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RADIATION SHIELDING INFORMATION CENTER:

A SOURCE OF COMPUTER CODES AND DATA FOR FUSION NEUTRONICS STUDIES*

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Abstract

The Radiation Shielding Information Center (RSIC),[1] established in 1962 to collect, package, analyze, and disseminate information, computer codes, and data in the area of radiation transport related to fission, is now being utilized to support fusion neutronics technology. The major activities include: (1) answering technical inquiries on radiation transport problems, (2) collecting, packaging, testing, and disseminating computing technology and data libraries, and (3) reviewing literature and operating a computer-based information retrieval system containing material pertinent to radiation transport analysis. The computer codes emphasize methods for solving the Boltzmann equation such as the discrete ordinates and Monte Carlo techniques, both of which are widely used in fusion neutronics. The data packages include multigroup coupled neutron-gamma-ray cross sections and kerma coefficients, other nuclear data, and radiation transport benchmark problem results.

Introduction

The solution of most radiation transport problems requires the use of large computer codes and their corresponding cross-section libraries, which together adequately treat the physics of neutron and gamma-ray interaction and production. RSIC is involved in various aspects of this general problem by interacting with the user community as they ask technical questions on radiation transport and discuss with RSIC staff members their particular problem and possible solutions, by helping to provide evaluated neutron and gamma-ray cross-section data in standard formats, by packaging and distributing computer codes and associated cross-section libraries, by packaging and distributing specialized libraries of various types, and by operating a computerized information retrieval system which contains material pertinent to radiation transport and fusion. This paper describes each of these aspects of RSIC services. These activities are sponsored by the Department of Energy (DOE), the Nuclear Regulatory Commission (NRC), and the Defense Nuclear Agency (DNA).

Interaction with Users

Communication channels between RSIC and users in the field are kept open and extensively used. It has always been RSIC policy to discuss with users their needs and recommend codes and data libraries. Insofar as possible, we assist those who are implementing codes and data at their installations. In cases where we lack experience to advise in particular problem areas, we call upon other experienced personnel locally and elsewhere to help them. This takes the form of offering advice on solving particular problems, as well as helping to diagnose problems in implementing calculational procedures. As always, RSIC relies on its user community to share their ideas, codes, and data so that technology as a whole may be advanced. Many RSIC projects have benefitted by the active participation of the Center's users. Seminar-workshops, topical meetings, review articles, and data collection activities point to an industry-wide cooperative enterprise.

An example of RSIC-community interaction is now in progress in regard to the MCNP[2] code. RSIC recently received from Los Alamos Scientific Laboratory the MCNP code, a general purpose Monte Carlo code which can be used for neutron, photon, or coupled neutron-photon transport, designed for the Livermore Time Sharing System (LTSS). A conversion of MCNP to standard CDC FORTRAN is now underway at Babcock and Wilcox, and the resulting version will also be made available for distribution by RSIC.

Another example of RSIC-community interaction is the RSIC Seminar-Workshop on "Theory and Application of Monte Carlo Methods" which was held in Oak Ridge during the week of April 21, 1980. The seminar lasted one and one-half days and featured invited and contributed presentations on the theory and application of Monte Carlo methods for radiation transport problems in shielding and reactor physics. The proceedings were published as an RSIC report.[3]

The workshop occupied one and one-half days and concentrated on the TRIPOLI-II[4] Monte Carlo system developed by the CEA/CEN/Saclay SERMA Shielding Laboratory, Gif-sur-Yvette, France. The TRIPOLI-II Monte Carlo code represents many years of development and successful application to a wide variety of problems in France. A particularly important recent development with this system is the interfacing of the code to ENDF/B formatted data and/or multigroup data in AMPX format. TRIPOLI-II is now available from RSIC (CCC-372).

A list of code packages which might be helpful to fusion neutronics work is included in this paper (see Table I).

The RSIC Newsletter has long been an effective means of communication between Center and users as it reaches more than 1200 persons each month.

Code and Data Packages

Radiation transport analysis requires the use of large computers and complex computer codes and associated data libraries. RSIC treats this kind of information as an inseparable part of information services. Effort is made to understand it, to use it to analyze other information, and to use it as a means of technology transfer. Computer code exchange has had a tremendous impact on radiation analysis technology.

RSIC has collected, tested, and packaged a large collection of computer codes (over 380 packages). Each code, packaged as a unit with pertinent auxiliary routines, and sometimes data, carries a unique Computer Code Collection (CCC) number. Availability of each package is announced in the RSIC Newsletter. Interested persons may acquire any code and associated documentation by supplying a reel of magnetic tape for transmittal, along with a request. The collection does not remain static but, as RSIC maintains contact with users, users often give feedback which is deemed possibly helpful to future users. When this is true, with approval of code contributor, RSIC opens the package, updates, extends or corrects it, repackages, then announces in the RSIC Newsletter information concerning the altered package. As this cycle is perpetuated, codes in the collection stay with state of the art and become progressively more valuable.

Since inception, RSIC has been deeply involved with the acquisition, check-out, packaging, and distribution of a computer code library. Subsequently, a data library collection has been developed which includes not only cross-section libraries, but other nuclear data and also radiation transport results. The data sets are packaged in a manner analogous to the RSIC code collection. Each data set, packaged as a unit, carries a Data Library Collection (DLC) number. As with the code packages, a particular data package does not remain static but is subject to revision, updating, and expansion as required. Such changes are announced in the RSIC Newsletter.

Data libraries were first announced as available from RSIC near the end of 1968. At that time it became evident that a collection of data libraries would be an extremely helpful companion to the RSIC Computer Code Collection.

The philosophy behind the packaging and distributing of these data libraries is to preserve and make available in an easily-usable form, data which may be useful to those utilizing or performing radiation transport calculations. Since this usually involves the use of large computer programs, several of these libraries are multigroup cross sections in the format utilized by many such programs.

Where available, RSIC seeks to obtain multigroup or other data sets which have been shown to handle adequately a particular problem or class of problems for which a standard multigroup approach is inadequate.

An example of RSIC special data activities is the coordination of the development of VITAMIN-C[5] (package number DLC-41), a general-purpose multigroup library with features useful for fusion applications. The data set contains 171 neutron, and 36 gamma-ray groups along with 66 isotopes and elements (ENDF/B-IV) with three temperatures and several values of σ_o for interpolation of f-factors. Neutron and gamma-ray cross sections in AMPX master library and CCCC interface formats are included. The VITAMIN-C general format allows the retention of all reaction cross sections and transfer arrays. A user may derive problem-dependent broad-group cross sections for particular applications.

MACKLIB-IV[6] (DLC-60) is a library of nuclear response functions generated with the MACK-IV[7] code (PSR-52) from ENDF-IV data. It contains 171 neutron, 36 gamma-ray group nuclear response functions and can be used interchangeably with VITAMIN-C.

User response to both MACKLIB-IV and VITAMIN-C was very favorable, and efforts are now underway to obtain sponsorship for updated versions based on ENDF/B-V.

Table II is a description of Data Library Packages which may be helpful to persons in fusion neutronics work.

Standards Activities

An RSIC staff member is currently the chairman of the ANS-6 Radiation Protection and Shielding Standards Subcommittee and another which serves on ANS-10 Computing Technology Standards Subcommittee.

Working groups in both American Nuclear Society subcommittees have developed and are continuing to develop American National Standards. These industry voluntary standards are frequently made part of contracts or government regulations. The ANS-6 subcommittee has developed procedures for testing reactor shields, determining direct and scattered gamma radiation from LWR power plants, radiation zoning of nuclear plants, and has developed a glossary, standard flux-to-dose-rate factors, and benchmark problem solutions for methods testing. The ANS-10 subcommittee has developed standards on programming practices and code documentation.

Another staff member heads the Shielding Data Testing and Applications Subcommittee of the U.S. Department of Energy Cross-Section Evaluation Working Group. The subcommittee is currently reviewing benchmarks, such as those done in the Fusion Integral Experiment Program at ORNL, which are useful for data testing of cross sections and methods important for fusion neutronics.

Information Retrieval System

Since 1962, RSIC has operated a computer-based information retrieval system which has evolved to represent fusion and other interests along with the original radiation shielding subjects. Two data bases are currently maintained by RSIC for nation-wide on-line searching through RECON (REmote CONsole), a computerized on-line interactive information storage and retrieval system which is designed to give users direct and fast access to bibliographic records stored on large on-line files at Oak Ridge, Tennessee.[8] The RSI data base is made up of bibliographic literature citations and abstracts. The RSC data base contains citations to literature which describe computer codes designed to do calculations within the scope of RSIC.

Summary

RSIC strives to build upon its more than 15 years of experience and continually reach out to increase its involvement with and usefulness to the fusion and other radiation transport communities. It solicits contributions, cooperation and suggestions from users and prospective users.

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8. DOE/RECON: Manual for the Dial-Up User, DOE Report TID-4587-R3 (1979), available from DOE Technical Information Center, P. O. Box 62, Oak Ridge, TN 37830.

TABLE I. COMPUTER CODE PACKAGES

PACKAGE NUMBER AND NAME	TITLE	CONTRIBUTOR
CCC-42/DTF-IV	Multigroup Neutron Transport Discrete Ordinates Code, One-Dimensional Anisotropic Scattering	Los Alamos Scientific Laboratory
CCC-46/OGRE	A General Purpose Monte Carlo Gamma-Ray Transport Code System	Oak Ridge National Laboratory
CCC-187/SAM-CE	Monte Carlo Time-Dependent Three-Dimensional Complex Geometry (combinatorial) Code System - ENDF Formatted Cross Sections	Mathematical Applications Group, Inc.
CCC-203/MORSE	General Purpose Monte Carlo Multigroup Neutron and Gamma-Ray Transport Code System	Science Applications, Inc.; Mathematical Applications Group, Inc.; Oak Ridge National Laboratory
CCC-222/TWOTRAN	Two-Dimensional Multigroup Discrete Ordinates Transport Code in (x,y), r-theta), and (r,z) Geometries	Los Alamos Scientific Laboratory
CCC-254/ANISN	ANISN: Multigroup One-Dimensional Discrete Ordinates Transport Code with Anisotropic Scattering	UCCND Computer Sciences Division, Oak Ridge National Laboratory; Nuclear Assurance Corp.
CCC-276/DOT 3.5	Two-Dimensional Discrete Ordinates Radiation Transport Code	Oak Ridge National Laboratory; Control Data Corp.
CCC-293/TRIDENT	Two-Dimensional Multigroup Discrete Ordinates Transport Code--(x,y) and r,z) Geometries	Los Alamos Scientific Laboratory
CCC-320/DOT IV	Two-Dimensional Discrete Ordinates Radiation Transport Code System	Oak Ridge National Laboratory
CCC-323/DKR	A Radioactivity Afterheat and Dose Rate Calculation Code for Fusion Reactors	University of Wisconsin
CCC-359/MAGIK	A Monte Carlo System for Computing Induced Residual Activation Dose Rates	Oak Ridge National Laboratory
CCC-371/ORIGEN2	Isotope Generation and Depletion Code - Matrix Exponential Method	Oak Ridge National Laboratory
CCC-372/TRIPOLI II	Three-Dimensional Monte Carlo Radiation Transport Code	CEA, France
CCC-376/KIM	A Two-Dimensional Monte Carlo Program for Linear Neutron Transport Calculations	CNEN, Bologna, Italy
CCC-377/TRIDENT-CTR	Two-Dimensional X-Y and R-Z Geometry Multigroup Transport Code	Los Alamos Scientific Laboratory
CCC-380/PALLAS-PL,SP	Multigroup Time-Independent Neutron Transport Code for Plane or Spherical Geometry	Ship Research Institute, Tokyo, Japan

TABLE 1. (continued)

PACKAGE NUMBER AND NAME	TITLE	CONTRIBUTOR
PSR-63/AMPX-II	Multigroup Library Production and Handling	Oak Ridge National Laboratory
PSR-105/MINX	Multigroup Library Production	Oak Ridge National Laboratory, Los Alamos Scientific Laboratory
PSR-112/MAME	Multigroup Library Handling	
PSR-117/MARS	Multigroup Library Handling	Los Alamos Scientific Laboratory, Oak Ridge National Laboratory
PSR-118/NJOY	Multigroup Library Handling	Los Alamos Scientific Laboratory; UCCND Computer Sciences Division
PSR-132/MACK-IV	Kerma Factor Production	Argonne National Laboratory; Oak Ridge National Laboratory
PSR-137/MARLOWE	Computer Simulation of Atomic-Displacement Cascades in Solids in the Binary-Collision Approximation	UCCND Computer Sciences Division; Oak Ridge National Laboratory
PSR-143/BREESE II	Auxiliary Routines for Implementing the Albedo Option in the MORSE Monte Carlo Code	Oak Ridge National Laboratory
PSR-151/CHENDF	Codes for Checking ENDF/B-V Data	National Nuclear Data Center, Brookhaven National Laboratory
PSR-152/HAUSER-5	A Computer Code to Calculate Nuclear Cross Sections	Hanford Engineering Development Laboratory

TABLE II. DATA LIBRARY PACKAGES

DATA PACKAGE NUMBER AND NAME	TITLE	CONTRIBUTOR
DLC-29/MACKLIB	100-Group Neutron Kerma Factor and Reaction Cross Section Library	University of Wisconsin; Oak Ridge National Laboratory
DLC-31/FEWG 1	Coupled 37-Neutron, 21-Gamma-Ray Group, P3, Multi-group Library in ANISN Format	Oak Ridge National Laboratory
DLC-33/MONTAGE	100 Group Neutron Cross Sections for Important Structure or Coolant Activation Reaction	Los Alamos Scientific Laboratory
DLC-36/CLAW	Coupled Neutron, Gamma-Ray Multigroup Multi-table Cross Sections for 29 Materials	Los Alamos Scientific Laboratory
DLC-37/EPR	Coupled 100-Neutron, 21 Gamma-Ray Group P8 Cross Sections in ANISN Format, Based on ENDF/B-IV for Experimental Power Reactor (EPR) and Other Fusion Concepts	Oak Ridge National Laboratory
DLC-41/VITAMIN-C	Programs for AMPX Retrieval, Manipulation, Transport Codes, Sample Problem	Oak Ridge National Laboratory
DLC-47/BUGLE	Coupled 45-Neutron, 16 Gamma-Ray Group P3 Cross Section Library for ANS-6.1.2 Studies	Oak Ridge National Laboratory
DLC-55/RECOIL	Recoil Spectra: W, 10B, 11B, Nb, Ni, Cr, Fe, Al, Si, V, Mn	Oak Ridge National Laboratory
DLC-58/HELLO	47 Neutron, 21 Gamma-Ray Group Cross Sections for Radiation Transport for Neutron Energies up to 60 MeV	Oak Ridge National Laboratory
DLC-60/MACKLIB-IV	171 Neutron, 36 Gamma-Ray Group Nuclear Response Function Library Calculated with MACK-IV from Cross Section Data in ENDF/B-IV	Oak Ridge National Laboratory
DLC-64/UKCTR1	46-Group Neutron Cross Sections for Fusion Reactor Calculations	University of Birmingham
DLC-69/ACTL	Evaluated Neutron Activation Cross-Section Library	Lawrence Livermore Laboratory
DLC-70/JENDL1	Japanese Evaluated Nuclear Data Library, Version-1	Japan Atomic Energy Research Institute
DLC-71/GAMMON	The GAMMON Activation Library of 100-Group Neutron Cross Sections and 25 Gamma-Ray Group Spectra for Fusion Reaction Application and Other Design Studies	Los Alamos Scientific Laboratory
DLC-72/MONTUK	*UKCTR III Transmutation and Activation Data* 100-Group Neutron Activation Cross-Section Data for Fusion Reactor Structure and Coolant Materials	*United Kingdom Atomic Energy Agency*