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. 168 _____ *December 1978*

Always tell the truth. You may make a hole in one when you're alone on the golf course someday. . . . *Franklin P. Jones*

NEW YEAR — NEW TELEPHONE NUMBER
615-574-6176 Commercial
624-6176 FTS

A new telephone system for the Oak Ridge National Laboratory and other local DOE installations has been under installation for several months and will be in operation on **January 1, 1979**. The number under which RSIC has been accessed for the past 17 years will no longer be valid. The new system offers more flexibility, and after a shakedown period, there should be less difficulty in reaching RSIC staff members. We will be on a rotary system with enough lines to efficiently serve the RSIC user community without the delays caused by an overload such as that routinely experienced over the past several months. So, make a note of the new telephone number listed above and of the effective date (January 1, 1979). The first number listed above may be used by anyone. The 7-digit number is for the use of those on the Federal Telecommunications System.

SHIELDING SPECIALISTS TRAVEL ABROAD

Two November meetings outside the United States, when fully reported, will be of interest to the international shielding community. The OECD Nuclear Energy Agency Committee on Reactor Physics met at JAERI, Tokai-Mura, Japan November 6–10, 1978 and a USA team visited the USSR for a bilateral FBR seminar at Obninsk, Kaluga region.

F. C. Maienschein and P. B. Hemmig, U. S. representatives to NEACRP, informally report that emphasis in reactor physics is changing, at least in the NEA area, from its classical role of supporting or leading reactor core design. In the future, there are expected to be more physics (neutronics) studies related to efficient reactor operation and to other parts of the fuel cycle, including safeguards.

The papers from the bilateral discussions with the USSR shielding specialists will be formally reported in Volume 4 of the Soviet journal, "Radiation Safety and Protection in Atomic Electric Power Plants." The members of the U. S. team were W. L. Bunch of HEDL, R. K. Disney of WARD, P. B. Hemmig of DOE, F. C. Maienschein and E. O. Oblow of ORNL.

PATENT ISSUED ON NON-COMBUSTIBLE HYDROGENOUS SHIELDING MATERIAL

William C. Hall and John M. Peterson of Chemtree Corporation, Central Valley, N.Y. have announced their recent patent for "Non-Combustible Nuclear Radiation Shields with High Hydrogen Content." With increasing emphasis on fire-resistance, the ready availability of non-combustible hydrogenous material is a welcome development in the shielding field.

PERSONAL ITEMS

The following changes of address have been received:

Jean-Paul Renier *from* Nuclear Assurance of Atlanta *to* UCND Computer Sciences at the Oak Ridge National Laboratory; **V. Sundararaman** *from* the Kalpakkam Reactor Research Center in India *to* the European Shielding Information Service at Euratom, Ispra, Italy; **Memory Horatio Turner, III** *from* Dallas, Texas *to* Owens-Corning Fiberglas in Toledo, Ohio; **Tomas Lefvert** *from* the Swedish Research Institute of National Defence in Stockholm *to* the State Power Board in Vällingby; and **E. T. Boulette** *from* Stone & Webster in Boston *to* HEDL in Richland, Washington.

VISITORS TO RSIC

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

P. Barbucci, ENEL, Pisa, Italy; Ernst H. Brehm and Rolf Geradt, Brown, Boveri & Cie, Mannheim, Federal Republic of Germany; Anthony R. Buhl, Nuclear Regulatory Commission, Washington, D.C.; C. E. Clifford, Radiation Research Associates, Fort Worth, TX; J. C. Courtney, Louisiana State University, Baton Rouge; Masao Kitamura, Hitachi Ltd., Ibaraki, Japan; Wolfgang Osterhage, OECD-NEA Data Bank, Paris, France, and Richard D. McCulloch and Ellen Williams, Information Division, ORNL.

UPCOMING MEETINGS

January 1979

First Topical Meeting on Fusion Reactor Materials, January 29–31, 1979, Americana of Bal Harbour, Miami Beach, Florida. Contact: J. A. DeMastry, Florida Power & Light Co., P. O. Box 529100, Miami, Florida 33152; 305-552-3547.

February 1979

6th Energy Technology Conference & Exposition, February 26–28, 1979, Sheraton Park Hotel, Washington, D.C. Contact: Conference Registrar, Energy Technology Conference, Inc., Dept. B-24680, 4733 Bethesda Ave., N.W., Washington, D.C. 20014.

AMPX-II PACKAGE NOW AVAILABLE

The PSR-63/AMPX-II modular code system for generation and manipulation of coupled multigroup neutron and gamma-ray cross sections is now available for distribution. This version supersedes the original AMPX package and represents substantial improvements to most of the major modules of the original system as well as the inclusion of many valuable new modules for performing a variety of processing and handling functions.

Modules available in the new package (a total of 118,371 logical records) include:

DRIVER	—	the AMPX module manager.
XLACS-2	—	produces multigroup neutron cross sections from ENDF-formatted data (or NPTXS processed data).
NPTXS	—	prepares point cross sections for materials with resonance parameters.
LAPHNGAS	—	produces multigroup secondary gamma-ray production data (multiplicities or cross sections).
SMUG	—	produces multigroup gamma-ray interaction cross sections.
CHOX	—	performs cross-section interface management, e.g., combining output from XLACS-2, LAPHNGAS, and SMUG.

NITAWL	—	performs resonance self-shielding and produces working libraries.
ROLAIDS	—	treats resonance self-shielding in multiregion geometry.
BONAMI	—	performs resonance self-shielding using Bondarenko factors.
CHOXM	—	generates a self-shielded AMPX master library using SPHINX generated self-shielding factors.
XSDRNPM	—	provides one-dimensional S_n capability for spatial cross-section weighting.
AIM	—	provides BCD-Binary conversion of AMPX master libraries.
AJAX	—	merges, collects, assembles, reorders, joins and/or copies selected data from AMPX master libraries produced by XLACS-2, LAPHNGAS, and SMUG.
DIAL	—	edits data from AMPX master libraries.
VASELINE	—	plots multigroup and/or point data for visual comparisons.
RADE	—	checks AMPX master libraries for consistency and reasonableness.
MALOCS-2	—	collapses AMPX master libraries with input spectra.
PAL	—	punches reaction cross sections from AMPX master libraries.
DOSE	—	calculates multigroup neutron and gamma-ray dose factors.
UNITAB	—	selects portions of AMPX master libraries to prepare hybrid master sets.
CLAROL	—	replaces cross sections on an AMPX master library.
COMET	—	corrects selected portions of AMPX master libraries.
LAVA	—	converts an ANISN-formatted library into an AMPX working library.
CONVERT	—	converts old XSDRN-formatted library into an AMPX master library.
REVERT	—	converts AMPX master library into old XSDRN format.
CONTAC	—	converts AMPX working libraries into ANISN or CCCC ISOTXS format.
GERITOL	—	places group constants into a "string" library.

MULTIGROUP NUCLEAR CROSS-SECTION PROCESSING PROCEEDINGS NOW AVAILABLE

ORNL/RSIC-41, "A Review of Multigroup Nuclear Cross-Section Processing Proceedings of a Seminar-Workshop, Oak Ridge, Tennessee," compiled by D. K. Trubey and H. R. Hendrickson (March 14–16, 1978) is now available from National Technical Information Service, U. S. Department of Commerce, 5285 Port Royal Road, Springfield, Virginia 22161. Price: $10.75 (Printed Copy); $3.00 (Microfiche).

The Proceedings are also available here at RSIC while the supply lasts.

A NOTE ON ORIGEN DATA BASED ON ENDF/B-IV FISSION PRODUCT DATA

R. Hubner, Sargent and Lundy, has pointed out an error in an ORIGEN library based on decay data from the ENDF/B-IV fission product file. An incorrect value of 0.14 is given for the branching ratio for β-decay of ^{133}I to ^{133m}Xe. The value 0.029 in the original ORIGEN library is more correct. R. Schenter, HEDL, Chairman of the CSEWG Fission Product Subcommittee has confirmed that ENDF/B-V will have a corrected value for the branching ratio.

CHANGES IN THE COMPUTER CODE COLLECTION

CCC-222/TWOTRAN II

An extension was made and an error corrected in the CDC 7600 version (A) of this two-dimensional multigroup discrete ordinates transport code package. It was extended to include utility (LCM) routines, provided by UCND Computer Sciences Division (CSD) at ORNL. The LCM routines allow the code to be operational under FTN, and thereby make it a more generally compatible code. A further update was made to correct an error called to RSIC attention by E. T. Tomlinson of UCND-CSD at ORNL and Forrest W. Brinkley, Jr. of LASL. For those now using this TWOTRAN II version, the error may be corrected as follows:

In Subroutine INPUT11, in the first executable statement preceding statement 100, only one set of parentheses was inserted. The corrected statement reads: IF (EOF(NINP)) 100,110.

The IBM version of this package was not affected by either extension or the update. TWOTRAN II was a contribution of Los Alamos Scientific Laboratory, Los Alamos, New Mexico.

CCC-315/SAMSY

This one-dimensional multilayer multigroup neutron removal-diffusion and gamma-ray point kernel code package was updated to make corrections in Subroutine HEDO1 by the contributor, Dr. Lech Szymendera of the Institute of Nuclear Research at Swierk-Otwock, Poland. A statement of the error and its correction is available from RSIC for current SAMSY users.

CCC-325/KAMCCO

A three-dimensional Monte Carlo code package for use in solving fast neutron physics problems was contributed by the Institute for Neutron Physics and Reactor Technology, Karlsruhe, FRG, through the OECD NEA Data Bank. The code solves the inhomogeneous time-dependent transport equation or the homogeneous static equation. Reaction rates or integrations of the flux over specified phase-space volumes can be calculated. In the cross-section treatment, energy is continuous and may be computed by linear interpolation from Doppler-broadened Breit-Wigner resonances, or from probability tables. Reference: KFK-2190. FORTRAN IV; IBM 370.

CCC-326/TREEDE

A Monte Carlo transport code for the calculation of flux (current) at a point in slab geometry by the track rotation estimator method was contributed by Ben Gurion University of the Negev, Beer Sheva, Israel. The problem solved is that of the flux at a point due to a point source. Each particle's track crossing a sphere whose radius equals the source-to-detector distance is rotated such that the rotated track passes through the detector point and a score is made for the flux at the point. Compensating weights are applied, depending on the relation between the original and rotated random walks. FORTRAN IV; CDC.

CCC-327/PHOEL

A Monte Carlo calculation of electron energies from photon interactions was contributed by Oak Ridge National Laboratory. PHOEL generates the initial energies of Compton and photoelectrons in water irradiated by photons with an arbitrary energy spectrum. An infinite, homogeneous water phantom is assumed in which the photon spectrum is uniform throughout. The code was written specifically to provide a source term for a Monte Carlo electron energy degradation and transport code for liquid water being used to study the relative biological effectiveness (RBE) of low-LET radiations at low doses. Reference: ORNL/TM-6515. FORTRAN IV; IBM 360.

PSR-63/AMPX II

A new model (AMPX II), which reflects all current code development in the series by the Oak Ridge National Laboratory and the UCND Computer Sciences Division, replaces the modular code system for generating multigroup neutron, gamma-ray, or coupled neutron-gamma-ray cross-section libraries from ENDF/B-formatted data, originally packaged in September, 1973. The new model includes, in the interim,

updated existing modules and many new modules. AMPX II consists of 32 modules for: (1) basic neutron and gamma-ray production, and/or gamma-ray interaction multigroup cross-section generation, (2) resonance self-shielding, (3) spectral collapsing, (4) format conversion, (5) one-dimensional discrete ordinates calculations, and (6) miscellaneous cross-section operations. Also included in the package are data libraries, input and output for a series of AMPX II sample problems, revised input instructions for the modules, and descriptions of the sample problems which provide the novice user with hints for utilizing the system. The AMPX II package contains 113,755 logical records. One 9-track, or several 7-track tapes are required for transmittal. References: ORNL/TM-3706, ORNL/CSD/TM-9/RM, ORNL CF-74-12-2, ORNL/CSD/TM-72. FORTRAN IV and Assembler Language; IBM 360.

PSR-117/MARS

The MARS code package for manipulating multigroup cross-section libraries in the AMPX and CCCC formats was updated with additions, corrections, and other modifications. (1) FORTRAN versions of several service routines, formerly in IBM 360 Assembler Language only, were added for more hardware independence: CLOCK, ITIME, TIMFAC, ALOCAT, DA, REED, RITE, FND, RD, DATIME. (2) Five Assembler Language utility routines were added: ICLOCK, IAND, IOR, ERRO, ICOMPL. (3) A new version of MALOCS replaces the original which would not properly collapse an AMPX library if some of the original fine groups were not a subset of the final broad group library (e.g., trying to change the upper limit of the resulting library). The new MALOCS correctly labels reactions and some group boundaries, correcting a problem in the original. (4) Subroutine SCALE in BONAMI was modified to correct a problem encountered in CDC versions where negative integer and floating point numbers do not follow IBM machine patterns. Current users may request either or all of the new material (1)—(4). A written statement of changes required to update SCALE is also available. Credit for these updates is shared with John Ridihalgh of Ridihalgh Associates, David L. Chapin of Westinghouse Fusion Power Systems, attendees at the 1978 RSIC seminar-workshops, and the code originators at ORNL.

PSR-130/MATXUF

An on-line derivative method, fast-spectrum unfolding code for NE-213 liquid scintillation proton recoil data was contributed by the Nuclear Engineering Department of the University of Missouri, Columbia. MATXUF generates an "unfolding matrix" for the analysis of NE-213 liquid scintillation proton recoil spectra which can be used in either an off-line or an on-line unfolding environment. For off-line application, unfolding can be accomplished by simple multiplication of the proton recoil vector times the unfolding matrix. For on-line application, the unfolding is accomplished one detected recoil proton at a time by using a small computer (in which the matrix has been stored) interfaced to the detection system. The technique uses the derivative method of unfolding which assumes an ideal, theoretical step function relationship between incoming neutrons and the resulting proton recoil distribution. This ideal function is corrected for anisotropic scattering and proton leakage. It also includes a smoothing technique applied directly to the unfolded result which improves energy identification and results in the ultimate generation of an unfolding matrix. Reference: NIM 153 (1978), 535-541. FORTRAN IV; IBM 360/370.

PSR-131/CARP

CARP, a data processor for DOT angular flux tapes, producing an intermediate albedo output tape for input to BREESE, the MORSE albedo routine, was contributed by Oak Ridge National Laboratory. Reference: ORNL/TM-6503. FORTRAN IV; IBM 360.

CHANGES IN THE DATA LIBRARY COLLECTION

The following changes were made in the data collection.

DLC-47/BUGLE

The documentation for the data package was updated to correct an error in the energy and lethargy

values for neutron group 17, printed incorrectly in Table 2 of the document. For those who have the document, the correct values are 1.0026 MeV and 2.300 for the group 17 upper bound values of energy and lethargy, respectively.

DLC-58/HELLO

The coupled 47 neutron, 21 gamma-ray group cross-section library in ANISN format was developed for use in analyzing a Li(D,n) neutron radiation damage facility. The upper energy limit for neutrons is 60 MeV. A P_5 Legendre expansion is used above 14.9 MeV and P_3 below that energy. Below 14.9 MeV, the data are provided from a subset (35n, 21g) library based on DLC-41/VITAMIN-C. Above 14.9 MeV, differential elastic scattering data based on optical model calculations and differential nonelastic scattering data based on the intranuclear-cascade-evaporation model are used. Data are provided for H, ^{10}B, ^{11}B, C, O, Si, Ca, Cr, Fe, Ni, and heavy concrete. A retrieval code is provided for conversion from card image to unformatted form. A single reel of magnetic tape is required for transmittal. Reference: ORNL/TM-6486. IBM 360/91.

RSIC GRAB BAG

We offer the following documents on a first-come, first-served basis. Please order by document number.

ORNL-RSIC-3—*A Comparison of First- and Last-Flight Expectation Values Used in an 05R Monte Carlo Calculation of Neutron Distributions in Water*—D. K. Trubey and M. B. Emmett (May 1965).

ORNL-RSIC-4—*Some Calculations of the Fast-Neutron Distribution in Ordinary Concrete from Point and Plane Isotropic Fission Sources*—D. K. Trubey and M. B. Emmett (June 1965).

ORNL-RSIC-5 Vol. II, III, IV, and V—*Bibliography, Subject Index, and Author Index of the Literature Examined by the Radiation Shielding Information Center (Reactor and Weapons Shielding)*.

ORNL-RSIC-7—*Tabulated Values of Scattered Gamma-Ray Fluxes in Iron Interpolated from Moments-Method Calculations*—D. K. Trubey (May 1965). (Microfiche only)

ORNL-RSIC-8—*Survey of Methods for Calculating Gamma-Ray Heating*—H. C. Claiborne (June 1965). (Microfiche only)

ORNL-RSIC-9—*A Comparison of Three Methods Used to Calculate Gamma-Ray Transport in Iron*—D. K. Trubey, S. K. Penny, and K. D. Lathrop. (October 1965). (Microfiche only)

ORNL-RSIC-10—*A Survey of Empirical Functions Used to Fit Gamma-Ray Buildup Factors*—D. K. Trubey (February 1966). (Microfiche only)

ORNL-RSIC-11—*Bibliography, Subject Index, and Author Index of the Literature Examined by the Radiation Shielding Information Center (Space and Accelerator Shielding) (Rev. II, May 1970)*.

ORNL-RSIC-12—*Abstracts of the Literature Examined by the Radiation Shielding Information Center (Space and Accelerator Shielding)*. (Microfiche only)

ORNL-RSIC-13, Vol. I, II, III, and IV—*Abstracts of Digital Computer Codes Assembled by the Radiation Shielding Information Center*—Betty F. Maskewitz, Betty L. McGill, Hemma E. Comolander, Marie Anthony, and Henrietta R. Hendrickson. (Vol. I and II, Microfiche only)

ORNL-RSIC-14—*The Exponential Transform as an Importance-Sampling Device – A Review*—Francis H. Clark (Jan. 1966).

ORNL-RSIC-16—*Use of ICRU-Defined Units in Shielding*—D. K. Trubey (October 1968).

ORNL-RSIC-17—*Comparisons of Results Obtained with Several Proton Penetration Codes*—W. Wayne Scott and R. G. Alsmiller, Jr. (July 1967).

ORNL-RSIC-18—*Estimates of Primary and Secondary Particle Doses Behind Aluminum and Polyethylene*

Slabs Due to Incident Solar-Flare and Van Allen Belt Protons—W. Wayne Scott (July 1967).

ORNL-RSIC-19—*A Review of the Discrete Ordinates S_n Method of Radiation Transport Calculations*—D. K. Trubey and Betty F. Maskewitz (March 1968).

ORNL-RSIC-20—*Weapons Radiation Shielding Handbook – Chapter 5: Methods for Calculating Effects of Ducts, Access Ways, and Holes in Shields*—Wade E. Selph and H. Clyde Claiborne (Jan. 1968).

ORNL-RSIC-21—*Weapons Radiation Shielding Handbook – Chapter 4: Neutron and Gamma-Ray Albedos*—Wade E. Selph (Feb. 1968).

ORNL-RSIC-22—*Comparisons of Results Obtained with Several Proton Penetration Codes – Part II*—W. Wayne Scott and R. G. Alsmiller, Jr. (June 1968).

ORNL-RSIC-24—*Compilation of Data on Experimental Shielding Facilities and Tests of Shields of Operating Reactors*—compiled by: European American Committee on Reactor Physics, European Nuclear Energy Agency (Nov. 1968).

ORNL-RSIC-25—*Shielding Benchmark Problems*—A. E. Profio, Editor.

ORNL-RSIC-26—*The Attenuation Properties of Concrete for Shielding of Neutrons of Energy Less Than 15 MeV*—F. A. R. Schmidt (Aug. 1970). **(Microfiche only)**

ORNL-RSIC-27—*A Review of Multigroup Nuclear Cross Section Preparation – Theory, Techniques, and Computer Codes*—compiled by D. K. Trubey and J. Gurney (Jan. 1970). **(Microfiche only)**

ORNL-RSIC-28—*Comparisons of the Results Obtained with Several Electron-Penetration Codes*—W. Wayne Scott (March 1970).

ORNL-RSIC-29—*A Review of the Monte Carlo Method for Radiation Transport Calculations*—compiled by Betty F. Maskewitz and Vivian Z. Jacobs (February 1971). **(Microfiche only)**

ORNL-RSIC-30—*Abstracts of the Data Library Packages Assembled by the Radiation Shielding Information Center*—R. W. Roussin (March 1972).

ORNL-RSIC-31—*Abstracts of Peripheral Shielding Code Packages Assembled by the Radiation Shielding Information Center*—Betty F. Maskewitz.

ORNL-RSIC-32—*Recent Developments in the Shielding of Neutron Sources*—H. Clyde Claiborne (June 1971). **(Microfiche only)**

ORNL-RSIC-33—*A Review of Calculations of Radiation Transport in Air – Theory, Techniques, and Computer Codes*—compiled by D. K. Trubey and H. E. Comolander (May 1972).

ORNL-RSIC-34, Vol. I—*Defense Nuclear Agency Working Cross Section Library – Description and Contents*—R. W. Roussin (October 1972).

ORNL-RSIC-35—*Shielding of Manned Space Vehicles Against Protons and Alpha Particles*—R. G. Alsmiller, Jr., R. T. Santoro, J. Barish, H. C. Claiborne (Nov. 1972). **(Microfiche only)**

ORNL-RSIC-36—*Shielding Against Initial Radiations from Nuclear Weapons*—Lorraine S. Abbott (July 1973).

ORNL-RSIC-38, Vol. I, II, III—*Techniques for Efficient Monte Carlo Simulation*—E. J. McGrath, *et al.* (April 1975). **(Microfiche only)**

ORNL-RSIC-39—*The Development of Radiation Shielding Standards in the American Nuclear Society*—D. K. Trubey (November 1975).

ORNL-RSIC-40—*A Review of Radiation Energy Spectra Unfolding*—compiled by D. K. Trubey (July 1976). **(Microfiche only)**

ECN-77-103, "Cross Section Library DOSCROS77," (in the SAND-II format) by Nico J.C.M. van der Borg, Henk J. Nolthenius, and Willem L. Zijp (Aug. 1977), ECN, The Netherlands.

ECN-77-104, "Nuclear Data Guide for Reactor Neutron Metrology," by Willem L. Zijp (Aug. 1977), ECN, The Netherlands.

ECN-77-111, "Comparison of Neutron Spectrum Unfolding Codes," by Willem L. Zijp, Jan H. Baard, and Henk J. Nolthenius (Sept. 1977), ECN, The Netherlands.

NOVEMBER 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

AED-Conf-76-556-008 (In German)
Damage Model and Neutron Spectrometry Applied in Reactor Steel Irradiation Programmes.
Weise, L.
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Using the Finite Element Method to Calculate Three-Dimensional Problems in Reactor Physics.
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The Effects of Radiation on Electrical Insulators in Fusion Reactors.
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BNL-NCS-23375/R
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Barton, B.J.; Tuli, J.K.
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Perry, R.T.; McKinnon, M.A.; Teofilo, V.L.; Aase, D.T.
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Uncertainties in the Breeding Ratio of a Large LMFBR.
Marable, J.H.; Weisbin, C.R.
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CONF-780546-1
    Review of Cross Section Data Important to the Uranium–Plutonium Fuel Cycle in Thermal Reactors.
    Weston, L.W.
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    Dep., NTIS

CONF-780615-3
    Computers in Technical Information Transfer.
    Price, C.E.
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    Sensitivity Analyses of Fast Reactor Systems Including Thorium and Uranium.
    Marable, J.H.; Weisbin, C.R.
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    Version 4 of ATR (Air Transport of Radiation).
    Huszar, L.; Woolson, W.A.; Straker, E.A.
    January 1976
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EPRI-NP-196, Vol.2
    Clean Critical Experiment Benchmarks for Plutonium Recycle in LWRs.
    Smith, R.J.; Konzek, G.J.
    September 1978
    Battelle Pacific Northwest Laboratories, Richland, Washington 99352

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    Dosimeter Based Adjustments of Fast Neutron Spectra.
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JAERI-M-7567
    Evaluation of Gamma-Ray Intensities.
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JAERI-M-7734
    Level Schemes for Some Fission Product Nuclides. Comparison of Level Schemes Used by JAERI and Petten.
    Matumoto, Z.; Murata, T.; Nakasima, R.
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    Japan Atomic Energy Research Inst., Tokyo

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    A Hybrid Method for the Numerical Solution of the Electron Transport Equation: The Reduced Source Method.
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    The Application of a Library of Processed ENDF/B-IV Fission-Product Aggregate Decay Data in the Calculation of Decay-Energy Spectra.
    LaBauve, R.J.; England, T.R.; George, D.C.; Stamatelatos, M.G.
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    Nucleonic Aspects of Synfuel Blankets.
    Dudziak, D.J.; Woodruff, G.L.
    October 1978
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LA-UR-78-1288; CONF-780508-44
    TRIDENT-CTR: A Two-Dimensional Transport Code for CTR Applications.
    Seed, T.J.
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    Dep., NTIS

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    Radiation Dose Estimates from Timepieces Containing Tritium or Promethium-147 in Radioluminous Paints.
    McDowell-Boyer, L.M.; O'Donnell, F.R.
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Dose-Response Curves from Incomplete Data.
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Ewbank, W.B.
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Estimates of Internal Dose Equivalent to 22 Target Organs for Radionuclides Occurring in Routine Releases from Nuclear Fuel-Cycle Facilities. Vol.1.
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Two- and Three-Dimensional Neutronics Calculations for the TFTR Neutral Beam Injectors.
Santoro, R.T.; Lillie, R.A.; Alsmiller, R.G.,Jr.; Barnes, J.M.
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And best wishes for the coming year