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COMPILAÇÃO DOS RESUMOS DOS CÓDIGOS NUCLEARES DISPONÍVEIS NO CPD/IPEN

Ana Granzotto, Antonio Soares de Gouvêa e Edna Maria Lourenção

INFORMAÇÃO IPEN 7 IPEN - Inf - 7	JUNHO/1981

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INFORMAÇÃO IPEN 7
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CENTRO DE PROCESSAMENTO DE DADOS
ÁREA DE PESQUISAS

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SÃO PAULO - BRASIL

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SUMÁRIO

COMPILAÇÃO DOS RESUMOS DOS CÓDIGOS NUCLEARES DISPONÍVEIS NO CPD/IPEN

Ana Granzotto[1], Antonio Soares de Gouvêa[2] e Edna Maria Lourenção[3]

0 – INTRODUÇÃO

A publicação IEA-INF-67 tornou-se obsoleta com a edição da 'Compilação dos Resumos dos Códigos Nucleares Disponíveis no CPD/IPEN'.

Nesta nova versão incluiram-se novos códigos e os resumos se encontram na lingua original do trabalho.

Os códigos foram classificados de acordo com a categoria atribuida na publicação 'Nuclear Program Abstracts', de Novembro de 1980, proveniente da Nuclear Energy Agency.

As palavras-chaves se encontram em lingua inglesa para facilitar o usuário na procura de bibliografia correlata e foram estabelecidas de acordo com o 'Thesaurus ERDA Rev. 3'.

Observe-se que as referências arroladas nos códigos correspondem apenas aos relatórios 'full-size' disponíveis no CPD. A bibliografia exaustiva de cada código se encontra armazenada em relatórios microficha e 'full-size' na Divisão de Informação e Documentação Científica, do IPEN.

1 – NOTAS EXPLICATIVAS

Esta compilação dos resumos dos códigos disponíveis no Centro de Processamento de Dados do IPEN, é o início da Biblioteca de Programas.

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 dos 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. Resumos - reune todos os relatórios de informação dos códigos disponíveis no C.P.D.

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(1) Bibliotecária do Departamento de Informação e Documentação Científica, lotada no Centro de Processamento de Dados do Instituto de Pesquisas Energéticas e Nucleares.
(2) Analista de Sistema – Supervisor do Centro de Processamento de Dados do IPEN.
(3) Analista de Sistemas-Júnior do Centro de Processamento de Dados do IPEN.

Os índices Alfabéticos de Assuntos e Alfabéticos de Códigos disponíveis no C.P.D., remetem para o resumo que fornece informações relativas a cada código.

Encontra-se armazenado 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.

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.

2 – ÍNDICE ALFABÉTICO DOS CÓDIGOS DISPONÍVEIS NO CPD/IPEN

TÍTULO	CÓDIGO
ACRAII	IPEN027
ALGAM-97	IPEN028
AMPX-1	IPEN048
ANACROM	IPEN054
ANISN	IPEN013
ARC-SYSTEM	IPEN060
ASAAM23	IPEN059
BEVINGTO	IPEN080
BLOOST-5	IPEN033
BLOOST-6	IPEN023
BRT-1	IPEN002
CELULAR	IPEN070
CHECK4	IPEN051
CHETAH	IPEN067
CITATION	IPEN049
COBRA 3C	IPEN034
COBRA 3M	IPEN035
CONTEMPT-LT	IPEN029
CRECT	IPEN061
CSMP	IPEN081
CSMPTEK	IPEN076
DALP	IPEN055
DICT4	IPEN051
DISCOMP	IPEN025
DOMINO	IPEN046
DOT II	IPEN014
DOT-3.5	IPEN015
ENDF/B IV	IPEN072
ETCHICAN	IPEN036
ETOE	IPEN052
EXPANDA-70	IPEN016
EXTERMINATOR-2	IPEN017
FEAST1-65	IPEN041
GACOST	IPEN020
GADOSE/DOSET	IPEN030
GAPOTKIN	IPEN024
GARGOYLE	IPEN021
GARLA	IPEN071
GAUSS 5	IPEN003
GGC-4	IPEN004
GRETEL	IPEN056
HAMMER	IPEN005
JAERI FAST SET	IPEN073

JANE	IPEN006
JFUSER	IPEN053
LASER	IPEN007
LEOPARD	IPEN008
LISTFC	IPEN051
LOAD-FLOW	IPEN068
LTFR-70	IPEN053
MACABRE-II	IPEN037
MACH-1	IPEN018
MARS	IPEN042
MATEXP	IPEN062
MC^2	IPEN009
MDCN	IPEN022
MEDLIST	IPEN074
MINITAB	IPEN063
MORSE	IPEN047
NATOF-2D3	IPEN038
NBS-576	IPEN057
NONSAP	IPEN043
ORIGEN	IPEN031
O5S	IPEN058
PHOEL	IPEN010
PV2MRD	IPEN044
QX1	IPEN026
RELAP 4	IPEN032
RESEND	IPEN001
RIGEL 4	IPEN051
SAAM25	IPEN064
SAFE-2D	IPEN045
SAND-2	IPEN019
SAS	IPEN065
SFR3	IPEN066
SOLMNEQ	IPEN069
STATIS	IPEN077
TEMAX	IPEN039
TOODEE-2	IPEN040
XLACS	IPEN011
XSDRN	IPEN012
ZZ-DLC-2D	IPEN076

3 – CLASSIFICAÇÃO DOS PROGRAMAS

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

A. Cálculos de Seções de Choque e Integrais de Ressonância.

Cálculo de seções de choque de reação a partir da teoria nuclear, tais como modelos óticos ou Hauser-Feshbach, seções de choque de ressonância pela teoria multigrupo ou Breit-Wigner, determinação de seções de choque de espalhamento diferencial, avaliação de seções de choque e compilação de programas.

B. Cálculos de Espectro, Geração de Constantes de Grupo, Problemas de Célula e Malha.

Determinação da densidade de moderação do espectro térmico, ponderação e avaliação de seções de choque e quantidades correlatas para a produção de constantes de grupo e avaliação de parâmetros de projetos para cálculo de célula e de malha.

C. Estudos para Projetos Estáticos.

Cálculo de reatividade e distribuição de fluxo do sistema do reator e ajuste dos parâmetros do projeto para as especificações formuladas, isto é, procedimentos para pesquisa de criticalidade e distribuição de potência.

D. Depleção, Gerência de Combustível, Análise de Custos e Economia de Reatores.

Inclui programas de queima, aparecimento de isótopos e produtos de fissão e cálculos de decaimento, e estudos de otimização.

E. Cinética Independente do Espaço.

Estudos do comportamento temporal dos reatores, incluindo efeitos de neutrons atrasados e mecanismos de realimentação e avaliação da função transferência.

F. Cinética Temporal e Espacial, Neutrônica Acoplada, e Simulações Hidrodinâmicas e Termodinâmicas.

Programas que consideram características espaciais de projeto e os efeitos que acompanham o estudo do comportamento temporal do reator.

G. Proteção Radiológica, Análise de Acidentes e Riscos.

Cálculo de taxas de dose interna e externa, determinação das propriedades termodinâmicas de um reator que acompanham um acidente, por exemplo, liberação de materiais radioativos, queda do sistema de refrigeração, ruptura do gerador de vapor.

H. Transferência de Calor Estacionário e Transiente.

Inclui estudos de escoamento de fluidos e cálculos de propriedades termodinâmicas.

I. Cálculos de Distribuição de Tensões e Deformações, Análise Estrutural e Estudos de Projetos de Engenharia.

Inclui avaliações do projeto do elemento combustível, estudos de configuração do núcleo, e análise de estruturas compostas.

J. Aquecimento Gama e Programas para Projeto de Blindagem.

Cálculos de taxas de geração de calor, cálculos de fuga e análise de penetração para blindagem de reatores.

K. Análise Global de Sistemas.

Conjuntos de soluções de problemas correlatos tirados de diversas categorias, projetados e utilizados como sistemas.

L. Preparação de Dados.

Geração de parâmetros de programas; verificação, impressão e colocação, em formato adequado, das informações de entrada do programa.

M. Gerência de Dados.

Confecção, manutenção e recuperação de arquivos de dados, por exemplo, bibliotecas de seções de choque.

N. Cálculos Auxiliares.

Traçado de gráficos, impressão e apresentação de rotinas que processam dados de saída de outros programas.

O. Processamento de Dados Experimentais.

Programas projetados para processar dados coletados diretamente de uma situação experimental para auxiliar o pesquisador na montagem de uma experiência.

P. Matemática e Computação em Geral.

Cálculos de funções matemáticas e rotinas com linguagem especial com possibilidade de processamento de dados gerais.

Q. Materiais.

Medidas e cálculos de propriedades físicas e mecânicas de materiais, simulação de processos de danos por radiação, estudos de corrosão, e determinação de funções cristalográficas.

R. Ciência da Terra e do Meio Ambiente.

Estudos de impacto no meio ambiente, cálculos de geologia, sismologia e geofísica, estudos de hidrologia e água subterrânea, análises de sistemas bioambientais, cálculos meteorológicos relacionados à atmosfera e seus fenômenos, estudos de material transportado pelo ar, climatologia, etc.

S. Ciências Espaciais.

Análise de órbitas e trajetórias, cálculos de astronomia e astrofísica, estudos de propagação de ondas, e cálculos de parâmetros de reentrada.

T. Equipamento de Engenharia e Eletrônica.

Projeto de equipamento eletrônico automatizado, cálculos de engenharia para máquinas, ferramentas controladas numericamente e programas para controle de processos.

U. Química

Análise química, espectroscopia de massa, química das radiações, estudos de radiólise, etc.

V. Aceleradores de Partículas e Máquinas de Alta Voltagem.

Programas relacionados ao projeto, desenvolvimento e operação de máquinas de alta voltagem e aceleradores de partículas, tais como geradores Van der Graaf, aceleradores lineares, cíclotrons, síncIotrons, etc.

W. Física.

Cálculos relacionados a teoria de propriedades da estrutura atômica ou molecular, estudos de colisão de partículas carregadas que envolvem fenômenos tais como troca de carga, excitação, ionização, dissociação, etc., teorias e modelos de partículas elementares, estudos de eletrodinâmica quântica, teoria de espalhamento, teoria quântica de campo, cálculos da teoria geral da relatividade e gravitação.

X. Pesquisa Termonuclear Controlada.

Fenômeno de descarga elétrica e cálculos de física de plasmas, estudos de eletrodinâmica e magnetohidrodinâmica.

Y. Bilogia e Medicina.

Estudos biológicos, médicos e radiológicos de estrutura, funções, química, biofísica, reprodução e hereditariedade de bactérias, plantas, animais de laboratórios e sêres humanos.

Z. Dados Nucleares.

Dados preparados em programas, com formatos especificados para teste e avaliações de programas, estudos de problemas - gabarito , ou uso em bibliotecas.

4 – ÍNDICE ALFABÉTICO DE CLASSIFICAÇÃO DE ASSUNTO

A. Cálculos de Seções de Choque e Integrais de Ressonância.

RESEND	IPEN001
STATIS	IPEN077

B. Cálculos de espectro, geração de constantes de grupos, problemas de célula e malha.

BRT-1	IPEN002
EXPANDA-70	IPEN016
GGC-4	IPEN004
HAMMER	IPEN005
LASER	IPEN007
LEOPARD	IPEN008
MC^2	IPEN009
XLACS	IPEN011
XSDRN	IPEN012

C. Estudos para Projetos Estáticos.

ANISN	IPEN013
ARC-SYSTEM	IPEN050
CITATION	IPEN049
DOT II	IPEN014
DOT-3.5	IPEN015
EXPANDA-70	IPEN016
EXTERMINATOR-2	IPEN017
HAMMER	IPEN005
MACH-1	IPEN018

D. Depleção, Gerência de Combustível, Análise de Custos e Economia de Reatores.

CITATION	IPEN049
GACOST	IPEN020
GARGOYLE	IPEN021
LASER	IPEN007
MDCN	IPEN022

E. Cinética Independente do Espaço.

BLOOST-6	IPEN023
GAPOTKIN	IPEN024

F. Cinética Temporal e Espacial, Neutrônica Acoplada e Simulações Hidrodinâmicas e Termodinâmicas.

CITATION	IPEN049
QX1	IPEN026

G. Proteção Radiológica, Análise de Acidentes e Riscos.

ACRA-II	IPEN027
ALGAM-97	IPEN028
COBRA-3C	IPEN034
CONTEMPT-LT	IPEN029
GADOSE/DOSET	IPEN030
ORIGEN	IPEN031
RELAP4	IPEN032

H. Transferência de Calor Estacionário e Transiente.

BLOOST-5	IPEN033
BLOOST-6	IPEN023
COBRA-3C	IPEN034
COBRA-3M	IPEN035
DISCOMP	IPEN025
ETCHICAN	IPEN036
MACABRE II	IPEN037
NATOF-2D3	IPEN038
RELAP4	IPEN032
TEMAX	IPEN039
TOODEE	IPEN040

I. Cálculo de Distribuição de Tensões e Deformações, Análise Estrutural e Estudos de Projetos de Engenharia

FEAST1-65	IPEN041
MARS	IPEN042
NONSAP	IPEN043
PV2MRD	IPEN044
SAFE-2D	IPEN045

J. Aquecimento Gama e Programas para Projetos de Blindagem.

DOMINO	IPEN046
MORSE	IPEN047

K. Análise Global de Sistemas.

AMPX-1	IPEN048
ARC-SYSTEM	IPEN050
CITATION	IPEN049

M. Gerência de Dados.

CHECK4	IPEN051
CRECT	IPEN051
DICT4	IPEN051
ETOE	IPEN052

JFUSER	IPEN053
LISTFC	IPEN051
LTFR-70	IPEN053
RIGEL4	IPEN051

O. Processamento de Dados Experimentais.

ANACROM	IPEN054
DALP	IPEN055
GAUSS 5	IPEN003
GRETEL	IPEN056
JANE	IPEN006
NBS-576	IPEN057
O5S	IPEN058
SAND-2	IPEN019

P. Matemática e Computação em Geral.

ASSAM23	IPEN059
BEVINGTO	IPEN060
CSMP	IPEN061
CSMPTEK	IPEN076
MATEXP	IPEN062
MINITAB	IPEN063
SAAM25	IPEN064
SAS	IPEN065
SFR-3	IPEN066

T. Equipamento de Engenharia e Eletrônica.

LOAD-FLOW	IPEN068

U. Química

CHETAH	IPEN067
SOLMNEQ	IPEN069

W. Física

CELULAR	IPEN070
PHOEL	IPEN010

Y. Biologia e Medicina

GARLA	IPEN071

Z. Dados Nucleares.

ENDF/B-IV	IPEN072
JAERI FAST SET	IPEN073
MEDLIST	IPEN074
ZZ-DLC-2D	IPEN075

5 – RESUMOS

1. Título
   RESEND

2. Nome do Autor
   Bhat, M. R. & Ozer, O.

3. Estabelecimento de Origem
   Brookhaven National Laboratory, New York, NY, (USA). National Nuclear Data Center.

4. Resumo
   RESEND generates infinitely dilute, unbroadened point cross sections in the ENDF format by combining ENDF file 3, background cross sections, with points calculated from ENDF file 2, resonance parameter data.

5. Palavras Chaves
   CROSS SECTIONS, RESONANCE INTEGRALS, DOPPLER BROADENING.

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

7. Categoria
   A

8. Referência
   OZER, O. *Program RESEND.* New York, NY, Brookhaven National Laboratory, 1972. (BNL-17134)

   BHAT, M. R. *Processing codes for the resonance region.* New York, NY, Brookhaven National Laboratory, 1971. (BNL-50296)

IPEN002

1. 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. Resumo
   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 interation, 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
   TRANSPORT THEORY, SLABS, CYLINDERS, CROSS SECTIONS, NEUTRON DENSITY, ONE-DIMENSIONAL CALCULATIONS.

6. Linguagem de Codificação e Computador
   FORTRAN IV.

7. Categoria
   B

8. Referência
   BENNETT, C. L. & PURCELL, W. L. *BRT-1: Battelle Revised Thermos.* Richland, WA, Battelle Nortwest, June 1970. (BNWL - 1434)

1. 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. Resumo
   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
   SPECTRA, SPECTROMETERS, GAUSS FUNCTIONS.

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

7. Categoria
   O

8. Referência
   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, Jan. 1972 (ANCR-1043).

IPEN004

1. Título
   GGC-4

2. Nome do Autor
   Matheus, D. R. & Drawbaugh, D. W.

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

4. Resumo

   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 sections and resonance parameters is read from a data tape.

5. Palavras Chaves
   DOPPLER BROADENING, ANGULAR DISTRIBUTION, MULTIGROUP THEORY, FAST NEUTRONS, THERMAL NEUTRONS, CROSS SECTIONS, GROUP CONSTANTS, DANCOFF CORRECTIONS.

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

7. Categoria
   B

8. Referência

   ADIR, J.; CLARK, S. S.; FROEHLICH, R.; TODT, L. J. *User's and programmer's manual for the GGC-3 multigroup cross sections code part 1 (user's part).* San Diego, CA, General Atomic, July 1967. (GA-7157, I and II).

   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)

   OAK RIDGE NATIONAL LABORATORY. *GGC-3: multigroup cross sections code diffusion and transport calculations.* Oak Ridge, TN. (GA-7156) (RSIC Computer Code Collection)

1. 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. Resumo
   HAMMER performs infinite lattice, one-dimensional cell multigroup calculations, followed (optionally) by one-dimensional, few-group, multiregion reactor calculations with neutron balance edits. HAMMER combines THERMOS, MUFT, ZUT, TUZ, FOG, 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
   GROUP CONSTANTS, REACTOR CELLS, NEUTRON FLUX, MULTIGROUP THEORY.

6. Liguagem de Codificação e Computador
   FORTRAN IV; IBM/360 e IBM/370

7. Categoria
   C, B

8. Referência
   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).

IPEN006

1. 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. Resumo
   A program for large computer (IBM 370/155) applicable to many different purpose and equipments is described. The program can do any possible evaluation of gamma-ray 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
   QUANTITATIVE CHEMICAL ANALYSIS, QUALITATIVE CHEMICAL ANALYSIS SPECTROMETERS.

6. Linguagem de Cofificação e Computador
   FORTRAN IV; IBM/360 e IBM/370

7. Categoria
   O

8. Referência
   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).

1. Título
   LASER

2. Nome do Autor
   Poncelet, C. G.

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

4. Resumo
   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 1.855 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 rod 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.

5. Palavras Chaves
   REACTORS CELLS, SLOWING DOWN, SPECTRA, CYLINDERS, CRITICALITY, BURNUP, ONE-DIMENSIONAL CALCULATIONS.

6. Liguagem de Codificação e Computador
   FORTRAN IV; IBM/360 e IBM 7090

7. Categoria
   D, B

8. Referência
   PONCELET, C. G. *Burnup physics of heterogeneous reactor lattices.* Pittsburgh, PA, Westinghouse Electric Corp., June 1965. (WCAP)-6069).

   PONCELET, C. G. *LASER: a depletion program for lattice calculations based on MUFT and THERMOS.* Pittsburgh, PA, Westinghouse Electric Corp., April 1966. (WCAP.6073).

IPEN008

1. 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 Division.

4. Resumo
   LEOPARD is a unit cell homogenization and spectrum generation (MUFT-SOFOCATE) program with a fuel depletion option. The MUFT-SOFOCATE homogeneous medium spectrum analysis 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
   REACTORS CELLS, BURNUP, SPECTRA, GROUP CONSTANTS.

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

7. Categoria
   B

8. Referência
   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-Revised Aug. 1968).

1. Título
   MC^2

2. Nome do Autor
   Toppel, B. J.; Rago, A. L.; Oshea, D. M.; Stenberg, C.

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

4. Resumo
   MC^2 is used to calculate multigroup cross sections using an evaluate nuclear data file (ENDF) and these cross sections are suitable for direct use by neutronics codes without performing ancillary calculations.

5. Palavras Chaves
   GROUP CONSTANTS, CROSS SECTIONS, RESONANCE, DOPPLER BROADENING, INELASTIC SCATTERING, ELASTIC SCATTERING

6. Linguagem de Codificação e Computador
   FORTRAM IV; IBM/370

7. Categoria
   B

8. Referência
   TOPPEL, B. J.; RAGO, A. L.; OSHEA, D. M. *MC^2 - a code to calculate multigroup cross sections.* Argonne, ILL, Argonne National Laboratory, June 1967. (ANL-7318)

IPEN010

1. Título
   PHOEL

2. Nome do Autor
   Turner, J. E.; Módolo, J. T.; Sordi, G. M. A. A.

3. Estabelecimento de Origem
   Instituto de Pesquisas Enérgeticas e Nucleares, São Paulo, SP (BR)

4. Resumo
   O código PHOEL emprega o método de Monte Carlo para gerar as energias de foto-elétrons e elétrons Compton num meio aquoso irradiado por fótons com um espectro arbitrário de energia. O meio aquoso é suposto infinito e homogêneo e o espectro de fótons uniforme. A produção de pares é desprezada, o que não representa uma omissão significativa para fótons com energias abaixo de 10.000 Kev. O programa foi escrito especificamente para ser um têrmo de fonte para códigos de transporte e degradação da energia de elétrons em meio aquoso.

5. Palavras Chaves
   MONTE CARLO METHOD, COMPTON EFFECT, PHOTO-ELECTRON INTERACTIONS, ENERGY LOSSES.

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

7. Categoria
   W

8. Referência
   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, IPEN, March 1978.

1. Título
   XLACS

2. Nome do Autor
   Greene, N. M.; Lucius J. L.; Wright, R. Q; Craven, C. W.; Tobias, M. L.

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

4. Resumo
   XLACS calculates fine-group average neutron cross sections from ENDF/B data. Its primary purpose is to produce full range multigroup libraries for the XSDRN program. Provisions are included 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
   CROSS SECTIONS, NEUTRONS, MULTIGROUP THEORY, GROUP CONSTANTS.

6. Linguagem de Codificação e Coputador
   FORTRAN IV; IBM/360 e IBM/370

7. Categoria
   8

8. Referência
   GREENE, N. M.; LUCIUS, J. L.; WHITE, J. E.; WRIGHT, R. Q.; CRAVEN, C. C. W.; TOBIAS, M. L. *XLACS: a program to procedure weighted multigroup neutron cross sections from ENDF/B.* Oak Ridge, TN, Oak Ridge National Laboratory, April 1972. (ORNL-TM-3648).

IPEN012

1. Título
   XSDRN - X-Section Dynamic for Reactor Nucleonics.

2. Nome do Autor
   Greene, N. M. & Craven, C. W.

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

4. Resumo
   XSDRN is a discrete ordinates, spectral code for the generation of nuclear multigroup constants in the fast, resonance, and thermalization energy regions. A variable dimensioning technique is employed which optimizes the use of core storage. The code calculates an arbitrary number of flux moments for zero or one-dimensional systems. Several problem type are provided for including fixed source, eigenvalue, and criticality search. Extensive cross sections libraries are available which can be reduced, using calculated fine-grow fluxes, to arbitrary broad-group structures. Provisions are available to produce library tapes for several codes, and modifications can be easily made to include other codes.

5. Palavras Chaves
   REACTOR CELLS, GROUP CONSTANTS, TRANSPORT THEORY, MULTIGROUP THEORY, ONE-DIMENSIONAL CALCULATIONS, DISCRETE ORDINATES METHOD, ANGULAR DISTRIBUTIONS, RESONANCE, SLABS, CYLINDERS, SPHERES.

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

7. Categoria
   B

8. Referência
   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)

   OAK RIDGE NATIONAL LABORATORY. *XSDRN: a discrete ordinates spectral averaging code.* Oak Ridge, TN, October 1973. (RSIC-CCC-123)

1. 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. Resumo
   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 abiiity 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 correction 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
   ANISOTROPY, SCATTERING, TRANSPORT THEORY, MULTIGROUP THEORY, ONE-DIMENSIONAL CALCULATIONS, DISCRETE ORDINATES METHOD.

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

7. Categoria
   C

8. Referência
   OAK RIDGE NATIONAL LABORATORY. *ANISN-ORNL: multigroup one-dimensional discrete ordinates transport code with anisotropic scattering.* Oak Ridge, TN, October 1975. (RSIC-CCC-254).

   SOLTESZ, R. G. & DISNEY, R. K. *One dimensional, discrete ordinates transport technique.* Pittsburgh, PA, Westinghouse Astronuclear Laboratory. August 1970 (WANL-PR-(LL)-034-Vol.4).

   OAK RIDGE NATIONAL LABORATORY. . *ISN: multigroup one-dimensional discrete ordinates transport code with anisotropic scattering.* Oak Ridge, TN, January 1968. (RSIC-CCC-82)

IPEN014

1. Título
   DOT-Discrete Ordinate Transport

2. Nome do Autor
   Mynatt, F. R.

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

4. Resumo
   DOT is a FORTRAN program which solves the linear, energy dependent Boltzman transport equation for two-dimensional R-Z, X-Y, and R-Theta geometries. The gradient convection term in the Boltzman 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
   ANISOTROPY, SCATTERING, TRANSPORT THEORY, MULTIGROUP THEORY, TWO-DIMENSIONAL CALCULATIONS, DISCRETE ORDINATES METHOD.

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

7. Categoria
   C

8. Referência
   OAK RIDGE NATIONAL LABORATORY. *DOT: two-dimensional discrete ordinates transport code.* Oak Ridge, TN, October 1969. (RSIC-CCC-89/K-1694)

1. Título
   DOT-3.5

2. Nome do Autor
   Mynatt, F. R.; Engle Jr, W. W.; Gritzner, M. L.; Rhoades, W. A.; Rodgers, R. J.

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

4. Resumo
   DOT Solves the Boltzmann tranport equation in two-dimensional geometries. Principal applications are to neutron and/or photon transport, although the code can be applied to tranport problems for any particles not subject to external forces fields. Both homogeneous and external-source problems can be solved. Searches on multiplication factor, time absorption, nuclide concentration, and zone thickness are available for reactor problems. Numerous edits and output data sets for subsequent use are available.

5. Palavras Chaves
   SCATTERING, NEUTRON TRANPORT, BOLTZMANN EQUATION, DISCRETE ORDINATES METHOD

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

7. Categoria
   C

8. Referência
   RHOADES, W. A. & MYNATT, F. R. *The DOT-III two-dimensional discrete ordinates transport code.* Oak Ridge, TN, Oak Ridge National Laboratory, Sept. 1973. (ORNL-TM-4280). In: CCC-276/DOT 3.5 Code Package.

IPEN016

1. Título
   EXPANDA-70

2. Nome do Autor
   Hasegawa, Akira

3. Estabelecimento de Origem
   Japan Atomic Energy Research Institute, Tokai-Mura (JAPAN). Division of Reactor Engineering.

4. Resumo
   EXPANDA-70 is a one-dimensional, real or adjoint solution for a slab, cylinder or sphere, and at the same time dimensional critical search may be made, if requested. In the numerical solution, the methods are the same as EXPANDA-4. Therefore EXPANDA-70 is a extended programme of EXPANDA-4. The central difference approximation is adopted for the differential coefficients. The three points difference equation is solved by the ordinary sets of two points recurrence formula.

5. Palavras Chaves
   FAST REACTORS, ADJOINT FLUX, REACTIONS KINECTICS.

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

7. Categoria
   C, B

8. Referência
   SUZUKI, T. *EXPANDA-4, one-dimensional diffusion equation code for fast reactor using the JAERI-FAST SET.* 1969 (JAERI-MEMO-3660).

1. 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. Resumo
   The multigroup two-dimensional neutron diffusion equations are solved in X-Y, R-Z ou R-Theta geometry. The EQUIPOISE method is employed to solve 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 wich 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
   DIFFUSION, MULTIGROUP THEORY, NEUTRON FLUX, NEUTRON DIFFUSION EQUATION, TWO-DIMENSIONAL CALCULATIONS, REACTIVITY.

6. Linguagem de Codificação e computador
   FORTRAN IV; IBM/360

7. Categoria
   C

8. Referência
   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, TN, Oak Ridge National Laboratory, April 1967. (ORNL-4078)

IPEN018

1. Título
   MACH-1

2. Nome do Autor
   Meneley, D. A.; Kuitek, L. C.; Oshea, D. M.

3. Estabelecimento de Origem
   Argonne National Laboratory, ILL (USA). Reactor Phisics Division.

4. Resumo
   The MACH-1 diffusion-theory package embodies the basic features of the AIM-6 one-dimensional diffusion code, the DEL perturbation 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 MAIM-6 are the DEL perturbation 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 calculations, alpha search, and microscopic cross section modifications 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
   DIFFUSION, MULTIGROUP THEORY, SLABS, CYLINDERS, CRITICALITY, ONE-DIMENSIONAL CALCULATIONS, REACTION RATE, PERTURBATION THEORY.

6. Linguagem de Codificação e Computador
   FORTRAN IV e 3600; CDC6500 e CDC3600

7. Categoria
   C

8. Referência
   MENELEY, D. A.; KUITEK, L. C.; OSHEA, D. M. *MACH-1 - a one dimensional diffusion theory package.* Argonne, ILL, Argonne National Laboratory, June 1966. (ANL-7223)

1. Título
   SAND-2

2. Nome do Autor
   Mcelroy, W. N.; Berg, S.; Crocket, T.; Hawkins, R. G.

3. Estabelecimento de Origem
   Oak Ridge National Laboratory, TN (USA). Radiation Shielding Information Center.

4. Resumo
   Neutron energy spectra are determined by an analysis of experimental activation detector data. An iterative pertubation method is used to obtain a 'best fit' neutron flux spectrum for a given input set of infinitely dilute foil activities, The calculation procedure consists of the selection of a known flux spectrum form to serve as the initial approximation to the solution, and subsequent iteration to a form acceptable as an appropriate solution. The solution is specified either as time-integrated flux (fluence) for a pulsed environment or as a flux for a steady-state neutron environment.

5. Palavras Chaves
   NEUTRON FLUX, NEUTRON SPECTRA.

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

7. Categoria
   O

8. Referência
   OAK RIDGE NATIONAL LABORATORY. *SAND: neutron flux spectra determination by multiple foil activation - iterative method.* Oak Ridge, TN, May 1969. (CCC-112) (RSIC Computer Code Collection)

IPEN020

1. Título
   GACOST

2. Nome do Autor
   Asmussem, K. E.; Brogli, R. H.; Nowick, S. E.; Southworth, B. W.

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

4. *Resumo*
   The GACOST code is a very flexible computer program for calculating fuel cycle costs. It had as its origin the fuel cycle cost 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 revelized basic. Yearly cash flows are also edited, as are the mass flows and fuel burnup (in MWD/MTI. 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
   FUEL CYCLE, ECONOMICS.

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

7. Categoria
   D

8. Referência
   ASMUSSEM, K. E.; BROGLI, R. H.; NOWICKI, S. E. *A manual on fuel cycle cost calculations and a description of the code: GACOST.* San Diego, CA, General Atomic, Oct. 1972. (GULF-GA-A10693)

1. Título
   GARGOYLE

2. Nome do Autor
   Todt, F. W.

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

4. Resumo
   GARGOYLE is a infinite medium depletion code for fuel cycle calculations on reactors employing partial refueling. The burnups of up to 12 discret fuel compositions (regions or stages of irradiation) may be followed simultaneously. The burnup calculation may be interrupted periodically to remove the contents of and to refuel one or more regions. Fuel may be completely or partially recycled. A control poison search may be performed at eachs time-step. Feed fuel searchs are permitted at the end of each burnup cycle before refueling. Concentration dependent self-shielding factors may be applied to any nuclide except moderators. The nuclide scheme employed, althouth not completely general, is flexible.

5. Palavras Chaves
   FUEL CYCLE, BURNUP, FUEL MANAGEMENT.

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

7. Categoria
   D

8. Referência
   TODT, F. W. *GARGOYLE: A infinite medium fuel cycle analysis code with fuel and poison searches.* San Diego, CA, General Atomic, Feb. 1966. (GA-6622)

IPEN022

1. **Título**
   MDCN

2. **Nome do Autor**
   Azevedo, J. B. L.

3. **Estabelecimento de Origem**
   Centrais Elétricas Brasileiras, Rio de Janeiro, RJ (BR).

4. **Resumo**
   O programa MDCN tem a finalidade de determinar o custo do combustível nuclear empregado em usinas nucleares para geração de energia elétrica ou calor.
   Os fundamentos econômicos utilizados são: recuperação de capital proporcional à geração de energia, equivalência de custos feita pelo método do valor presente. O modelo apresenta inovações em comparação com outros modelos já empregados, uma vez que nos cálculos leva em conta os períodos de recarga e manutenção e além disso possue as etapas do ciclo do combustível (processos industriais e serviços) não fixas. A primeira inovação conduz à determinação de um custo mais real, além de permitir que o modelo seja utilizado em conjunto com simuladores de sistemas de potência hidrotérmicos; a segunda permite uma flexibilidade maior no seu uso, como comparação econômica de ciclos.

5. **Palavras Chaves**
   FUEL CYCLE, COST, NUCLEAR FUELS.

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

7. **Categoria**
   D

8. **Referência**
   AZEVEDO, J. B. L. *Modelo de determinação do custo do combutível nuclear.* Rio de Janeiro, COPPE-UFRJ, 1979. (Tese).

1. Título
   BLOOST-6

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

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

4. Resumo
   BLOOST-6 consists of three parts: a space-independent kinectics 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 kinectics code, which does a fourth-order Runge-Kutta integration of the reactor kinectic equations using a variable time step based on the speed of the change in the reactor power level.

5. Palavras Cheves
   TEMPERATURE DISTRIBUTION, THERMODYNAMICS, TWO-DIMENSIONAL CALCULATION, HEAT TRANSFER, FUEL ELEMENTS, HTGR TYPE REACTORS.

6. Linguagem de Codificação e Computador
   FORTRAN IV; UNIVAC 1108 e IBM/370

7. Categoria
   H, E

8. Referência
   DAHLBERG, R. C. & MERRILL, M. H. *BLOOST-6: kinectics code containning a thermodynamic model coated particles for HTGR applications.* San Diego, CA, General Atomic Div., July 1967. (GAMD-8119).

IPEN024

1. Título
   GAPOTKIN-General Atomics Points Kinectics Code.

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

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

4. Resumo
   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 aliows 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 or events such as a sudden loss of coolant followed reactivity insertions due to energy build-up.

5. Palavras Chaves
   KINECTICS, FEEDBACK, REACTIVITY.

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

7. Categoria
   E

8. Referência
   HANSEN, K. F. & KOCH, P. K. *GAPOTKIN: a point kinectics code for the UNIVAC-1108.* San Diego, CA, General Atomic Div., Oct. 1967. (GA-8204)

1. Título
   DISCOMP - Three-dimensional heated surface discharge computations.

2. Nome do Autor
   Stolzenbach, K. D.; Adams, E. E.; Harleman, D. R. F.

3. Estabelecimento de Origem
   Massachusetts Inst. of Tech. Cambridge (USA). Dept. Of Civil Enginnering

4. Resumo
   The code considers a discharge of heated water from a rectangular open channel at the surface of an ambient body of infinite extent. The three-dimensional temperature distribution depends upon the mixing between the discharged and ambient water and upon the rate of heat transfer to the atmosphere at the water surface. The discharge is assumed to be a free turbulent jet with a well defined turbulent region in which velocity and temperature are related to centerline values by similarity functions. A major contribution of this work is the treatment of lateral buoyant spreading by incorporating an assumed distribution for the lateral velocity into the set of integrated governing equations.

5. Palavras Chaves
   TEMPERATURE DISTRIBUTION, DISCHARGE CANALS, HIDRODYNAMICS, SURFACES.

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

7. Categoria
   H

8. Referência
   STOLZENBACH, K. D.; ADAMS, E. E.; HARLEMAN, D. R. F. *A user's manual for three-dimensional heated surface discharge computations*. Cambridge (USA), Massachusetts Inst. of Tech., Sept. 1972. (Report nº 156)

IPEN026

1. Título
   QX1

2. Nome do Autor
   Meneley, D. A. & Fuller, B. L.

3. Estabelecimento de Origem
   Argonne National Laboratory, ILL (USA). Applied Physics Division.

4. Resumo
   QX1 solves the multigroup, one-dimensional, time-dependent diffusion equations. Problem geometry may be plane, cylindrical or spherical. Steady-state initial conditions may be established either for a source-free system or for a system with an external neutron source. The reactor may be perturbed by changing material volume fractions and/or temperatures or by changing the neutron source level. A first-collision pulsed source distribution may be specified. Resonance absorption feedback is calculated by groupwise interpolation in a cross-section versus temperature table. A highly simplified fuel temperature model is included.

5. Palavras Chaves
   DIFFUSION, MULTIGROUP THEORY, FAST REACTORS, SPACE-TIME, ONE-DIMENSIONAL CALCULATIONS, KINECTCS.

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

7. Categoria
   F

8. Referência
   MENELEY, D. A.; OTT, K. O.; WIENER, E. S. *Fast reactor kinetics - the QX1 code.* Argonne, ILL, Argonne National Laboratory, March 1971. (ANL-7769).

1. 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. Resumo
   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 three-dimensional normal distribution. The radioactivity of the material is described throughout the space domain as a function of time.

5. Palavras Chaves
   REACTOR ACCIDENTS, RADIATION DOSES, INHALATION.

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

7. Categoria
   G

8. Referência
   OAK RIDGE NATIONAL LABORATORY. *ACRA II: Kernel integration code estimation of radiation dose caused by a hypothetical reactor accident.* Oak Ridge, TN, July 1974. (CCC-213).

IPEN028

1. 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. Resumo
   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 use 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 phantom. Doses are determined from photon energy deposited at each interaction point.

5. Palavras Chaves
   DOSE RATES, MONTE CARLO METHOD, PHANTOMS.

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

7. Categoria
   G

8. Referência
   OAK RIDGE NATIONAL LABORATORY. *ALGAM: a Monte Carlo estimation of internal doses from gamma-ray sources in a phantom man.* Oak Ridge, TN, 1973. (CCC-152)

1. Título
   CONTEMPT-LT

2. Nome do Autor
   Wheat, L. L.; Wagner, R. J.; Niederauer, G. F.; Obenchain, C. F.

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

4. Resumo
   CONTEMPT-LT was developed to describe the long-term behavior of water-cooled nuclear reactor containment systems subjected to postulated loss-of-coolant accident (LOCA) conditions. The program calculates the time variation of compartment pressures, te.nperatures, mass and energy inventories, heat structure temperature distributions, and energy exchange with adjacent compartments. The program is capable of describing the effects of leakage on containment response. Models are provided to describe fan cooler and cooling spray enginnering safety systems. The program can be used to model all current boiling water reactor pressure suppression systems including containment with either vertical or horizontal vent systems. CONTEMPT-LT can also be used to model pressurized water reactor dry containments, subatmospheric containments, and dual volume containments with an annulus region, and can be used to describe containment responses in experimental containment systems.

5. Palavras Chaves
   REACTOR ACCIDENTS, LOSS OF COOLANT, CONTAINMENT, PRESSURE, SUPPRESSION.

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

7. Categoria
   G

8. Referência
   WHEAT, L. L.; WAGNER, R. J.; NIEDERAUER, G. F.; OBENCHAIN, C. F. *CONTEMPT-LT: a computer program for predicting containment pressure temperature response to a loss-of-coolant accident.* Idaho Falls, ID, Aeroject Nuclear Company, June 1975. (ANCR-1219)

IPEN030

1. Título
   GADOSE/DOSET

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

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

4. Resumo
   The GADOSE program calculates radioactivity and doses resulting from instantaneous accidental release of activity while the companion program DOSET includes the effects of a time-dependent accidental fission product release for the HTGR type of plant. Given an initial fission product inventory released into a reactor vessel or containment volume. The quantity of each isotope is calculated at a number of times, at a number of locations (in the containment, containment recirculating cleanup filters, leak collector, plant exhaust filters and in the atmosphere at a number of distances from the plant). Each decay chain is calculated separately considering buildup and decay of each isotope. Radiological doses are calculated for any three body organs and the whole body based on metereological and physiological input paramenters. Fallout and rainout are included.

5. Palavras Chaves
   RADIOACTIVITY, DOSE RATES, FISSION PRODUCTS, HTGR TYPE REACTORS.

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

7. Categoria
   G

8. Referência
   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)

1. Título
   ORIGEN - ORNL Isotope Generation and Depletion Code.

2. Nome do Autor
   Bell, M. J.

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

4. Resumo
   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
   CHAINS, RADIATIVE DECAY, FISSION PRODUCTS.

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

7. Categoria
   G

8. Referência
   OAK RIDGE NATIONAL LABORATORY. *ORIGEN: isotope generation and depletion code matrix method.* Oak Ridge, TN, June 1977. (CCC-217)

IPEN032

1. Título
   RELAP4

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

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

4. Resumo
   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 subjected 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
   HYDRODYNAMICS, HEAT TRANSFER, ACCIDENTS, BLOWDOWN, EXCURSIONS, POWER PLANTS, REACTOR SAFETY.

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

7. Categoria
   H, G

8. Referência
   MOORE, K. V. &RETTIG, W. H. *RELAP4: a computer program for transient thermal-hydraulic analysis.* Idaho Falls, ID, Aeroject Nuclear Company. (ANCR-1127)

1. 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. Resumo
   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
   THERMODYNAMICS, HEAT TRANSFER, SPACE-INDEPENDENT, TWO-DIMENSIONAL CALCULATIONS, TEMPERATURE DISTRIBUTION, KINECTICS.

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

7. Categoria
   H

8. Referência
   MERRILL, M. H. & TROOST, M. *BLOOST5: a combined reactor kinectics heat transfer code for the IBM-7044*. San Diego, CA, General Atomic Div., Aug. 1965. (GAMD-6644)

IPEN034

1. Título
   COBRA-3C

2. Nome do Autor
   Rowe, D. S. & Curtis, R. L.

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

4. Resumo
   COBRA-3C calculates the steady-state and transient flow, enthalpy and pressure drop in the subchannels of rod bundle nuclear fuel elements during both boiling and nonboiling conditions. The program uses a mathematical model that includes the effects of turbulent and diversion crossflow mixing between the subchannels. A thermal model option is included to calculate radial fuel temperature distribution as a function of user-specfied power generation and heat transfer correlations. Critical heat flux calculations can also be performed by user option. Forced crossflow mixing due to diverter vanes or wire wraps can be considered.

5. Palavras Chaves
   THERMODYNAMICS, FLUID FLOW, FUEL ELEMENTS, LIQUIDS, VAPORS, PRESSURE DROP, ENTHALPY.

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

7. Categoria
   G, H

8. Referência
   ROWE, D. S. *COBRA-3C: a digital computer program for steady state and transient thermal-hydraulic analysis of rod bundle nuclear fuel elements.* Richland, WA, Battelle-Nortwest, March 1973. (BNWL-1695).

1. Título
   COBRA-3M

2. Nome do Autor
   Marr, W. W.

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

4. Resumo
   COBRA-3M calculates the steady-state and transient flow, coolant temperature and pressure drop in the subchannels of nuclear fuel pin bundles. In addition, it calcualtes temperatures of fuel, cladding and duct wall. COBRA-3M is a variation of the COBRA-3C program.

5. Palavras Chaves
   THERMODYNAMICS, FLUID FLOW, FUEL ELEMENTS, LIQUIDS, VAPORS, PRESSURE DROP, DUCTS, ENTHALPY.

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

7. Categoria
   H

8. Referência
   ROWE, D. S. *COBRA III - a digital computer program for steady state and transient thermal - hydraulical subchannel analysis of rod bundle nuclear fuel elements.* Richland, WA, Battelle Nortwest, July 1971. (BNWL-B-82)

   MARR, W. W. *COBRA 3M - a digital computer code for analyzing thermal hydraulic behavior in pin bundles.* Argonne, ILL, Argonne National Laboratory, March 1975. (ANL-8131)

IPEN036

1. Título
   ETCHICAN - Programa para análise termohidráulica em regime permanente de um trocador de calor de carcaça e tubos 'U' com chicanas segmentais.

2. Nome do Autor
   Baptista F9, B. D.

3. Estabelecimento de Origem
   Instituto de Pesquisas Energéticas e Nucleares, São Paulo, SP (BR)

4. Resumo
   O código ETCHICAN é a implementação do método de análise de subcanais fornecendo as distribuições de fluxo, pressões e temperaturas de estado estacionário para os flúidos de carcaça e tubos escoando ao longo de um trocador de calor de carcaça e tubos 'U' com chicanas segmentais.

5. Palavras Chaves
   THERMODYNAMICS, HEAT TRANSFER, BAFFLES, FLUID FLOW, TEMPERATURE DISTRIBUTION

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

7. Categoria
   H

8. Referência
   BAPTISTA F9, B. D. *Modelo numérico para solução termohidráulica de um trocador de calor de carcaça e tubos 'U' com chicanas segmentais.* São Paulo, IPEN, 1979. (Dissertação de Mestrado)

1. Título
   MACABRE-II

2. Nome do Autor
   Griebenow, M. L. & Richert, K. D.

3. Estabelecimento de Origem
   Idaho Nuclear Corp., Idaho Falls (USA). Phillips Petroleum Company.

4. Resumo
   MACABRE-II is capable of solving the isothermal hidraulics and nonisothermal hydraulics and heat transfer of most plate-type reactor fuel assemblies cooled by a subcooled fluid in forced convection. It can accommodate two-dimensional, time dependent (quasi steady-state) heat generation rates and hydraulic communication between the parallel coolant channels. In addition, MACABRE-II will compute the corrosion of the alumnium-clad, heat-transfer surfaces (when present) and can handle partial sloughing of the Boehmite corrosion product. MACABRE-II has been employed in MTR, ETR, ATR and EOCR fuel assembly analysis.

5. Palavras Chaves
   FUEL PLATES, HEAT TRANSFER, ISOTHERMAL PROCESSES.

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

7. Categoria
   H

8. Referência
   GRIEBENOW, M. L. & RICHERT, K. D. *MACABRE-II.* Idaho Falls, (ID, Idaho Nuclear Corp., Sept. 1967. (IN-1107)

IPEN038

1. Título
   NATOF-2D3

2. Nome do Autor
   Granziera, M. R.

3. Estabelecimento de Origem
   Instituto de Pesquisas Energéticas e Nucleares, São Paulo, SP (BR)

4. Resumo
A two-dimensional numerical model for the simulation of sodium boiling transient using the two fluid set of conservation equations. A semi-implicit numerical differencing scheme, capable of handling the problems associated with the ill-posedness implied by the complex characteristic roots of the two fluid problem, was used, which took advantage of the dumping effect of the exchange terms. Of particular interest in the development of the model was the identification of the numerical problem caused by the strong disparity between the axial and radial dimensions of fuel assemblies. A solution to this problem was found which used the particular geometry of fuel assemblies to accelerate the convergence of the iterative technique used in the model.

The most important feature of the model was its ability to simulate severe conditions of sodium boiling, in particular flow reversal which was shown in the tests performed with the model.

5. Palavras Chaves
   FUEL ASSEMBLIES, HEAT TRANSFER, TRANSIENTS, TWO-DIMENSIONAL CALCULATIONS, BOILING

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

7. Categoria
   H

8. Referência
GRANZIERA, M. R. *A two-dimensional two fluid model for sodium boiling in LMBFR fuel assemblies.* Cambridge, (USA), Massachusetts Inst. of Technology, May 1980 (Tese)

1. Título
   TEMAX

2. Nome do Autor
   Stegemann, D. & Zurita, A.

3. Estabelecimento de Origem
   Technische Univ., Hannover, (F. R. Germany). Inst. fuer Kerntechnick.

4. Resumo
   The TEMAX code gives the temperature distribution for a fuel plate of one research reactor for the steady and nonsteady states. It is possible to see the influence of the enrichment in the fuel plate design considering a invariable power density. The code analyses the puntal enrichment evolution in the meat and gives the maximal temperature points in the cladding and meat. The code also gives the minimal subcooling and the DNBR as safety margin with respect to the thermohydraulical design limits.

5. Palavras Chaves
   TEMPERATURE DISTRIBUTION, FUEL PLATES, HEAT TRANSFER, THERMODYNAMICS.

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

7. Categoria
   H

8. Referência
   ZURITA, A. *Description of the code TEMAX, a thermohydraulic code for plate-type fuel elements.* Hannover, Inst. fuer Kerntechnik, Sept. 1979. (IKH 110/79)

IPEN040

1. Título
   TOODEE-2

2. Nome do Autor
   Lauben, G. N.

3. Estabelecimento de Origem
   Nuclear Regulatory Commission, Washington, DC (USA). Division of Technical Review.

4. Resumo
   TOODEE-2 is a two dimensional, time dependent fuel element thermal analysis program developed by NCR staff the TOODEE code. TOODEE-2 was developed primarily as an evaluation tool to calculated fuel element thermal response during post-LOCA refill and reflood in a PWR. For small breaks the code may be used for the entire transient. The code as originally written allowed three optional solution geometries: slab (X-Y), axisymmetric (R-Z) and polar (R-Theta). The changes to the original TOODEE program have been considered only axisymmetric geometry. Two general classes of changes have been made to the original code: (1) those changes made for general Regulatory use since receiving the code in 1967, (2) those changes made directed toward its use as an evaluation tool.

5. Palavras Chaves
   TWO-DIMENSIONAL CALCULATIONS, PWR TYPE REACTORS, FUEL ELEMENTS, THERMAL ANALYSIS, HEAT TRANSFER, LOSS OF COOLANT.

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

7. Categoria
   H

8. Referência
   LAUBEN, G. N. *TOODEE 2: a two dimensional time dependent fuel element thermal analysis program.* Washington, DC, Nuclear Regulatory Commission, 1975. (NUREG-75/057)

1. Título
   FEAST1-65

2. Nome do Autor
   Wilson, E. L.; Christian, J. T.

3. Estabelecimento de Origem
   Massachusetts Inst. of Tech., Cambridge (USA). Dept. of Civil Enginnering

4. Resumo
   The purpose of this computer program is to determine the deformations and stresses within certain types of stressed bodies. The program will analyse problem situations in any of the following categories: axial symmetry, plane stress, plane strain. Elastic non-linear material properties are considered by a successive approximation technique. The effects of displacement or stress boundary condition, concentrated loads, gravity forces and temperature changes are included.

5. Palavras Chaves
   AXIAL SYMMETRY, DEFORMATION, STRESSES, STRAINS.

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

7. Categoria
   I

8. Referência
   WILSON, E. L. *Program FEAST1-65 user's manual.* Cambridge (USA), Massachusetts Inst. of Tech., Sept. 1966.

IPEN042

1. Título
   MARS

2. Nome do Autor
   Stegemann, D. & Wiesenack, W.

3. Estabelecimento de Origem
   Technische Univ., Hannover, (F. R. Germany). Inst. fuer Kerntechnick.

4. Resumo
   MARS is a FORTRAN IV computer code designed to predict the in-pile perfomance of cylindrical, light-water-reactor fuel element under normal operanting conditions. The code predicts temperature distribution, thermoelastic, creep and plastic deformation of the fuel and clad, fuel volume alterations due to swelling and densification and fission gas release during load cycles. Either a gap or fuel clad interface. Central void formation is not allowed. The stress-strain calculations are performed for uncracked or cracked fuel and clad. Plane strain conditions and axial summetry in the fuel element are assumed. Power is input as a function of time, allowing analysis of power time.

5. Palavras Chaves
   TEMPERATURE DISTRIBUTION, STRESSES, STRAINS, CRACKS.

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

7. Categoria
   I

8. Referência
   STEGEMANN, D. & WIESENACK, W. *Modeling fuel element performance of light water reactors.* Hannover, Inst. fuer Kerntechnik, Aug. 1978

1. Título
   NONSAP

2. Nome do Autor
   Bathe, K. J.; Wilson, E. L.; Iding, R. H.

3. Estabelecimento de Origem
   California Univ., Berkeley (USA). Structural Engineering Lab.

4. Resumo
   THE NONSAP is a structural analysis program for static and dynamic response of nonlinear systems. It's not an extension of the linear analysis program SAP, but rather a completely new development. Program NONSAP is designed with two primary objectives.
   1 – efficient solution of variety of practical nonlinear problems with the current capabilities of nonlinear analysis procedures and computer equipment.
   2 – use effective in the various research areas pertainig to nonlinear analysis.

5. Palavras Chaves
   NONLINEAR PROBLEMS, STRUCTURE

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

7. Categoria
   I

8. Referência
   BATHE, K. J.; WILSON, E. L.; IDING, R. H. *NONSAP: a structural analysis program for static and dynamics response of nonlinear systems.* Berkeley, CA, California Univ., Feb 1974. (UCSESM 74-3)

IPEN044

1. Título
   PV2MRD

2. Nome do Autor
   Nagamati, M.; Tamura, M.; Dieguez, J. A.; Camargo, P. B.; Davidson, I.

3. Estabelecimento de Origem
   Instituto de Pesquisas Energéticas e Nucleares, São Paulo, SP (BR)

4. Resumo
   O código emprega o método da relaxação dinâmica utilizando o método de diferenças finitas. Calcula as pressões em vasos de concreto protendido de reatores nucleares e permite o cálculo do crescimento de fissuras no concreto.

5. Palavras Chaves
   REACTOR VESSELS, PRESTESSED CONCRETE, FINITE DIFFERENCE METHOD.

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

7. Categoria
   I

8. Referência
   NAGAMATI, M.; TAMURA, M.; DIEGUEZ, J. A.; CAMARGO,P. B.; DAVIDSON, I. *Manual de utilização do código PV2MRD.* São Paulo, IPEN, 1979. (Relatório datilografado)

1. Título
   SAFE-2D

2. Nome do Autor
   Cornell, D. C.

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

4 Resumo
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
   STRESS ANALYSIS, FINITE ELEMENT METHOD

6. Linguagem de Codificação e Computador
   FORTRAN IV; UNIVAC 1108 e IBM/370

7. Categoria
   I

8. Referência
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)

IPEN046

1. Título
   DOMINO - Discrete ordinates Monte Carlo interface operation

2. Nome do Autor
   Emmett, M. B.; Burgart, C. E.; Hoffman, T. J.

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

4. Resumo
   A method of coupling two-dimensional discrete ordinates calculations with Monte Carlo Calculations is incorporated in the DOMINO code. In particular. DOT a two-dimensional discrete ordinates code, has been coupled with MORSE, a Monte Carlo code. The discrete ordinates calculations is limited to an r-z geometry. Either the DOT boundary angular fluxes or the angular fluxes at internal surfaces may be input to DOMINO. Any amount of surface may be used in DOMINO; i.e., from all mesh intervals down to a portion of one interval the radii, axes and quadrature set must be read in to DOMINO tape. The DOMINO tape contains all necessary coupling information for use by the Monte Carlo codes.

5. Palavras Chaves
   DISCRETE ORDINATES METHOD, MONTE CARLO METHOD, TWO-DIMENSIONAL CALCULATIONS.

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

7. Categoria
   J

8. Referência
   EMMETT, M. B.; BURGART, C. E.; HOFFMAN, T. J. *DOMINO: a general purpose code for coupling discrete ordinates and Monte Carlo radiation transport calculations.* Oak Ridge, TN, Oak Ridge National Laboratory, July 1973. (ORNL-4853). In: CCC-203/MORSE-CG Code Package.

1. Título
   MORSE-CG - A general purpose Monte Carlo multigroup neutron and gamma ray tranport code with combinatorial geometry

2. Nome do Autor
   Emmett, M. B.

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

4. Resumo
   The MORSE code is a multipurpose neutron and gamma ray transport Monte Carlo code. It has been designed as a tool for solving most shielding problems. Through the use of multigroup cross sections, the solution of neutron, gamma ray or coupled neutron-gamma-ray problems may be obtained in either the forward or adjoint mode. Time dependence for both shielding and criticality problems is provided. General three-dimensional geometry, as well as specialized one dimensional geometry descriptions, may be used with an albedo option available at any material surface. Isotropic and anisotropic scattering up P16 expansion of the angular distribution is allowed.

5. Palavras Chaves
   MONTE CARLO METHOD, TRANSPORT THEORY, NEUTRONS, ALBEDO, SHIELDING, CRITICALITY, TIME-DEPENDENCE

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

7. Categoria
   J

8. Referência
   EMMETT, M. B *The MORSE Monte Carlo radiation transport code system.* Oak Ridge, TN, Oak Ridge National Laboratory, February 1975. (ORNL-4972). In: CCC-203/MORSE-CG Code Package.

IPEN048

1. Título
   AMPX-1

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

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

4. Resumo
   AMPX-1 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-1 is flexibly dimensioned, neutron group structures, gamma group structures, expansion order to represent anisotropic processes are all arbitrary and limited only by available computer core and budget.

5. Palavras Chaves
   CROSS SECTIONS, NEUTRONS, GROUP CONSTANTS, DISCRETE ORDINATES METHOD, RESONANCE.

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

7. Categoraia
   K

8. Referência
   GREENE, N. M.; LUCIUS, J. L.; WHITE, J. E.; WRIGHT, R. Q.; PETRIE, L. M.; FORD, W. E. *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)

1. Título
   CITATION

2. Nome do Autor
   Fowler, T. B.; Vondy, D. R.; Cunninghan, G. W.

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

4. Resumo
   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.

5. Palavras Chaves
   ONE-DIMENSIONAL CALCULATIONS, TWO-DIMENSIONAL CALCULATIONS, THREE-DIMENSIONAL CALCULATIONS, DIFFUSION, CRITICALITY, BUCKLING, FUEL MANAGEMENT, MULTIGROUP THEORY, BURNUP.

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

7. Categoria
   K, F, C, D

8. Referência
   FOWLER, T. B.; VONDY, D. R.; CUNNINGHAN, C. W. *Nuclear reactor core analysis code: CITATION.* Oak Ridge, TN, Oak Ridge National Laboratory. Rev. 2, July 1971. Suppl. 1, October 1971. Suppl. 2, March 72. Suppl. 3, July 1972. (ORNL-TM-2496)

IPEN050

1. Título
ARC SYSTEM-Argonne Reactor Computation System.

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

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

4. Resumo
The ARC-SYSTEM consists of a library of computacional modules for fast reactor design calculations, along with a set of system modules which provide an environment for the operation of the computational modules. The modules are used in a selected order by directive programs called "paths". Intermodule communication is through ARC System data sets. The data sets are defined such that output from one computational module can be used as input to others. This allows for the automation into a single run of what would normally be a sequence of runs.

5. Palavras Chaves
ONE-DIMENSIONAL CALCULATIONS, TWO-DIMENSIONAL CALCULATIONS, THREE-DIMENSIONAL CALCULATIONS, TRANSPORT THEORY, DIFFUSION, BURNUP, FUEL MANAGEMENT, CROSS-SECTIONS, PERTURBATIONS THEORY.

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

7. Categoria
K, C

8. Referência
WOODRUFF, W. L.; DALY, T. A.; HENRYSON, H.; KIER, P. H.; REGIS, J. P.; STENBERG, C. G., TOPPEL, B. J. *The ARC System standard paths and cataloged procedures.* Argonne, ILL, Argonne National Laboratory, December 1971. (ANL-7712)

KOVALSKY, E. A.; ZAPATKA, J.; HENRYSON, H.; HOOVER, J.; WALKER, P. M. *The ARC System cross-sections homogenizations and modification capabilities.* Argonne, ILL, Argonne National Laboratory, June (1971). (ANL-7714)

DALY, T. A.; LEAF, G. K.; KENNEDY, A. S. *The ARC System two dimensional diffusion theory capability,* DARC2D. Argonne, ILL, Argonne National Laboratory, May 1972. (ANL-7716).

KOVALSKY, E. A. & NEAL, D. E. *THE ARC System neutronics input processors.* Argonne, ILL, Argonne National Laboratory, January 1971. (ANL-7713)

KRYANT, L. T. & LEAF, G. K. *The ARC System one-dimensional transport theory capability,* SNARC2D. Argonne, ILL, Argonne National Laboratory, January 1971. (ANL-7717).

STENBERG, C. G. & LINDEMAN, A. *The ARC System cross-sections generation capabilities,* ARC-MC^2. Argonne, ILL, Argonne National Laboratory, June 1973. (ANL-7722)

1. Título
   CHECK4, CRECT, LISTFC, RIGEL4, DICT4

2. Nome do Autor
   Honeck, H. C.; Felberbaum, J.; Culler, D. E.

3. Estabelecimento de Origem
   Brookhaven National Laboratory, New York, (USA). National Nuclear Data Center.

4. Resumo
   CHECK4 checks the structure, consistency and formats of ENDF/B data files.
   CRECT corrects ENDF/B data files.
   LISTFC produces interpreted listings of ENDF/B data.
   RIGEL4 retrieves ENDF/B data and changes mode (BCD-BIN) or arrangement of data on ENDF/B tapes.
   DICT4 constructs a new section dictionary (file 1 section 451) for an entire ENDF/B tape.

5. Palavras Cheves
   DATA PROCESSING, CROSS SECTIONS, LIBRARIES, MAINTENANCE

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

7. Categoria
   M

8. Referência
   OZER, O. ed. *ENDF-110 Description of the ENDF/B processing codes and retrieval routines.* Upton, New York, Brookhaven National Laboratory, 1971. (BNL-50300) (TID-4500)

IPEN052

1. Título
   ETOE

2. Nome do Autor
   Green, D. M. & Pitterle, T. A.

3. Estabelecimento de Origem
   Atomic Power Development Associates, Inc., Detroit, MI (USA)

4. Resumo
   ETOE (ENDF/B to MC^2 data conversion) accepts cross section data from a mode 2 ENDF/B tape and prepares the binary cross section and Legendre polynomial tape for the MC^2 code written Argonne National Laboratory. The ETOE program processes formats I, II, III and IV ENDF/B data tapes.

5. Palavras Chaves
   CROSS SECTIONS, LIBRARIES.

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

7. Categoria
   M

8. Referência
   GREEN, D. M. & PITTERLE, T. A. *ETOE - a program for ENDF/B to MC^2 data conversion.* Detroit, MI, Atomic Power Development Associates, June 1968. (APDA-219)

1. Título
   JFUSER, LTFR-70

2. Nome do Autor
   Hasegawa, A. & Suzuki, T.

3. Estabelecimento de Origem
   Japan Atomic Energy Research Institute, Tokai-Mura (JAPAN). Division of Reactor Engineering.

4. Resumo
   JFUSER is a collapsing and data format conversion routine for LTFR-70. Colapsing is done by simple flux weighting.
   LTFR-70 processes JAERI-FAST group constants sets of up to 70 group and prepares a binary library tape for efficient usage by a series of related fast reactor-design calculation programmes.

5. Palavras Chaves
   FAST REACTORS, DATA PROCESSING.

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

7. Categoria
   M

8. Referência
   TAKANO, J.; HASEGAWA, M.; NAKAGAWA, M.; ISHIGURO, Y.; KATSURAGI, S. *JAERI fast reactor group constants set, version II.* Tokai-Mura, Japan Atomic Energy Research Institute, Nov. 1977. (JAERI-1225)

   TUZUKI, T. EXPANDA-4 one-dimensional diffusion equation code for fast reactor using the JAERI-FAST SET. 1969 (JAERI-MEMO-3660).

IPEN054

1. Título
   ANACROM - Programa Computacional para a Análise de Cromatogramas.

2. Nome do Autor
   Gouvêa, A. S. & Mesquita, C. H.

3. Estabelecimento de Origem
   Instituto de Pesquisas Energéticas e Nucleares, São Paulo, SP (BR)

4. Programa
   O programa faz uma pesquisa automática de picos do cromatograma e determina para cada pico a posição central, altura, área, largura na meia altura e a resolução. As estimativas numéricas de cada parâmetro é seguida do seu erro assintótico. A pesquisa dos picos está baseada no conceito da mudança de sinal da derivada primeira. Após a subtração do 'back-ground' é feito o ajuste pelo método dos mínimos quadrados não linear, empregando o algoritmo de Marquadt-Bevington. A função ajustada tem a forma de uma soma de guassianas simples ou soma de gaussianas modificadas à esquerda ou à direita pela inclusão de uma parte exponencial. São obtidos os gráficos: (a) valores experimentais e preditos versus número da observação, (b) resíduos versus número da observação, (c) gráfico probabilístico dos resíduos.

5. Palavras Chaves
   LEAST SQUARE FIT, GAUSS FUNCTION, CHROMATOGRAPHY.

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

7. Categoria
   O

8. Referência
   GOUVÊA, A. S. & MESQUITA, C. H. *ANACROM: programa computacional para análise de cromatogramas.* São Paulo, IPEN, Janeiro 1981. (IPEN-Inf-5)

1. Título
   DALP

2. Nome do Autor
   Mazzaro, A. C.

3. Estabelecimento de Origem
   Instituto de Pesquisas Energéticas e Nucleares, São Paulo, SP (BR)

4. Resumo
   O programa DALP simula as técnicas experimentais que possibilitam a obtenção de distribuição de alturas de pulsos devido a neutrons mono-energéticos incidentes em um cintilador orgânico líquido (NE-213).

5. Palavras Chaves
   LIQUID SCINTILATORS, PULSE ANALYSERS, SIMULATION.

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

7. Categoria
   O

8. Referência
   INSTITUTO DE PESQUISAS ENERGÉTICAS E NUCLEARES. *DALP - relatório do programa.* São Paulo, 1979. (Relatório datilografado)

IPEN056

1. Título
   GRETEL. A computer program for gamma-ray spectrometry with Ge(Li) detectors.

2. Nome do Autor
   Guzzi, G. & Guypers, J.

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

4. Resumo
   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) detector, using special 'oriented libraries', which are prepared for each particular problem. The computer routines which detect and evaluated peak areas performs 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 possiblity 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
   SPECTRA, RADIATIONS DETECTORS, QUANTITATIVE CHEMICAL ANALYSIS

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

7. Categoria
   O

8. Referência
   GUZZI, G. & GUYPERS, J. *GRETEL: a computer program for gamma-ray spectrometry with Ge(Li) detectors*. Ispra, Joint Nuclear Research Center, 1974. (EUR-5117)

1. *Título*
   NBS-576

2. Nome do Autor
   Spiegel Jr., V. & Murphey, W. M.

3. Estabelecimento de Origem
   National Bureau of Standards, Washington, D. C. (USA)

4. Resumo
   It is a computer code for the calculation of thermal neutron absorption in spherical and cylindrical neutron sources. The source may be composed of up to three cladding and one possibly fissionable core material. The effects of elastic and inelastic scattering of thermal neutrons are neglected. The thermal-neutron flux is assumed to be isotropic, which enables one to carry out the computation as a sum of mono-directional fluxes from different directions. All integrations are performed with Weddle's formula.

5. Palavras Cheves
   NEUTRON SOURCES. CALIBRATION STANDARDS.

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

7. Categoria
   O

8. Referência
   SPIEGEL, V. & MURPHEY, W. M. *Computer code for the calculation of thermal neutron absorption in spherical and cylindrical neutron sources.* Washington, D. C., National Bureau of Standards, May 1971. (NBS Technical Note 576)

IPEN058

1. Título
   O5S

2. Nome do Autor
   Textor, R. E. & Verbinsky, V. V.

3. Estabelecimento de Origem
   Oak Ridge National Laboratory, TN (USA). Computing Technology Center.

4. Resumo
   O5S is designed to directly simulate the experimental techniques used to obtain the pulse height distribution for a parallel beam of monoenergetic neutrons incident on organic scintillator systems. Developed to accurately calibrate the nominaly 2 in- by 2 in- liquid organic scintillator NE-213 (composition CH-1.2) the code should be readily adaptable to many similar problems.
   O5S is a Monte Carlo code patterned after the general-purpose Monte Carlo neutron transport code system O5R. The O5S Monte Carlo 'experiment' follows the course of each neutron through the scintillator and abtains the energy-deposits of the ions produced by elastic scattering and reactions. The light pulse produced by the neutron is obtained by summing up the contributions of the various ions with the use of appropriate light vs. ion-energy table.

5. Palavras Chaves
   MONTE CARLO METHOD, LIQUID SCINTILATORS.

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

7. Categoria
   O

8. Referência
   OAK RIDGE NATIONAL LABORATORY. *O5S response function generator: an O5R Monte Carlo code for calculating pulse height distribution due to monoenergetic neutrons incident on organic scintillator.* Oak Ridge, TN, Oak Ridge National Laboratory, April 1972. (ORNL-TM-3646).

1. Título
   ASAAM23 - Simulation, Analysis and Modeling

2. Nome do Autor
   Angell, M. K.

3. Estabelecimento de Origem
   National Institute of Health, Bethesda, MD (USA). Department of Biomathematics at Cornell University, Medical College.

4. Resumo
   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. 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.

5. Palavras Chaves
   ANALYSIS, SIMULATION.

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

7. Categoria
   P

8. Referência
   BERMAN, M. & WEISS, M. F. *User's manual for SAAM23.* Bethesda, M. D., National Institute of Arthrists Metabolic Diseases, May 1966. (SAAM23)

IPEN060

1. Título
   BEVINGTO

2. Nome do Autor
   Módolo, J. T.

3. Estabelecimento de Origem
   Instituto de Pesquisas Energéticas e Nucleares, São Paulo SP (BR)

4. Resumo
   O código BEVINGTO é um sistema que envolve técnicas de ajuste de funções não lineares nos parâmetros que as compõem, a pares de pontos experimentais, usando o método dos mínimos quadradcs 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.

5. Palavras Chaves
   LEAST SQUARE FIT, NONLINEAR PROGRAMMING.

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

7. Categoria
   P

8. Referência
   MÓDULO, J. T.; *BEVINGTO: Ajuste de função usando método dos mínimos quadrados não linear.* São Paulo, IPEN, Janeiro 1978. (Relatório datilografado)

1. Titulo
   CSMP - Continuous System Modeling Program

2. Nome do Autor
   Internacional Business Machines Corp., White Plains, N. Y. (USA)

3. Estabelecimento de Origem
   Internacional Business Machines Corp., White Plains, N. Y. (USA)

4. Resumo
   The S/360 Continuous System Modelling Program is a problem-oriented program designed to facilitate the digital simulation of continuous processes on large-scale digital machines. The program provides an application-oriented language that allows these problems to be prepared directly and simply from either a block-diagram representation or a set of ordinary differential equations. The program includes a basic set of functional blocks with which the components of a continuous system may be represented, and accepts application oriented statements for defining the connections between those functional blocks. This program is based on the Digital Simulation Languagem (DSL 190).

5. Palavras Chaves
   SIMULATION, SYSTEM ANALYSIS.

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

7. Categoria
   P

8. Referência
   INTERNATIONAL BUSINESS MACHINES CORP. *S/360 Continuous System Modeling Program User's Manual.* White Plains, N. Y., Jan. 1972. (83GH20-0367-4)

IPEN062

1. 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. Resumo
   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 essentialy 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
   DIFFERENTIAL EQUATIONS

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

7. Categoria
   P

8. Referência
   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)*

1. Título
   MINITAB STATISTICAL COMPUTING SYSTEM

2. Nome do Autor
   Ryan Jr., T. A.; Joiner, B. L.; Ryan, B. F.

3. Estabelecimento de Origiem
   Pennsylvania State University, (USA)

4. Resumo
   MINITAB is designed to allow students in introductory statistical courses to "speak" to the computer in commands similar to English sentences. These commands generally correspond to the major steps a students might follow in solving a problem by hand. The MINITAB system is not limited to use in elementary courses. Many students use it in course in fields of applications, for term papers, thesis and others projects. It is also useful in more advanced data analysis like: simulation, confidence intervals, correlations and regression, analysis of variance, chi-square tests and contingency tables, nonparametric statistics.

5. Palavras Chaves
   STATISTICS, SIMULATION

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

7. Categoria
   P

8. Referência
   RYAN JR., T. A.; JOINER, B. L.; RYAN, B. F. *MINITAB Student Handbock* North Scituate, MA, Duxbury Press. 1976.

IPEN064

1. Título
   SAAM25 - Simulation, Analysis and Modeling

2. Nome do Autor
   Berman, M. & Weiss, M. F.

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

4. Resumo
   SAAM is a digital computer program developed for 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 kinectics 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
   SIMULATION, BIOLOGICAL MODELS.

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

7. Categoria
   P

8. Referência
   BERMAN, M. & WEISS, M. F. *User's manual for SAAM, version SAAM25.* Bethesda, M. D., National Institute of Arthrists and Metabolic Diseases, Dec. 1971. (Relatório datilografado) (SAAM25)

1. Título
   SAS - Statistical Analysis System

2. Nome do Autor
   SAS Institute Inc.

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

4. Resumo
   SAS is a computer system for data analysis. In one easy-to-use system, it provides all the tools needed for data analysis:
   – information storage and retrieval
   – data modification and programming
   – report writing
   – statistical analysis
   –file handling

5. Palavras Chaves
   DATA MANAGEMENT, STATISTICS, LEAST SQUARE FIT.

6. Linguagem de Codificação e Computador
   IBM/370

7. Categoria
   P

8. Referência
   BARR, J. A.; GOODNIGHT, J. H.; SALL, J. P. *SAS user's guide.* Raleigh, N. C. SAS Institute Inc., 1979.

   BARR, J. A.; GOODNIGHT, J. H.; SALL, J. P.; HELWIG, J. T. *SAS programmer's guide.* Raleigh, N. C., SAS Institute Inc., Sept. 1977.

   HELWIG, J. T., ed. *SAS supplemental library user's guide.* Raleigh, N. C., 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, N. C., SAS Institute Inc., July 1977.

   BARR, J. A.; GOODNIGHT, J. H.; SALL, J. P.; BLAIR, H. W.; CHILKO, D. M. *SAS user's guide 1979 edition.* Raleigh, N. C., SAS Institute Inc., 1979.

IPEN066

1. Título
   SFR-3

2. Nome do Autor
   Kerlin, T. W. & Lucius, J. L.

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

4. Resumo
   The SFR-3 code is a program for calculating the frequency response of a multivariable system and its sensitivity to parameter changes.

5. Palavras Chaves
   SENSITIVITY, FREQUENCY RESPONSE TESTING.

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

7. Categoria
   P

8. Referência
   KERLIN, T. W. & LUCIUS, J. L. *THE SFR-3 code - a FORTRAN program for calculating the frequency response of a multivariable system and its sensivity to parameter changes.* Oak Ridge, TN, Oak Ridge National Laboratory, Oct. 1977. (ORNL-TM-1575)

1. *Título*
   CHETAH - The ASTM chemical thermodynamic and energy release evaluation program.

2. Nome do Autor
   American Society for Testing and Materials

3. Estabelecimento de Origem
   American Society for Testing and Materials, Philadelphia, PA (USA)

4. Resumo
   The program first estimates the thermodynamic properties of the reactants. Then it defines the reaction by which the chemical, or mixture of chemicals, can react, in a way consistent with the laws of staichiometry so as to release the maximum possible amount of energy. It then calculates this maximum heat of reaction. The program evaluates the potential energy release of the system according various criteria on a scale of three: high, medium or low, using empirical correlations of impact sensivity with either the maximum heat of reaction, the oxygen balance, or the difference between the heat of combustion and the maximum heat of reaction. The program can also be used just to calculate estimated values of the chemical thermodynamic properties, heat capacity, enthalpy, entropy and heat of combustion at any temperature between 290 e 1510 °K, either for individual compounds or for balanced chemical equations.

5. Palavras Chaves
   ORGANIC COMPOUNDS, REACTION HEAT, MIXTURES, ENTALPHY, PROPERTIES, ENTROPY, THERMODYNAMICS .

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

7. Categoria
   U

8. Referência
   SEATON, W. H.; FRIEDMAN, E.; TREWEEK, D. N. *CHETAH - the ASTM chemical thermodynamic and energy release evaluation program.* Philadelphia, PA, ASTM, Nov. 1974. (ASTM-DS-51)

IPEN068

1. Título
   LOAD-FLOW CESP

2. Nome do Autor
   Petracioli, J. R. & Laudanna, P. R.

3. Estabelecimento de Origem
   Companhia Energética de São Paulo, São Paulo, SP (BR). Divisão de Desenvolvimento de Modelos de Sistemas

4. Resumo
   O programa resolve o problema de fluxo de carga (load-flow) determinando as tensões (módulo e fase) e as potências (ativa e reativa) nas divesas barras de um sistema bem como o fluxo de potência (ativa e reativa) nas linhas sob determinadas condições inicialmente especificadas de geração e carga. E utilizado o método de Newton-Raphson Polar.

5. Palavras Chaves
   POTENTIAL FLOW, FLOW MODELS.

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

7. Categoria
   T

8. Referência
   COMPANHIA ENERGÉTICA DE SÃO PAULO. *LOAD-FLOW CESP - relatório de manutenção do programa.* (Relatório datilografado)

1. Título
   SOLMNEQ - Solution - mineral equilibrium computation

2. Nome do Autor
   Kharaka, Y. K. & Barnes, I.

3. Estabelecimento de Origem
   University of California, Berkeley (USA). Department of Geology & Geophysics.

4. Resumo
   SOLMNEQ computes the equilibrium distribution of 162 inorganic aqueous species generally present in natural waters over the temperature range of 0° to 350°C from the reported chemical analysis, temperature, PH, and EH (optional). Interpolated dissociation constants of the aqueous complesxes and the computed activity coefficients are also used in these computations. States of reactions of the aqueous solutions with respect to 158 solid phases (minerals) are computed from the distribution of aqueous species and an internally consistent set of thermodynamic data. Ionic proportions and subsurface temperature estimates are computed.

5. Palavras Chaves
   CHEMICAL REACTIONS, WATER CHEMISTRY, GEOTHERMOMETRY, EQUILIBRIUM, THERMODYNAMICS, CHEMICAL PROPERTIES

6. Linguagem de Codificação e Computador
   PL/I; IBM/360 e IBM/370

7. Categoria
   U

8. Referência
   KHARAKA, Y. K. & BARNES, I. *SOLMNEQ: Solution-Mineral Equilibrium Computation.* Menlo Park, CA, Department of Geology and Geophysics, February 1973. (USGS-WRD-73-002-NTIS-PB-215-899)

IPEN070

1. Título
   CELULAR

2. Nome do Autor
   Lima, M. A. P. & Ferreira, L. G.

3. Estabelecimento de Origem
   Universidade de São Paulo, São Paulo, SP (BR). Instituto de Física.

4. Resumo
   O programa emprega o método celular auto - consistente para moléculas diatômicas.

5. Palavras Chaves
   SCHROEDINGER EQUATION, SELF-CONSISTENTT FIELD, YUKAWA POTENTIAL, THOMAS-FERMI MODEL, MOLECULAR STRUCTURE.

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

7. Categoria
   W

8. Referência
   FERREIRA, L. G. *Método Celular auto-consistente para moléculas diatômicas.* São Paulo, IFUSP, Agôsto 1979. (Manual datilografado)

1. Título
   GARLA - Sistema computacional para a análise de ensaios radioligantes.

2. Nome do Autor
   Mesquita, C. H. & Gouvea, A. S.

3. Estabelecimento de Origem
   Instituto de Pesquisas Energéticas e Nucleares, São Paulo, SP (BR)

4. Resumo
   O Sistema está projetado para efetuar os cálculos da curva padrão utilizando o modelo empírico logístico de quatro parâmetros estimados por procedimentos ponderados de acordo com Rodbart e colaboradores. A sensibilidade do método é apreciada pela dose mínima detectável que considera as dispersões do tubo de concentração zero e a da própria amostra. A análise de Scatchard é efetuada segundo esquema de Priore e Rosenthal. A atividade específica do traçador é estimada segundo o método do auto-deslocamento de acordo com a reavaliação metodológica de Morris. A análise termodinâmica das reações é elaborada de acordo com Keane e colaboradores. O sistema oferece ao usuário outros recursos tais como o teste de Qui-quadrado para averiguar a qualidade das mediads radioativas e permite traçar gráficos de dados de diversas naturezas.

5. Palavras Chaves
   RADIOIMMUNOASSAY, ANTIGENS, ANTIBODIES, TRACER TECHNIQUES.

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

7. Categoria
   Y

8. Referência
   INSTITUTO DE PESQUISAS ENERGÉTICAS E NUCLEARES. *GARLA - relatório do programa.* São Paulo, 1980. (Relatório datilografado)

IPEN072

1. Título
   ENDF/B-IV: Evaluated Nuclear Data File

2. Nome do Autor
   National Nuclear Data Center.

3. Estabelecimento de Origem
   Brookhaven National Laboratory, New York, NY (USA). National Nuclear Data Center.

4. Resumo
   The ENDF system was designed for the storage and retrieval of evaluated nuclear data that are required for neutronics, photonics and decay heat calculations. This system is composed of several parts that include a serie of data processing codes and neutron and photon cross sections nuclear structure libraries. The ENDF/B library contains only one evaluation of the cross sections for each material in the library, but each material contains cross sections for all significant reactions. The data set selected for the ENDF/B library is the data set recommended by the Cross Sections Evaluation Working Group (CSEWG).

5. Palavras Chaves
   NUCLEAR DATA COLLECTIONS, NEUTRONS, PHOTONS.

6. Linguagem de Codificação e Computador
   IBM/370

7. Categoria
   Z

8. Referência
   GARBER, O.; DUNFORD, C.; PEARLSTEIN, S. *ENDF-102 data formats and procedures for the evaluated nuclear data file, ENDF.* New York, NY, Brookhaven National Laboratory, 1975. (BNL-NCS-50496)

1. Título
   JAERI FAST SET (JFS-Version II)

2. Nome do Autor
   Takano, H.; Hasegawa, A.; Nakagawa, M.; Ishiguro, Y.; Katsuragi, S.

3. Estabelecimento de Origem
   Japan Atomic Energy Research Institute, Tokai-Mura (JAPAN). Division of Reactor Engineering.

4. Resumo
   The JAERI FAST SET version II (JFS-V-II) has been produced by revising the original JFS-V-I on a large scale. The revision works have been performed on the basis of the benchmark calculation for many fast critical assemblies, the cross section adjustment of using the least squares method and the evaluation of nuclear data. In JFS-V-II, the group constants of 25 nuclides, Be, B-10, B-11, C, O, Na, Al, Si, Cr, Mn, Fe, Ni, Cu, Mo, Th-232, U-233, U-234, U-235, U-236, U-238, Pu-239, Pu-240, Pu-241, Pu-242, Am-241 and FP group constants have been provided for both the standard seventy and twenty five group structures.

5. Palavras Chaves
   FAST REACTORS, GROUP CONSTANTS, CROSS SECTIONS, CRITICALITY, DOPPLER EFFECT.

6. Linguagem de Codificação e Computador
   IBM/370

7. Categoria
   Z

8. Referência
   TAKANO, H.; HASEGAWA, A.; NAKAGAWA, M.; ISHIGURO, Y., KATSURAGI, S. *JAERI fast reactor group constans set, version II.* Tokai-Mura, Japan Atomic Energy Research Institute, Nov. 1977. (JAERI-1255)

IPEN074

1. Título
   MEDLIST

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

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

4. Resumo
   The source of data is the Evaluated Nuclear Structure File (ENSDF) of the ORNL Nuclear Data Project. The initial set includes 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
   ISOTOPES, RADIOISOTOPES.

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

7. Categoria
   Z

8. Referência
   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)

1. Título
   ZZ-DLC-2D

2. Nome do Autor
   Wright, R. Q.

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

4. Resumo
   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 sections 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 ANISN, DOT, DTF–IV codes, or any computer codes using data in the ANISN or DTF-IV format.

5. Palavras Chaves
   NEUTRON TRANSPORT, CROSS SECTIONS, GROUP CONSTANTS.

6. Linguagem de Codificação e Computador

7. Categoria
   Z

8. Referência
   OAK RIDGE NATIONAL LABORATORY. *100G: 100 Groups Neutron Cross Sections data bases on ENDF/B*. Oak Ridge, TN, July 1972. (DLC2RP e APRFX-I Data Retrieval Programs). (RSIC Data Collection). (DLC-2)

IPEN076

1. Título
   CSMPTEK

2. Nome do Autor
   Gouvêa, A. S. & Bello, R. S.

3. Estabelecimento de Origem
   Instituto de Pesquisas Energéticas e Nucleares, São Paulo, SP (BR)

4. Resumo
   O programa conversacional CSMPTEK torna disponível aos usuários do sistema de simulação 'Continuous System Modeling Program', CSMP a obtenção de soluções gráficas para as variáveis do problema no terminal gráfico interativo TEKTRONIX-4010. O programa utiliza as informações armazenadas em arquivo gerado pelo comando 'prepare' do CSMP.

5. Palavras Chaves
   SIMULATION, GRAPHIC.

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

7. Categoria
   P

8. Referência
   GOUVÊA, A. S. & BELLO, R. S. *CSMPTEK - Solução do CSMP no terminal gráfico interativo TEKTRONIX-4010.* São Paulo, IPEN, 1981. (Relatório datilografado)

1. Título
   STATIS

2. Nome do Autor
   Stokstad, R.

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

4. Resumo
   STATIS calculates total and differential cross sections for the reaction A + B → C + D, with no restrictions on the spins, parities or identity of the nuclei involved. The code enables the calculation of reactions for any combination of entrance and exit channel for a given compound system with only minor changes in the input data. Cross sections to a particular excitation energy in the residual nucleus may also be integrated over the density of final states in the residual nucleus. An option is provided for printing the various factors in the Hauser - Feshbach expression for each value of the total angular momentum.

5. Palavras Chaves
   DIFFERENTIAL CROSS SECTIONS, TOTAL CROSS SECTIONS, HAUSER FESHBACH THEORY, ANGULAR MOMENTUM

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

7. Categoria
   A

8. Referência
   STOKSTAD, R. *STAT1, STAT2 Hauser - Feshbach computer code description and user's manual.* Oak Ridge, TN, Oak Ridge National Laboratory. (Relatório datilografado)

ABSTRACT

This paper is a listing of the nuclear codes available at the data processing center of IPEN (CPD-IPEN). 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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