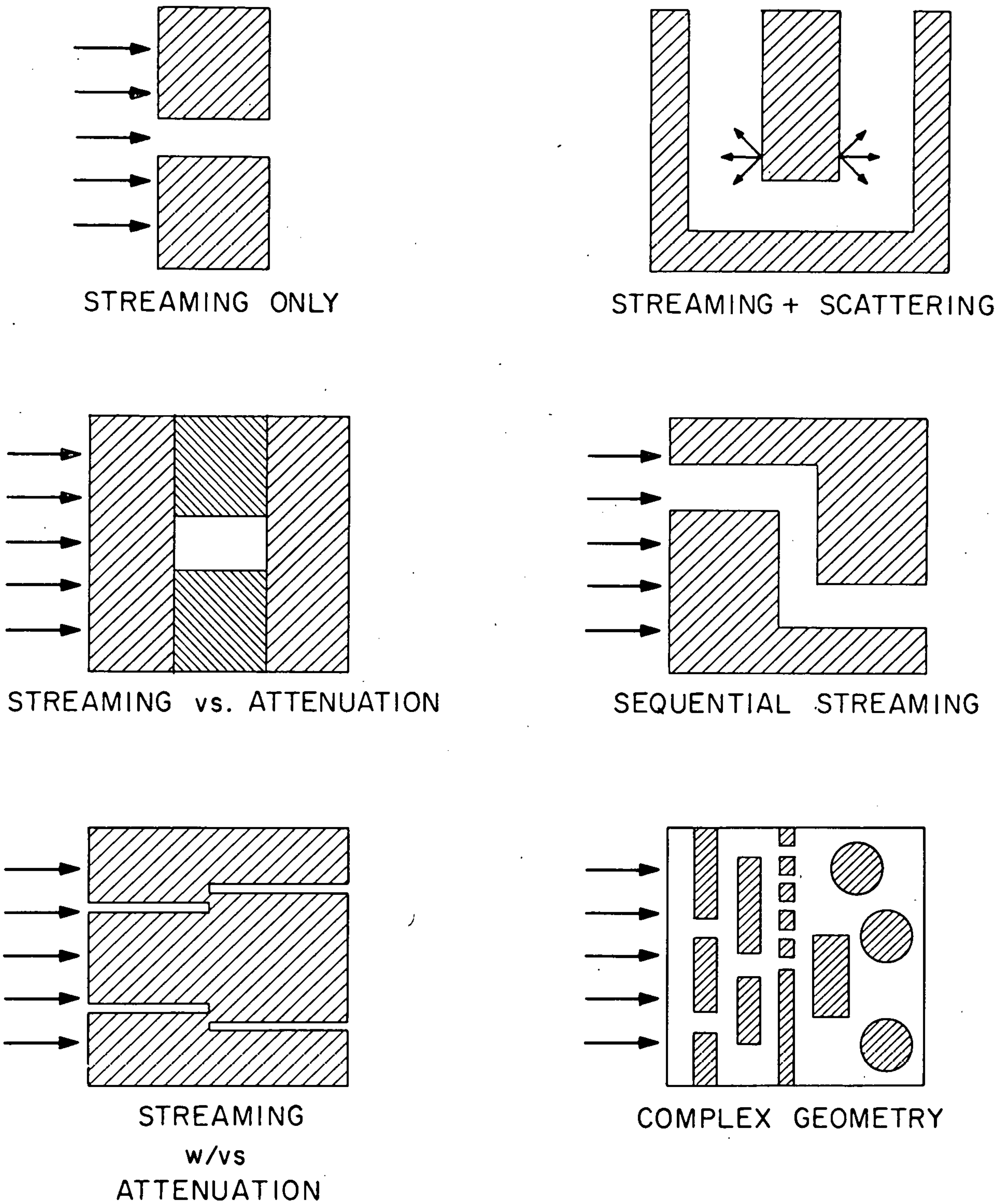

Fig. 2. Typical Streaming Configurations.

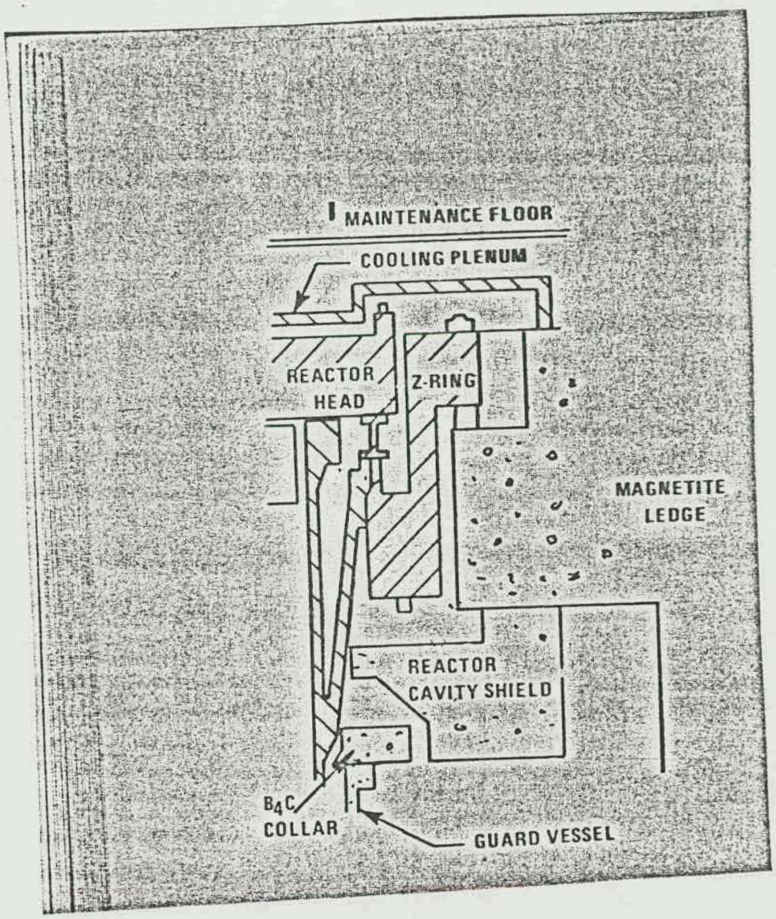

Fig. 3. Simplified Sketch of FFTF Vessel Support Ring and Reactor Cavity Shield.

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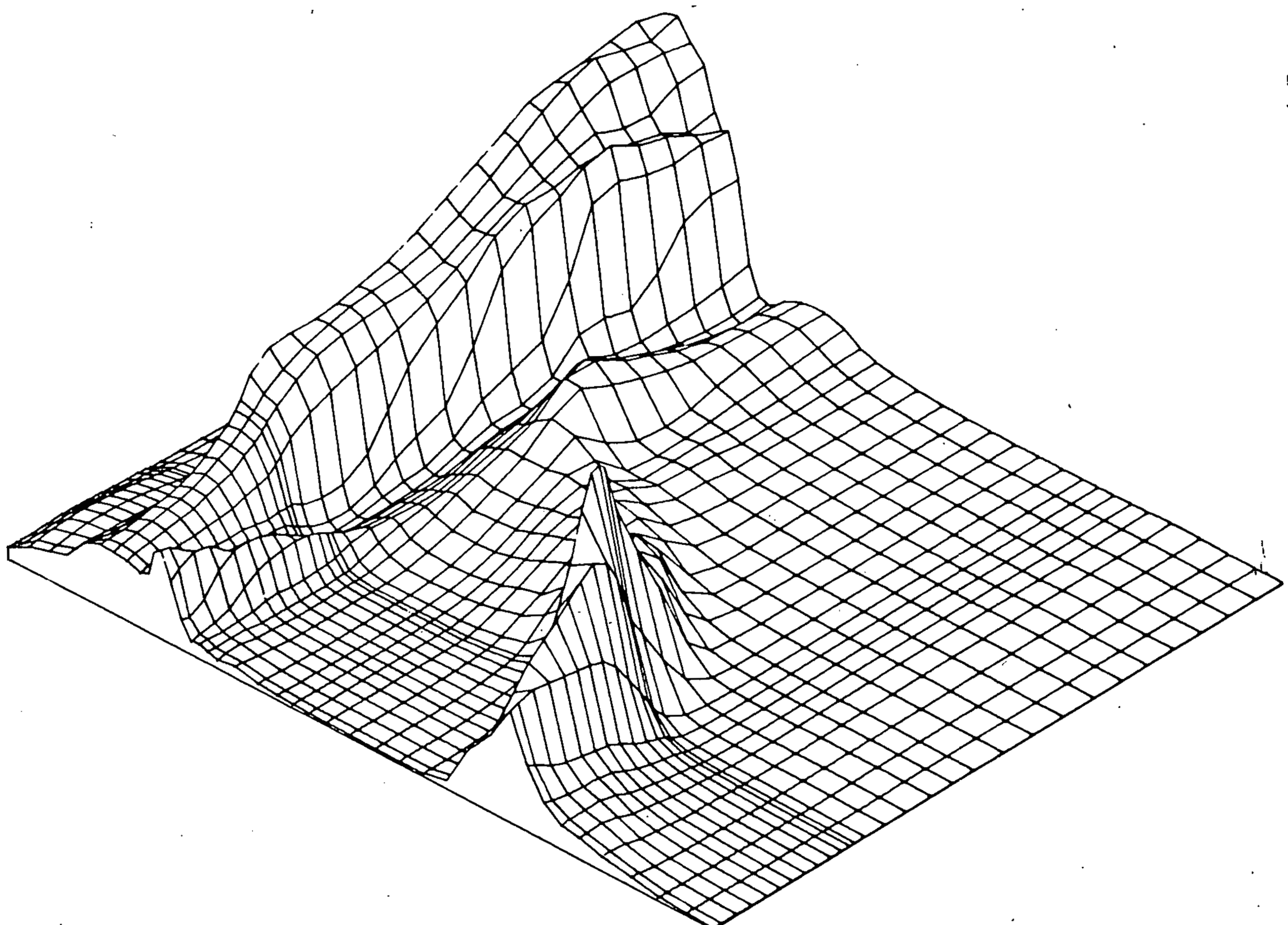

Fig. 4. Channel Theory Plot of Maintenance Floor Dose Contribution from Reactor Cavity Shield Area. The ridges show streaming paths on either side of the B_4C Collar shown in Fig. 3. The two paths merge above the collar to form one streaming path between the vessel and the reactor cavity shield.