RSIC Newsletter

RADIATION SHIELDING INFORMATION CENTER

OAK RIDGE NATIONAL LABORATORY

OPERATED BY UNION CARBIDE CORPORATION FOR THE U.S. DEPARTMENT OF ENERGY

POST OFFICE BOX X •
OAK RIDGE, TENNESSEE 37830

No. 167 *November 1978*

Nothing dies faster than a new idea in a closed mind. . . . *Arnold Glascow*

RSIC FY 1978 USER STATISTICS

RSIC user statistics have been compiled for the period October 1, 1977 through September 30, 1978. Information dissemination activities were as follows:

A total of 2911 separate letters/telephone calls (about 11.6 each working day) requesting a variety of products and services (7604 total) were processed during FY 1978.

On an average, the following dissemination of activities took place each working day:

Activities/Working Day

4.1 Code/data packages were shipped to requesters.

5.2 Shielding documents (RSIC reports, handbooks, code and data documentation in addition to those included in above packages) were mailed.

20.9 Responses to inquiries for information (citing possible solutions to problems; recommendations of calculational methods, computer codes, nuclear data sets, or literature specimens for study; trouble-shooting problems when requester had difficulties using RSIC materials; and miscellaneous consultation and advisory services.)

0.2 Special retrospective searches.

30.4 Total of separate activities required daily to satisfy the 2911 letters of request.

In addition to the above daily activities, the following special products or services were given.

The RSIC Newsletter was mailed each month to a peak of 1479 people. Maintenance of the RSIC-user directory resulted in 582 changes during the year.

A total of 133 people (54 foreigners) came for an orientation visit and/or to use the Center's facilities during the year.

FIRST CALL FOR PAPERS, THIRD ASTM–EURATOM SYMPOSIUM ON REACTOR DOSIMETRY AT ISPRA, ITALY

A call for papers is hereby issued for the 3rd ASTM-Euratom Symposium on Reactor Dosimetry to be held October 1–5, 1979, at the Joint Research Centre of the Commission of the European Communities, Ispra (Varese), Italy. The Symposium is sponsored by the Commission of the European Communities, the ASTM E10 Committee on Nuclear Applications and Measurements of Radiation Effects, the U.S. Nuclear Regulatory Commission (NRC) and the U.S. Electric Power Research Institute (EPRI), all in cooperation with the International Atomic Energy Agency (IAEA). Simultaneous translations of the presentations in English, French and German will be through the courtesy of the Commission of the European Communities.

The theme of the Symposium is intended to be the interface between materials experts and dosimetry metrologists. Papers are solicited for presentation, and publication in the proceedings, on the following general topics. These papers are expected to form the basis for workshops and panel discussions.

1. Radiation damage correlations of structural materials and damage analyses techniques.
2. Materials radiation dosimetry outside the reactor core, including the calculation of flux densities in power reactor cavities.
3. Dosimetry for fusion materials, including the most pressing nuclear data needs, planned programs, and calculational models of damage and of Be or Li(d,n) neutron fields.
4. Dosimetry needs for new fuel cycles.
5. Dosimetry and metallurgy for fuels irradiations.
6. Neutron benchmark measurements, calibrations and validations of dosimetry and metallurgy methodology, including applications of light water reactor pressure vessel simulators.
7. Recent advances in dosimetry data and techniques, including cross section evaluations, most recent advances in solid state track dosimetry techniques and helium accumulation fluence monitors.
8. Fundamental research and new concepts for materials damage and reactor dosimetry.
9. Neutron metrology methods guided by safety considerations and safety regulations.

It is the intent of the Symposium Program Committees to have authors participate in workshops and panel discussions each afternoon following plenary preparatory sessions that morning. Efforts will be made to avoid parallel sessions. Persons interested in chairing the afternoon sessions, emphasizing users' needs, are requested to identify themselves at the earliest possible date. Prospective authors should send, by December 31, 1978, a 500-word abstract to the applicable program committee secretary. For U.S. authors this is: E. D. McGarry, National Bureau of Standards, Gaithersburg, Maryland 20760, USA; for all other authors this is: H. Röttger, Joint Research Centre, HFR Division, Postbus 2, 1755 ZG PETTEN (N.H.), The Netherlands.

OVERVIEW OF RSIC CODE AND DATA PACKAGES AVAILABLE

We attempt to keep up with the RSIC inventory of code and data packages by entering titles, keywords, and other comments into a computer file and getting printouts frequently. The printouts contain indexes which can help us locate particular packages for specific needs. We keep on hand copies of these printouts, and we can mail a copy upon request to anyone needing such an overall list of the RSIC code and data collection.

NOW AVAILABLE: ANNUAL ASTM STANDARDS 1978, PART 45, NUCLEAR

The American Society for Testing and Materials (ASTM) is a nonprofit organization that provides a management system in which producers, users, ultimate consumers, and representatives of government and academia develop technical information in the published form of agreed upon documents called "voluntary consensus standards."

These standards—test methods, specifications, definitions, practices, and classifications—are written by those having expertise in specific areas, who choose, voluntarily, to work within the ASTM system. Current membership is 26,000 organizations and individuals, with a total unit participation of well over 65,000 in 130 Technical Committees.

Part 45 contains over 140 standards which can assist your organization when researching, testing, specifying, buying, or selling your products or services.

The increased use of nuclear energy to generate electricity has made it imperative for engineers, designers and scientists concerned with nuclear reactors to have reliable standards. Operators and utilities are equally concerned. Now ASTM provides its standards in a single volume which can be of immense value.

Part 45 of the 1978 Annual Book of ASTM Standards contains all ASTM Standards dealing with nuclear materials and materials related to nuclear reactors, 25.5% of which are new, revised, or changed in status since the 1977 edition. Thirteen standards have been approved by ASME and four by other organizations.

New standards include: Test for Sulfur in Graphite by Combustion-Iodometric Method; Definitions of Terms Relating to Nuclear Materials; Specification for Nuclear-Grade Gadolinium Oxide (Gd_2O_3) Powder;

Chemical and Spectrochemical Analysis of Nuclear-Grade Gadolinium; Calibration of Germanium Detectors for Measurement of Gamma-Ray Emission Rates of Radionuclides; Recommended Practice for Systematizing the Development of (ASTM) Voluntary Consensus Standards for the Solution of Nuclear and Other Complex Problems; and Recommended Guide for Developing the (ASTM) Voluntary Consensus Standards Required to Help Implement the National Energy Plan.

The standards in Part 45 cover: Concrete Products for Nuclear Applications; Graphite Products for Nuclear Applications; Metal Products for Nuclear Applications-Nickel and Nickel Alloys, Steel, Tantalum, Titanium and Titanium Alloys, Zirconium and Zirconium Alloys; Nuclear-Grade Materials; Radiation Effects in Organic Materials; Radioactivity, Inorganic Materials in Water; Analysis, Dosimetry and Radiation Effects in Metals; and Temperature Measurement.

You may order from : American Society for Testing and Materials, 1916 Race Street, Philadelphia, PA 19103, Attn: Sales Service Department. The price is $39.00, and the publication code number is 01-045078-35.

MONTE CARLO RADIATION TRANSPORT IN NOVEL APPLICATION

Harry D. Smith, Jr. and Ward E. Schultz of Texaco, Inc. have reported novel uses of Monte Carlo techniques in SPE 7432, *Computer Simulation of Two Nuclear Well Logging Methods* at the 53rd Annual Fall Technical Conference and Exhibition of the Society of Petroleum Engineers of AIME, held in Houston, TX in October.

SAM-C and, later, SAM-CE (packaged in RSIC as CCC-187, a MAGI contribution) was utilized to simulate the response of a multi-window natural gamma-ray spectral log to varying borehole conditions and to investigate the potential of a porosity logging concept employing a ratio of fast/epithermal neutrons. The authors conclude that Monte Carlo radiation transport techniques, which have long been used by scientists involved in nuclear reactor shielding design and radiation dosimetry, provide perhaps the best methods for fulfilling the theoretical nuclear logging objectives. They believe that the Monte Carlo method is well suited to assisting in the development, calibration, and general understanding of a wide range of nuclear logging systems.

The article (SPE 7432) is copyrighted by the American Institute of Mining, Metallurgical, and Petroleum Engineers, Inc. Requests for copies may be addressed to 6200 N. Central Expy., Dallas, TX 75206.

PERSONAL ITEMS

S. Ganesan, after working as a Guest Scientist on reactivity coefficients in fast power reactors at Kernforschungszentrum, Karlsruhe, Federal Republic of Germany, has returned to the Reactor Research Centre, Kalpakkam, Tamil Nadu, India.

An announcement has been received from Nichols Research Corporation (NRC) that **Dr. Rick Byrn** has joined NRC as Vice President of Research. NRC is an Alabama corporation located in Huntsville. The corporation is dedicated to research and development programs related to optical and infrared technology. NRC's capabilities are centered in seven optical technology areas: (1) optical sensor signal processing and functional algorithm development; (2) optical system simulation and analysis; (3) optical target signature and background phenomenology and models; (4) reduction and analysis of optical sensor test data; (5) data processing subsystem definition and real-time software development; (6) optical sensor performance analysis; and (7) nuclear analysis. Rick was formerly associated with SAI in Huntsville.

Dr. Jehudah Wagschal, Racah Institute of Physics of The Hebrew University, and **Dr. Jacob Barhen** of the Technion—Israel Institute of Technology, Haifa, Israel, are spending a sabbatical year working in the ORNL Reactor Methods and Data Development Group of the Engineering Physics Division.

Dr. Jacob Kastner has transferred within the Nuclear Regulatory Commission (NRC) from the

Environmental Standards Branch of the Office of Standards Development to the Transportation Branch of the Division of Fuel Cycle & Material Safety.

Yasushi Seki, who spent one year in the Engineering Physics Division at ORNL working on the analysis of the fusion integral experiment using sensitivity methods, has recently returned to JAERI, Tokai-mura, Ibaraki-ken, Japan.

Mohamed Abdou has left Argonne National Laboratory, Argonne, Illinois, to join the School of Nuclear Engineering at Georgia Institute of Technology, Atlanta, Georgia.

VISITORS TO RSIC

The following persons came for an orientation vist and/or to use RSIC facilities during the month of October:

Moselle Rankin Bell and P. R. Bell, consultants, ORNL, Oak Ridge, Tenn.; Vic Cain, Science Applications, Inc., Oak Ridge, Tenn.; Y. Gohar, Argonne National Laboratory, Argonne, Ill.; Harry Hubbell, Health and Safety Research Division, ORNL, Oak Ridge, Tenn.; Jacob Neufeld, consultant, ORNL, Oak Ridge, Tenn.; G. Dan Robbins, Information Division, ORNL, Oak Ridge, Tenn.; Bill Rhyne, Science Applications, Inc., Oak Ridge, Tenn.; Yaakov Shima, Soreq Research Center, Yavne, Israel; and Charles D. Swanson, Control Data Corporation, Arden Hills, Minn.

CHANGES IN THE COMPUTER CODE COLLECTION

Several changes were made in the computer code collection during September.

CCC-121/SABINE-3

The Spinney (removal-diffusion) shielding code package in one-dimensional geometry (SABINE) was extended to include a CDC-7600 version (C), contributed by the OECD NEA Data Bank. SABINE-3 was developed by the European Shielding Information Service (ESIS), Euratom, Ispra, Italy. FORTRAN IV; IBM 360 (A), CDC-6600 (B) and CDC-7600 (C) above.

CCC-203/MORSE-CG

The IBM 360 package of the general purpose Monte Carlo multigroup neutron and gamma-ray transport code system has been extended by the addition of the printer plot package, P^3 to be used for collision site plotting and a non-standard systems routine, ERROR, contributed by the original ORNL code developers. Neither of the additions affect the CDC (B) or UNIVAC (A) versions. FORTRAN IV AND Assembler Language; IBM 360.

CCC-204/SWANLAKE

The code package, utilizing ANISN radiation transport calculations for cross section sensitivity analysis, was extended to include a CDC 6600/7600 version (CCC-204B), a contribution of the University of California, Berkeley. SWANLAKE was developed at the Oak Ridge National Laboratory. Reference: ORNL-TM/3809.

CCC-254/ANISN-ORNL

The multigroup one-dimensional discrete ordinates transport code package (ANISN) was extended to include a non-standard systems routine, ICLOCK, in the IBM version (A) supplied by the ORNL code originators, and a new CDC 6600/7600 version (C) of the complete code system contributed by the University of California—Berkeley. The package also includes a UNIVAC 1108 version (B).

CCC-276/DOT 3.5

The IBM 360 DOT 3.5 two-dimensional discrete ordinate radiation transport code package (CCC-276A) was extended to include perturbation code modules DGRAD and TPERT, contributed by the ORNL code originators. DGRAD calculates the directional flux gradients from DOT diffusion theory flux tapes. TPERT obtains exact and first-order reactivity changes. These modules have not yet been implemented in the CDC version (B). Reference: ORNL/CSD/TM-71. FORTRAN IV; IBM 360.

CCC-306/DINT

DINT, a routine for computing the gamma-dose integrals I_1 and I_2 for the finite-cloud sector-average model in Regulatory Guide 1.109 for the purpose of evaluating compliance with 10 CFR Part 50, Appendix I, was contributed by Nuclear Services Division, Yankee Atomic Electric Company, Westborough, Mass. The functions are integrations of Gaussian plume concentrations multiplied by exponential attenuation functions for air. References: YAEC-1105 and Rev. 1, NRC REGULATORY GUIDE 1.109. FORTRAN IV; IBM 360 and CDC.

CCC-316/XOQDOQ

A code package for the meteorological evaluation of routine effluent releases at nuclear power stations was contributed by the U.S. Regulatory Commission and the National Oceanic and Atmospheric Administration. Using a "straight-line" airflow model, XOQDOQ implements the assumptions outlined in Section C (excluding Cla & Clb) of Regulatory Guide 1.111, *Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors.* It calculates average relative effluent concentrations (X/Q) and average relative deposition values (D/Q) at specified locations and at standard radial distances and segments for downwind sectors, and calculates these values at the specified locations for intermittent releases. Reference: NUREG-0324. FORTRAN IV; IBM 360.

CCC-323/DKR

A radioactivity afterheat and dose rate calculation code system for fusion reactors was contributed by the Fusion Technology Program, Nuclear Engineering Department, University of Wisconsin at Madison. A point activity calculation code, DKR constructs the linear decay chains using nuclear data from Decay Chain Library Data (DCDLIB, included in the package) and solves them to compute the activity of a fusion reactor. The calculation of radioactivity, biological hazard potential (BHP), afterheat due to β- and gamma-rays, and that due to β-rays only is performed. A decay gamma-ray source may be produced as one of the optional output. Neutron flux from ANISN (CCC-254) and transmutation data from DCDLIB are essential input for DKR. An auxiliary code, DOSE (included in the package), with either forward or adjoint gamma-ray flux, computes spatially-dependent or time-dependent dose rates respectively. References: UWFDM-170 and -171. FORTRAN IV; UNIVAC 1110.

PSR-64/DOMINO

The B (CDC-6400) and C (UNIVAC-1110) versions of this general purpose code package developed at ORNL for coupling discrete ordinates and Monte Carlo radiation transport calculations were extended to include a sample problem and selected MORSE routines, taken from the ORNL-developed A (IBM 360) version. The three versions now have a consistent set of contents. FORTRAN IV; CDC 6400 and UNIVAC 1110.

PSR-123/FEDGROUP

A neutron multigroup cross section processing code system using various evaluated data formats (KEDAK, UKNDL, ENDF/B, LENDL, etc.) was contributed by the Central Research Institute for Physics in Budapest, Hungary, the Kurchatov Institute of Atomic Energy in Moscow, USSR, and the Institute of Nuclear Research at Swierk-Otwock, Poland. The system was designed to process the IAEA-disseminated evaluated nuclear data files into group-averaged constants, transfer matrices, etc., applicable to any multigroup neutronic calculation. It is designed to be relatively independent of hardware—is written in FORTRAN for BESM-6 (FORTRAN-CERN), for CDC-3300, CYBER-72, etc. The packaged version was tested on the IBM 360 computers at ORNL. Reference: INDC(HUN)-13/L+Sp. by P. Vértes of CRIP in Budapest.

SCALE-01/HEATING 5

The generalized heat conduction code system package was extended to add sample problem plotting data produced by HEATING 5 for use by the plotting package HEATPLOT. The ORNL Computer Science Division, Union Carbide Nuclear Division, contributed the code and this recently added plot data set. FORTRAN IV; IBM 360.

CHANGES IN THE DATA LIBRARY COLLECTION

The following changes were made in the data collection.

DLC-35/EURLIB-III

The 100-group neutron cross-section library generated for use in the European shielding benchmark program was updated to change ID numbers, making them consistent with the retrieval routine which is included in the package. Mahmoud Metghalchi, Atomic Energy Organization of Iran on assignment in RSIC, called attention to the need for this change. European Shielding Information Service, Ispra, Italy, contributed the data library to the RSIC collection. The updated library may be requested as DLC-35B.

DLC-45/SENPRO

The data package (compilation of multigroup sensitivity profiles in SENPRO format) has been extended to include additional profiles for fast reactor benchmarks and to add profiles for two thermal reactor benchmarks and a fast reactor shielding benchmark. The ORNL 126-neutron-group set now includes a total of 650 profiles for CSEWG fast reactor benchmarks, ZPR6-7, ZPR6-6A, ZPR3-56B, ZPR3-11,Godiva and Jezebel. A set of 12-neutron-group profiles was contributed by Argonne National Laboratory for ZPR3-48, ZPR6-7, ZPR6-6A, and ZPR9-31. ORNL added a 32-group set used in the LCCEWG studies and a 4-group set for ZPR6-7. Profiles from all the benchmarks mentioned above were collapsed or expanded to 26 groups and the resulting set added. Profiles for the CRBR upper axial shield experiment (171-neutron group) have also been added to the package. Also included is a 131-group set for the TRX-2 thermal lattice benchmark and a 57-group set for the U-L212 mixed-oxide lattice benchmark. The resulting package is denoted DLC-45B. The SENDIN retrieval code is now provided for BCD to binary conversion and editing. A full, blocked, magnetic tape is required for transmittal (46,000 records). References: ORNL-5262, ZPR-TM-240, ZPR-TM-280, ORNL-5356, ORNL/TM-5946, plus two papers and informal notes. IBM 360/91.

DLC-54/LAFPX-E

A package of 154-neutron-group cross sections for fission products generated with NJOY from ENDF/B-IV was contributed by Los Alamos Scientific Laboratory in New Mexico. The data describe 824 nuclides. Cross sections, given for 181 of these nuclides, have been processed into 154 neutron energy groups. The fission-product library includes radioactive decay, neutron reaction, and fission yield data for the 824 nuclides. Of these 824 nuclides, 181 have neutron cross-section evaluations for total, elastic, total inelastic, and radiative capture reactions from 10^{-5}eV to 20 MeV. Additional cross-section evaluations are included for 36 of the 181 nuclides for other neutron absorption reactions, e.g., (n,2n), (n,p), etc.

DLC-56/FTF

A compilation of the results of neutron and gamma-ray transport calculations of dose transmission factors for concrete was contributed by the Research Institute of National Defense (FOA), Stockholm, Sweden. The factors were calculated with ANISN in the adjoint mode. The cross sections were a 22,18-group set collapsed from the DLC-27/AMPX01 104,22-group set. The retrieval and folding code included in the package reads the data tables and the user's source information, performs folding and, if necessary, interpolates in slab thickness. Reference: FOA Report C-20195-A2. FORTRAN IV; IBM 360.

OCTOBER ACCESSION OF LITERATURE

The following literature cited has been ordered for review, and that selected as suitable will be placed in the RSIC Information Storage and Retrieval Information System (SARIS). This early announcement is made as a service to the shielding community. Copies of the literature are not distributed by RSIC. They may generally be obtained from the author or from a documentation center such as the National Technical Information Service (NTIS), Department of Commerce, Springfield, Virginia 22151.

RSIC maintains a microfiche file of the literature entered into SARIS, and duplicate copies of out-of-print reports may be available on request. Naturally, we cannot fill requests for literature which is copyrighted (such as books or journal articles) or whose distribution is restricted.

THIS LITERATURE IS ON ORDER. IT IS NOT IN OUR SYSTEM. PLEASE ORDER FROM NTIS OR OTHER AVAILABLE SOURCE AS INDICATED.

REACTOR AND WEAPONS RADIATION SHIELDING LITERATURE

AAEC/E-426
Reactor Calculations and Nuclear Information.
Lang, D.W.
December 1977
Dep., NTIS (U.S. Sales Only)

AECL-5965
The Concrete Canister Program.
Ohta, M.M.
February 1978
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AERE-R8903
Radioactive-Nuclide Decay Data for Reactor Calculations: Activation Products and Related Isotopes.
Nichols, A.L.
1978
HMSO

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Argonne National Laboratory Papers Presented at Third ANS Topical Meeting on the Technology of Controlled Nuclear Fusion, May 9-11, 1978, Sante Fe, New Mexico.
Argonne National Laboratory
July 1978
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ANL/NDM-40
Statistical Theory of Neutron Nuclear Reactions.
Moldauer, P.A.
February 1978
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The Fusion Reactor Materials Program Plan. Section II. Damage Analysis and Fundamental Studies.
Stiegler, J.O.; Doran, D.G.; Bauer, W.; Scott, J.L.
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Nuclear Data for Actinide Production and Depletion Calculations.
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ECN-36
Damage Cross Section Library (DAMSIG77).
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Netherlands Energy Research Foundation (ECN), 3 Westerduinweg, Petten (NH), The Netherlands

FRA-TM-114
The VIM Cross Section Library Processing System: Description and User Information.
Prael, R.E.
June 14, 1978
Applied Physics Division, Argonne National Laboratory, Argonne, Illinois 60439

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Some Comments on Measurements of Reaction Rates by Gamma-Ray Spectrometry.
Helmer, R.G.
August 1976
Hanford Engineering Development Lab., Richland, Wash.

HEDL-TME-75-130, pp.BNL.1–BNL.13
Evaluated Cross Section Libraries.
Magurno, B.A.
August 1976
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HEDL-TME-78-39
Subcritical Reactivity Monitoring: Neutron Yields from Spontaneous (alpha,n) Reactions in FFTF Fuel.
Ombrellaro, P.A.; Johnson, D.L.
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ICRU-27
An International Neutron Dosimetry Intercomparison.
ICRU
1978
ICRU Publications, P.O. Box 30165, Washington, D.C. 20014

INDC(CCP)-107/L
Neutron Cross-Sections of Deuterium in the Energy Range 0.0001 eV-15 MeV.
Bazazyants, N.O.; Zabrodskaya, A.S.; Larina, A.F.; Nikolaev, M.N.
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IAEA Nuclear Data Section, Karntner Ring 11, A-1010 Vienna

INDC(CCP)-124/LV
Preparation of Nuclear Data for Detailed Calculation of the Neutron Spectrum in Fast Systems.
Vozyakov, V.V.; Voropaev, A.I.; Abdullaev, Kh.Sh.; Vorotyntsev, M.F.; Van'kov, A.A.; Krivtsov, A.S.; Pivovarov, V.A.
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IAEA Nuclear Data Section, Karntner Ring 11, A-1010 Vienna

LA-7174-MS
Multigroup and Few-Group Cross Sections for ENDF/B-IV Fission Products; the TOAFEW Collapsing Code and Data File of 154-Group Fission-Product Cross Sections.
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The Time Dependence of the Compton Current and Energy Deposition from Scattered Gamma Rays.
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Los Alamos Scientific Lab., P.O. Box 1663, Los Alamos, New Mexico 87545

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Background Cross Section Method as a General Tool for Reactor Analysis.
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Integral Decay-Heat Measurements and Comparisons to ENDF/B-IV and V.
England, T.R.; Schenter, R.E.; Schmittroth, F.
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Los Alamos Scientific Lab., P.O. Box 1663, Los Alamos, New Mexico 87545

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Development of Depletion Perturbation Theory for Coupled Neutron/Nuclide Fields.
Williams, M.L.
September 1978
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Revised Uranium-Plutonium Cycle PWR and BWR Models for the ORIGEN Computer Code.
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Dickson, H.W.; Hill, G.A.; Perdue, P.T.
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Technical Specifications for the Bulk Shielding Reactor.
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Dep., NTIS $4.00

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Varela, D.W.; Philbin, J.S.
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Theory and Practice of Solving Ordinary Differential Equations (ODEs).
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Lord, M.E.
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TID-28061
Effects of Nuclear Weapons. Third Edition.
Glasstone, S.; Dolan, P.J.
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Program Linear (Version 78-1): Linearize Data in the Evaluated Nuclear Data File/Version B(ENDF/B) Format.
Cullen, D.E.
July 4, 1978
NTIS

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Program SIGMA1 (Version 78-1): Doppler Broaden Evaluated Cross Sections in the Evaluated Nuclear Data File/Version B (ENDF/B) Format.
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Description of the Response of Reactor First Walls to Pulsed Thermonuclear Radiation (Part I).
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March 1977
Fusion Research Program, Nuclear Engineering Dept., University of Wisconsin, Madison, Wisconsin 53706

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October 1977
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SOLACE - A Laser Fusion Reactor Study. Chapter I. Overview.
Conn, R.W.; Abdel-Khalik, S.I.; Moses, G.A.; Beranek. F.; Cheng, E.T.; Cooper, G.W.; Droll, R.B.; Henderson, T.; Howard, J.; Hunter, T.O.; Larsen, E.M.; Kulcinski, G.L.; Lovell, E.G.; Magelssen, G.R.; Maynard, C.W.; Okula, K.R.; Ortman, M.; Raghev, M.M.H.; Rensel, W.B.; Solomon, D.
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Fusion Research Program, Nuclear Engineering Dept., University of Wisconsin, Madison, Wisconsin 53706

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Miura, T.; Yamaji, A.; Takeuchi, K.
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