COMPILAÇÃO DOS CÓDIGOS NUCLEARES DISPONÍVEIS NO CPD/IEA

Ana Granzotto e Antonio Soares de Gouvêa

INFORMAÇÃO IEA 067 IEA - Inf - 067	JANEIRO/1979

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CENTRO DE PROCESSAMENTO DE DADOS
CPD 008

INSTITUTO DE ENERGIA ATÔMICA
SÃO PAULO – BRASIL

COMPILAÇÃO DOS CÓDIGOS NUCLEARES DISPONÍVEIS NO CPD/IEA

Ana Granzotto[1] e Antonio Soares de Gouvêa[2]

0 – NOTAS EXPLICATIVAS

Esta compilação dos "Abstracts" dos códigos disponíveis no Centro de Processamento de Dados do I.E.A., é o início da Biblioteca de Programas em Planejamento.

A compilação está dividida em quatro partes, a saber:

1. Índice Alfabético dos códigos disponíveis no C.P.D. – reune e identifica os códigos disponíveis no C.P.D.

2. Classificação de Programas – Por meio das tabelas do Argonne National Laboratory e Nuclear Energy Agency, identifica os assuntos tratados nos códigos existentes nesses centros de pesquisa.

3. Índice Alfabético de Classificação de Assuntos – Por meio das Tabelas citadas relacionam os códigos disponíveis no C.P.D.

4. Abstracts – reune todos os relatórios de informação dos códigos disponíveis no C.P.D., por ordem alfabética de classificação de assuntos.

Os Índices Alfabéticos de Assunto e Alfabético de Códigos disponíveis no C.P.D., remetem para o "Abstracts" que fornece informações relativas a cada código.

Sendo o I.E.A. membro associado da Nuclear Energy Agency e uma vez que essa entidade reune códigos de vários centros europeus de pesquisa e dos americanos Oak Ridge National Laboratory e Argonne National Laboratory, o pesquisador que necessitar algum programa, poderá requisitá-lo por intermédio do C.P.D.

Encontra-se em processo de armazenamento no C.P.D., o seguinte material bibliográfico de códigos nucleares:

A. Bibliografia do Argonne National Laboratory.

B. Documentação proveniente da Nuclear Energy Agency:
   1. Nuclear Program Abstracts.
   2. Newsletter – seminários e experiência sobre a atualização dos códigos nucleares por diferentes usuários.

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(1) Bibliotecária do Departamento de Informação e Documentação Científica, lotada no Centro de Processamento de Dados do Instituto de Energia Atômica.
(2) Analista de Sistema – Supervisor do Centro de Processamento de Dados do I.E.A.
Aprovada para publicação em Dezembro/1978.

C. Relatórios.

Solicitamos aos pesquisadores que possuam algum documento referente a códigos (relatórios, listagens, cartões, fitas) que façam doação ao C.P.D. a fim de ser tratado e armazenado.

Acreditamos que somente com a informação centralizada poderemos atender as necessidades de todos os usuários.

1 – ÍNDICE ALFABÉTICO DOS CÓDIGOS DISPONÍVEIS NO CPD/IEA

2 – CLASSIFICAÇÃO DOS PROGRAMAS

Esta tabela de classificação dos programas é a reprodução do Argonne Code Center: Compilation of Program Abstract, ANL 7411.

A. Cross-Section and Resonance-Integral Calculations

Computation of reaction cross sections from nuclear theory such as the optical or Hauser-Feshabach models, resonance cross sections by Breit-Wigner or multilevel theory, determination of differential scattering cross-sections, cross-section evaluation, and compilation programs.

B. Spectrum Calculations, Generation of Group Constants, Lattice and Cell Problems.

Determination of the slowing-down density of thermal spectrum, weighting and averaging of cross-sections and related quantities for the production of group constants, and evaluation of design parameters by lattice and cell calculation.

C. Static Design Studies

Calculation of the reactivity and flux distribution of the reactor system, and adjustment of design parameters to prescribed specifications, i.e., criticality and power distribution search procedures.

D. Depletion, Fuel Management, Cost Analysis and Reactor Economics

Includes burnup programs, isotope and fission-product buildup and decay computations, and optimization studies.

E. Space-Independent Kinetics

Studies of the time behavior of reactors, including delayed-neutron effects and feedback mechanisms, and transfer-function evaluation.

F. Space-Time Kinetics, Coupled Neutronics, Hydrodynamics, Thermodynamics and Excursion Simulations.

Programs that consider spatial design characteristics and accompanying effects in studying the time behavior of the reactor.

G. Radiological Safety, Hazard and Accident Analysis

Calculation of internal and external dose rates, determination of reactor thermodynamic and hydrodynamic properties following an accident, e.g., release of radioactive materials, coolant system blow down, steam generator rupture.

H. Steady-State and Transient Heat Transfer

Includes fluid-flow studies and calculations of thermodynamic properties.

I. Deformation and stres Distribution Computations, Structural Analysis, and Engineering Design Studies

Includes fuel element design evaluations, core configuration studies, and composite structure analysis.

J. Gamma Heating and Shield Design Programs

Computation of heat generation rates, and penetration analysis and leakage calculations for reactor shields.

K. Total Systems Analysis

Collections of solutions to correlated problems elicited from several categories, designed and used as systems.

L. Data Preparation

Generation of program parameters; checking, editing, and formatting the problem input information.

M. Data Management

Construction, maintenance, and retrieval of data files, e.g., cross-section libraries.

N. Subsidiary Calculations

Plotting, editing, and display routines that process output data from other programs.

O. Experimental Data Processing

Programs designed to process data directly acquired from an experimental situation or to assist experimenter in the design of the experiment.

P. General Mathematical and Computing System Routines

Calculation of mathematical functions, and special language routines with general data processing capabilities.

Q. Materials

Measurements and computation of the physical and mechanical properties of materials, simulation of radiation damage process, corrosion studies, and determination of crystallographic functions.

R. Environmental and Earth Sciences

Environmental impact studies, geology, seismology, geophysics calculations, hydrology and ground water studies, bioenvironmental systems analyses, meteorological calculations relating to the atmosphere and its phenomena, studies of airborne particulate matter, climatology, etc.

S. Space Sciences

Analysis of orbits and trajetories, astronomy and astrophysics computations, wave propagation studies, and the calculation or reentry parameters.

T. Electronics and Engineering Equipment

Automated design of electronics equipment, engineering computations for numerically controlled machine tools and process control programs.

U. Chemistry

Chemical analysis, mass spectroscopy, radiation chemistry, radiolysis studies, etc.

V. Particle Accelerators and High Voltage Machines

Programs relating to the design, development and operation of high-voltage machines and particle accelerators such as Van de Graaff generators, linear accelerators, cyclotrons, synchrotrons, etc.

W. Physics

Calculations relating to theory of atomic or molecular structure or properties, charged particle collision studies that involve phenomena such as charge exchange, excitation, ionization, dissociation, etc., elementary particle theories and models, scattering theory quantum field theory and quantum electrodynamics studies, general relativity and gravitation theory computations.

X. Controlled Thermonuclear Research

Electric discharge phenomena and plasma physics computations, electrodynamics and magnetic hydrodynamics studies.

Y. Biology and Medicine

Biological, medical and radiological studies of the structure, functions, Chemistry, biophysics, reproduction, and heredity of bacteria, plants, laboratory animals, and humans.

Z. Data

Data prepared in specified program formats for program testing and evaluation, benchmark studies, or library. use.

3 – ÍNDICE ALFABÉTICO DE CLASSIFICAÇÃO DE ASSUNTO

BLOOST-5	H.01
ORCENT	H.02

I. Deformation and Strees Distribution Computations, Structural Analysis and Engineering Design Studies.

SAFE-2D	I.01

K. Total Systems Analysis

AMPX-1	K.01
CITATION	K.02

M. Data Management.

ARC-System	M.01

P. General Mathematical and Computing System Routines.

ASAAM23	P.01
BEVINGTO	P.02
MATEXP	P.03
SAAM25	P.04
SAS	P.05

Z. Nuclear Data

MEDLIST	Z.01
ZZ-DLC-2D	Z.02

4 – ABSTRACTS

B. SPECTRUM CALCULATION, GENERATION OF GROUP CONSTANTS, LATTICE AND ALL PROBLEMS.

RELATÓRIO PARA INFORMAÇÃO SOBRE O CÓDIGO "BRT-I"

1. Sigla e Título:
   BRT-I, Battelle Revised Thermos.

2. Nome do Autor:
   Bennett, C. L. & Purcell, W. L.

3. Estabelecimento de Origem:
   Battelle Nortwest, Richland, WA (USA).

4. Abstract:
The code computes the space dependent thermal neutron density, flux and current spectra over the energy range 0 to 0.683 ev in either slab or cylindrical geometry.
The neutron density is computed from the collision probability form of the integral transport-theory matrix equation using either a combination of power iteration, over relaxation and extrapolation or straight power iteration. The neutron currents are computed from either the gradient of the scaler flux or the uncollided flux matrix. The flux and current spectra is used to weight point thermal cross sections over an arbitrary thermal energy range for use in multigroup transport or diffusion theory codes.

5. Palavras Chaves:
   BRT-I, SPECTRUM CALCULATIONS, GENERATION OF GROUP CONSTANTS, LATTICE AND CELL PROBLEMS, THERMAL SPECTRA, TRANSPORT THEORY, SLABS, CYLINDERS, CROSS SECTIONS, NEUTRON DENSITY, NEUTRON CURRENT.

6. Linguagem de Programação e Computador:
   FORTRAN-IV.

7. Estado Operacional do Código:

8. Material Disponível:
   a) BENNETT, C. L. & PURCELL, W. L. *BRT-I: Battelle Revised Thermos.* Richland, WA, Battelle Northwest, June 1970. (BNWL-1434).

   b) Informação em fita:
   DEN003, DEN005, IT3001.

B.02

RELATÓRIO PARA INFORMAÇÃO SOBRE O CÓDIGO "GAUSS 5"

1. Sigla e Título:
   GAUSS 5.

2. Nome do Autor:
   Helmer, R. G. & Putnam, M. H.

3. Estabelecimento de Origem:
   Aeroject Nuclear Company, Idaho Falls, ID (USA).

4. Abstract:
   GAUSS 5 is used to determine GAMMA-RAY energies and intensities from spectra obtained with a Ge(Li) detector and multichannel pulse-height analysis system.

5. Palavras Chaves:
   GAUSS 5, GAMMA-RAY, SPECTRA, GE(LI) SPECTROMETERS, GAUSS FUNCTION, LOTAPE CODES, GAUSS 3 CODES.

6. Linguagem de Codificação e Computador:
   FORTRAN IV, IBM 370.

7. Estado Operacional:
   Em uso.

8. Material Disponível:
   a) HELMER, R. G. & PUTNAM, M. H. *GAUSS V: A Computer Program for the Analysis of Gamma-ray Spectra from (Ge(Li) Spectrometers.* Idaho Falls, ID, Aeroject Nuclear Company, January 1972. (ANCR-1043).

   b) Informação em fita:
   IT2116 e IT3106.
   – programa fonte.
   – problema amostra.

   c) Informação em listagem:
   – programa fonte.
   – execução problema amostra.

RELATÓRIO PARA INFORMAÇÃO SOBRE O CÓDIGO "GGC-4"

1. Sigla e Título:
GGC-4.

2. Nome do Autor:
Mahteus, D. R. & Drawbaugh, D. W.

3. Estabelecimento de Origem:
General Atomic Div., General Dynamic Corp., San Diego, CA, (USA).

4. Abstract:
The GGC-4 program solves the multigroup spectrum equations with spatial dependence represented by a single input buckling. Broad group cross sections (shielded or unshielded) are prepared for diffusion and transport codes by averaging with the calculated spectra over input-designated energy limits. The code is divided into three main parts. A fast (GAM) section which covers the energy range from 14.9 mev to 0.414 ev, α thermal (GATHER) section which covers the energy range from 0.001 to 2.38 ev, and a combining (COMBO) section which combines fast and thermal cross sections into single sets. Basic nuclear data for the fast sections which consists of fine group-averaged cross section and resonance parameters is read from a data tape.

5. Palavras Chaves:
GGC-4, SPECTRUM CALCULATIONS, GENERATION OF GROUP CONSTANTS, LATTICE AND CELL PROBLEMS, MULTIGROUP CROSS SECTIONS, AVERAGES, FAST, THERMAL, SPECTRA, DOPPLER BRADENING, DANCOFF CORRECTION, ANGULAR DISTRIBUTION, RESONANCE PARAMETERS, MAKE CODES, MST CODES, PRINT CODES, MIXER CODES, WTFG CODES, MGT3 CODES, SPRINT CODES, COMBIN CODES, GRAND2 CODES.

6. Linguagem de Codificação e Computador:
FORTRAN IV, UNIVAC.

7. Estado Operacional: .

8. Material Disponível:
a) ADIR, J.; CLARK, S. S.; FROEHLICH, R.; TODT, L. J. *User's and Programmer's Manual for the GGC-3 Multigroup Cross Section Code – Part 1 (User's Part).* San Diego, CA, General Atomic, July 1967. (GA-7157).

DRAKE, M. K.; SMITH, C. V.; TODT, L. J. *Description of Auxiliary Codes Used in the Preparation of Data for the GGC-3 Code.* San Diego, CA, General Atomic, Aug. 1967. (GA-7157).

ADIR, J. *GGC-4 code.* San Diego, CA, General Atomic, June 1968. (Memorandum).

b) Informação em fita:
de IPR300 à IPR306, de IT2012 à IT2015, IT2024, IT1004.

B.04

RELATÓRIO PARA INFORMAÇÃO SOBRE O CÓDIGO "GRETEL"

1. Sigla e Título:
   GRETEL.

2. Nome do Autor:
   Guzzi, G. & Cuypers, J.

3. Estabelecimento de Origem:
   European Atomic Energy Community, Ispra (Italy) Joint Nuclear Research Center.

4. Abstract:
A computer program set up for routine batchwise of spectrometric data, is presented.
The program performs the quantitative analysis of gamma-ray spectra obtained by Ge(Li) detectors, using special "oriented libraries", which are prepared for each particular problem.
The computer routines which detect and evaluate peak areas perform the following operations:

- Local smoothing of the spectrum;
- First derivative of the smoothed spectrum;
- Peak location according to the change of sign of the first derivative;
- Computation of the net area of each peak found.

The possibility of detecting and computing also double peaks is one of the features of the program. Results in PPM are obtained in the form of digital list. On request a drawing of the spectrum can also be produced, showing the way in which the spectrum has been processed.

5. Palavras Chaves:
   GAMMA-RAY, SPECTRA, GE(LI) SPECTROMETERS, LI-DRIFTED GE DETECTORS, PROGRAMMING, QUANTITATIVE CHEMICAL ANALYSIS.

6. Linguagem de codificação e computador:
   FORTRAN IV; IBM-370.

7. Material Disponível:
   a) GUZZI, G. & GUYPERS, J. *GRETEL: A Computer Program for Gamma-Ray Spectrometry with Ge(Li) Detectors.* Ispra, Italy, Joint Nuclear Research Center, 1974. (EUR-5117 e).

   b) Informação em fita:
   (IT1064 e IT1065) programa fonte, problema amostra, módulo de carga e outros dados.

   c) Informação em listagem:
   programa fonte, execução problema amostra.

RELATÓRIO PARA INFORMAÇÃO SOBRE O CÓDIGO "HAMMER"

1. Sigla e Título:
   HAMMER, – Heterogeneous Analysis by Multigroup Methods of Exponential and Reactors.

2. Nome do Autor:
   Suich, J. E. & Honeck, H. C.

3. Estabelecimento de Origem:
   Du Pont de Nemours (E.I.) and Co., Aiken, SC (USA). Savannah River Laboratory.

4. Abstract:
   Hammer perfoms infinite lattice, one-dimensional cell multigroup calculations, followed (optionally) by one-dimensional, few-group, multiregion reactor calculations with neutron balance edits. HAMMER combines Thermos (ACC Abstract 184), MUFT, ZUT (ACC Abstract 41), TUZ (ACC Abstract 42), FOG (ACC Abstract 28), plus thermal Fourier transform, epithermal integral transport, space-dependent group fluxes, and neutron balance edits from a single set of input data as well as a library of cell results, compatible reactor/exponential assembly criticality searches from the library.

5. Palavras Chaves:
   HAMMER, SPECTRUM CALCULATIONS, GENERATION OF GROUP CONSTANTS, LATTICE AND CELL PROBLEMS, CELL CALCULATION, 1-DIMENSIONAL, THERMAL, EPITHERMAL, FLUX DISTRIBUTION, LITHE CODES, HELP CODES.

6. Linguagem de Codificação e Computador:
   FORTRAN IV, IBM 360 e IBM 370.

7. Estado Operacional:

8. Material Disponível:
   a) SUICH, J. E. & HONECK, H. C. *The HAMMER System: Heterogeneous Analysis by Multigroup Methods of Exponentials and Reactors.* Aiken, SC, Savannah River Lab., Jan. 1967. (DP-1064).

   b) Informação em fita:
   IT3166 e IT3167.

B.06

RELATÓRIO PARA INFORMAÇÃO SOBRE O CÓDIGO "JANE"

1. Sigla e Título:
   JANE

2. Nome do Autor:
   Schubiger, P. A.; Chakraborty, S.; Wyttenbach, A. & Blaser, W.

3. Estabelecimento de Origem:
   Swiss Federal Institute for Reactor Research, (Switzerland).

4. Abstract:
   A program for large computer (IBM 370/155) applicable to may diferent purpose and equipments is described. The program can do any possible evaluation of gamma-spectra up to a complete identification of an unknown spectra and/or calculations of the amounts if the nuclides concerned by comparison to a standard or by an absolute method.

5. Palavras Chaves:
   GAMMA-RAY, SPECTRA, PROGRAMMING.

6. Linguagem de Codificação e Computador:
   FORTRAN IV; IBM 360/155 e 370.

8. Material Disponível:
   a) SCHUBIGER, P. A.; CHAKRABORTY, S.; WYTTENBACH, A.; BLASER, W. *JANE: An Easily to Handle Computer Program for Different Levels of Qualitative and Quantitative Gamma-Spectra Analysis*. São Paulo, SP, Instituto de Energia Atômica (Relatório datilografado).

   b) Informação em fita (IT2008).
   Programa fonte, biblioteca dados;

   c) Informação em cartão:
   Programa fonte, biblioteca dados;

   d) Informação em listagem:
   Programa fonte, execução amostra.

RELATÓRIO PARA INFORMAÇÃO SOBRE O CÓDIGO "LASER"

1. Sigla e Título:
   LASER.

2. Nome do Autor:
   Poncelet, C. G.

3. Estabelecimento de Origem:
   Westinghouse Electric Corporation, Pittsburgh, PA (USA), Atomic Power Div.,

4. Abstract:
   LASER is a one-dimensional (cylindrical), multi-energy (85 groups) lattice-cell program which is based on the MUFT and THERMOS codes. The thermal cutoff is 1855 ev. A burnup option is provided. The program will, at option, account for the non-linear effects in the burnup equations. The spatial burnup distribution within the fuel rods is explicitly calculated. The output is quite extensive and includes, in addition to the normal edits, an edit for the energy range $0 \leqslant E \leqslant 0.625$ ev. LASER is written in FORTRAN IV for IBM 7090/3094.

5. Palavras Chaves:
   LASER, SPECTRUM CALCULATIONS, GENERATION OF GROUP CONSTANTS, LATTICE AND CELL PROBLEMS, SLOWING DOWN, THERMALIZATION, SPECTRA, DEPLETION, 1-DIMENSIONAL, CYLINDERS, CRITICALITY SEARCHES, LATTICES, LIBP4 CODES, SIG1 CODES.

6. Linguagem de Codificação e Computador:
   FORTRAN IV, IBM 360.

7. Estado Operacional.

8. Material Disponível:
   a) PONCELET, C. G. *Burnup Physics of Heterogeneous Reactor Lattices.* Westinghouse Electric Corp., Pittsburgh, PA, Atomic Power Div., June 1965. (WCAP-6069).

   b) PONCELET, C. G. *LASER: A depletion Program for Lattice Calculations Based on MUFT and THERMOS.* Pittsburgh, PA, Westinghouse Electric Corp., April 1966. (WCAP-6073).

   c) Informação em fita:
   DEN003, DEN005, IT3001.

B.08

RELATÓRIO PARA INFORMAÇÃO SOBRE O CÓDIGO "LEOPARD"

1. Sigla e Título:
   LEOPARD – Lifetime Evaluating Operations Pertinent to the Analysis of Reactor Designs.

2. Nome do Autor:
   Barry, R. F.

3. Estabelecimento de Origem:
   Westinghouse Electric Corporation, Pittsburgh, PA (USA), Atomic Power Div.

4. Abstract:
   LEOPARD is a unit cell homogenization and spectrum generation (MUFT-SOFOCATE) program with a fuel depletion option. The MUFT-SOFOCATE homogeneous medium spectrum analyses with heterogeneous corrections are used. The monoenergetic Amouyal-Benoist thermal disadvantage factor is applied at each of 172 SOFOCATE energy levels up to 0.625 ev. The U238 resonance integral is forced to agree with a generalized Helsstrand correction.

5. Palavras Chaves:
   LEOPARD, SPECTRUM CALCULATIONS, GENERATION OF GROUP CONSTANTS, LATTICE AND CELL PROBLEMS. DEPLETION, CELL CALCULATIONS, SPECTRA, DISADVANTAGE FACTORS, SPOTS CODES.

6. Linguagem de Codificação e Computador:
   FORTRAN II, IBM 370.

7. Estado Operacional:

8. Material Disponível:
   a) BARRY, R. F. *LEOPARD: A Spectrum Dependent Non-Spatial Depletion Code for the IBM 7094.* Pittsburgh, PA, Westinghouse Electric Corp., September 1963. (WCAP-3269-26).

   b) Informação em fita:
   IT3167.

RELATÓRIO PARA INFORMAÇÃO SOBRE O CÓDIGO "PHOEL"

1. Sigla e Título:
   PHOEL.

2. Nome do Autor:
   TURNER, J. E.; MÓDOLO, J. T.; SORDI, G. M. A. A.

3. Estabelecimento de Origem:
   Instituto de Energia Atômica, São Paulo, SP (BR).

4. Abstract:
A Monte Carlo Code, PHOEL, is described for generating the initial energies of photoelectrons and Compton electrons in liquid 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 neglect of pair production, which could be added, does not represent a significant omission for photons wigt energies below about 10.000 kev. This code PHOEL, 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 RBE of low-LET radiations at low doses. The basic numerical data used in PHOEL and their mathematical treatment are described as well as the operation of the code. Some numerical computation which have been made for verification of the code are brief described. The code uses FORTRAN IV language on IBM S/370-155 using a simple overlay structure the program has been using less than 500 K-bytes of storage. It is necessary card reader and disc unite.

5. Palavras Chaves:
   MONTE CARLO, PHOTOELECTRONS, COMPTON, ENERGY DEGRADATION AND TRANSPORT.

6. Linguagem de Codificação e Computador:
   FORTRAN IV level H, IBM/370.

7. Estado Operacional:
   Em uso.

8. Material Disponível:
   a) TURNER, J. E.; MÓDOLO, J. T.; SORDI, G. M. A. A. *PHOEL: A Monte Carlo code for generating initial energies of photoelectrons and compton electrons produced by photons in liquid water.* São Paulo, S.P., Instituto de Energia Atômica, March 1978.

   b) Informação em fita:
      IT2028, IT2035.

   c) Informação em listagem:
      – gravação do módulo de carga.
      – cópia das fitas.
      – execução problema amostra.

B.10

RELATÓRIO PARA INFORMAÇÃO SOBRE O CÓDIGO "RIFF-RAFF"

1. Sigla e Título:
   RIFF-RAFF.

2. Nome do Autor:
   Kier, P. H.

3. Estabelecimento de Origem:
   Argonne National Laboratory, ILL (USA), Reactor Physics Div.

4. Abstract:
   A method of computing resonance absorption by a rod in a two-region cylindricalized cell that is not based on the assumption of an asymptotic neutron source distribution is developed and used to write a FORTRAN program RIFF RAFF (RP 359X), for the CDC-3600. An energy range of interest is divided into extremely narrow intervals of equal lethargy width. For each interval, the source distribution in each region is taken to be a three-term polynomial. By using this form for the source and the assumption that neutrons enter the rod isotropically, the reaction rates and the flux distribution for the interval are obtained. The reaction rates are accumulated over specified energy intervals to obtain resonance integrals; the flux distribution is used to obtain the source distribution for intervals at lower energies.
   RIFF RAFF allows for the materials in each region. There can be four resonance absorbers in the rod, each having 60 resonances. The nonresonance materials can have 1/v absorption cross section, with the restriction that for the moderator, the macroscopic absorption cross section must be small compared to the scattering cross section.

5. Palavras Chaves:
   RIFF/RAFF, SPECTRUM CALCULATIONS, GANERATION OF GROUP CONSTANTS, LATTICE AND CELL PROBLEMS, RESONANCE INTEGRALS, CELL CALCULATIONS, FLUX DISTRIBUTION, ISOTROPIC SCATTERING, FLUX CODES.

6. Linguagem de Codificação e Computador:
   Fortran, CDC-3600.

7. Estado Operacional do Código:

8. Material Disponível:
   a) KIER, P. H. *RIFF RAFF: A Program for Computation of Resonance Integrals in Two-Region Cell.* Argonne, ILL, Argonne National Laboratory, Aug. 1965. (ANL-7033).

RELATÓRIO PARA INFORMAÇÃO SOBRE O CÓDIGO "THERMOS"

1. Sigla e Título do Código:
   THERMOS.

2. Nome do Autor:
   Honeck, H. C.

3. Estabelecimento em que Foi Desenvolvido:
   Argonne National Laboratory, ILL (USA).

4. Resumo:
   THERMOS is written in FORTRAN for the IBM 704/709/7090. The code computer the scalar thermal neutron spectrum as a function of position in a lattice by solving the integral transport equation with isotropic scattering. One dimensional slab or cylindrical geometry can be used. As output the code supplies flux-averaged values of SIGMA A, SIGMA F, NU SIGMA F, SIGMA S, and D for the cell composition and the values of SIGMA A, SIGMA F, NU SIGMA F, SIGMA S, and SIGMA TR for the isotopic constituents. The method used in solving the transport integral equation is the power iteration method. To accelerate convergence the code uses a combination of GAUSS iteration, renormalization over-relaxation, and extrapolation procedures.

5. Palavras Chaves:
   THERMOS, SPECTRUM CALCULATIONS, GENERATION OF GROUP CONSTANTS, LATTICE AND CELL PROBLEMS, THERMAL SPECTRA, TRANSPORT THEORY, SLABS, CYLINDERS, 1-DIMENSIONAL, ISOTROPIC SCATTERING, CROSS SECTIONS, LIBP CODES, GAKER CODES, RLITHE CODES.

6. Linguagem de Codificação e Computador:
   FORTRAN II, IBM 370.

7. Estado Operacional:

8. Material Disponível:
   a) HONECK, H. C. *A Thermalization Transport Theory Code for Reactor Lattice Calculations.* Argonne, ILL, Argonne National Laboratory. (BNL-5826).

   b) Informação em fita:
   IT3167, IT3166, IT3001, DEN003, DEN005.

   c) TOPPEL, B. J. & BAKSYS, I. *The Argonne-Revised Thermos Code.* Argonne, ILL, Argonne National Laboratory, March 1965. (ANL-7023).

B.12

RELATÓRIO PARA INFORMAÇÃO SOBRE O CÓDIGO "XLACS"

1. Sigla e Título:
   XLACS – XSDRN Library of Averaged Cross Sections.

2. Nome do Autor:
   Greene, N. M. & Lucius, J. L. & White, J. E. & Wright, R. Q. & Craven, C. W. & Tobias, M. L.

3. Estabelecimento de Origem:
   Oak Ridge National Laboratory, TN, (USA).

4. Abstract:
   XLACS calculates fine-group average neutron cross section from ENDF/B data. Its primary purpose is to produce full range multigroup libraries for the XSDRN program. Provisions are include for treating fast, resonance, and thermal ENDF/B data. Fine-group energy structures and expansion orders used to represent differential cross sections for XSDRN can be arbitrarily set by the user. Cross section can be averaged over an arbitrary input weighting functions or by any several "build-in" functions.

5. Palavras Chaves:
   XLACS, SPECTRUM CALCULATIONS, NEUTRON CROSS SECTION, MULTIGROUP LIBRARIES, ENDF/B, XSDRN CODES, SAD CODES, MC**2 CODES, FLANGE2 CODES, GENERATION OF GROUP CONSTANTS, LATTICE AND CELL PROBLEMS.

6. Linguagem de Codificação e Computador:
   FORTRAN IV, IBM 360 e 370.

7. Estado Operacional:

8. Material Disponível:
   a) Greene, N. M.; LUCIUS, J. L; WHITE, J. E.; WRIGHT, R. Q.; CRAVEN JR, C. W.; TOBIAS, M. L. *XLACS: A Program to procedure Weighted Multigroups Neutron Cross Sections from ENDF/B* Oak Ridge, TN, Oak Ridge National Laboratory, April 1972. (ORNL-TM-3646).

   b) Informação em fita:
   IT1009 e IT3058.

RELATÓRIO PARA INFORMAÇÃO SOBRE O CÓDIGO "XSDRN"

1. Sigla e Título
   XSDRN, X-Section Dynamic for Reactor Nucleonics.

2. Nome do Autor:
   Greene, N. & CRAVEN, C. W.

3. Estabelecimento de Origem:
   Oak Ridge National Laboratory, TN (USA).

4. Abstract:
   XSDRN is a discrete ordinates spectral averaging code to generate microscopic multigroup cross sections. The code is based on ANISN and concentrates on a basic rod group. It has been designed specifically to generate the microscopic cross section tape for the CITATION code.

5. Palavras Chaves:
   XSDRN, SPECTRUM CALCULATIONS, GENERATION OF GROUP CONSTANTS, LATTICE AND CELL PROBLEMS, TRANSPORT THEORY, MULTIGROUP, 1-DIMENSIONAL, SN METHOD, RESONANCE, SLABS, CYLINDERS, SPHERES, FLUX DISTRIBUTION, ANGULAR DISTRIBUTION, XLACS CODES, ANISN CODES, DOT CODES, CITATION CODES, ROD CODES, EXTERMINATOR2 CODES.

6. Linguagem de Codificação e Computador:
   FORTRAN IV level H, IBM 360/65-75 e IBM 370.

7. Estado Operacional:
   Em uso.

8. Material Disponível:
   a) GREENE, N. M. & CRAVEN, C. W. *XSDRN: A Discrete Ordinates Spectral Averaging Code.* Oak Ridge, TN, Oak Ridge National Laboratory, July 1969. (ORNL-TM-2500).

   b) OAK RIDGE NATIONAL LABORATORY *XSDRN: A Discrete Ordinates Spectral Averaging Code.* Oak Ridge, TN, october 1973. (RSIC. Computer Code Colection, CCC-123).

   c) Informação em fita:
   IT2023, IT3088, DEN009, DEN010 e DEN012.

C. STATIC DESIGN STUDIES

RELATÓRIO PARA INFORMAÇÃO SOBRE O CÓDIGO "ANISN"

1. Sigla e Título:
   ANISN.

2. Nome do Autor:
   Computing Technology Center and Oak Ridge National Laboratory, Union Carbide Corporation, Nuclear Division, Oak Ridge, TN, (USA).

3. Estabelecimento de Origem:
   Oak Ridge National Laboratory, TN, (USA).

4. Abstract:
   ANISN solves the one-dimensional multigroup neutron transport equation in slab, cylindrical and spherical geometry using the SN method. It has a large number of options, including higher order anisotropic scattering and the ability to use any order of SN calculation desired. Boundary conditions include vacuum, reflection, periodic and white/albedo options. A complete shell source option description by group, position and angle is available, and void streaming corrections can be made. Fixed source, keff(*) calculation, concentration search, zone width search outer radius search and buckling search for criticality options are available. Cross-sections can be input on cards or from a tape prepared by a suplementary program, and the code will collapse cross sections to any desired few-group scheme.

5. Palavras Chaves:
   ANISN, STATIC DESIGN STUDIES, DISCRETE ORDINATES, ANISOTROPIC SCATTERING, NEUTRON, GAMMA-RAY, 1-DIMENSIONAL TRANSPORT THEORY, MULTIGROUP, SN METHOD, BOLTZMANN.

6. Linguagem de Codificação e Computador:
   FORTRAN IV; IBM 360.

7. Estado Operacional:
   Em uso com OS/VS-2 da IBM.

8. Material Disponível:
   a) OAK RIDGE NATIONAL LABORATORY. *ANISN-ORNL: Multigroup One-Dimensional Ordinates Tranport Code with Anisotropic Scattering*. Oak Ridge, TN, October 1975. (RSIC Compute Code Collection) (CCC-254).

   SOLTESZ, R. G. & DISNEY, R. K. *One-Dimensional, Discrete Ordinates Transport Technique*. Pittsburgh, PA, Westinghouse Astronuclear Laboratory, Augusto 1970. vol. 4. (WANL-PR-(LL)-034).

   OAK RIDGE NATIONAL LABORATORY. *ANISN: Multigroup One Dimensional Discrete Ordinates Transport Code with Anisotropic Scattering*. Oak Ridge, TN, January 1968. (RSIC Computer Code Collection). (CCC-82).

   b) Informação em fita:
   DEN004, DEN008.

(*) k = fator de multiplicação.
eff = effective.

C.02

RELATÓRIO PARA INFORMAÇÃO SOBRE O CÓDIGO "DOT"

1. Sigla e Título:
   DOT, Discrete Ordinates Transport.

2. Nome do Autor:
   Mynatt, F. R.

3. Estabelecimento de Origem:
   Oak Ridge National Laboratory, TN (USA).

4. Abstract:
   DOT is a FORTRAN IV program which solves the linear, energy dependent, Boltzmann transport equation for two-dimensional R-Z, X-X, and R-Theta geometries. The gradient convection term in the Boltzmann equation is approximated by a finite difference technique known as discrete ordinates or Carlson's SN method. The inscatter integral is approximated by expanding the differential cross sections in a Legendre series which allows the integral to be computed by quadrature. DOT will solve forward or adjoint, homogeneous or inhomogeneous problems. The inhomogeneous problems may have a volume distributed source or a specified angular flux at the right or top boundaries; fissions may be included for subcritical systems. Homogeneous (eigenvalued) problems will determine the following: (a) static multiplication factor, "K", (b) time absorption, "Rossi alpha", (c) concentration for a specified "K", (d) zone thickness for a specified "K".

5. Palavras Chaves:
   STATIC DESIGN STUDIES.

6. Linguagem de Codificação:
   FORTRAN IV, IBM/370.

7. Estado Operacional:

8. Material Disponível:
   a) OAK RIDGE NATIONAL LABORATORY. *DOT: Two Dimensional Discrete Ordinates Transport Code.* Oak Ridge, TN, oct. 1969. (RSIC Computer Code Collections) (CCC-89/K-1694).

   b) Informação em fita:
   DEN004 e DEN008.

RELATÓRIO PARA INFORMAÇÃO SOBRE O CÓDIGO "EXTERMINATOR-2"

1. Sigla e Título:
   EXTERMINATOR-2.

2. Nome do Autor:
   Fowler, T. B.; Tobias, M. L.; Vondy, D. R.

3. Estabelecimento de Origem:
   Oak Ridge National Laboratory, TN (USA), Reactor Division.

4. Abstract:
   The multigroup two-dimensional neutron diffusion equations are solved in X-Y, R-Z or R-Theta geometry. The EQUIPOISE method is employed to solve the finite-difference analogs of the multigroup neutron diffusion equations with certain modifications made to acelerate convergence and stability. Basically the method consists of sweeping the mesh using over or under-relaxation to compute new fluxes from an eigenvalue estimate and the most recently computed fluxes wherever possible. At the end of a sweep of all points and all groups, the eigenvalue is recomputed so that the number of neutrons produced equals the number lost. With this new eigenvalue, a new mesh sweep is started. Either point relaxation or single line relaxation may be used in the solution of a problem. Convergence is accelerated by means of the extrapolated Liebmann process in which the extrapolated Liebmann coefficient is made to vary as necessary during the course of a calculation to obtain reasonable convergence rates. When the convergence rate drops below a certain level, the fluxes are extrapolated by means of the Aitken-Delta squared process. In cell problems and which the groups are not well coupled, a group rebalancing procedure is used to accelerate convergence.

5. Palavras Chaves:
   EXTERMINATOR-2, STATIC DESIGN STUDIES, DIFFUSION EQUATIONS, 2-DIMENSIONAL, MULTIGROUP, X-Y, R-Z, R-THETA, REACTIVITY, FLUX DISTRIBUTION.

6. Linguagem de Codificação e Computador:
   FORTRAN; IBM 360.

7. Estado Operacional:

8. Material Disponível:
   a) FOWLER, T. B.; TOBIAS, M. L.; VONDY, D. R. EXTERMINATOR-2: A FORTRAN IV Code for *Solving Multigroup Neutrons Diffusion Equations in Two Dimensions.* Oak Ridge National Laboratory, April 1967. (ORNL-4078).

   b) Informação em fita:
   DEN001 e DEN002.

C.04

RELATÓRIO PARA INFORMAÇÃO SOBRE O CÓDIGO "MACH 1"

1. Sigla e Título:
   MACH-1.

2. Nome do Autor:
   Meneley, D. A.; Kuitek, L. C.; O'Shea, D. M.

3. Estabelecimento de Origem:
   Argonne National Laboratory, ILL (USA). Reactor Physics Div.

4. Abstract:
   The MACH 1 diffusion-theory package embodies the basic features of the AIM-6 one-dimensional diffusion code, the DEL pertubation code, and the 1188/RE delayed-neutron code, as well as several additional ancillary subroutines. MACH-1 is the result of further modifications of and additions to the MAIM-6 code. The purpose of these modifications and additions was to allow single-step calculation of many reactor parameters of interest, with minimum required input data and at reasonably high speed.
   The main additions to MAIN-6 are the DEL pertubation code, beta-effective and lifetime calculations, inhour-equation solution, expanded reaction summary, group collapsing, point-by-point reaction rates and ratios, card flux dump, external-source calculation, alpha search, and microscopic cross section modification at run time. Transfer may also be made to any subroutine written by a user. This subroutine would then have complete control of the MACH-1 code.

5. Palavras Chaves:
   MACH-1, STATIC DESIGN STUDIES, DIFFUSION EQUATIONS, 1-DIMENSIONAL, MULTIGROUP, SLABS, CYLINDERS, CRITICALITY SEARCHES. REACTION RATES, AVERAGES, PERTUBATION THEORY.

6. Linguagem de Programação e Computador:
   FORTRAN IV e 3600, CDC6500 e CDC3600.

7. Estado Operacional do Código:

8. Material Disponível:
   a) MENELEY, D. A.; KUITEK, L. C.; O'SHEA, D. M. *MACH-1, A One Dimensional Diffusion Theory Package*. Argonne, ILL, Argonne National Laboratory, June 1966. (ANL-7223).

   b) Informação em fita:
   DEN001, DEN002.

D. DEPLETION, FUEL MANAGEMENT, COST ANALYSIS AND REACTOR ECONOMICS

RELATÓRIO PARA INFORMAÇÃO DO CÓDIGO "EREBUS"

1. Sigla e Título:
   EREBUS, Evaluation of Reactor Evalutions with Burn-Up and Searches.

2. Nome do Autor:
   Console, M.; Daneri, A.; Salina, E.

3. Estabelecimento de Origem:
   FIAT, Sexione Energia Nucleare, Turim (Italy).

4. Abstract:
   EREBUS is a burn-up program incorporating the multigroup two-dimensional neutron diffusion code SQUID. The burnup routine makes use of variable fitted self-shielding factors.

5. Palavras Chaves:
   EREBUS, DEPLETION, FUEL MANAGEMENT, COST ANALYSIS, REACTOR ECONOMICS, BURN-UP, NEUTRON DIFFUSION, SELF-SHIELDING FACTORS.

6. Linguagem de Codificação e Computador:
   FORTRAN, IBM 370.

7. Estado Operacional:

8. Material Disponível:
   a) CONSOLE, M.; DANERI, A.; SALINA, E. *EREBUS: A Multigroup Diffusion Depletion Program in Two Dimensions for the IBM 360.* Turin, Italy, Fiat, Sezione de Energia Nucleare, Nov. 1967. (FN-E-88).

   b) Informação em fita:
   IT3167.

D.02

RELATÓRIO PARA INFORMAÇÃO SOBRE O CÓDIGO "GACOST"

1. Sigla e Título:
   GACOST.

2. Nome do Autor:
   Asmussem, K. E. & Brogli, R. H. & Nowicki, S. E. & Southworth, B. W.

3. Estabelecimento de Origem:
   Gulf Radiation Technology, San Diego, CA (USA).

4. Abstract:
   The GACOST code is a very flexible computer program for calculating fuel cycle costs. It had as its origin the fuel cycle cost code PWCOST. The code calculates total fuel cycle coast and running costs on the basic of given massa flows, in-cove resident times, capacity factors, reactor power, ore costs, and fuel handling costs (e.g., enrichment, fabrication, shipping, and reprocessing costs). An extensive edit is provided by the program, including running costs, working capital costs, and total fuel cycle costs on a per segment, per reload interval and levelized basic. Yearly cash flows are also edited, as are the mass flows and fuel burnup (in MWD/MT).
   In addition to fuel cycle cost calculations, GACOST has the capability of calculating fuel fabrication cost based on the concept of learning curves. The code is also equipped to account for the escalation of essentially all components of the fuel cycle.

5. Palavras Chaves:
   GACOST, DEPLETION, FUEL MANAGEMENT, COST ANALYSIS, REACTOR ECONOMICS, FUEL CYCLE, ECONOMICS, PWCOST CODES.

6. Linguagem de Programação e Computador:
   FORTRAN IV.

7. Estado Operacional do Código:

8. Material Disponível:
   a) ASMUSSEN, K. E.; BROGLI, R. H.; NOWICKI, S. E.; SOUTHWORTH, B. W. *A Manual on Fuel Cycle Cost Calculations and a Description of the Code: GACOST.* San Diego, CA, General Atomic Co., Oct. 1972. (GULF-GA-A10593).

   b) Informação em fita:
      IT2048.

RELATÓRIO PARA INFORMAÇÃO SOBRE O CÓDIGO "GARGOYLE"

1. Sigla e Título:
   GARGOYLE.

2. Nome do Autor:
   Todt, F. W.

3. Estabelecimento de Origem:
   General Atomic Div., General Dynamic Corp., San Diego, CA (USA).

4. Abstract:
   GAFFE family of nuclear depletion codes. The initial incentive for developing this code was to completely automate the process of determining feed requirements for partial refuelings during the approach to equilibrium fuel cycle. The resulting code contains many useful options, but is not particularly efficient.
   GARGOYLE employs a flexible nuclide scheme (specified by input) and depletes up to 12 fuel regions (fuel types or stages of irradiation) by subjecting them all to the same average flux. A control poison search may be performed at each time step. Feed material searches may be performed at the end of each burnup cycle before refueling. Any nuclide or combination of nuclides may be included in the feed search. Any nuclide or combination of nuclides may be completely or partially recycled in accordance with input specifications. Concentration-dependent self-shielding factors may be applied to any nuclide.

5. Palavras Chaves:
   GARGOYLE, DEPLETION, FUEL MANAGEMENT, COST ANALYSIS, REACTOR ECONOMICS, FUEL CYCLE, INFINITE MEDIA, CONTROL SEARCHES.

6. Linguagem de Codificação e Computador:
   FORTRAN IV; IBM 360 e 370.

7. Estado Operacional:

8. Material Disponível:
   a) TODT, F. W. *GARGOYLE: An Infinite Medium Fuel Cycle Analysis Code with Fuel and Poison Searches*. San Diego, CA, General Atomics, Feb. 1966. (GA-6622).

   b) Informação em fita:
   DEN003, DEN005, IT3001.

D.04

RELATÓRIO PARA INFORMAÇÃO SOBRE O CÓDIGO "REFCO/POW76"

1. Sigla e Título:
   REFCO/POW76.

2. Nome do Autor:
   Salmon, R.

3. Estabelecimento de Origem:
   Oak Ridge National Laboratory, TN (USA).

4. Abstract:
   REFCO and POW76 are outgrowths of the POWERCO code. All three use the discounted cash flow method. REFCO and POW76 are designed to calculate the fuel cycle cost from the discounted cash flow equation given in ORNL-4116. The interest rate used for discounting is the weighted average interest rate on debt and equity, adjusted downward to account for bond interest. REFCO can also be used to calculate the equilibrium fuel cycle cost of a reactor by supplying input data for a single equilibrium batch.

5. Palavras Chaves:
   REFCO/POW76, DEPLETION, FUEL MANAGEMENT, COST ANALYSIS, REACTOR ECONOMICS, DISCOUNTED CASH FLOW METHOD, POWERCO CODES, FUEL CYCLE.

6. Linguagem de Codificação e Computador:
   FORTRAN IV, IBM 370.

7. Estado Operacional:

8. Material Disponível:
   a) SALMON, R. *Two Computer Codes (REFCO POW76) Calculating the Fuel Cycle Cost of a Nuclear Power Reactor.* Oak Ridge, ILL, Oak Ridge National Laboratory, Aug. 1971. (ORNL-4695).

E. SPACE INDEPENDENT KINETICS

RELATÓRIO PARA INFORMAÇÃO SOBRE O CÓDIGO "BLOOST6"

1. *Sigla e Título:*
   BLOOST6.

2. Nome do Autor:
   Merrill, M. H. & Dahlberg, R. C.

3. Estabelecimento de Origem:
   General Atomic Div., General Dynamics Corp., San Diego, CA (USA).

4. Abstract:
   BLOOST-6 consists of three parts: a space-independent kinetics code, a two-dimensional time-dependent heat transfer code called RAT, and a spherical geometry time-dependent heat transfer code for fuel particles called FUGRA.
   BLOOST-6 operates under the control of the kinetics code, which does a fourth-order Runge-Kutta integration of the reactor kinetic equations using a variable time step based on the speed of the change in the reactor power level.

5. Palavras Chaves:
   BLOOST6, SPACE-INDEPENDENT KINETICS, TEMPERATURE DISTRIBUTION, THERMO DYNAMICS, 2-DIMENSIONAL, R-Z GEOMETRIES, HEAT TRANSFER, FUEL ELEMENTS, RAT CODES, FUGRA CODES, HTGR REACTORS.

6. Linguagem de Programação e Computador:
   FORTRAN IV; UNIVAC 1108.

7. Estado Operacional do Código:

8. Material Disponível.
   a) DAHLBERG, R. C. & MARRIL, M. H. *BLOOST6: Kinetics Codes Containing a Thermodynamic Model od Coated Particles for HTGR Aplications.* San Diego, CA, General Atomic Div., July 1967. (GAMD-8119).

   b) Informação em fita:
      BLOOST6, IT2117, IT1009 e IT3058.

E.02

RELATÓRIO PARA INFORMAÇÃO SOBRE O CÓDIGO "GAPOTKIN"

1. Sigla e Título:
   GAPOTKIN, General Atomics Point Kinetics Code.

2. Nome do Autor:
   Hansen, K. F. & Koch, P. K.

3. Estabelecimento de Origem:
   General Atomics Div., General Dynamics Corp., San Diego, CA (USA).

4. Abstract:
   GAPOTKIN is a point kinetics code that solves the space independent kinetics equations for a very general form of the reactivity function.
   The method of integration of the equations allows rapid operation of the code and varying size time steps. The reactivity is specified in terms of step, linear, and quadratic functions of time plus feedback coefficients proportional to the reactor power and/or total energy release. A unique feature of the energy feedback is the option for including time delays in the reactivity. There is also provision for including time varying external sources.
   The code is sufficiently versatile to handle complex sequences os events such as a sudden loss of coolant followed reactivity insertions due to energy build-up.

5. Palavras Chaves:
   GAPOTKIN, SPACE-INDEPENDENT, KINETICS, FEEDBACK, REACTIVITY.

6. Linguagem de Codificação e Computador:
   FORTRAN IV; IBM 360 e 370.

7. Estado Operacional do Código:

8. Material Disponível:
   a) HANSEN, K. F. & KOCH, P. K. *GAPOTKIN: A Point Kinetics Code for the UNIVAC 1108.* San Diego, CA, General Atomics, oct. 1967. (GA-8204).

   b) Informação em fita:
      DEN001, DEN002, IT3056, TAP112.

G. RADIOLOGICAL SAFETY, HAZARD AN ACCIDENT ANALYSIS

RELATÓRIO PARA INFORMAÇÃO SOBRE O CÓDIGO "ACRA II"

1. Sigla e Título:
   ACRA II.

2. Nome do Autor:
   Stallmann, F. W. & Kam, F. B. K.

3. Estabelecimento de Origem:
   Oak Ridge National Laboratory, TN, (USA).

4. Abstract:
   The ACRA code calculates the internal dose from inhalation of radionuclides in the air and the gamma radiation dose from a passing cloud of finite size. The assumption is made that the radioactive material is released to the atmosphere at ground, the material is transported away by the wind and is dispersed to a 3 dimensional normal distribution. The radioactivity of the material is described throughout the space domain as a function of time.

5. Palavras Chaves:
   ACRA II, RADIOLOGICAL SAFETY, HAZARD AND ACCIDENT ANALYSIS.

6. Linguagem de Codificação e Computador:
   FORTRAN IV; IBM 360.

7. Estado Operacional:

8. Material Disponível:
   a) OAK RIDGE NATIONAL LABORATORY. *ACRA II: Kernel Integration code estimation of radiation doses caused by a hypothetical reactor accident.* Oak Ridge T.N., July 1974. (CCC-213) (RSIC Computer code Collection).

   b) Informação em fita:
   IT1081 e ACRAME, IT1009, IT3058, IT3167.

   c) Informação em Listagem.

G.02

RELATÓRIO PARA INFORMAÇÃO SOBRE O CÓDIGO "ALGAM-97"

1. Sigla e Título:
   ALGAM-97.

2. Nome do Autor:
   Warner, G. C. & Graig Jr., A. M.

3. Estabelecimento de Origem:
   Oak Ridge National Laboratory, TN (USA).

4. Abstract:
   ALGAM computes dose deposited by gamma rays from an external or internal source in various sections of a phantom approximating the body of a man. The body is considered to be made up of three kinds of tissue: soft tissue, bone, and lung. The phantom is divided into a number of sections and each is given a reasonably typical tissue composition. The user can specify a source arbitrarily through a subroutine. Monte Carlo calculations using the OGRE System are employed for particle transport. In addition to options available in OGRE, Woodcock's scheme of the artificial "nothing" cross section is employed to make the total cross section a constant throughout the phanton. Doses are determined from photon energy deposited at each interaction point.

5. Palavras Chaves:
   ALGAM, RADIOLOGICAL SAFETY, HAZARD AND ACCIDENT ANALYSIS, DOSE RATES, MONTE CARLO METHOD, GAMMA-RAYS, PHANTOM MAN.

6. Linguagem de Codificação e Computador:
   FORTRAN-IV, IBM 370.

7. Estado Operacional:
   Em uso, com OS/VS-2 da IBM 370.

8. Material Disponível:
   a) OAK RIDGE NATIONAL LABORATORY. *ALGAM: A Monte Carlo Estimation of Internal Dose from Gamma-ray Sources in a Phantom Man*. Oak Ridge, TN, 1973. (RSIC Computer Code Collection) (CCC-152).

   b) Informação em fita:
      IT3030, BRAZ1.

RELATÓRIO PARA INFORMAÇÃO SOBRE O CÓDIGO "GADOSE/DOSET"

1. Sigla e Título:
   GADOSE/DOSET.

2. Nome do Autor:
   Lee, E. & Mack, R. J. & Sedgley, D. B.

3. Estabelecimento de Origem:
   General Atomic Div., General Dynamics Corp., San Diego, CA (USA).

4. Abstract:
   GADOSE was written specifically for the High Temperature Gas-cooled Reactor (HTGR) type of plant, but no specific accident results are given.
   GADOSE is a FORTRAN program which calculates the radioactivity in various reactor plant locations and the doses to the public resulting from instantaneous accidental release of activity into any of the plant spaces. The calculations can be performed for any combination of isotopes and include consideration of decay, buildup, filtration, atmospheric dilution, fallout, and rainout. The companion program DOSET is also available to include the effects of a time dependent accidental fission product release into the various plant spaces. A typical calculation of the activities and doses at five times and five distances from the plant with 134 isotopes requires approximately 4 centihours of IBM 7044 machine time for the basic GADOSE program.

5. Palavras Chaves:
   GADOSE/DOSET, RADIOLOGICAL SAFETY, HAZARD AND ACCIDENT ANALYSIS, RADIOACTIVITY, DOSE RATES, FISSION PRODUCTS, HTG2 REACTORS, ATMOSPHERE, ISOTOPES, DECAY, PRODUCTION, INSTANTANEOUS RELEASE.

6. Linguagem de Programação e Computador:
   FORTRAN IV; IBM 360.

7. Estado Operacional do Código:

8. Material Disponível:
   a) LEE E.; MACK, R. J.; SEDGLEY, D. B. *GADOSE and DOSET: Programs to Calculate Environmental Consequences of Radioactivity Release*. San Diego, CA, General Atomics Div., April 1966. (GA-6511).

   b) Informação em fita:
   DEN009 e DEN010.

G.04

RELATÓRIO PARA INFORMAÇÃO SOBRE O CÓDIGO "ORIGEN"

1. Sigla e Título:
   ORIGEN, the ORNL Isotope Generation and Depletion Code.

2. Nome do Autor:
   Bell, M. J.

3. Estabelecimento de Origem:
   Oak Ridge National Laboratory, TN, (USA).

4. Abstract:
   The code solves the equations of radioactive growth and decay allowing continuous first order chemical processing and a neutron flux described by a three-region spectrum. Complex decay and transmutation schemes can be treated. An example of a problem which can be solved by ORIGEN is calculating decay heat in discharged fuel sub-assemblies as a function of time from discharge.

5. Palavras Chaves:
   ORIGEN, RADIOLOGICAL SAFETY, HAZARD AND ACCIDENT ANALYSIS, RADIOACTIVE DECAY CHAINS, FISSION PRODUCTS.

6. Linguagem de Codificação e Computador:
   FORTRAN IV, IBM 360 e IBM 370.

7. Estado Operacional:
   Em uso.

8. Material Disponível:
   a) OAK RIDGE NATIONAL LABORATORY. *ORIGEN: Isotope Generation and Depletion Code Matrix Exponential Method.* Oak Ridge, TN, June 1977. (RSIC Computer Code Collection, CCC-217).

   b) Informação em fita:
   (IT3049 e IT3235): programa fonte, problema amostra, bibliotecas nucleares etc.

   c) Informação em listagem:
   programa fonte, execução problema amostra.

RELATÓRIO PARA INFORMAÇÃO SOBRE O CÓDIGO "RELAP4"

1. Sigla e Título:
   RELAP4.

2. Nome do Autor:
   Moore, K. V. & Rettig, W. H.

3. Estabelecimento de Origem:
   Aerojet Nuclear Company, Idaho Falls, ID (USA).

4. Abstract:
   RELAP4 is a computer program, written almost entirely in FORTRAN IV, that was developed primarily to describe the transient behavior of water-cooled nuclear reactor subejected to postulated accident such as those resulting from loss of coolant, pump failure, or nuclear power variations. Since features of the program that describe the nuclear reactor are optional, the program can be applied to experimental water-reactor simulators.

5. Palavras Chaves:
   RELAP4, RADIOLOGICAL SAFETY, HAZARD AND ACCIDENT ANALYSIS, ACCIDENTS, BLOWDOWN, EXCURSIONS, HEAT TRANSFER, HYDRODYNAMICS, POWER PLANTS, REACTOR SAFETY, TRANSIENTS, PLOTR4 CODES.

6. Linguagem de Codificação e Computador:
   FORTRAN IV, IBM 370.

7. Estado Operacional do Código:
   Em uso com OS/VS-2 da IBM 370.

8. Material Disponível:
   a) MOORE, K. V. & RETIG, W. H. *RELAP4: A Computer Program for Transient Thermal-Hydraulic Analysis*. Idaho Falls, ID, Aerojet Nuclear Company. (ANCR-1127).

   b) Informação em fita:
      IT3189.

   c) Informação em listagem.

H. STEADY STATE AND TRANSIENT HEAT TRANSFER

RELATÓRIO PARA INFORMAÇÃO SOBRE O CÓDIGO "BLOOST-5"

1. Sigla e Título:
   BLOOST5.

2. Nome do Autor:
   Merrill, M. H. & Troost, M.

3. Estabelecimento de Origem:
   General Atomic Div., General Dynamics Corp., San Diego, CA, (USA).

4. Abstract:
   It is essentially a combination of an earlier kinetics code called BLOOST-3 and a two-dimensional heat transfer code called RAT, and is applicable to problems for which the space-independent form of the reactor kinetics equation is valid. Considerable flexibility in fuel element geometry is possible in the heat transfer subroutines although experience has only been obtained so far for a few HTGR designs.
   Three types of problems can be run, which will be referred to subsequently as temperature calculations, kinetics calculations, and power table calculations.

5. Palavras Chaves:
   BLOOST5, STEADY-STATE AND TRANSIENT HEAT TRANSFER, SPACE-INDEPENDENT KINETICS, 2-DIMENSIONAL, TEMPERATURE DISTRIBUTION.

6. Linguagem de Codificação e Computador:
   FORTRAN IV, IBM 370.

7. Estado Operacional:

8. Material Disponível:
   a) MERRIL, M. H. & TROOST, M. *BLOOST5: A Combined Reactor Kinetics-Heat Transfer Code for the IBM-7044*. San Diego, CA, General Atomic Div., Aug. 1965. (GAMD-6644).

   b) Informação em fita:
      DEN001, DEN002, IT1009 e IT3058.

H.02

RELATÓRIO PARA INFORMAÇÃO SOBRE O CÓDIGO "ORCENT"

1. Sigla e Título:
   ORCENT – Oak Ridge Cycle Efficiency for Nuclear Turbine,

2. Nome do Autor:
   Bowers, H. I.

3. Estabelecimento de Origem:
   Oak Ridge National Laboratory, TN, (USA).

4. Abstract:
   The program will perform full-load, design point calculations for steam turbine cycle supplied with throttle steam characteristic of contemporary water reactor power plants and will handle both condensing and back-pressure turbine exhaust arrangements. Turbine performance calculations are based on the General Electric Company Method for large steam turbine-generators operating with saturated and low superheat throttle steam.

5. Palavras Chaves:
   ORCENT, AEC SPONSORED, COUPLING, DUAL-PURPOSE PLANT, FEEDWATER HEATING, FLOWSHEETS, HEAT BALANCE, NUCLEAR DESALATION, SINGLE-PURPOSE PLANTS, PROGRAMS COMPUTERS, STEAM CYCLE, TURBINE-GENERATORS.

6. Linguagem de Codificação e Computador:
   FORTRAN IV; IBM 360.

7. Estado Operacional:

8. Material Disponível:
   a) BOWERS, H. I. *ORCENT: A Digital Computer Program for Saturated and Low Superheat Turbine Cycle Analysis.* Oak Ridge, TN, Oak Ridge National Laboratory, January 1969. (ORNL-TM-2395).

I. DEFORMATION AND STRESS DISTRIBUTION COMPUTATIONS; STRUCTURAL ANALYSIS, AND ENGINEERING DESIGN STUDIES

RELATÓRIO PARA INFORMAÇÃO SOBRE O CÓDIGO "SAFE-2D"

1. Sigla e Título:
   SAFE-2D.

2. Nome do Autor:
   Cornell, D. C.

3. Estabelecimento de Origem:
   Gulf General Atomic Incorporated, San Diego, CA, (USA).

4. Abstract:
   SAFE-2D performs the elastic stress analysis of general axisymmetric, plane, and combined axisymmetric and plane composite structures.
   The finite element variational method is used.
   Equilibrium equations are solved by block or tri-diagonalization of the stiffness matrices.

5. Palavras Chaves:
   SAFE-2D, DEFORMATION AND STRESS DISTRIBUTION COMPUTATIONS, STRUCTURAL ANALYSIS, ENGINEERING DESIGN STUDIES, ELASTIC, STRESS, FINITE-ELEMENT, SAFE CODES.

6. Linguagèm de Codificação e Computador:
   FORTRAN IV; UNIVAC 1108.

7. Estado Operacional:

8. Material Disponível:
   a) CORNELL, D. C. *SAFE-2D: A computer Program for the Stress Analysis of Plane and Axisymmetric Composite Structures, A User's Manual*. San Diego, CA, General Atomic Co, Feb. 1969. (GA-9076).

   b) Informação em fita:
   DEN001, DEN002, TAP112.

K. TOTAL SYSTEMS ANALYSIS

RELATÓRIO PARA INFORMAÇÃO SOBRE O CÓDIGO "AMPX-1"

1. Sigla e Título:
   AMPX-1.

2. Nome do Autor:
   Greene, N. M.; Lucius, J. L.; Petrie, L. M.; Ford, W. E.; White, J. E.; Wright, R. Q.

3. Estabelecimento de Origem:
   Oak Ridge National Laboratory, TN, (USA).

4. Abstract:
   AMPX is a modular system for producing coupled multigroup neutron-gamma cross section sets. Basic neutron and gamma cross section data for AMPX are obtained from ENDF/B libraries. Most commonly used operations required to generate and collapse multigroup cross section sets are provided in the system.
   AMPX is flexibly dimensioned, neutron group structures, gamma group structures, expansion order to represent anisotropic processes are all arbitraty and limited only by available computer core and budget.

5. Palavras Chaves:
   AMPX, TOTAL SYSTEM ANALYSIS, MODULAR SYSTEMS, NEUTRONS, GAMMA RAYS, MULTIGROUP CROSS SECTIONS, SN METHOD, RESONANCES, SELF-SHIELDING; ENDF/B.

6. Linguagem de Codificação e Computador:
   FORTRAN IV e ASSEMBLY; IBM 370.

7. Estado Operacional:
   Em uso.

8. Material Disponível:
   a) GREENE, N. M.; LUCIUS, J. L.; PETRIE, L. M.; FORD, W. E.; WHITE, J. E.; WRIGHT, R. Q. *AMPX: A Modular Code System for Generating Coupled Multigroup Neutron-gamma Libraries from ENDF/B.* Oak Ridge, TN, Oak Ridge National Laboratory, March 1976. (ORNL-TM-3706).

   b) Informação em fita (IT3185 e IT2069):
   programa fonte, biblioteca ENDF/B, módulo de carga, outros dados.
   IT3183: secções de choque para 122 elementos com 123 grupos de energia. Contém os mesmos elementos de biblioteca do código XSDRN acrescidos do U-235 processado pelo módulo XLACS a partir de dados do arquivo ENDFB.

   c) Informação em listagem:
   Programa fonte, execuções problema amostra.

   d) Relatório "Instruções para Uso" (preparação).

   e) Informação em cartão:
   bibliotecas usadas com a procedure AMPXLG – (AMPXOVLY: estrutura de overlay; AMPXLIBR; 54 membros de biblioteca.

K.02

RELATÓRIO PARA INFORMAÇÃO SOBRE O CÓDIGO "CITATION"

1. Sigla e Título:
   CITATION.

2. Nome do Autor:
   Fowler, T. B.; Vondy, D. R. e Cunningham, G. W.

3. Estabelecimento de Origem:
   Oak Ridge National Laboratory, TN, (USA).

4. Abstract:
   CITATION is designed to attack the reactor core depletion and dynamics problems in a general sense. The program will solve, in one-, two- or three- dimensional geometry, eigenvalue problems within a finite-difference representation of the diffusion theory aproximation to neutron transport in multigroup form with arbitrary scattering. The eigenvalue problem may involve the determination of the effective multiplication factor, or be a direct criticality search on buckling, on 1/v absorber, or on the concentrations of specified nuclides which may include fuel. The adjoint problem may be solved and perturbation calculations performed.
   Up to three geometric dimensions may be solved and pertubation calculations performed. Up to three geometric dimensions may be considered including X-Y-Z, R-Theta-Z and Hexagonal-Z coordinate systems. CITATION is a FORTRAN IV program designed for the IBM 360/91 with at least 300.000 4-bytes words directly adressable core storage and eight I/O devices, excluding input and output devices and system requeriments.

5. Palavras Chaves:
   CITATION, TOTAL SYSTEM ANALYSIS, 1-DIMENSIONAL, 2-DIMENSIONAL, 3-DIMENSIONAL, MULTIGROUP, DIFFUSION, CRITICALITY SEARCHES, BUCKLING, X–Y–Z, R–THETA, HEXAGONAL, DEPLETION, FUEL MANAGEMENT.

6. Linguagem de Codificação e Computador:
   FORTRAN IV, IBM 370.

7. Estado Operacional:

8. Material Disponível:
   a) FOWLER, T. B.; VONDY, D. R.; CUNNINGHAM, C. W. *Nuclear Reactor Core Analysis Code: CITATION*. Oak Ridge, TN, Oak Ridge National Laboratory, Revision 2, July 1971, Supplement 1, October 1971, Supplement 2, March 1972, Supplement 3, July 1972. (ORNL-TM-2496).

   b) Informação em fita:
      IT3166, IT3105.

M. DATA MANAGEMENT

RELATÓRIO PARA INFORMAÇÃO SOBRE O CÓDIGO "ARC–SYSTEM"

1. Sigla e Título:
   ARC-SYSTEM, Argonne Reactor Computation System.

2. Nome do Autor:
   Just, L. C.; Henryson, II, H.; Kennedy, A. S.; Sparck, S. D.; Toppel, B. J. & Walker, P. M.

3. Estabelcimento de Origem:
   Argonne National Laboratory, ILL, (USA).

4. Abstract:
   ARC-System is the collection of systems subprograms and modules which provide the software environment necessary to execute the modular computational capabilities of the Argonne Reactor Computation (ARC) System on IBM 360.

5. Palavras Chaves:
   ARC–SYSTEM, DATA MANAGEMENT, MODULAR SYSTEM, DATA PROCESSING, ARC CODES.

6. Linguagem de Codificação e Computador:
   FORTRAN IV, ASSEMBLER e MACRO definições; IBM 360.

7. Estado Operacional:
   Em uso com sistema OS/VS-2 da IBM 370.

8. Material Disponível:
   a) JUST, L. C.; HENRYSON, H.; KENNEDY, A. S.; SPARCK, S. D.; TOPPEL, B. J.; WALKER, P. M. *The system Aspect and Interface Data Sets of the Argonne Reactor Computation (ARC) System.* Argonne, ILL, Argonne National Laboratory, April 1971. (ANL-7711).

   b) Informação em fita:
   NMG-01, IT3185.

   c) Informação em listagem.

P. GENERAL MATHEMATICAL AND COMPUTING

RELATÓRIO PARA INFORMAÇÃO SOBRE O CÓDIGO "ASAAM23"

1. Sigla e Título:
   ASAAM23 – Simulation, Analysis and Modeling.

2. Nome do Autor:
   M. K., Angell.

3. Estabelecimento de Origem:
   Departament of Biomathematics at Cornell University Medical College.

4. Abstract:
   ASAAM23 is an Abreviated version of the large, multipurpose SAAM23 program written by Dr. Mones Berman and Mrs. Marjory F. Weiss at the National Institute of Health Bethesda, Maryland. SAM is a digital computer program developed for the analysis of data in terms of models. It permits simulation and data fitting and contains various techniques encountered in model building.

5. Palavras Chaves:
   ANALYSIS, MODELING, SIMULATION, FITTING.

6. Linguagem de Codificação e Computador:
   FORTRAN IV; IBM 360 e 370.

7. Material Disponível:
   a) BERMAN, M. & WEISS, M. F. *User's manual for SAAM 23.* Bethesda, M. D., National Institute of Arthristis an Metabolic Diseases, may 1966. (SAAM 23).

   b) Informação em fita:
   (IT2029 e IT2004) programa fonte e módulo de carga.

   c) Informação em listagem:
   Programa fonte.

   d) Relatório "Instrução para Uso" (sendo preparado).

P.02

RELATÓRIO PARA INFORMAÇÃO SOBRE O CÓDIGO "BEVINGTO"

1. Sigla e Título:
   BEVINGTO.

2. Nome do Autor:
   MÓDOLO, J. T.

3. Estabelecimento de Origem:
   Instituto de Energia Atômica, São Paulo, SP. (BR). Centro de Processamento de Dados.

4. Abstract:
   O código BEVINGTO é um sistema que envolve técnicas de ajuste de funções, não linear nos parâmetros que as compõem, a pares de pontos experimentais, usando método e mínimos quadrados não linear, com modificações sugeridas por BEVINGTON.
   As funções devem satisfazer as condições do princípio de mínimos quadrados; sendo definidas pelo usuário no momento da execução, bem como as derivadas parciais em relação aos parâmetros em cada ponto experimental.

5. Palavras Chaves:
   NON LINEAR LEAST-SQUARES FIT.

6. Linguagem de Codificação e Computador:
   FORTRAN IV level H, IBM/370.

7. Estado Operacional:
   Em uso.

8. Material Disponível:
   a) MÓDOLO, J. T. *BEVINGTO: Ajuste de funções usando métodos de mínimos quadrados não linear.* São Paulo, SP, Instituto de Energia Atômica, Janeiro 1978.

   b) Informação em fita:
      IT2028, IT2035.

   c) Informação em listagem:
      – gravação do módulo de carga.
      – cópia das fitas.
      – execução do problema amostra.

RELATÓRIO PARA INFORMAÇÃO SOBRE O CÓDIGO "MATEXP"

1. Sigla e Título:
   MATEXP, Matrix Exponential.

2. Nome do Autor:
   Ball, S. J. & Adams, R. K.

3. Estabelecimento de Origem:
   Oak Ridge National Laboratory, TN, (USA).

4. Abstract:
   MATEXP has several advantages over standard numerical integration routines. It gives virtually exact solutions to constant-coefficient homogeneous equations and to nonhomogeneous equations for which the forcing functions are constant during the computation interval. The speed at which the equations are solved and the accuracy of the solution are essentially unaffected either by the degree of cross coupling of the equations or by whether or not the coefficient matrix is nonsingular or that its eigenvalues are distinct.
   The method has been extended to nonlinear equations and equations with time-varying coefficients; this use is very effective for engineering systems analysis problems.

5. Palavras Chaves:
   MATEXP, GENERAL MATHEMATICAL AND COMPUTING SYSTEM ROUTINES, DIFFERENTIAL EQUATIONS, HOMOGENEOUS.

6. Linguagem de Codificação e Computador:
   FORTRAN IV; IBM 360 e IBM 370.

7. Estado Operacional:
   Em uso com OS/VS-2 do IBM 370.

8. Material Disponível:
   a) BALL, S. J. & ADAMS, R. K. *MATEXP: A General Purpose Digital Computer Program for Solving Ordinary Differential Equations by the Exponential Method.* Oak Ridge, TN, Oak Ridge National Laboratory, August 1967. (ORNL-TM-1933).

   b) Informação em listagem:
      programa fonte, problema amostra.

   c) Informação em cartão:
      programa fonte.

P.04

RELATÓRIO PARA INFORMAÇÃO SOBRE O CÓDIGO "SAAM25"

1. Sigla e Título:
   SAAM25 – Simulation, Analysis and Modeling.

2. Nome do Autor:
   Mones Berman and Marjory F. Weiss.

3. Estabelecimento de Origem:
   National Institute of Health, Bethesda, MD (USA).

4. Abstract:
   SAAM25 is a version of program SAAM.
   SAAM is a digital computer program developed for the analysis of data in terms of models. It permits simulation and data fitting and contains various techniques encountered in model building. Although developed primarily for biological systems and more specifically for kinetics models, the program is of general utility. It differs from other simulation and data fitting program in that its "language" is geared to the bio-medical "system" investigator, and its elements and computational procedures are counterparts of conceptualizations and experimental methodologies employed by the investigator.

5. Palavras Chaves:
   SIMULATOR, ANALYSIS, MODELING, FITTING.

6. Linguagem de Codificação e Computador:
   FORTRAN IV, IBM 360 e 370.

7. Material Disponível:
   a) BERMAN, M. & WEISS, M. F. *User's manual for SAAM, version SAAM 25.* Bethesda, M.D., National Institute of Arthrists and Metabolic Diseases, Dec. 1971. (Relatório datilografado) (SAAM 25).

   b) Informação em fita:
   (IT2029 e IT2004) Módulo de carga.

   c) Informação em listagem:
   Execução de problemas amostra.

   d) Relatórios "Instrução Para Uso" (sendo preparado).

RELATÓRIO PARA INFORMAÇÃO SOBRE O CÓDIGO "SAS"

1. Sigla e Título:
   SAS – Statistical Analysis System.

2. Nome do Autor:
   SAS, Institute, Inc.

3. Estabelecimento de Origem:
   SAS Institute, Ic., Raleigh, NC, (USA).

4. Abstract:
   SAS in an integrated system for data management and statistical analysis. Highlighting SAS's statistical capabilities are its versatile least squares procedures, which produces a wide variety of linear and non-linear regression analyses, of variance and covariance and multivariet analyses of variance.
   SAS can also produce multiple and partial correlation coefficients. It can perform discriminant analyses, factor analyses, and cluster analyses.

5. Palavras Chaves:
   SAS, GENERAL MATHEMATICAL AND COMPUTING SYSTEM ROUTINES, DATA MANAGEMENT, STATISTICS, LEAST SQUARES.

6. Linguagem de Codificação e Computador:
   IBM 370.

7. Estado Operacional:
   Em uso com OS/VS-2 da IBM 370.

8. Material Disponível:
   a) Informação em fita
      IT2046.

   b) BARR, J. A.; GOODNIGHT, J. H.; SALL, J. P.; HELWIG, J. T. *SAS Programmer's Guide.* Raleigh, NC, SAS Institute, INC., Sept. 1977.

      HELWIG, J. T., ed. *SAS Supplemental Library User's Guide.* Raleigh, NC, SAS Institute, Inc., July 1977.

      BARR, J. A.; GOODNIGHT, J. H.; SALL, J. P.; HELWIG, J. T. *A User's Guide to SAS 76.* Raleigh, NC, SAS Institute, Inc., July 1977.

Z. NUCLEAR DATA

RELATÓRIO PARA INFORMAÇÃO SOBRE O CÓDIGO "MEDLIST"

1. Sigla e Título:
   MEDLIST.

2. Nome do Autor:
   Ewbank, W. B. & Martin, M. J.

3. Estabelecimento de Origem:
   Oak Ridge National Laboratory, TN, (USA).

4. Abstract:
   The source of the data is the Evaluated Nuclear Structure File (ENSDF) of the ORNL Nuclear Data Project. The initial include nearly 200 nuclides from ^{3}H to ^{239}Np. Listed are recommended values for half-lives, energies, and intensities (probabilities per decay) for each of the atomic and nuclear radiations emitted by the radioactive nuclides in a format similar to ENDF/B-V.

5. Palavras Chaves:
   MEDLIST, DATA LIBRARY, ENDF/B-V, RADIOACTIVE NUCLIDES.

6. Linguagem de Codificação e Computador:
   FORTRAN IV; IBM S/370.

7. Estado Operacional:
   Em uso.

8. Material Disponível:
   a) OAK RIDGE NATIONAL LABORATORY. *MEDLIST: radionuclide radiation data from the evaluated nuclear structure data file (ENSDF) for interest in medical, health physics, nuclear power, environmental impact, and Industrial applications.* Oak Ridge; TN, March 1977. (DLC-46). (RSIC Data Library Collections).

   b) Informação em fita:
   (IT1081, (ACRAME).

   c) Informação em listagem.

Z.02

RELATÓRIO PARA INFORMAÇÃO SOBRE O CÓDIGO "ZZ DLC-2D"

1. Sigla e Título:
   ZZ DLC-2D.

2. Nome do Autor:
   Wright, R. Q.

3. Estabelecimento de Origem:
   Oak Ridge National Laboratory, TN, (USA).

4. Abstract:
   Neutron transport calculations can be performed with DLC-2 data. Since the data are intended for use in multigroup discrete-ordinates or Monte Carlo transport codes which treat anisotropic scattering, possible cross section angular expansion is limited only by the options available in the particular code used. Specifically, the retrieval program manipulates DLC-2 such that it conforms to input requirements of the CCC-82/ANISN, CCC-89/DOT, CCC-42/DIF-IV codes, or any computer codes using data in the ANISN or DIF-IV format.
   There are two retrieval programs packaged with DLC-2D, DLC2RP and APRFX-I:
   DLC2RP will retrieve DLC-2 data from a maximum of 46 data sets and merge these data into one data set. The program will then, by input option, edit the data, punch cards in either the ANISN or DTF-IV format, or write an unformatted tape for use by ANISN.
   APRFX-I collapses the fine group cross sections to a broad group structure according to a flux spectrum either input by the user or calculate by the code. The code will average the fine group cross sections to form either macroscopic or microscopic isotope cross sections and any combination of macroscopic mixtures of these cross sections on the same problem. It also determines the broad group input source and generates averaged neutron velocities for use with transport calculations.

5. Palavras Chaves:
   ZZ DLC-2D, NUCLEAR DATA, NEUTRON TRANSPORT, MULTIGROUP CROSS SECTIONS, DATA LIBRARIES.

6. Linguagem de Codificação e Computador:

7. Estado Operacional:

8. Material Disponível:
   a) OAK RIDGE NATIONAL LABORATORY. *100G: 100 Groups Neutron Cross-Section Data Bases on ENDF/B*. Oak Ridge, TN, July 1972. (DLC2RP e APRFX-I Data Retrieval Programs). (RSIC Data Library Collection). (DLC-2).

   b) Informação em fita:
   DEN004, DEN008.

ABSTRACT

This paper is a listing of the nuclear codes available at the data processing center of IEA (CPD–IEA). This compilation is divided in four parts: 1) alphabetic list of the codes; 2) Subject classification; 3) alphabetic index and 4) abstracts.

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