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RECENT TRENDS IN RADIATION SHIELDING

AN RSIC PERSPECTIVE*

D. K. Trubey, R. W. Roussin, and B. F. Maskewitz

Radiation Shielding Information Center
Oak Ridge National Laboratory
Oak Ridge, Tennessee 37830, USA

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ABSTRACT

The subject of radiation transport and shielding in the nuclear power industry is reviewed, and advances in the state of the art are described. These fall into the areas of computational methods, nuclear cross sections, industry practices, and standards. Computer codes and data available from the Radiation Shielding Information Center (RSIC) representing recent advances are also described.

Introduction

In attempting to compile and analyze recent trends, one realizes that any description is highly dependent on one's perspective. This paper will be based on observations from the Radiation Shielding Information Center (RSIC), which has gathered information from the international shielding community for some 16 years.

The trends to be listed here are partly technical and partly socio-organizational. They are in the research and development sector, principally for fission reactors and fusion radiation transport problems, and in the commercial nuclear power sector. The trends to be discussed are not listed in order of importance.

Computational Methods

Discrete Ordinates and Monte Carlo

As evidenced in the literature, at recent conferences, and in requests for codes at RSIC, it is clear that there is a diminishing use of empirical and approximate computational methods. These have generally been the removal-diffusion method and the point-kernel method, which generally incorporates buildup factors. The use of buildup factors, of course, is still useful in many applications, especially in nuclear plant "balance-of-plant" shielding.

The Fifth International Conference on Reactor Shielding (Fifth ICRS) was typical of recent conferences in showing the widespread use of one- and two-dimensional discrete ordinates (S_n) codes and Monte Carlo codes to solve radiation transport problems. In applications requiring complicated geometry descriptions, successive runs are made with changes of mesh and quadrature. Coupling the discrete ordinates results to Monte Carlo calculations allows detailed three-dimensional treatment. DOMINO is a typical code for the application.[2] Routine use is made of variance-reduction techniques, such as direction-dependent biasing.

Channel Theory

A new concept for transport calculations, called "channel theory," has been introduced to solve for the neutrons, or other types of particles, which contribute to the response of a detector. The ones which contribute are called "contributons."[3-5] The method can determine streaming paths through a non-multiplying medium. It is the hypothesis of channel theory that the presence of both a source and a detector produces a response continuum. With this interpretation, response originates at a source and flows throughout the system until it disappears in the detector sink. The computer code FANG[6] has been developed to perform channel theory analysis using flux moments computed by the DOT transport code. Recently, work has been performed to extend this technique to channels in energy. This would contribute significantly to an understanding of the transport mechanism in deep-penetration problems.

Albedo Data

There have been many calculations over the years of transmission through ducts and voids using albedo methods. The earliest well-known method was that of Simon and Clifford.[7] This method involves integration of reflection probabilities, or albedos, over surfaces, and is generally best treated by Monte Carlo. The most important limitation, up until now, has been the lack of detailed albedo data. Recently, two data bases [8-9] and their corresponding computer codes have become available. The Oak Ridge data base, (Ref. 8) which includes data for water, ordinary concrete, steel, and 1.8-cm-thick steel-over-concrete, is meant for use with the well-known MORSE multigroup Monte Carlo code.[9] The Science Applications, Inc. data for concrete and iron has been expanded in spherical harmonics to reduce the computer storage requirements. These requirements have been very large since the detailed albedos are a function of five variables (two energy and three angle variables). This technique has resulted in a tenfold reduction in data storage requirements and a fivefold increase in computational efficiency.[9] Both data bases were computed with the two-dimensional discrete ordinates code DOT.[11]

Cross Sections

In radiation transport calculations, cross sections are at least as important as methods. For this reason, RSIC began making multigroup libraries available, separately from code packages in 1969. This first library, DLC-2, which was first based on ENDF/B-I, but having been updated several times, has enjoyed world-wide use. Since 1969, many other libraries have become available. During this period, the reliability, accuracy, and usefulness all have improved. There has been greater international cooperation in evaluation and data exchange, greater emphasis on elaborate, documented evaluations, and greater attention to photon production and higher neutron energies. Using coupled neutron gamma-ray libraries, a complete neutron and gamma-ray analysis can be made in one computer run. With common formats and data centers providing testing, distribution services, and common reference libraries, the sources for multigroup processing has improved. In addition, sensitivity analysis is being increasingly used to guide evaluation, experiments, and multigroup cross-section preparation.

Sensitivity Analysis

With the announced availability of the FORSS sensitivity computing system,[12] RSIC held a seminar-workshop on sensitivity analysis[13] which explored methodology and many applications for fast and thermal fission reactor parameters, fusion reactors, and dosimetry. Considerable discussion was held on the subject of data adjustment which may be considered to be a way of using integral experiments to improve differential data.

Application-Independent Libraries

Multigroup cross-section libraries, in general, are application dependent since the weighting function for one application may be quite poor for other applications. This has led to the generation of fine-group libraries which have sufficient fine-structure that the weighting function is not important and has correction factors to account for resonance self-shielding. One common approach to self shielding is to use interpolated Bondarenko factors, which are slowly varying functions of a background cross section and temperature, to account for self-shielding. The user can acquire the fine-group library and collapse to any group-structure and nuclide mixture that he chooses, using an appropriate weighting function. Such a library is the RSIC-distributed DLC-41/VITAMIN-C. This library, with 171 neutron groups and 36 gamma-ray groups, contains data for 61 nuclides.

The subject of multigroup cross sections was reviewed at a recent seminar-workshop, and the proceedings are available.[14] The AMPX cross-section processing system[14] was the subject of the workshop.

Recently, the use of multiband cross sections has improved the ability of the multigroup cross section libraries to treat resonance structure.[16]

Nuclear Power Industry

The technical staff of the architect-engineering firms who do most of the nuclear plant shielding design are now using the most advanced methods. This was evident at the Fifth ICRS and other recent conferences such as the special sessions on radiation streaming in power reactors held at the American Nuclear Society meeting in November of 1978.[17] The proceedings of these sessions will be published as an RSIC-ANS report.[18]

The shielding engineer has to be ever aware of the government requirement to keep radiation exposures "as low as reasonably achievable" (ALARA). The guidelines to be followed are published by the Nuclear Regulatory Commission.[19] The subject of occupational exposures has become a sufficient problem that a session was devoted to it at the Fifth ICRS.

Standards

As the industry matures, the technology tends to become standardized. The object is to require higher levels of excellence throughout the industry and to document the practices of experienced designers. A published standard is a special type of technical literature which has undergone a process whereby there is input from, and consensus of, leading experts across the industry, and which documents the state of the art. In the USA, there are a number of organizations developing nuclear standards. The American Nuclear Soceity is typical in having a standards subcommittee, ANS-6, Radiation Protection and Shielding, developing shielding standards. The subcommittee has eight working groups: (1) cross sections, (2) benchmarks, (3) shield testing, (4) materials, (5) glossary, (6) direct and scattered gamma radiation, (7) nuclear plant radiation zoning, and (8) radiation monitoring (area and process). These working groups interact to promote the availability and use of benchmark data, integral data testing, and computer code and method validation.

RSIC has begun the maintenance and distribution of standardized computing technology for the Nuclear Regulatory Commission. The first system, SCALE,[20-21] is intended to be used by applicants for licensing of spent fuel cask designs.

In summary, the last few years have seen considerable advances in the state of the art of shielding and radiation transport.

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