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UNITED STATES ATOMIC ENERGY COMMISSION

**A BIBLIOGRAPHY OF AVAILABLE DIGITAL
COMPUTER CODES FOR NUCLEAR REACTOR
PROBLEMS**

**By
A. Radkowsky
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October 14, 1955

**Naval Reactors Branch
Division of Reactor Development
United States Atomic Energy Commission
Washington, D. C.**

Technical Information Service, Oak Ridge, Tennessee

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A BIBLIOGRAPHY
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INTRODUCTION

This bibliography provides a ready listing of reactor digital computer codes presently available or in preparation, and is arranged in such a manner as to enable the engineer or physicist to determine which codes will be of assistance to him in his work. The codes are for digital computers of the size of the Card Programmed Calculator (CPC) or larger. The information listed in these pages was obtained directly from the laboratories in which these programs were prepared. A minimum of editing went into the preparation of this report to enable immediate publication.

The following format is used for presenting the information:

<u>1</u> TITLE	<u>2</u> COG. ENGR.	<u>3</u> LABORATORY
<u>4</u> COMPUTER	<u>5</u> PROBLEM STATUS	<u>6</u> CATEGORY
<u>7</u> GEOMETRY & DIMENSIONS	<u>8</u> TYPICAL COMPUTING TIME	<u>9</u> PARAMETERS SOLVED
<u>10</u> PROBLEM DESCRIPTION		
<u>11</u> LIMITATIONS OF THE CODE		
<u>12</u> REFERENCES		

To provide simplicity in the referencing of individual subjects, provisions have been made for referencing specific categories in space 6. The categories used are as follows:

Reactor and Nuclear Physics (P)*

Reactor Survey (RS)*

One Dimension

One Group
Two Group
Three Group
Multigroup

Two Dimension

Few Group
Multigroup

Three Dimension

Two Group

Shielding (S)*

Reactor Kinetics (K)*

Reactor Burnout (RB)*

Reactor Engineering (E)*

Miscellaneous (M)*

It can be seen that there is some tendency for overlap between categories. It is, therefore, suggested that for the greatest utility and convenience the potential user familiarize himself with the nature of these subdivisions.

The responsible laboratory (Space 3) and the cognizant scientist or engineer (Space 2) at that laboratory are also listed. This information plus the available references (Space 12) should simplify the procurement of additional data on a given code.

* All pages in a single category are numbered and prefixed by the category initial.

Identification of the status of the code is accomplished by the use of a box labeled PROBLEM STATUS (Space 5). Listed below are the categories used:

Analyzed	Debugged
Programmed	In Use
Coded	Obsolete

When a code has been classified obsolete, every effort has been made to give the code that replaces it. Other information presented herein is self-explanatory.

It is believed that this bibliography contains a majority of the codes available to date. However, as there undoubtedly are some codes which are not listed, it is requested that laboratories and other organizations having appropriate codes and who have not been previously contacted, forward information concerning these codes to:

Dr. Harry Polachek
Applied Mathematics Laboratory
David Taylor Model Basin
Carderock, Maryland

A supplement will be issued in the near future in an attempt to bring this bibliography up to date. As new codes are developed, regular supplements will be issued.

A. RADKOWSKY
R. BRODSKY
Naval Reactors Branch
Division of Reactor Development
United States Atomic Energy Commission
Washington 25, D. C.

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AECU-5978

Resonance
Absorption
Integrals

D. D. Dix

Phillips Petroleum
National Reactor Testing
Station

TITLE

COG. ENGR.

LABORATORY

CPC

Coded

Reactor and Nuclear Physics

COMPUTER

PROBLEM STATUS

CATEGORY

GEOMETRY &
DIMENSIONS

1 min./integral

 $\int \sigma/E \, dE$

TYPICAL COMPUTING TIME

PARAMETERS SOLVED

The resonance absorption integral $\int \sigma/E \, dE$

$$\left\{ \sigma = \sigma(E) = \frac{\sigma_0 \sqrt{\Gamma} (E_0/E)^{1/2}}{4 (E-E_0)^2 + (\Gamma/2)^2} \right\}$$

is evaluated directly
PROBLEM DESCRIPTION

Limited only by applicability of the physical assumptions

LIMITATIONS OF THE CODE

None published. Refer to D. D. Dix, Phillips Petroleum Company, Idaho Falls, Idaho.

REFERENCES

REP (Resonance Escape Probability)	Dr. R. Richtmyer R. L. Hellens (Phys.) G. Crane (Math.)	AEC Computing Facility New York University Westinghouse Atomic Power Division
TITLE	COG. ENGR.	LABORATORY
UNIVAC	In Use	Reactor and Nuclear Physics
COMPUTER	PROBLEM STATUS	CATEGORY
Hexagonal lattice of round fuel rods with (vacuum) cladding in moderator (water) GEOMETRY & DIMENSIONS	3 hours for 10,000 histories TYPICAL COMPUTING TIME	1) Resonance Escape 2) Pu Production distribution 3) Capture Spectrum PARAMETERS SOLVED

Monte Carlo treatment yields resonance escape probability for a Doppler-broadened Breit-Wigner resonance. Taken into account are: resonance scattering, energy losses in all scatterings. Main moderator is assumed to be H, but one can have up to 5 isotopes present in moderator and 5 in fuel. Yields also energy distribution of scatterings and of absorptions and spatial distribution of absorptions. 10,000 histories gives 1-p to about 2% accuracy.

PROBLEM DESCRIPTION

Treats only one resonance at a time.

(See Rep. Mod. 1)

LIMITATIONS OF THE CODE

WAPD P-439	R. Hellens
KAPL 1241	G. M. Roe
NYU 6479	R. D. Richtmyer

REFERENCES

REP MOD. 1	Dr. R. Richtmyer	AEC Computing Facility New York University
TITLE	COG. ENGR.	LABORATORY
UNIVAC	In Use	Reactor and Nuclear Physics
COMPUTER	PROBLEM STATUS	CATEGORY
Hexagonal lattice of round fuel rods with cladding, in water moderator	8-10 hours for 15,000 histories 18 resonances (H ₂ O mod'd.)	1) Resonance Escape 2) Pu Production distribution 3) Capture Spectrum
GEOMETRY & DIMENSIONS	TYPICAL COMPUTING TIME	PARAMETERS SOLVED

Same as REP except; (1) takes any number of resonances into account simultaneously, (2) uses more accurate treatment of far wings of the resonances, (3) energy and spatial distributions are printed out in more detail, (4) main moderator need not be hydrogen.

PROBLEM DESCRIPTION

LIMITATIONS OF THE CODE

NYU 6479 R. D. Richtmyer

REFERENCES

MONTE CARLO FAST
FISSION EFFECT

FIN

R. Hellens (Phys.)
G. Crane (Math)

Westinghouse Atomic Power
Division

TITLE

COG. ENGR

LABORATORY

UNIVAC

Analyzed

Reactor and Nuclear Physics

COMPUTER

PROBLEM STATUS

CATEGORY

Rods in
hexagonal lattice

- - - - -

- 1) Fast fission factor
- 2) Average fast fission cross
section
- 3) Average fast capture
cross section

GEOMETRY &
DIMENSIONS

TYPICAL COMPUTING TIME

PARAMETERS SOLVED

Code is, in principle, similar to REP except for the presence of inelastic scattering and the accounting of events. Energy range from 10 Mev to 0.8 Mev. Rods in hexagonal lattice with void gap between rod and moderator.

Analysis of problem is partially completed.

May possibly be dropped since "Bunny" may provide information required.

PROBLEM DESCRIPTION

LIMITATIONS OF THE CODE

REFERENCES

MONTE CARLO RESONANCE
ESCAPE - WREP

R. L. Hellens (Phys.)
G. E. Crane (Math.)

Westinghouse Atomic Power
Division

TITLE

COG. ENGR.

LABORATORY

NORC

Debugged

Reactor and Nuclear Physics

COMPUTER

PROBLEM STATUS

CATEGORY

Rods in hexagonal
or square lattice

1.2 minutes per single resonance

- 1) Resonance escape
- 2) Pu production distribution
- 3) Capture spectrum

GEOMETRY &
DIMENSIONS

TYPICAL COMPUTING TIME

PARAMETERS SOLVED

NORC code set up for hexagonal or square rod lattice with void gap between rod and moderator. Treats one resonance or multiple resonances with Doppler broadening. Can be used to obtain self-shielding effect of rods.

PROBLEM DESCRIPTION

Admits mixtures of five elements in both moderator and rod; however, only one element may have a resonance.

LIMITATIONS OF THE CODE

NYO-6479

REFERENCES

S-APPROXIMATION-II
MUFT-II

R. L. Hellens (Phys)
B. H. Mount (Math)

Westinghouse Atomic Power
Division

TITLE

COG. ENGR.

LABORATORY

CPC	In Use	Reactor and Nuclear Physics
-----	--------	-----------------------------

COMPUTER

PROBLEM STATUS

CATEGORY

Fourier trans-
form one
dimension

1-1/2 hours

See below

GEOMETRY &
DIMENSIONS

TYPICAL COMPUTING TIME

PARAMETERS SOLVED

54 energy group code for calculating Fourier transform of slowing down distribution in water-metal mixtures. Has been extensively applied to water and water-uranium for fission neutrons. Compares four approximations:

- 1) P-1 with additional Selengut-Goertzel app. in either case.
- 2) B-1

Provides:

- 1) ϵ, ρ , two-group age, fast diffusion constant.
- 2) moments of the slowing down distribution.

PROBLEM DESCRIPTION

- 1) Inelastic spectrum independent of incident neutron energy.
- 2) Awkward for more than three elements in a given mixture.
- 3) No edit for other than two-group constants.
- 4) Slow operation

LIMITATIONS OF THE CODE

WAPD-PM-12

REFERENCES

MUFT-III

R. L. Hellens (Phy.)
B. H. Mount (Math.)

Westinghouse Atomic Power
Division

TITLE

COG. ENGR.

LABORATORY

IRM-650

Coded

Reactor and Nuclear Physics

COMPUTER

PROBLEM STATUS

CATEGORY

Fourier-trans-
form-one-
dimension

Approximately five minutes

"See Below"

GEOMETRY &
DIMENSIONS

TYPICAL COMPUTING TIME

PARAMETERS SOLVED

Multi-group (199 maximum) code for calculating Fourier transform of slowing down distribution. Computes any or all of the four approximations:

- 1) P-1
- 2) B-1
- 3) P-1 Selengut-Goertzel
- 4) B-1 Selengut-Goertzel

Provides: Few-group constants

PROBLEM DESCRIPTION

LIMITATIONS OF THE CODE

Report to be written.

REFERENCES

SPH - H I

Brookhaven National Laboratory

TITLE	COG. ENGR.	LABORATORY
UNIVAC	In Use	Reactor and Nuclear Physics
COMPUTER	PROBLEM STATUS	CATEGORY
Cylindrical		Flux and Thermal Utilization
GEOMETRY & DIMENSIONS	TYPICAL COMPUTING TIME	PARAMETERS SOLVED

Survey of cylindrical reactor with fuel rods in lattice arrangement. Volume ratio, enrichment, and other parameters varied from case to case. Set of simultaneous linear equ.solved using P_3 spherical harmonics.

PROBLEM DESCRIPTION

This code could probably not be used generally because of the way it stores input.

LIMITATIONS OF THE CODE

REFERENCES

Temperature
Coefficients of
Heterogenous
Reactors (I-24184)

G. W. Stuart
E. R. Astley
J. R. Triplett

Numerical Analysis Unit
HAPO
GE
Richland

TITLE

COG. ENGR.

LABORATORY

702 EDPM

In Use

Reactor and Nuclear Physics

COMPUTER

PROBLEM STATUS

CATEGORY

1 min. per Neutron Temp.

Lattice Dimensions, Fuel
Exposure, Neutron Temp.

GEOMETRY &
DIMENSIONS

TYPICAL COMPUTING TIME

PARAMETERS SOLVED

Computes average of Eta, thermal utilization, migration area, absorption & fission neutron cross sections of U235 and Pu239, for various values of neutron temperature, and variation of effective multiplication constant by

$$\frac{\Delta K_{eff}}{K_{eff}} = \frac{\Delta f}{f} + \frac{\Delta \eta}{\eta} - B^2 M^2 \left(\frac{\Delta M^2}{M^2} \right)$$

Where Δ indicates difference of quantity at given temperature from thermal temperature. Averages are computed by integrating over maxwellian distribution with blackness weighting in lieu of weighting by macroscopic absorption cross section.

PROBLEM DESCRIPTION

Up to 300 energy points.

LIMITATIONS OF THE CODE

HW-36825 (Confidential)

REFERENCES

Program H - Cylinder Transport Equation	E. D. Nix	Aircraft Nuclear Propulsion Department General Electric Company Cincinnati 15, Ohio
TITLE	COG. ENGR.	LABORATORY
IBM-701	Programmed	Reactor and Nuclear Physics
COMPUTER	PROBLEM STATUS	CATEGORY
Cylinder	30 Minutes (Estimated)	Disadvantage factors, flux, angular distribution of neutrons.
GEOMETRY & DIMENSIONS	TYPICAL COMPUTING TIME	PARAMETERS SOLVED

Numerically integrates the monoenergetic Boltzmann equation in cylindrical geometry.

PROBLEM DESCRIPTION

Up to 20 space intervals, each of which may contain a different material.

LIMITATIONS OF THE CODE

ANL-5049

REFERENCES

SLAV-I		Brookhaven National Laboratory
TITLE	COG. ENGR.	LABORATORY
UNIVAC	In Use	Reactor and Nuclear Physics
COMPUTER	PROBLEM STATUS	CATEGORY
Slab		Flux Distribution
GEOMETRY & DIMENSIONS	TYPICAL COMPUTING TIME	PARAMETERS SOLVED

Study of one velocity transport equ. in a slab reactor consisting of alternating fuel and moderator slabs with uniform source in moderator.

PROBLEM DESCRIPTION

The way in which this problem stores input would make it difficult for general use.

LIMITATIONS OF THE CODE

REFERENCES

SPOT - I

Brookhaven National Laboratory

TITLE

COG. ENGR.

LABORATORY

UNIVAC

Coded

Reactor and Nuclear
Physics

COMPUTER

PROBLEM STATUS

CATEGORY

Bare Homogeneous
Reactors

Reactor Bucklings and
Fluxes

GEOMETRY &
DIMENSIONS

TYPICAL COMPUTING TIME

PARAMETERS SOLVED

Code solves energy-dependent Boltzmann equ. of bare reactor using Lagransian interpolation expansions of the scattering densities. Moderator can be anything not containing hydrogenous material.

PROBLEM DESCRIPTION

LIMITATIONS OF THE CODE

REFERENCES

P-12

RE-18	D. Satkus	Argonne National Laboratory
TITLE	COG. ENGR.	LABORATORY
AVIDAC	Programmed	Reactor and Nuclear Physics
COMPUTER	PROBLEM STATUS	CATEGORY
		Solves for critical buckling
GEOMETRY & DIMENSIONS	TYPICAL COMPUTING TIME	PARAMETERS SOLVED

This program is the fundamental mode calculation for buckling with 16 energy groups. With a specified material composition, this program computes the critical material buckling and the real and adjoint fluxes for each of the 16 groups, as well as related quantities.

PROBLEM DESCRIPTION

Flux $< 10^3$

LIMITATIONS OF THE CODE

ANL Memorandum - D. Satkus to H. Hummel
SUBJECT: Multigroup Fundamental Mode Calculation - 11/4/54

REFERENCES

Program I - Analytic
 P_3 Calculation

R. R. McCready

Aircraft Nuclear Propulsion
Department
General Electric Company
Cincinnati 15, Ohio

TITLE

COG. ENGR.

LABORATORY

IBM-701

Programmed

Reactor and Nuclear
Physics

COMPUTER

PROBLEM STATUS

CATEGORY

Cylinder

Disadvantage factor, flux
angular distribution

GEOMETRY &
DIMENSIONS

TYPICAL COMPUTING TIME

PARAMETERS SOLVED

Evaluates the analytic solution of the P_3 spherical harmonics equations for cylindrical geometry.

PROBLEM DESCRIPTION

Two regions, constant source terms in either or both regions.

LIMITATIONS OF THE CODE

REFERENCES

Program J - Numerical
 P_3 Calculations

R. R. McCready

Aircraft Nuclear Pro-
pulsion Department
General Electric Co.
Cincinnati 15, Ohio

TITLE

COG. ENGR.

LABORATORY

IBM-701

Programmed

Reactor and Nuclear
Physics

COMPUTER

PROBLEM STATUS

CATEGORY

Cylinder

Disadvantage factor,
flux, angular
distribution

GEOMETRY &
DIMENSIONS

TYPICAL COMPUTING TIME

PARAMETERS SOLVED

Numerically integrates the P_3 spherical harmonics equations for cylindrical geometry, assuming an arbitrary source.

PROBLEM DESCRIPTION

Five regions, up to 40 space intervals.

LIMITATIONS OF THE CODE

REFERENCES

RE-24

M. Butler

Argonne National Laboratory

TITLE

COG. ENGR.

LABORATORY

AVIDAC

In Use

Reactor and Nuclear Physics

COMPUTER

PROBLEM STATUS

CATEGORY

Slab

~ 0.1 minute for F
 ~ 1 minute for F and flux
 distribution

Solves for
~~F~~ Average Flux in Region II
 Average Flux in Region I
 and if desired the flux
 distribution

GEOMETRY &
DIMENSIONS

TYPICAL COMPUTING TIME

PARAMETERS SOLVED

This program is the P_3 spherical harmonics calculation for 2-region slab.

Given: $\Sigma_s(\text{cm}^{-1})$ = scattering cross section,

$\Sigma(\text{cm}^{-1})$ = total cross section, $(= \Sigma_s + \Sigma_a)$,

$\bar{\mu}$ = average cosine of scattering angle

$X(\text{cm})$ = half thickness of region

$Q/\Sigma_a(\text{cm})$ = source/absorption cross section for each region

program computes F, the disadvantage factor, the average flux in each region, and if desired, the flux distribution in each region.

PROBLEM DESCRIPTION

$$0 < \bar{\mu} < \Sigma < 32$$

$$0 \leq \bar{\mu} < 1$$

$$0 < X < 64$$

$$10^{-8} < X < 10$$

$$0 \leq Q/\Sigma_a < 1$$

LIMITATIONS OF THE CODE

RED Memorandum - M. Butler to J. A. Thie

SUBJECT: P_3 Calculation for a 2-region Slab (RE-24)

7/28/55

REFERENCES

WIGNER-WILKINS
CALCULATION

H. Amster

Westinghouse Atomic Power
Laboratory

TITLE

COG. ENGR.

LABORATORY

IBM-650

Coded

Reactor and Nuclear
Physics

COMPUTER

PROBLEM STATUS

CATEGORY

Five hours

Flux energy distribution
Ave. cross sections

GEOMETRY &
DIMENSIONS

TYPICAL COMPUTING TIME

PARAMETERS SOLVED

Numerical integration of Wigner-Wilkins Riccati differential equation
generalized to include arbitrarily--varying cross sections.
Absorption cross section is averaged over thermal flux.

PROBLEM DESCRIPTION

Only hydrogen moderator

LIMITATIONS OF CODE

AECD-2275

REFERENCES

TEMP - I	J. A. Fleck	Origin. Brookhaven National Laboratory Prog. & Code - New York University
TITLE	COG. ENGR.	LABORATORY
UNIVAC	In Use	Reactor and Nuclear Physics
COMPUTER	PROBLEM STATUS	CATEGORY
Cylindrical		Neutron Density, Exit Temp.
GEOMETRY & DIMENSIONS	TYPICAL COMPUTING TIME	PARAMETERS SOLVED

The differential equ. of a circulating fuel reactor are replaced by a difference scheme. Parameters such as reactivity and intial temp. can be varied from case to case. Has been used in survey of reactor with Neg. temperature coefficient and no mechanical control rods.

PROBLEM DESCRIPTION

LIMITATIONS OF THE CODE

J. A. Fleck - Brookhaven National Laboratory

"The Temperature Dependent Kinetics of Circulating Fuel Reactors"

REFERENCES

COM-CEN-CH-I

Orig. Brookhaven National
Laboratory
Prog. & Code. New York University

TITLE

COG. ENGR.

LABORATORY

UNIVAC

In Use

Reactor and Nuclear
Physics

COMPUTER

PROBLEM STATUS

CATEGORY

Cylindrical

20-30 sec./case

Reflector Savings

GEOMETRY &
DIMENSIONS

TYPICAL COMPUTING TIME

PARAMETERS SOLVED

This is a survey of a cylindrical reactor surrounded by infinite lateral reflector. The loading radius, reactor buckling, and other reactor parameters are varied from case to case and the reflector savings are solved for by iterating on a transcendental equation involving Bessel functions.

PROBLEM DESCRIPTION

This problem is coded fixed point and will probably require rescaling when used with numbers other than those for which it was coded.

LIMITATIONS OF THE CODE

BNL C-7368

REFERENCES

NED	D. B. MacMillan	Knolls Atomic Power Laboratory
TITLE	COG. ENGR.	LABORATORY
IBM 650	In Use	Reactor and Nuclear Physics
COMPUTER	PROBLEM STATUS	CATEGORY
The equation is derived on the assumption of an infinite moderator GEOMETRY & DIMENSIONS	For a 25 point mesh, 1/2 hour to compute the kernel and 5 min. to solve the equation. TYPICAL COMPUTING TIME	Atomic Mass No. 9. PARAMETERS SOLVED

This program is in two parts. The first part computes the values of the kernel function in the Wigner-Wilkins integral equation (see AECD-2275. The equation is concerned with energy distribution near thermal energy of neutrons in a moderator.) on a finite mesh. The second part solves the integral equation, on a finite mesh, by a standard iterative scheme. Scattering, absorption, and source functions must be hand-computed or otherwise provided for the program.

PROBLEM DESCRIPTION

The finite mesh may have no more than 35 points (in which case the kernel will be evaluated at 35 X 35 points.)

LIMITATIONS OF THE CODE

"Effect of the Temperature of the Moderator on the Velocity Distribution of Neutrons with Numerical Calculations for H as Moderator" by E. P. Wigner and J. E. Wilkins, Jr., AECD-2275.

REFERENCES

MONTE CARLO REACTOR CALCULATION	S. Podgor	Naval Research Laboratory
TITLE	COG. ENGR.	LABORATORY
NAREC	In Use	Reactor and Nuclear Physics
COMPUTER	PROBLEM STATUS	CATEGORY
Spherical Homogeneous; 25 cm. radius GEOMETRY & DIMENSIONS	2 hours per 1500 histories TYPICAL COMPUTING TIME	Spatial distribution of thermal neutrons PARAMETERS SOLVED

Spherical homogeneous mixture of U 235 and H₂O of 25 cm. radius. Neutrons of 2 Mev energy are started at a certain distance from the center of the sphere and are followed until they (1) become thermal, or (2) escape from the sphere, or (3) are absorbed. For (1) the spatial distribution of thermal neutrons is obtained. For (2) the energy spectrum of escape neutrons is obtained. For (3) we have the total number absorbed.

PROBLEM DESCRIPTION

Limitations

- (1) Time dependence is not considered.
- (2) Inelastic scattering by U not considered.
- (3) Oxygen and uranium do not slow down neutrons.
- (4) Oxygen and uranium scatter isotropically in the laboratory system.

LIMITATIONS OF THE CODE

Written up in

"Report of Shielding Meeting of May, 1955, at Fort Belvoir, Va."

REFERENCES

INTEGRAL NETWORK
METHOD

Daniel Schiff
E. H. Bareiss (DTMB)

Westinghouse Atomic Power
Division
David Taylor Model Basin

TITLE

COG. ENGR.

LABORATORY

UNIVAC

Obsolete

Reactor and Nuclear
Physics

COMPUTER

PROBLEM STATUS

CATEGORY

One-dimensional
10.5 cm thick

16 hours

GEOMETRY &
DIMENSIONS

TYPICAL COMPUTING TIME

PARAMETERS SOLVED

Numerical solution of the integral form of the Transport equation. Two problems completed in multi-group, multi-region, one-dimensional slab cases, for both neutrons and gamma rays. Now doing analyses of numerical approximations in lattice spacing and angular divisions, machine calculation of test cases. Above figures refer to problem finished.

PROBLEM DESCRIPTION

This was a single problem, not an available code.

LIMITATIONS OF THE CODE

WAPD-TN-513; WAPD-P-663

WAPD-108 (obsolete)

REFERENCES

FLEXIBLE INTEGRAL
NETWORK

Daniel Schiff
E. H. Bareiss

Westinghouse Atomic Power
Division
David Taylor Model Basin

TITLE

COG. ENGR.

LABORATORY

NORC (NRL)

Programmed

Reactor and Nuclear
Physics

COMPUTER

PROBLEM STATUS

CATEGORY

One-dimensional
slab; Variable
thickness

30 minutes

GEOMETRY &
DIMENSIONS

TYPICAL COMPUTING TIME

PARAMETERS SOLVED

Numerical solution of transport equation in one energy; isotropic scattering, isotropic sources. Number of space points variable to a maximum value of 1000. Scattering and absorption cross sections and source strength variable at each space point. Space points separated by constant optical depth, ξ , which can assume a fixed value for a problem in the range $0.05 \leq \xi \leq 0.25$. Eight angular divisions. Sample problems have been completed (56-54 and 56-55), flexible code is ready for coding. Primarily for cell problems.

PROBLEM DESCRIPTION

Geometry and energy limitations given above.

LIMITATIONS OF THE CODE

REFERENCES

RE-23

E. McNulty

Argonne National Laboratory

TITLE

COG. ENGR.

LABORATORY

AVIDAC

Programmed

Reactor and Nuclear
Physics

COMPUTER

PROBLEM STATUS

CATEGORY

 τ = asymptotic pile period
as a function of the
excess reactivityGEOMETRY &
DIMENSIONS

TYPICAL COMPUTING TIME

PARAMETERS SOLVED

This program calculates a table of τ , for specified values of k_{ex} = excess reactivity, using the equation:

$$k_{ex} = \frac{\frac{l}{\tau} + \beta \sum_{i=1}^6 \frac{a_i}{1 + \lambda_i \tau}}{1 - \beta \sum_{i=1}^6 \frac{a_i}{1 + \lambda_i \tau}}$$

l = neutron lifetime
 β = fraction of neutrons delayed
 a_i = fraction of delayed neutrons
 emitted in i th group
 λ_i = decay constant for delayed
 neutron precursors

PROBLEM DESCRIPTION

LIMITATIONS OF THE CODE

REFERENCES

P-24

Multiple-Level Decay Chains	D. D. Dix	National Reactor Testing Station, Phillips Petroleum
TITLE	COG. ENGR.	LABORATORY
CPC	In Use	Reactor and Nuclear Physics
COMPUTER	PROBLEM STATUS	CATEGORY
Up to 5th level	45 sec./chain	Quantities of substances in multi-level decay chains
GEOMETRY & DIMENSIONS	TYPICAL COMPUTING TIME	PARAMETERS SOLVED

Radioactive decay chain quantities N_i described by the differential equations

$$\frac{dN_i}{dt} = k_i N_{i-1} - \lambda_i N_i \quad (i = 1, \dots, 5)$$

are computed directly.

PROBLEM DESCRIPTION

Limited R 5th level

LIMITATIONS OF THE CODE

None published. Refer to D. D. Dix, Phillips Petroleum Company, Idaho Falls, Idaho.

REFERENCES

Non linear Isotope Buildup and Decay (I-23021)	J. R. Triplett	Numerical Analysis Unit HAPO GE Richland
TITLE	COG. ENGR.	LABORATORY
702	Analyzed	Reactor and Nuclear Physics
COMPUTER	PROBLEM STATUS	CATEGORY
Heterogeneous Lattice with Cylindrical Fuel Rods GEOMETRY & DIMENSIONS	7 sec. per mesh point TYPICAL COMPUTING TIME	Reactivity, Temperature Coefficients, Yields of U-234, U-236, U-238, Pu 239, Pu-240, Pu-241, Pu242, fissions PARAMETERS SOLVED

Non linear isotope yield equations are solved by Runge-Kutta method. Resonance absorption is handled by an n-group modification of age theory formulas (an expanded version of the method of HW-32912). In addition, thermal cross sections are calculated by the energy-dependent blackness weighting method.

Reactivity and temperature coefficients may also be obtained as functions of exposure.

PROBLEM DESCRIPTION

Application is restricted to thermal reactors with cylindrical rod fuel elements or fuel elements representable as cylindrical rods, containing, apart from structural materials only uranium isotopes 235, 236, and 238 and plutonium isotopes 239 through 242; plus fission products. Radioactive decays of Np-239 and Pu-241 are not included. No restriction on initial composition or exposure, LIMITATIONS OF THE CODE or upon neutron temperature.

HW-33912 (Secret)
HW-36825 (Confidential)
Hanford Work Order I-24184

REFERENCES

Xenon & Iodine Concentration Calculation for Hanford Reactors (I-24119, 24179)

R. E. McGrath

Numerical Analysis Unit
HAPO
GE
Richland

TITLE

COG. ENGR.

LABORATORY

702

Debugged

Reactor and Nuclear Physics

COMPUTER

PROBLEM STATUS

CATEGORY

GEOMETRY & DIMENSIONS

TYPICAL COMPUTING TIME

Time Power Level

PARAMETERS SOLVED

Computes saturated xenon and iodine concentrations for different power levels, and corresponding transient concentrations at various times for each power level.

PROBLEM DESCRIPTION

Constants peculiar to Hanford reactors are used. However, it is possible to modify these constants to apply to other reactors.

LIMITATIONS OF THE CODE

HW 25565, Parts I through VI (Secret)
HW 32721 (Secret)
HW 33537 (Secret)

REFERENCES

PHY-8	J. F. Hall	Argonne National Laboratory
TITLE	COG. ENGR.	LABORATORY
AVIDAC	In Use	Reactor and Nuclear Physics
COMPUTER	PROBLEM STATUS	CATEGORY
	5 minutes	Buckling, Reflector Savings
GEOMETRY & DIMENSIONS	TYPICAL COMPUTING TIME	PARAMETERS SOLVED

This program fits the flux distributions $f(x) = \alpha [e^{-\beta x} - \gamma e^{\beta x}]$ or $f(x) = \alpha \cos \beta(x-\gamma)$ to experimental data: x_i, f_i, w_i where w_i is the weight of the experimental point. It finds the "best" (least squares sense) parameters, α, β, γ and an estimate of their probable errors.

PROBLEM DESCRIPTION

$N \leq 68$, where N is the number of data points

LIMITATIONS OF THE CODE

ANL-MH-98 - N. Rosenzweig and B. I. Spinrad to Distribution - July 26, 1954
 SUBJECT: Curve Fitting by Least Squares

REFERENCES

CLOUDY CRYSTAL BALL NEUTRON SCATTERING	B. H. Mount	Westinghouse Atomic Power Division
TITLE	COG. ENGR.	LABORATORY
CPC	In Use	Reactor and Nuclear Physics
COMPUTER	PROBLEM STATUS	CATEGORY
GEOMETRY & DIMENSIONS	For moments, 7 minutes Moments and complete angular distribution, 25 minutes TYPICAL COMPUTING TIME	1) Shape elastic and compound elastic scattering at given angle. 2) Partial elastic cross sections PARAMETERS SOLVED

- 1) Computes angular distribution of elastic scattering as a function of angle, initial energy, well depth, well radius, and complex potential component.
- 2) Computes differential cross sections and their first four Legendre polynomial expansion coefficients.
- 3) Computes penetrabilities to be used in inelastic scattering theory of Hauser and Feshback.

PROBLEM DESCRIPTION

Treats angular momentum up to $L = 7$.

LIMITATIONS OF THE CODE

Westinghouse Analytical Sect. Report #54-60.

REFERENCES

CLOUDY CRYSTAL BALL NEUTRON SCATTERING	H. Amster (Phys) B. H. Mount (Math)	Westinghouse Atomic Power Division
TITLE	COG. ENGR.	LABORATORY
UNIVAC	In Use	Reactor and Nuclear Physics
COMPUTER	PROBLEM STATUS	CATEGORY
GEOMETRY & DIMENSIONS	Roughly 6 sec per point of variables mentioned in (1) below TYPICAL COMPUTING TIME	1) Shape elastic and compound elastic scattering at given angle 2) Partial elastic cross sections PARAMETERS SOLVED

- 1) Computes angular distribution of elastic scattering as a function of angle, initial energy, well depth, well radius, and complex potential component.
- 2) Computes differential cross sections and their first four Legendre polynomial expansion coefficients.
- 3) Computes penetrabilities to be used in inelastic scattering theory of Hauser and Feshbach.

PROBLEM DESCRIPTION

- 1) Treats angular momentum up to $L = 12$.

LIMITATIONS OF THE CODE

REFERENCES

SURFACE-MODIFIED OPTICAL MODEL	H. Amster (Phys) L. M. Culpepper (Math)	Westinghouse Atomic Power Laboratory
TITLE	COG. ENGR.	LABORATORY
NORC	Analyzed	Reactor and Nuclear Physics
COMPUTER	PROBLEM STATUS	CATEGORY
GEOMETRY & DIMENSIONS	No estimate	Neutron cross sections
	TYPICAL COMPUTING TIME	PARAMETERS SOLVED

- 1) Computes cloudy-crystal-ball differential cross sections and their Legendre polynomial expansion coefficients.
- 2) Model has a complex square-well core with arbitrary complex potential in a surface layer.

PROBLEM DESCRIPTION

LIMITATIONS OF THE CODE

REFERENCES

End and Fast Source Corrections to Neutron Measure- ment in an exponential pile (I-24176)	D. E. Davenport	Numerical Analysis Unit HAPO GE Richland
TITLE	COG. ENGR.	LABORATORY
702	Debugged	Reactor and Nuclear Physics
COMPUTER	PROBLEM STATUS	CATEGORY
Parallelopiped		Diffusion Length
GEOMETRY & DIMENSIONS	TYPICAL COMPUTING TIME	PARAMETERS SOLVED

Computes corrections which must be made to flux measurement in an exponential pile due to fast neutrons and end effects. Pile must be parallelopiped with source(s) situated in certain manner.

PROBLEM DESCRIPTION

LIMITATIONS OF THE CODE

Hanford Work Order I-24176

REFERENCES

Cell Flux Calculations	Helen Rutgers	N.A.A., INC. Nuclear Engineering and Manufacturing Division
TITLE	COG. ENGR.	LABORATORY
I.B.M. 701 (Duet Code) Floating Point Arithmetic	In Use	Reactor Survey (One Dimension) (One Group)
COMPUTER	PROBLEM STATUS	CATEGORY
Infinite Co-axial Cylinder	One case is computed in approx. 2.00 minutes	Flux distribution & average flux in each region.
GEOMETRY & DIMENSIONS	TYPICAL COMPUTING TIME	PARAMETERS SOLVED

Applicable for One Group - Ten Region Diffusion Theory.

PROBLEM DESCRIPTION

Assumes all thermal neutron source equals zero except in the outer region.

LIMITATIONS OF THE CODE

None

REFERENCES

RS-1

RE 2-Group Spherical
Reactor Problem

Argonne National Laboratory

TITLE	COG. ENGR.	LABORATORY
IBM CPC	Obsolete Replaced by: RE-1	Reactor Survey (Two Group) (One Dimension)
COMPUTER	PROBLEM STATUS	CATEGORY
Sphere		Finds the root of the critical determinant
GEOMETRY & DIMENSIONS	TYPICAL COMPUTING TIME	PARAMETERS SOLVED

This program solves analytically the general 2-group, 3-region spherical reactor problem. Given the radii of the three regions and their material composition, it determines the root of the critical determinant.

PROBLEM DESCRIPTION

LIMITATIONS OF THE CODE

REFERENCES

RS-2

RE-1	M. Butler	Argonne National Laboratory
TITLE	COG. ENGR.	LABORATORY
AVIDAC	In Use	Reactor Survey (Two Group) (One Dimension)
COMPUTER	PROBLEM STATUS	CATEGORY
Sphere	45 Minutes	Solves for critical composition
GEOMETRY & DIMENSIONS	TYPICAL COMPUTING TIME	PARAMETERS SOLVED

This program solves analytically the general 2-group, 3-region spherical reactor problem. Given the radii of the three regions, their material composition in the form of volume fractions for up to and including 11 materials, and the fast and thermal cross sections for each material this program varies γ , the ratio in the core of the volume fraction of material #1/volume fraction of material #1 + volume fraction of material #2 until the critical solution is determined.

PROBLEM DESCRIPTION

The initial γ must be less than the critical γ . No more than 11 materials may be used.

LIMITATIONS OF THE CODE

RED Memoranda - M. Butler to R. Avery

SUBJECT: Programming of the Analytic Two-Group, Three-region Spherical
Geometry Reactor Calculation for the AVIDAC
Part I Criticality Determination 5/28/53
Part II Flux and Current Calculations

REFERENCES

Program B - Two-Region
Reflector Savings

D. S. Selengut

Aircraft Nuclear Propulsion
Department
General Electric Company
Cincinnati 15, Ohio

TITLE

COG. ENGR.

LABORATORY

IBM-701

In Use

Reactor Survey
(One dimension)
(Two Group)

COMPUTER

PROBLEM STATUS

PARAMETERS SOLVED

Sphere, slab, or
cylinder

1/2 - 1-1/2 minutes

Core dimension required for
criticality; flux and power
distribution

GEOMETRY &
DIMENSIONS

TYPICAL COMPUTING TIME

PARAMETERS SOLVED

Evaluates the analytic solution of the two-region, two-group diffusion equations. Output consists of tabulated flux and power distributions which are punched on cards for automatic plotting if requested.

PROBLEM DESCRIPTION

Two regions, with fissionable material in core only; two energy groups.

LIMITATIONS OF THE CODE

XDC 55-4-44

REFERENCES

RS-4

Program F - Digital Space Simulator	D. S. Selengut	Aircraft Nuclear Propulsion Department General Electric Company Cincinnati 15, Ohio
TITLE	COG. ENGR.	LABORATORY
IBM-701	In Use	Reactor Survey (One Dimension) (Two Group)
COMPUTER	PROBLEM STATUS	CATEGORY
Sphere, slab or cylinder (one dimensional)	3 minutes	Reactivity, flux and power distribution
GEOMETRY & DIMENSIONS	TYPICAL COMPUTING TIME	PARAMETERS SOLVED

Numerically solves the direct or adjoint two-group equations, subject to arbitrary boundary conditions for each group. Used for multiple reflectors, non-uniform fuel distribution, disadvantage factors, and central control rod calculations.

PROBLEM DESCRIPTION

Up to 20 materials may be distributed arbitrarily among up to 40 regions. Diffusion theory is assumed; strong absorbers are treated either as boundary conditions or by adjusting their cross sections on the basis of transport calculations.

LIMITATIONS OF THE CODE

XDC 55-4-44

REFERENCES

LMFR-I		Origin - Brookhaven National Laboratory Prog. & Code - New York University
TITLE	COG. ENGR.	LABORATORY
UNIVAC	In Use	Reactor Survey (One Dimension) (Two Group)
COMPUTER	COG. ENGR.	CATEGORY
Spherical, multiregion		Fast and slow flux critical core radius
GEOMETRY & DIMENSIONS	TYPICAL COMPUTING TIME	PARAMETERS SOLVED

This is a survey of a spherically symmetrical reactor with as many as five regions. The flux distributions are approximated by two group diff. equ. which are replaced by a diff. scheme. Initial guess is taken for core radius which is then solved for by iterating.

PROBLEM DESCRIPTION

LIMITATIONS OF THE CODE

REFERENCES

RS-6

Two Group - Two Region Axial Problem	Helen Rutgers	North American Aviation, Inc. Nuclear Engineering & Manu- facturing Division
TITLE	COG. ENGR.	LABORATORY
IBM 701 (Duet Code) (Floating point Arithmetic)	In Use	Reactor Survey (One Dimension) (Two Group)
COMPUTER	PROBLEM STATUS	CATEGORY
Finite cylinder with a finite reflector on top and bottom	One case is computed in approx. 1.85 min.	Height of the core required for criticality.
GEOMETRY & DIMENSIONS	TYPICAL COMPUTING TIME	PARAMETERS SOLVED

Two group diffusion equations are solved by technique of separation of variables with the assumption that axial flux is given by a $\cos z$ function. The result is the critical height of the core and the flux.

PROBLEM DESCRIPTION

The code is limited to Thermal Reactors

LIMITATIONS OF THE CODE

NAA-SR-MEMO-1021 Fillmore, Dr. F. L.

REFERENCES

Two Group - Two
Region Radial
Problem

Helen Rutgers

North American Aviation
Inc. Nuclear Engineering
and Manufacturing Division.

TITLE

COG. ENGR.

LABORATORY

IBM 701 Duet
Code. (Floating
point arithmetic)

In Use

Reactor Survey
(one dimension)
(Two Group)

COMPUTER

PROBLEM STATUS

CATEGORY

Finite
Cylinder with a
finite reflector
on side

One case is computed in 4.00 min.

Radius of core which is
required for criticality

GEOMETRY &
DIMENSIONS

TYPICAL COMPUTING TIME

PARAMETERS SOLVED

Two group diffusion equations are solved by technique of separation of variables with the assumption that radial flux is given by a $J_0(R)$ function.

The result is critical radius of the core and the flux.

PROBLEM DESCRIPTION

The code is limited to Thermal Reactors.

LIMITATIONS OF THE CODE

NAA-SR-MEMO-1021 Fillmore, Dr. F. L.

REFERENCES

RS-8

Two-Group Critical Mass Reactivity Calculations		National Reactor Testing Station (Phillips Petroleum)
TITLE	COG. ENGR.	LABORATORY
CPC	In Use	Reactor Survey (One Dimension) (Two Group)
COMPUTER	PROBLEM STATUS	CATEGORY
N-Regions Spherical; Cylindrical; slab	Cylinder 2-Reg. 45 min. 3-Reg. 3 hrs.	Other 15 min. 1 hr.
GEOMETRY & DIMENSIONS	TYPICAL COMPUTING TIME	PARAMETERS SOLVED

N-region reactor properties are calculated on the basis of the two neutron group, diffusion theory model. The boundary value problem is treated by calculating a series of matrix inversions, products and determinants.

PROBLEM DESCRIPTION

Arguments of Bessel Functions in cylindrical problems must be less than 30; Outer reflector assumed infinite, although finite reflector case is programmed.

LIMITATIONS OF THE CODE

D. D. Dix, IDO-16221 "Reactor Calculations on the CPC"
 IDO-16223 "Finite Representations of Bessel Functions"

A. S. Householder, and Garabedian, H. L., MonP-246, "Multi-Group, multi-Reflector Pile Theory"

REFERENCES

ONE-DIMENSIONAL TWO-GROUP CODE	R. P. Eddy (DTMB) R. R. Schiff	Westinghouse Atomic Power Division
TITLE	COG. ENGR.	LABORATORY
UNIVAC	In Use	Reactor Survey (One Dimension) (Two Group)
COMPUTER	PROBLEM STATUS	CATEGORY
One-dimensional, any coordinates, variable mesh spacing	Varies with problem 30 minutes average	Fast and slow flux distributions, eta Poison for criticality (optional)
GEOMETRY & DIMENSIONS	TYPICAL COMPUTING TIME	PARAMETERS SOLVED

Solutions of the steady-state two group diffusion equations with variable coefficients in one-dimension. A maximum of twenty-nine mesh points is available.

PROBLEM DESCRIPTION

LIMITATIONS OF THE CODE

WAPD-105

REFERENCES

RS-10

Program A - Bare Reactor Criticality	D. S. Selengut	Aircraft Nuclear Propulsion Department General Electric Company Cincinnati 15, Ohio
TITLE	COG. ENGR.	LABORATORY
IBM-701	Obsolete	Reactor Survey (One Dimension) (Two Group)
COMPUTER	Replaced by: Program C	PARAMETERS SOLVED
Unreflected cylinder	1-2 minutes	Reactivity, flux, slowing down density, two-group constants
GEOMETRY & DIMENSIONS	TYPICAL COMPUTING TIME	PARAMETERS SOLVED

Computes criticality using 18 lethargy intervals; includes lethargy dependent buckling and a transport leakage correction. Cross sections entered from punched cards.

PROBLEM DESCRIPTION

Core assumed homogeneous and unreflected; slowing down due mainly to hydrogen; up to ten elements present.

LIMITATIONS OF THE CODE

XDC 55-3-18

REFERENCES

RS-11

Program C - Bare
Reactor Criticality

D. S. Selengut

Aircraft Nuclear Propulsion
Department
General Electric Company
Cincinnati 15, Ohio

TITLE

COG. ENGR.

LABORATORY

IHM-701

In Use

Reactor Survey
(One Dimension)
(Two Group)

COMPUTER

PROBLEM STATUS

CATEGORY

Unreflected
Sphere, slab or
cylinder

1-1/2 - 6 minutes

Reactivity, flux, slowing down
density, two-group constants.

GEOMETRY &
DIMENSIONS

TYPICAL COMPUTING TIME

PARAMETERS SOLVED

Similar to program A, but includes alternative slowing down and two-group constant weighting procedures. Disadvantage factors and other heterogeneity corrections can be applied. Cross sections and subroutines are stored on magnetic tape.

PROBLEM DESCRIPTION

Unreflected core; up to ten elements present in fuel or moderator.

LIMITATIONS OF THE CODE

XDC 55-4-44

REFERENCES

RS-12

IDA	G. J. Habetler	Knolls Atomic Power Laboratory
TITLE	COG. ENGR.	LABORATORY
UNIVAC	in Use	Reactor Survey (one dimension) (three group)
COMPUTER	PROBLEM STATUS	CATEGORY
One-space-dimensional (slab, cylinder, sphere); three groups	20 seconds per source iteration for approximately 40 space points.	✓, fluxes (power curve, neutron balance quantities)
GEOMETRY & DIMENSIONS	TYPICAL COMPUTING TIME	PARAMETERS SOLVED

Finite-difference solution of the three-group problem

$$-D_i \left(\frac{d\phi_i}{dr} + \frac{r}{r} \frac{d\phi_i}{dr} \right) + A_i \phi_i = \chi_i \lambda + S_{i-1} \phi_{i-1} \quad i=1,2,3$$

$$\phi_i(r) = 0; \quad \phi_0(r) = 0; \quad \lambda = \nu \sum F_i \phi_i$$

Where the range of integration can be subdivided into at most 6 regions in each of which D_i , A_i , S_i , and F_i are known constants.

PROBLEM DESCRIPTION

1. Interfaces between regions of constant coefficients must fall on the finite-difference mesh.
2. Number regions ≤ 6 , only the first five of which may be cores.
3. Number of mesh points plus number of regions ≤ 58 .
4. Δr constant within a region; $\Delta r > \sqrt{r}$
5. At the present time, the coefficients D_i , A_i , etc. are input.

LIMITATIONS OF THE CODE

REFERENCES

PHY-4	D. Satkus	Argonne National Laboratory
TITLE	COG. ENGR.	LABORATORY
AVIDAC	In Use	Reactor Survey (one dimension) (three group)
Sphere	2 hours	Solves for the critical radius
GEOMETRY & DIMENSIONS	TYPICAL COMPUTING TIME	PARAMETERS SOLVED

This program is a 3-group Serber-Wilson calculation for a 2-region spherical reactor. Starting with a guessed source, and core and reflector radii, the program iterates on the source to compute the critical core radius.

PROBLEM DESCRIPTION

Core radius < 20 cm

LIMITATIONS OF THE CODE

ANL-5440 - AVIDAC Setup for Multigroup Serber-Wilson Calculations - H. Hummel

ANL Memorandum - D. Satkus to H. Hummel

SUBJECT: AVIDAC Method of Solution of Finite Tamper Serber-Wilson
Problem - 1/26/55

REFERENCES

RS-14

RE-2	H. Greenspan	Argonne National Laboratory
TITLE	COG. ENGR.	LABORATORY
IBM CPC	Obsolete	Reactor Survey (One Dimension) (Multi-Group)
COMPUTER	Replaced by: RE-2A PROBLEM STATUS	CATEGORY
Sphere	3 hours	Finds flux and source distribution
GEOMETRY & DIMENSIONS	TYPICAL COMPUTING TIME	PARAMETERS SOLVED

This program solves the one-dimensional multigroup reactor equations. Given a source distribution this program uses the diffusion equation $\nabla D \nabla \phi - \sigma_a \phi + S = 0$ with D , the diffusion coefficient, $(= 1/3 \sigma_{tr})$, $\sigma_a = \sigma_s + \sigma_{in} + \sigma_c + \sigma_f$, ϕ = neutron flux and S = the source term, replaced by a set of J coupled difference equations, to solve for the flux and the source produced by this flux distribution.

PROBLEM DESCRIPTION

Number of regions ≤ 10 , however, this may be changed without too much difficulty.

$$\Delta l^r = \Delta l^{r-1} \quad \Delta l^r = \text{interval size of region } l$$

LIMITATIONS OF THE CODE

Memoranda by J. Alexander and D. Okrent

SUBJECT: - Part I Theory for IBM Setup of Multigroup

II Instructions for IBM Setup of Multigroup in Spherical
Coordinates for Uniform Regions

1-23-53

REFERENCES

RS-15

RE-2A	H. Greenspan	Argonne National Laboratory
TITLE	COG. ENGR.	LABORATORY
IBM CPC	Obsolete	Reactor Survey (One Dimension) (Multi Group)
COMPUTER	PROBLEM STATUS Replaced by: RE-4, 7	CATEGORY
Sphere	3 hours	Finds flux and source distribution
GEOMETRY & DIMENSIONS	TYPICAL COMPUTING TIME	PARAMETERS SOLVED

This program solves the one-dimensional multigroup reactor equations. Given a source distribution this program uses the diffusion equation $\nabla D \nabla \phi - \sigma_a \phi + S = 0$ with D , the diffusion coefficient, ($= 1/3 \sigma_{tr}$), $\sigma_a = \sigma_s + \sigma_{in} + \sigma_c + \sigma_f$, ϕ = neutron flux and S = the source term, replaced by a set of J coupled difference equations, to solve for the flux and the source produced by this flux distribution.

PROBLEM DESCRIPTION

Number of regions ≤ 10 , however, this may be changed without too much difficulty.

LIMITATIONS OF THE CODE

REFERENCES

RS-16

RE-4	M. Butler E. Jutzi L. Sommerfeld	Argonne National Laboratory
TITLE	COG. ENGR.	LABORATORY
AVIDAC	In Use	Reactor Survey (One Dimension) (Multigroup)
Sphere R < 256 GEOMETRY & DIMENSIONS	1 hour TYPICAL COMPUTING TIME	Solves for critical composition PARAMETERS SOLVED

This program solves the one-dimensional multigroup reactor equations. The diffusion equation $\nabla D \nabla \phi - \sigma_a \phi + S = 0$ with D, the diffusion coefficient, ($= 1/3\sigma_{tr}$), $\sigma_a = \sigma_s + \sigma_{in} + \sigma_c + \sigma_f$, ϕ = neutron flux, and S = the source term, is replaced by a set of J coupled difference equations which are solved by iteration. Inelastic scattering is considered.

PROBLEM DESCRIPTION

$12 L + 8N \leq 306$	$1 \leq G \leq 4$	N = number of points L = number of regions
$\Delta \ell_r \leq \Delta \ell + 1^r$	$1 \leq Z \leq 10$	$\Delta \ell_r$ = interval size in region $\mathcal{L}$
$1 \leq J \leq 15$		J = number of energy groups
		G = number of energy groups into which inelastic scattering is considered
LIMITATIONS OF THE CODE		Z = number of materials

RED Memorandum - M. Butler to F. W. Thalgott

SUBJECT: Programming of the Multigroup Spherical Geometry Reactor
Calculation for the AVIDAC - 2-19-54

REFERENCES

RE-6	J. Hall M. Butler H. Gray	Argonne National Laboratory
TITLE	COG. ENGR.	LABORATORY
UNIVAC	In Use	Reactor Survey (One Dimension) (Multigroup)
COMPUTER	PROBLEM STATUS	CATEGORY
Cylinder $R < 300$ $H < 1000$	15 - 20 minutes	Solves for critical composition
GEOMETRY & DIMENSIONS	TYPICAL COMPUTING TIME	PARAMETERS SOLVED

This program solves the one-dimensional multigroup reactor equations. The diffusion equation $\nabla D \nabla \phi - \sigma_a \phi + S = 0$ with D , the diffusion coefficient, ($= 1/3 \sigma_{tr}$), $\sigma_a = \sigma_s + \sigma_{in} + \sigma_c + \sigma_f$, ϕ = neutron flux, and S = the source term, is replaced by a set of J coupled difference equations which are solved by iteration. Inelastic scattering is considered.

PROBLEM DESCRIPTION

$(4+G)N + (4+2G)L \leq 553$	$1 \leq G \leq 6$	N = number of points L = number of regions
$\Delta \ell_r \leq \frac{4}{L} 1^r$	$1 \leq Z \leq 10$	$\Delta \ell_r$ = interval size in region ℓ
$2 \leq J \leq 20$		J = number of energy groups
		G = number of energy groups into which inelastic scattering is considered.

LIMITATIONS OF THE CODE Z = number of materials

- ANL-5437 - UNIVAC Programs for the Solution of One-Dimensional Multigroup reactor Equations - M. Butler and J. Cook.
 ANL-5260 - RED Quarterly Report, 12/1/53-3/30/54, p.64
 ANL-5345 - RED Quarterly Report, 7/1/54-9/30/54, p.47
 ANL-5371 - RED Quarterly Report, 10/1/54-12/31/54, p.13
 ANL-5321 - On the application of Multigroup Diffusion Theory to Fast Critical Assemblies - D. Okrent

REFERENCES

RE-6X	M. Butler	Argonne National Laboratory
TITLE	COG. ENGR.	LABORATORY
UNIVAC	In Use	Reactor Survey (One Dimension) (Multigroup)
COMPUTER	PROBLEM STATUS	CATEGORY
Cylinder $R < 300$ $H < 1000$	15-20 minutes	Solves for critical composition
GEOMETRY & DIMENSIONS	TYPICAL COMPUTING TIME	PARAMETERS SOLVED

This program is a modification of RE-6, the general one-dimensional multigroup program for this geometry; it allows an unlimited number of materials and, if desired, the use of disadvantage factors in the computation of the effective thermal group cross sections.

PROBLEM DESCRIPTION

$$(4+G)N + (4+2G)L \leq 545$$

$$0 \leq \text{disadvantage factor} \leq 1$$

$$\Delta r \leq \Delta r^*$$

$$2 \leq J \leq 20$$

$$1 \leq G \leq 6$$

N = number of points, L = number of regions

Δr = interval size in region

J = number of energy groups

LIMITATIONS OF THE CODE G = number of energy groups into which inelastic scattering is considered

RED Memorandum - M. Butler to F. W. Thalgott

SUBJECT: UNIVAC Programs RE-6X, 7X, 8X - 5-31-55

REFERENCES

RE-7	J. Hall H. Gray L. Sommerfeld M. Butler	Argonne National Laboratory
TITLE	COG. ENGR.	LABORATORY
UNIVAC	In Use	Reactor Survey (One Dimension) (Multigroup)
COMPUTER	PROBLEM STATUS	CATEGORY
Sphere $R < 300$	15 - 20 minutes	Solves for critical composition
GEOMETRY & DIMENSIONS	TYPICAL COMPUTING TIME	PARAMETERS SOLVED

This program solves the one-dimensional multigroup reactor equations. The diffusion equation $\nabla D \nabla \phi - \sigma_a \phi + S = 0$ with D , the diffusion coefficient, ($= 1/3 \sigma_{tr}$), $\sigma_a = \sigma_s + \sigma_{in} + \sigma_c + \sigma_f$, ϕ = neutron flux, and S = the source term, is replaced by a set of J coupled difference equations which are solved by iteration. Inelastic scattering is considered.

PROBLEM DESCRIPTION

$$(4+G)N + (4+2G)L \leq 521$$

$$\Delta \ell^r < 10 \Delta \ell_{+1}^r$$

$$2 \leq J \leq 20$$

$$1 \leq G \leq 6$$

$$1 \leq Z \leq 10$$

N = number of points L = number of regions

$\Delta \ell^r$ = interval size in region $\mathcal{L}$

J = number of energy groups

G = number of energy groups into which inelastic scattering is considered

Z = number of materials

LIMITATIONS OF THE CODE

ANL-5437 - UNIVAC Programs for the Solution of One-Dimensional Multigroup Reactor Equations - M. Butler and J. Cook

ANL-5260 - RED Quarterly Report, 12/1/53-3/30/54, p. 64

ANL-5345 - RED Quarterly Report, 7/1/54-9/30/54, p. 47

ANL-5371 - RED Quarterly Report, 10/1/54-12/31/54, p. 13

ANL-5321 - On the Application of Multigroup Diffusion Theory to Fast Critical Assemblies - D. Okrent

RE-7X	M. Butler	Argonne National Laboratory
TITLE	COG. ENGR.	LABORATORY
UNIVAC	In Use	Reactor Survey (One Dimension) (Multigroup)
COMPUTER	PROBLEM STATUS	CATEGORY
Sphere $R < 300$	15-20 Minutes	Solves for critical Composition
GEOMETRY & DIMENSIONS	TYPICAL COMPUTING TIME	PARAMETERS SOLVED

This program is a modification of RE-7, the general one-dimensional multigroup program for this geometry. It allows an unlimited number of materials and, if desired, the use of disadvantage factors in the computation of the effective thermal cross sections.

PROBLEM DESCRIPTION

$$(4+G)N + (4+2G)L \leq 512 \quad 0 \leq \text{disadvantage factor} < 1$$

$$\Delta r < 10 \Delta r_1$$

$$2 \leq J \leq 20$$

$$1 \leq G \leq 6$$

G = number of energy groups
into which inelastic
scattering is considered

N = number of points
L = number of regions
 Δr = interval size in
region
J = number of energy
groups

LIMITATIONS OF THE CODE

RED Memorandum - M. Butler to F. W. Thalgott

SUBJECT: UNIVAC Programs RE-6X, 7X, 8X - 5/31/55

REFERENCES

RS-21

RE-8	M. Butler H. Gray E. Jutzi	Argonne National Laboratory
TITLE	COG. ENGR.	LABORATORY
UNIVAC	In Use	Reactor Survey (One Dimension) (Multigroup)
COMPUTER	PROBLEM STATUS	CATEGORY
Slab $Z < 300$	15-20 minutes	Solves for critical composition or lateral buckling
GEOMETRY & DIMENSIONS	TYPICAL COMPUTING TIME	PARAMETERS SOLVED

This program solves the one-dimensional multigroup reactor equations. The diffusion equation $\nabla D \nabla \phi - \sigma_a \phi + S = 0$ with D , the diffusion coefficient ($= 1/3 \sigma_{tr}$), $\sigma_a = \sigma_s + \sigma_{in} + \sigma_c + \sigma_f$, ϕ = neutron flux, and S = the source term, is replaced by a set of J coupled difference equations which are solved by iteration. Inelastic scattering is considered.

PROBLEM DESCRIPTION

$(4+G)N + (4+2G)L \leq 553$ $1 \leq Z \leq 10$ N = number of points
 $\Delta/r < 10 \Delta_{l+1}^r$ L = number of regions
 $2 \leq J \leq 20$ Z = number of materials Δ/r = interval size in region l
 $1 \leq G \leq 6$ J = number of energy groups
 G = number of energy groups into which inelastic scattering is considered.

LIMITATIONS OF THE CODE

ANL-5437 - UNIVAC Programs for the Solution of One-Dimensional Multigroup Reactor Equations - M. Butler and J. Cook.
 ANL-5260 - RED Quarterly Report - 13/1/53-3/30/54, p. 64
 ANL-5345 - RED Quarterly Report - 7/1/54-9/30/54, p. 47
 ANL-5371 - Red Quarterly Report - 10/1/54-12/31/54, p. 13
 ANL-5321 - On the Application of Multigroup Diffusion Theory to Fast Critical Assemblies - D. Okrent

RE-8X	M. Butler	Argonne National Laboratory
TITLE	COG. ENGR.	LABORATORY
UNIVAC	In Use	Reactor Survey (One Dimension) (Multigroup)
COMPUTER	PROBLEM STATUS	CATEGORY
Slab Z < 300	15-20 minutes	Solves for critical composition
GEOMETRY & DIMENSIONS	TYPICAL COMPUTING TIME	PARAMETERS SOLVED

This program is a modification of RE-8, the general one-dimensional multigroup program for this geometry. It allows an unlimited number of materials and, if desired, the use of disadvantage factors in the computation of the effective thermal group cross sections.

PROBLEM DESCRIPTION

$$(4+G)N + (4+2G)L \leq 544$$

$$\Delta l_r < 10 \Delta l + 1r$$

$$2 \leq J \leq 20$$

$$1 \leq G \leq 6$$

N = number of points L = number of regions

Δl_r = interval size in region ℓ

J = number of energy groups

G = number of energy groups into which inelastic scattering is considered

LIMITATIONS OF THE CODE $0 < \text{disadvantage factor} < 1$

RED Memorandum - M. Butler to F. W. Thalgott

SUBJECT: UNIVAC Programs RE-6X, 7X, 8X - 5/31/55

REFERENCES

RE-26	M. Butler E. Jutzi	Argonne National Laboratory
TITLE	COG. ENGR.	LABORATORY
UNIVAC	In Use	Reactor Survey (One Dimension) (Multigroup)
COMPUTER	PROBLEM STATUS	CATEGORY
Cylinder R < 300 H < 1000 GEOMETRY & DIMENSIONS	15 - 20 Minutes TYPICAL COMPUTING TIME	Solves for critical composition PARAMETERS SOLVED

This program solves the one-dimensional multigroup adjoint equations. This is the adjoint equivalent of RE-6.

PROBLEM DESCRIPTION

$$(4+G)N + (4+2G)L \leq 546$$

$$\Delta x_r \leq \Delta x + 1^r$$

$$2 \leq J \leq 20$$

$$1 \leq G \leq 6$$

$$1 \leq Z \leq 10$$

LIMITATIONS OF THE CODE

N = number of points L = number of regions

Δx_r = interval size in region r

J = number of energy groups

G = number of energy groups into which inelastic scattering is considered

Z = number of materials

RED Memorandum - D. Okrent to PBR File

SUBJECT: Programming of Multigroup Adjoint Solution - 1/11/55

RED Memorandum - M. Butler to F. W. Thalgott

SUBJECT: Modification of UNIVAC Input Data Sheets for the Adjoint Solutions - 2/17/55

ANL-5437 - UNIVAC Programs for the Solution of One-Dimensional Multigroup Reactor Equations - M. Butler and J. Cook

REFERENCES

RS-24

RE-27	M. Butler E. Jutzi	Argonne National Laboratory
TITLE	COG. ENGR.	LABORATORY
UNIVAC	In Use	Reactor Survey (One Dimension) (Multigroup)
COMPUTER	PROBLEM STATUS	CATEGORY
Sphere $R \leq 300$	15 - 20 Minutes	Solves for critical composition
GEOMETRY & DIMENSIONS	TYPICAL COMPUTING TIME	PARAMETERS SOLVED

This program solves the one-dimensional multigroup adjoint equation. This is the adjoint equivalent of RE-7.

PROBLEM DESCRIPTION

$$(4+G)N + (4+2G)L \leq 514$$

$$\Delta \ell_r \leq 10 \Delta \ell_{r+1}$$

$$2 \leq J \leq 20$$

$$1 \leq G \leq 6$$

$$1 \leq Z \leq 10$$

N = number of points L = number of regions
 $\Delta \ell_r$ = interval size in region
 J = number of energy groups
 G = number of energy groups into which inelastic scattering is considered
 Z = number of materials

LIMITATIONS OF THE CODE

RED Memorandum - D. Okrent to PBR File

SUBJECT: Programming of Multigroup Adjoint Solution 1/11/55

RED Memorandum - M. Butler to F. W. Thalgott

SUBJECT: Modification of UNIVAC Input Data Sheets for the Adjoint Solutions - 2/17/55

REFERENCES: ANL-5437 - UNIVAC Programs for the Solution of One-Dimensional Multigroup Reactor Equations - M. Butler and J. Cook

RE-28	M. Butler E. Jutzi	Argonne National Laboratory
TITLE	COG. ENGR.	LABORATORY
UNIVAC	In Use	Reactor Survey (One Dimension) (Multigroup)
COMPUTER	PROBLEM STATUS	CATEGORY
Slab $Z \leq 300$	15-20 minutes	Solves for critical composition or lateral buckling.
GEOMETRY & DIMENSIONS	TYPICAL COMPUTING TIME	PARAMETERS SOLVED

This program solves the one-dimensional multigroup adjoint equations. This is the adjoint equivalent of RE-8.

PROBLEM DESCRIPTION

$$(4+G)N + (4+2G)L \leq 546$$

$$2 \leq J \leq 20$$

$$1 \leq G \leq 6$$

$$1 \leq Z \leq 10$$

LIMITATIONS OF THE CODE

N = number of points L = number of regions

Δl_r = interval size in region L

J = number of energy groups

G = number of energy groups into which
inelastic scattering is considered

Z = number of materials

RED Memorandum - D. Okrent to PBR File

SUBJECT: Programming of Multigroup Adjoint Solution - 1/11/55

RED Memorandum: M. Butler to F. W. Thalgott

SUBJECT: Modification of UNIVAC Input Data Sheets for the Adjoint
Solutions - 2/17/55

ANL-5437 - UNIVAC Programs for the Solution of One-Dimensional Multigroup

REFERENCES Reactor Equations - M. Butler and J. Cook

RS-26

Program D - Spherical Multigroup	E. D. Nix	Aircraft Nuclear Propulsion Department General Electric Company Cincinnati 15, Ohio
TITLE	COG. ENGR.	LABORATORY
IBM-701	In Use	Reactor Survey (One Dimension) (Multigroup)
COMPUTER	PROBLEM STATUS	CATEGORY
Sphere (One dimensional)	20 Minutes	Reactivity, flux, and power distribution
GEOMETRY & DIMENSIONS	TYPICAL COMPUTING TIME	PARAMETERS SOLVED

Computes the reactivity of a three-region spherical reactor.

PROBLEM DESCRIPTION

Continuous slowing down; fissionable material limited to one of the three regions; 18 lethargy intervals; up to 50 space intervals.

LIMITATIONS OF THE CODE

XDC 55-4-26

REFERENCES

RS-27

Program G - General Multigroup	D. S. Selengut	Aircraft Nuclear Propulsion Department General Electric Company Cincinnati, Ohio
TITLE	COG. ENGR.	LABORATORY
IBM 701	Coded	Reactor Survey (One Dimension) (Multigroup)
COMPUTER	PROBLEM STATUS	CATEGORY
Sphere, slab, or cylinder (one dimensional) GEOMETRY & DIMENSIONS	TYPICAL COMPUTING TIME	Reactivity, flux, and power distribution PARAMETERS SOLVED

Will be an 18 group version of Program F.

PROBLEM DESCRIPTION

Same as for Program F.

LIMITATIONS OF THE CODE

REFERENCES

RS-28

OMAR	G. J. Habetler	Knolls Atomic Power Laboratory
TITLE	COG. ENGR.	LABORATORY
UNIVAC	In Use	Reactor Survey (One Dimension) (Multigroup)
COMPUTER	PROBLEM STATUS	CATEGORY
One-space dimensional (slab, cylinder, sphere); fifteen groups GEOMETRY & DIMENSIONS	80 seconds per source iteration for 40 space points TYPICAL COMPUTING TIME	V, fluxes, (power curve, neutron balance quantities) PARAMETERS SOLVED

Finite-difference solution of

$$-D \left(\frac{d^2 \phi}{dr^2} + \frac{\phi}{r} \frac{d\phi}{dr} \right) + A\phi = - \left(\frac{\partial g}{\partial u} + \frac{\partial \eta}{\partial u} \right) + \chi \phi; 0 \leq r \leq R, 0 \leq u \leq u_T$$

$$\frac{\partial \phi}{\partial u} + \phi = \Lambda \cdot \phi \quad \phi(R, u) = 0$$

$$q = S \cdot \phi; \phi(r, 0) = 0$$

$$S = \nu \int_0^{u_T} F \phi du$$

where the range of integration can be subdivided into at most 6 regions in each of which D, A, S, F, and Λ are known functions of u alone.

PROBLEM DESCRIPTION

1. Interfaces between regions of constant coefficients must fall on finite-difference mesh.
2. Number of regions ≤ 6 , only the first 5 of which can be cores.
3. Number of mesh points plus number of regions ≤ 58 .
4. Δr constant within a region; $\Delta r > \sqrt{.1}$
5. Group-averaged microscopic cross sections on tape.

LIMITATIONS OF THE CODE

REFERENCES

650 MULTIGROUP	G. J. Habetler	Knolls Atomic Power Laboratory
TITLE	COG. ENGR.	LABORATORY
IBM 650	In Use	Reactor Survey (One Dimension) (Multigroup)
COMPUTER	PROBLEM STATUS	CATEGORY
One-Space-Dimension (slab, cylinder, sphere); lethargy	1 group of approximately 40 space points in 1 minute.	γ ; neutron flux (per space point and group); power curve.
GEOMETRY & DIMENSIONS	TYPICAL COMPUTING TIME	PARAMETERS SOLVED

Finite - Difference Solution of

$$-D\left(\frac{\partial \phi}{\partial r} + \frac{r}{r} \frac{\partial \phi}{\partial r}\right) + A\phi = -\left(\frac{\partial g}{\partial u} + \frac{\partial \eta}{\partial u}\right) + \chi g \quad 0 \leq r < R$$

$$\frac{\partial \phi}{\partial u} + \phi = \Lambda \cdot \phi \quad \phi(R, u) = 0 \quad 0 < u < u_T$$

$$g = S \cdot \phi \quad \phi(r, 0) = 0 \quad S = \nu \int_0^{u_T} F \cdot \phi du$$

Where the range of integration can be subdivided into regions in each of which D , A , S , F , and Λ are known functions of u alone. The inclusion of inelastic scattering is being undertaken at the present time.

PROBLEM DESCRIPTION

1. Interfaces between regions of constant coefficients must fall on finite-difference mesh.
2. Number of space-mesh points plus number of regions ≤ 100 .
3. Number of groups, i.e. subdivisions in u -direction ≤ 50 .
4. At the present time no coding is available for editing neutron balance quantities, i.e. total absorptions per region and group, etc.

LIMITATIONS OF THE CODE

Reports in preparation.

REFERENCES

Multigroup Reactor Analysis (Speed Co.)	Melba Nead	North American Aviation, Inc. Nuclear Engineering & Manufacturing Division
TITLE	COG. ENGR.	LABORATORY
701	Debugged	Reactor Survey (One Dimension) (Multigroup)
COMPUTER	PROBLEM STATUS	CATEGORY
Spherical GEOMETRY & DIMENSIONS	20 minutes TYPICAL COMPUTING TIME	 PARAMETERS SOLVED

Iterative method for obtaining fluxes and critical neutron multiplication ratio.

PROBLEM DESCRIPTION

This designation of 8 groups, 2 regions, and 32 space points is fixed.

LIMITATIONS OF THE CODE

NAA-SR-1034

REFERENCES

RS-31

Multigroup Reactor Analysis (Duet)	Melba Nead	North American Aviation Inc. Nuclear Engineering & Mfg. Division
TITLE	COG. ENGR.	LABORATORY
701	Coded	Reactor Survey (One Dimension) (Multigroup)
COMPUTER	PROBLEM STATUS	CATEGORY
Spherical	20 minutes	
GEOMETRY & DIMENSIONS	TYPICAL COMPUTING TIME	PARAMETERS SOLVED

Iterative method designed to include:

1. Listing of total absorption, inelastic absorption fission, leakage, degradation fission source, and inelastic scattering source for each group & each region.
2. Listing of final assumed & computed spatial fission distribution.
3. Listing of V_o and V_c
4. Listing of fluxes.

PROBLEM DESCRIPTION

This calculation allows a maximum of

15 groups
10 regions
90 sp. points.

LIMITATIONS OF THE CODE

REFERENCES

RS-32

EYEWASH

Oak Ridge National Laboratory

TITLE	COG. ENGR.	LABORATORY
UNIVAC	In Use	Reactor Survey (One Dimension) (Multigroup)
COMPUTER	PROBLEM STATUS	CATEGORY
Spherical, Multiregion	15 Min./Case	Critical Size
GEOMETRY & DIMENSIONS	TYPICAL COMPUTING TIME	PARAMETERS SOLVED

This is a general one-dimensional multi-group reactor code for calculating spherically symmetric reactors with elastic scattering and for all elements (including H₂).

PROBLEM DESCRIPTION

LIMITATIONS OF THE CODE

ANP 58
ORNL 1493

REFERENCES

RS-33

ONE-DIMENSIONAL- MULTI-GROUP CODE	R. B. Smith (Math) E. Gelbard (Phys)	Westinghouse Atomic Power Division
BUNNY		
TITLE	COG. ENGR.	LABORATORY
UNIVAC	Coded	Reactor Survey (One Dimension) (Multigroup)
COMPUTER	PROBLEM STATUS	PARAMETERS SOLVED
Slab, Cylinder, Sphere; one- dimension GEOMETRY & DIMENSIONS	TYPICAL COMPUTING TIME	PARAMETERS SOLVED

This code is to be used in the analysis of one-dimensional multi-region reactors, moderated primarily by hydrogen. Finite slabs and cylinders may be treated by inserting artificial buckling terms. Inelastic scattering and Breit-Wigner resonance absorption effects are included in the multi-group equations; self-shielding factors inserted as input data. Provisions have been made for calculation of slowing down densities due to point sources. Twelve separate regions with cross sections constant in each. Minimum lattice point separation is 0.1 cm. Roughly 400 space points maximum, 80 groups maximum.

PROBLEM DESCRIPTION

- 1) Selengut-Goertzel approximation used in analysis.
- 2) Symmetry about the origin required.

LIMITATIONS OF THE CODE

WAPD-PM-14

REFERENCES

RS-34

TWO-DIMENSIONAL FEW-GROUP CODE	E. Cuthill (DTMB) R. S. Varga	Westinghouse Atomic Power Division
TITLE	COG. ENGR.	LABORATORY
UNIVAC	In Use	Reactor Survey (Two Dimension) (Few Group)
COMPUTER	PROBLEM STATUS	CATEGORY
Cartesian coordinates, uniform spacing GEOMETRY & DIMENSIONS	Not yet determined	Flux distributions, eta
	TYPICAL COMPUTING TIME	PARAMETERS SOLVED

Solution of the steady-state few-group equations, with variable coefficients in a two-dimensional Cartesian geometry. A mesh of $59 \times n$ is expected, where n is at most 118. This code is intended to supersede the two-dimensional space simulators now in use at WAPD.

PROBLEM DESCRIPTION

See WAPD-LSR(P)-30

LIMITATIONS OF THE CODE

WAPD-105
WAPD-LSR(P)-30

REFERENCES

RS-35

MUG II	H. R. Stark	Knolls Atomic Power Laboratory
TITLE	COG. ENGR.	LABORATORY
UNIVAC	In Use	Reactor Survey (Two Dimension) (Multigroup)
COMPUTER	PROBLEM STATUS	CATEGORY
$(r, z), (r, \theta)$ two space, one lethargy	5 hours for 15 group problem on intermediate mesh (30 x 30).	Power surface. Flux Distribution Reactivity
GEOMETRY & DIMENSIONS	TYPICAL COMPUTING TIME	PARAMETERS SOLVED

Numerical solution of age diffusion equation

$$-D\nabla^2\phi + A\phi = -\left(\frac{\partial q}{\partial u} + \frac{\partial \eta}{\partial u}\right) + \chi\phi$$

ϕ and $D\nabla_n\phi$ are continuous across any surface

$\phi = 0$ or $-D\nabla_n\phi = \Sigma\phi$ at boundaries

q = slowing down by heavy atoms
 η = slowing down by hydrogen atoms

PROBLEM DESCRIPTION

1. Model must be separable by at most seven co-ordinate surfaces along each axis into regions of homogeneous composition.
2. Input consists of macroscopic group data. No routine exists for generating these.
3. Space interval is constant along each axis.
4. (r, θ) often gives convergence troubles.
5. No good edit exists.

LIMITATIONS OF THE CODE

KAPL 950 KAPL 1245
KAPL 1068 KAPL 1321
KAPL 1096
KAPL 1172

REFERENCES

RS-36

TOSPY	E. Wachspress	Knolls Atomic Power Laboratory
TITLE	COG. ENGR.	LABORATORY
NORC or UNIVAC	Analyzed	Reactor Survey (Two Dimension) (Multigroup)
COMPUTER	PROBLEM STATUS	CATEGORY
X-Y r-z r- θ	Unknown	Fluxes, power, reactivity, burnout, neutron accounting
GEOMETRY & DIMENSIONS	TYPICAL COMPUTING TIME	PARAMETERS SOLVED

This is a general two-space-dimension multigroup coding. Extension of MUG II to include thin regions such as H₂O gaps and control rods. Also includes variable mesh spacing. Will do any number of energy groups.

Pointwise burnout to be calculated by machine from power distribution.

PROBLEM DESCRIPTION

Material boundaries must fall on mesh lines. Values for slowing down parameters used in Scheme A must be supplied for each problem.

LIMITATIONS OF THE CODE

TOSPY Memoranda from KAPL and NYU. MUG II references applicable also. Thin region treatment in KAPL 1318.

REFERENCES

Three Dimensional Two Group Reactor Problem		National Bureau of Standards Computation Laboratory
TITLE	COG. ENGR.	LABORATORY
SEAC	Debugged	Reactor Survey (Three Dimension) (Two Group)
COMPUTER	PROBLEM STATUS	CATEGORY
Reactor has prismatic shape cartesian co- ordinates used GEOMETRY & DIMENSIONS	Not yet known	
	TYPICAL COMPUTING TIME	PARAMETERS SOLVED

The code solves a set of difference equations which approximate the two differential equations of elliptic type and the boundary conditions for the low and high energy level flux in a prismatic reactor consisting of three homogeneous regions.

PROBLEM DESCRIPTION

The mesh length of the difference net cannot be decreased since all the fluxes are stored internally so that the number of mesh points is limited by the size of the memory.

Code is intended to be experimental in nature and serves to develop techniques for the solution of three dimensional problems of this type.

LIMITATIONS OF THE CODE

REFERENCES

RS-38

ACHILLES
(Neutron Shielding
of a plane sym-
metric source by
hydrogen.)

H. B. Keller
Jack Heller

AEC Computing Facility
New York University

TITLE

COG. ENGR.

LABORATORY

UNIVAC

Analyzed:

Coded:

In Use: X

Programmed:

Shielding

COMPUTER

Debugged:

Obsolete:

PROBLEM STATUS

Replaced by:

CATEGORY

Plane Geometry
(Symmetric)

50 hours for 100 Cm. shield and
50 energy steps.

GEOMETRY &
DIMENSIONS

TYPICAL COMPUTING TIME

PARAMETERS SOLVED

Direct integration of neutron Boltzmann Equation. Solves for the distribution of neutrons in space, energy, and angle. During the edit, the first four spherical harmonic (expansion in angle) coefficients are evaluated.

PROBLEM DESCRIPTION

LIMITATIONS OF THE CODE

H. B. Keller & Jack Heller

NYO 6481 "On the numerical integration of the neutron transport equation."

REFERENCES

Neutron Diffusion	R. B. Theus	Naval Research Laboratory
TITLE	COG. ENGR.	LABORATORY
NAREC	Analyzed: Coded: In Use: ☒ Programmed: Debugged: Obsolete:	Shielding
COMPUTER	PROBLEM STATUS Replaced by:	CATEGORY
Infinite Medium	0.5 Sec/Hist.	
GEOMETRY & DIMENSIONS	TYPICAL COMPUTING TIME	PARAMETERS SOLVED

The one velocity neutron diffusion (with absorption) problem was selected to examine different methods of finding transport solutions. Moment method, biased and unbiased Monte Carlo, and exact solutions were compared for a variety of cases.

PROBLEM DESCRIPTION

LIMITATIONS OF THE CODE

Naval Research Laboratory Reports Nos. 4421, 4420, 4386, and 4514.

REFERENCES

Program O-G	W. E. Edwards J. E. MacDonald	Aircraft Nuclear Propulsion Department General Electric Company Cincinnati 15, Ohio
TITLE	COG. ENGR.	LABORATORY
IBM-701	Analyzed: Coded: In Use: ☒ Programmed: Debugged: Obsolete:	Shielding
COMPUTER	PROBLEM STATUS Replaced by:	CATEGORY
Cylindrical	5 sec per source-receiver path	Neutron dose rate in and around reactor shield
GEOMETRY & DIMENSIONS	TYPICAL COMPUTING TIME	PARAMETERS SOLVED

Combines infinite media point kernel with integration over cylindrical volume distributed sources. Evaluates path length and material compositions along path penetrating cylindrical regions coaxial with reactor axis so that thicknesses of each material encountered by a ray from the source are determined. This information is then combined with appropriate effective removal cross sections to determine the attenuation along the path. A point kernel based on the Welton-Albert simplified theory of neutron attenuation is currently in use. The program can distinguish shield boundaries.

PROBLEM DESCRIPTION

Only coaxial cylindrical geometry can be treated. Non-cylindrical regions must be handled by suitable approximations to cylindrical geometry. The program can accommodate up to 50 regions (a region is characterized by a composition different from the hydrogenous media comprising the bulk of the shield.) Also the product of the number of materials and the number of regions cannot exceed 500. Limitations inherent in the Welton-Albert theory necessarily hold true for the program.

LIMITATIONS OF THE CODE

XDC 55-6-109

REFERENCES

Program 1-G

W. E. Edwards
J. E. MacDonald

Aircraft Nuclear Propulsion
Department
General Electric Company
Cincinnati 15, Ohio

TITLE	COG. ENGR.	LABORATORY
IBM-701	Analyzed: Coded: In Use: ☒ Programmed: Debugged: Obsolete:	Shielding
COMPUTER	PROBLEM STATUS Replaced by:	CATEGORY
Cylindrical	5 sec per source-receiver path	Neutron dose rate in and around reactor shield
GEOMETRY & DIMENSIONS	TYPICAL COMPUTING TIME	PARAMETERS SOLVED

Same as program O-G except that integration over source distribution is not included -- coordinates of source point as well as coordinates of receiver points must be included.

PROBLEM DESCRIPTION

Same as for program O-G

LIMITATIONS OF THE CODE

XDC 55-6-109

REFERENCES

S-4

Program 2-F	W. E. Edwards J. E. MacDonald	Aircraft Nuclear Propulsion Department General Electric Company Cincinnati 15, Ohio
TITLE	COG. ENGR.	LABORATORY
IBM-701	Analyzed: Coded: In Use: ☒ Programmed: Debugged: Obsolete:	Shielding
COMPUTER	PROBLEM STATUS Replaced by:	CATEGORY
GEOMETRY & DIMENSIONS	4 seconds TYPICAL COMPUTING TIME	Neutron dose rates in and around reactor shield. PARAMETERS SOLVED

Calculates point kernel for program O-G. Path length and material composition must be provided.

PROBLEM DESCRIPTION

LIMITATIONS OF THE CODE

XDC 55-6-109

REFERENCES

Program 2-G

W. E. Edwards
J. E. MacDonald

Aircraft Nuclear Propulsion
Department
General Electric Company
Cincinnati 15, Ohio

TITLE	COG. ENGR.	LABORATORY
IBM-701	Analyzed: Coded: ☒	Shielding
COMPUTER	In Use: Programmed:	CATEGORY
	Debugged: Obsolete:	
	PROBLEM STATUS Replaced by:	
Cylindrical	Undetermined	Gamma dose rates in and around reactor shield
GEOMETRY & DIMENSIONS	TYPICAL COMPUTING TIME	PARAMETERS SOLVED

Combines infinite media point kernel with integration over cylindrical volume distributed source and source energy. Evaluates path lengths and material compositions along path penetrating cylindrical regions coaxial with reactor axis so that thicknesses of each material encountered by a ray from the source are determined. This information is then combined with appropriate gamma absorption coefficients to determine the attenuation along a path an uncollided point kernel in conjunction with NDA buildup factors is coded. The buildup factor data has been fitted using third degree polynomials. Buildup factors for different adjoining media may be either multiplied together or calculated for equivalent shielding thickness as desired.

PROBLEM DESCRIPTION

A maximum of ten energy groups is allowed. The program can accommodate up to 50 regions. Also, the product of the number of materials and the number of regions cannot exceed 50. A maximum product of five buildup factors can be used. Limitations inherent in the use of infinite media point kernels and NDA buildup factors hold.

LIMITATIONS OF THE CODE

XDC 55-6-109

REFERENCES

Gamma Ray Diffusion, Shallow Penetration	R. B. Theus	Naval Research Laboratory
TITLE	COG. ENGR.	LABORATORY
NAREC	Analyzed: Coded: In Use: ☒ Programmed: Debugged: Obsolete:	Shielding
COMPUTER	PROBLEM STATUS Replaced by:	CATEGORY
Semi-infinite and Slab	1 sec/hist	Plane parallel .66 Mev γ -Rays in water
GEOMETRY & DIMENSIONS	TYPICAL COMPUTING TIME	PARAMETERS SOLVED

The effect of boundaries on the intensity of the spectral and directional distributions of the diffusing radiation was investigated by Monte Carlo Techniques.

PROBLEM DESCRIPTION

The exact physical analogue was found to yield satisfactory statistics for penetration distances of 5 mean free paths or less.

LIMITATIONS OF THE CODE

Stochastic Estimates of γ -Ray Penetration.

NRL Report #4412

REFERENCES

Gamma Ray Diffusion, Deep Penetration	R. B. Theus	Naval Research Laboratory
TITLE	COG. ENGR.	LABORATORY
NAREC	Analyzed: Coded: In Use: ☒ Programmed: Debugged: Obsolete:	Shielding
COMPUTER	PROBLEM STATUS Replaced by:	CATEGORY
Infinite Media, Plane Parallel Source	1 Sec/Hist.	6 Mev Photons in Iron
GEOMETRY & DIMENSIONS	TYPICAL COMPUTING TIME	PARAMETERS SOLVED

Several methods of biased or optimum sampling techniques were investigated. These included sampling from biased distributions of scattering angles as well as free paths and the use of the "exponential transform."

PROBLEM DESCRIPTION

Spectral intensities were estimated for penetration distances as large as 16 mean free paths.

LIMITATIONS OF THE CODE

Stochastic Estimates of γ -Ray Penetration.

NRL Report #4412

REFERENCES

WAPD NEPTUNE	A. Foderaro	Westinghouse Atomic Power Division
TITLE	COG. ENGR.	LABORATORY
UNIVAC	Analyzed: Coded:	Shielding
	In Use: ☒ Debugged:	
	Programmed: Obsolete:	
COMPUTER	PROBLEM STATUS Replaced by:	CATEGORY
Infinite plane	9 sec per history	
GEOMETRY & DIMENSIONS	TYPICAL COMPUTING TIME	PARAMETERS SOLVED

This problem consists of a (1) main routine which generates histories of N^{17} neutrons in an infinite water medium simultaneously under four biasing schemes (2) a number of processing codes that extract interesting information from the histories.

PROBLEM DESCRIPTION

This was a single problem, not "an available code."

LIMITATIONS OF THE CODE

A description of the problem and the results is in the process of being written.

REFERENCES

Influence function for gamma ray emitting fission products	Melba Nead	North American Aviation, Inc. Nuclear Engineering and Manufacturing Division
TITLE	COG. ENGR.	LABORATORY
701	Analyzed: Coded: In Use: Programmed: Debugged: Obsolete X PROBLEM STATUS Replaced by:	Shielding CATEGORY
COMPUTER		
GEOMETRY & DIMENSIONS	2 min. TYPICAL COMPUTING TIME	PARAMETERS SOLVED

Designed to aid in determining a simplified engineering method which would give the shielding requirements for fission products.

PROBLEM DESCRIPTION

LIMITATIONS OF THE CODE

REFERENCES

Saturated activities of foils placed in Pile Shielding (I-24115)	R. L. Tomlinson	Numerical analysis Unit HAPO GE Richland
TITLE	COG. ENGR.	LABORATORY
702 EDPM	Analyzed: Coded: ☒	Shielding
COMPUTER	In Use: Programmed: Debugged: Obsolete: PROBLEM STATUS Replaced by:	CATEGORY
GEOMETRY & DIMENSIONS	250 measurements/min TYPICAL COMPUTING TIME	PARAMETERS SOLVED

Given: Pile down time, buildup factor, foil position, counting interval, total counts, background counts, instrument factor, foil weight, foil size correction, and coincidence correction;

Calculate & list: Identification and saturated activity.

PROBLEM DESCRIPTION

LIMITATIONS OF THE CODE

Sanford Work Order I-24115

REFERENCES

RE-9	E. Jutzi R. Brittan	Argonne National Laboratory
TITLE	COG. ENGR.	LABORATORY
AVIDAC	Obsolete	
COMPUTER	Replaced by: 12, 13, or 17	Reactor Kinetics
	PROBLEM STATUS	CATEGORY
GEOMETRY & DIMENSIONS	15 minutes	<p> p = neutron flux k_{ex} = excess reactivity C = delayed neutron precursor </p>
	TYPICAL COMPUTING TIME	PARAMETERS SOLVED

This program solves the one-energy reactor kinetics equations with one delay group.

$$\frac{dn}{dt} = \frac{1}{\ell} k_{ex} n - \frac{dC}{dt}$$

$$\frac{dk_{ex}}{dt} = Dn + E$$

$$\frac{dC}{dt} = (1 + k_{ex}) \beta$$

$$n = \lambda_j C$$

$$\lambda_j = \text{delayed neutron decay rate}$$

$$\beta = \text{delayed neutron fraction}$$

PROBLEM DESCRIPTION

LIMITATIONS OF THE CODE

ANL-5371 - RED Quarterly Report - 10/1/54 - 12/31/54, p. 66

REFERENCES

K-1

RE-12	H. Greenspan L. Sommerfeld	Argonne National Laboratory
TITLE	COG. ENGR	LABORATORY
AVIDAC	In Use	Reactor Kinetics
COMPUTER	PROBLEM STATUS	CATEGORY
GEOMETRY & DIMENSIONS	1 Hour	$x_v, (dx_v / dt, \text{ and } \int x_1 dt$ $v = 1, 2 \dots n+1 \text{ as a}$ function of time
	TYPICAL COMPUTING TIME	PARAMETERS SOLVED

This program solves the one energy group reactor kinetics equations with delay groups in a general Runge-Kutta form adaptable to other problems as well.
The system solved is:

$$\frac{dx_1}{dt} = \frac{x_{n+1} (1 - \beta_1) - \beta_1}{\ell} x_1 + \sum_{i=2}^n b_{1,i} x_i = f_1(x_1, \dots, x_{n+1})$$

$$\frac{dx_i}{dt} = \frac{(x_{n+1} + 1) \beta_i}{\ell} x_1 + \sum_{i=2}^n b_{j,i} x_i = f_j(x_1, \dots, x_{n+1})$$

$$i = 2, 3, \dots, n$$

$$\frac{dx_{n+1}}{dt} = A + Bx_1 + Cx_1^2 + Dx_1^3 + Et + Ft^2 + Gt^3 = f_{n+1}(x_1, t)$$

PROBLEM DESCRIPTION

and $b_j = \lambda_j$ = delayed neutron decay constants ℓ = neutron lifetime,

β_j = delayed neutron fraction

t = time for reactor kinetics use only.

$$n \leq 10$$

LIMITATIONS OF THE CODE

RED Memorandum - H. Greenspan, L. Sommerfeld, H. J. Wheeler to F. W. Thalgott
SUBJECT: Solution of the One Energy Group Reactor Kinetics Equations by Means of the AVIDAC (RE-12) - 11/15/54

REFERENCES ANL-5371 - RED QUARTERLY REPORT - 10/1/54-12/31/54 - p. 66

RE-13	H. J. Wheeler	Argonne National Laboratory
TITLE	COG. ENGR.	LABORATORY
AVIDAC	In Use	Reactor Kinetics
COMPUTER	PROBLEM STATUS	CATEGORY
GEOMETRY & DIMENSIONS	45 Minutes TYPICAL COMPUTING TIME	n = neutron flux k_{ex} = excess reactivity $\int_0^t n dt$ = integral of the flux and if desired, $n/dn/dt$ = pile period as a function of time PARAMETERS SOLVED

This program solves the one energy group reactor kinetics equations with 6 delay groups by the Runge-Kutta method as developed in RE-12

$$\frac{dn}{dt} = \frac{k_{ex}(1-\beta) - \beta}{\ell} n + \sum_{i=1}^6 \lambda_i C_i + S$$

$$\frac{dC_i}{dt} = \frac{(k_{ex} + 1)\beta a_i}{\ell} n - \lambda_i C_i; i = 1, 2, \dots, 6$$

PROBLEM DESCRIPTION

n = neutron flux
 C_i = delayed neutron precursor
 k_{ex} = excess reactivity
 β = fraction of neutrons
delayed
 a_i = fraction of neutrons
delayed emitted in i th
group
 λ_i = decay constant for delayed
neutron precursor
 ℓ = neutron lifetime
 t = time

LIMITATIONS OF CODE

RED Memorandum - Reactor Safety Group written by H. J. Wheeler to F. W. Thalgott
Subject: Programming of RE-13 - 1/24/55
ANL-5371 - RED Quarterly Report - 10/1/54 - 12/31/54, p. 66

REFERENCES

RE-17	L. Kassel L. Sommerfeld	Argonne National Laboratory
TITLE	COG. ENGR.	LABORATORY
AVIDAC	In Use	Reactor Kinetics
COMPUTER	PROBLEM STATUS	CATEGORY
GEOMETRY & DIMENSIONS	1 hour TYPICAL COMPUTING TIME	n = neutron flux C _i = delayed neutron precursor k _{ex} = excess reactivity as a function of time PARAMETERS SOLVED

This program solves the one energy group reactor kinetics equations with 6 delay groups by a Taylor series approximation. The equations solved are:

$$\frac{dn}{dt} = \frac{k_{ex}(1-\beta)-\beta}{\ell} n + \sum_{i=1}^6 \lambda_i C_i$$

$$\frac{dC_i}{dt} = \frac{(k_{ex} + 1)\beta a_i}{\ell} n - \lambda_i C_i$$

$$\frac{dk_{ex}}{dt} = f(n,t) = A + Bn$$

n = neutron flux
C_i = delayed neutron precursor
k_{ex} = excess reactivity
β = fraction of neutrons delayed
a_i = fraction of neutrons delayed
emitted in ith group
λ_i = decay constant for
delayed neutron precursor
ℓ = neutron lifetime
t = time

PROBLEM DESCRIPTION

LIMITATIONS OF THE CODE

RED Memorandum - L. Kassel, L. Sommerfeld to F. W. Thalgott
SUBJECT: Solution of Reactor Kinetics Equations by Taylor Series
(AVIDAC RE-17) - 2/25/55

REFERENCES

RE-21	H. J. Wheeler	Argonne National Laboratory
TITLE	COG. ENGR.	LABORATORY
AVIDAC	In Use	Reactor Kinetics
COMPUTER	PROBLEM STATUS	CATEGORY
	15 Minutes	k_{ex} = excess reactivity as a function of time
GEOMETRY & DIMENSIONS	TYPICAL COMPUTING TIME	PARAMETERS SOLVED

This program using the one energy group reactor kinetics equations with 6 delay groups computes k_{ex} as a function of time, given initial concentrations of delayed emitters, an initial k_{ex} , and values of n , (neutron flux), as a function of time.

PROBLEM DESCRIPTION

LIMITATIONS OF THE CODE

RED Memorandum - J. Wheeler to Distribution
SUBJECT: An AVIDAC Program for Analyzing BORAX Data (RE-21)

REFERENCES

RE-22

L. Sommerfeld
M. Butler

Argonne National Laboratory

TITLE

COG. ENGR.

LABORATORY

AVIDAC

In Use

Reactor Kinetics

COMPUTER

PROBLEM STATUS

CATEGORY

15 Minutes

 $k_{ex} \int_0^t n dt, \frac{dk_{ex}}{dt}$ and $d^2 k_{ex}/dt^2$ as a function of time.
GEOMETRY &
DIMENSIONS

TYPICAL COMPUTING TIME

PARAMETERS SOLVED

This program is similar to RE-21. It uses the one energy group reactor kinetics equations with up to 10 delay groups to compute k_{ex} , dk_{ex}/dt and $d^2 k_{ex}/dt^2$ as a function of time, given initial concentrations of delayed emitters, an initial k_{ex} , and values of n , (the neutron flux), as a function of time.

PROBLEM DESCRIPTION

LIMITATIONS OF THE CODE

REFERENCES

RE-19	L. Kassel	Argonne National Laboratory
TITLE	COG. ENGR.	LABORATORY
AVIDAC	In Use	Reactor Kinetics
COMPUTER	PROBLEM STATUS	CATEGORY
GEOMETRY & DIMENSIONS	1-1-1/2 minutes for each set of input parameters, for $m_0 \approx 20$ TYPICAL COMPUTING TIME	k_{ex} = excess reactivity and $\rho = \frac{1}{1 + k_{ex}}$ PARAMETERS SOLVED

This program computes k_{ex} and ρ for specified ℓ = neutron lifetime, β = fraction of neutrons delayed, a_i = fraction of delayed neutrons emitted in i th group, λ_i = decay constant for delayed neutrons precursors, and m_0 , from the equation.

$$k_{ex} = \frac{\frac{\ell}{2m} + \beta \sum_{i=1}^6 \frac{a_i}{1+2^m \lambda_i}}{1 - \beta \sum_{i=1}^6 \frac{a_i}{1+2^m \lambda_i}}$$

$$\rho = \frac{1}{1 + k_{ex}}$$

where m goes from $-m_0(1)$ m_0 and 2^m = asymptotic pile period

$$10^{-9} < \ell < 10^{-3}$$

$$10^{-3} < \beta < 10^{-1}$$

$$0 \leq a_i \leq 1$$

$$0 \leq \lambda_i < 10^2$$

$$1 \leq m_0 \leq 20$$

$$0 \leq k_{ex} \leq 0.2 \quad \text{LIMITATIONS OF THE CODE}$$

REFERENCES

CHAIN

Origin - Brookhaven National
Laboratory
Prog. & Code - New York University

TITLE

COG. ENGR.

LABORATORY

UNIVAC

In Use

Reactor Kinetics

COMPUTER

PROBLEM STATUS

CATEGORY

Cylindrical

Reactor Periods

GEOMETRY &
DIMENSIONS

TYPICAL COMPUTING TIME

PARAMETERS SOLVED

The code computes reactor periods of circulating fuel reactors by finding the eigenvalues of a system of differential equations. The method accommodates two energy groups of neutrons and five groups of delayed neutrons.

PROBLEM DESCRIPTION

LIMITATIONS OF THE CODE

Brookhaven National Laboratory 334 (T-57)

REFERENCES

K-8

TEMP	H. B. Keller J. Franklin	New York University (AEC Comptg. Fac.)
TITLE	COG. ENGR.	LABORATORY
UNIVAC	In Use	Reactor Kinetics
COMPUTER	PROBLEM STATUS	CATEGORY
One space & time	1 - 5 hours	Neutron density & temperature
GEOMETRY & DIMENSIONS	TYPICAL COMPUTING TIME	PARAMETERS SOLVED

The temperature and neutron density along the axis of a circulating fuel reactor are computed as functions of time and distance along the axis. The steady state distributions or several oscillations in time may be computed from general initial configurations. The length, flow speed, or zero temperature reactivity of the reactor may be changed independently. Thus the transient behavior from one steady state to another may be obtained.

PROBLEM DESCRIPTION

Maximum number of space points = 59.

LIMITATIONS OF THE CODE

Herbert B. Keller & Joel Franklin	- Steady states, boundedness, and numerical analysis of temperature and neutron density in circulating fuel reactors; N.Y.U, AEC Computing Facility Report # NYO-6480.
J. Fleck REFERENCES	- Temperature dependent kinetics of circulating fuel reactors, Brookhaven National Laboratory forthcoming report.

Long Term Irradiation of Nuclear Fuels (A)	Barbara Lemke	North American Aviation, Inc. Nuclear Engineering and Manufacturing Division
TITLE	COG. ENGR.	LABORATORY
CPC & IBM 701 Speedco and Duel Code (Floating point arithmetic) COMPUTER	In Use (The CPC and Speed Co programmed are obsolete) PROBLEM STATUS Replaced by:	Reactor Burnout CATEGORY
Cylindrical geometry GEOMETRY & DIMENSIONS	For a given value of each para- meter one case is computed in 0.04 min. TYPICAL COMPUTING TIME	With and without initial Pu in various fuel enrich- ments (U235) or a function of integrad flux-time. PARAMETERS SOLVED

The program computes the average and effective values of isotope concentrations, reactivity, exposure, and the fission poison effect.

PROBLEM DESCRIPTION

The program is limited by the geometry.

LIMITATIONS OF THE CODE

REFERENCES

Long term Irradiation of Nuclear Fuels (B)	Barbara Lemke	North American Aviation, Inc. Nuclear Eng. & Mfg. Division
TITLE	COG. ENGR.	LABORATORY
IBM 701 Duel Code (Floating point arithmetic)	In Use	Reactor Burnout
COMPUTER	PROBLEM STATUS Replaced by:	CATEGORY
Cylindrical geometry	For a given value of each para- meter one case is computed in 0.15 min.	Rod radius Lattice spacing enrichment
GEOMETRY & DIMENSIONS	TYPICAL COMPUTING TIME	PARAMETERS SOLVED

The program computes the multiplication factor, buckling, conversion ratio, thermal utilization, resonance escape probability, moderator utilization, moderator excess absorption, eta, thermal migration area, and age.

PROBLEM DESCRIPTION

The program is limited to rod type lattices with either square or triangular spacing.

LIMITATIONS OF THE CODE

none

REFERENCES

RB-2

BURPP	G. J. Habetler	Knolls Atomic Power Laboratory
TITLE	COG. ENGR.	LABORATORY
UNIVAC	Obsolete Replaced by: BURPPE	Reactor Burnout
COMPUTER	PROBLEM STATUS	CATEGORY
One-space-dimensional spherical model; time GEOMETRY & DIMENSIONS	Endurance study of typical SIR-type reactor takes 2 to 3 hours. TYPICAL COMPUTING TIME	Fuel, burnable poison, and fission product as a function of time; Endurance PARAMETERS SOLVED

Consists of a number of criticality calculations for each reactor in various stages of fuel and burnable poison burn-up and fission product build-up. The point at which a reactor can no longer be made critical determines its endurance.

Burn-up and build-up are assumed linear in a given time step Δt .

PROBLEM DESCRIPTION

1. Restricted to three regions of homogeneous composition, only the first of which contains fuel and burnable poison.
2. 15 predetermined groups in the steady-state multigroup calculations; group-averaged microscopic cross-sections on tape.
3. Number of space points in the finite-difference mesh ≤ 57 .
4. Number of burnable poisons present in a given problem ≤ 2 .
5. Number of fission products allowed ≤ 1 .

LIMITATIONS OF THE CODE

Memos JBN-5, JBN-6, JBN-10
Report KAPL 1171
Multigroup described in KAPL 975

REFERENCES

BURPPE	G. J. Habetler	Knolls Atomic Power Laboratory
TITLE	COG. ENGR.	LABORATORY
UNIVAC	Debugged	Reactor Burnout
COMPUTER	PROBLEM STATUS	CATEGORY
One-space dimensional (cylinder, sphere); time GEOMETRY & DIMENSIONS	Endurance study of typical SIR-type reactor takes 2 to 3 hours TYPICAL COMPUTING TIME	Fuel, burnable poison, and fission products as a function of time; endurance. PARAMETERS SOLVED

A generalized BURPP coding. Consists of a number of criticality calculations for each reactor in various stages of fuel and burnable poison burn-up and fission product build-up. The point at which a reactor can no longer be made critical determines its endurance.

Burn-up and build-up are considered linear in a given time step Δt .

The Selengut-Goertzel treatment for slowing down by hydrogen is included.

PROBLEM DESCRIPTION

1. Restricted to at most 6 regions of homogeneous composition, only the first five of which can be cores.
2. Predetermined groups widths; the group averaged microscopic cross-sections must be available on tape.
3. Number of mesh points plus number of regions ≤ 58 .
4. Δr constant within a region; $\Delta r > \sqrt{t}$
5. Number of burnable poisons present in a given problem ≤ 3 .
6. Number of fission products ≤ 2 .

LIMITATIONS OF THE CODE

REFERENCES

Burnout Code (Tentative)	R. P. Eddy W. R. Clancey	David Taylor Model Basin Westinghouse Atomic Power Division
TITLE	COG. ENGR.	LABORATORY
UNIVAC	Coded	Reactor Burnout
COMPUTER	PROBLEM STATUS	CATEGORY
Cartesian Cylindrical One-dimensional Spherical GEOMETRY & DIMENSIONS	Unknown TYPICAL COMPUTING TIME	Time variable coefficients of two-group diffusion equation PARAMETERS SOLVED

This code is used in conjunction with the AML-17-55 code of David Taylor Model Basin which solves the two-group diffusion equations in one-dimensional geometries. The effect of flux level and time is considered in computing the spatial variation of the coefficients. The effects of isotopic build-up and burn-up are explicitly included.

PROBLEM DESCRIPTION

The equations describing the Pu build-up give a dependence on the slow flux only in the first version of the code. A second version will incorporate dependence on both the slow and fast flux at each point. Only the slow group coefficients are assumed to change with time.

LIMITATIONS OF THE CODE

The difference equations used are given in WAPD-105.

REFERENCES

HEX 3	G. J. Habetler	Knolls Atomic Power Laboratory
TITLE	COG. ENGR.	LABORATORY
UNIVAC	In Use	Reactor Engineering
COMPUTER	PROBLEM STATUS	CATEGORY
One-space-dimensional	One hour per one minute thermal time.	Temperatures as funcns of time.
GEOMETRY & DIMENSIONS	TYPICAL COMPUTING TIME	PARAMETERS SOLVED

Calculation of transient temperature distribution in SIR reactor core. The system consists of a single liquid metal channel in a solid matrix structure. The flow rate is given as a function of time. Heat conduction is assumed perpendicular to the liquid metal flow only. Top and bottom temperatures for the channel are assumed known.

PROBLEM DESCRIPTION

1. The code, as such, is applicable only to this system.

LIMITATIONS OF THE CODE

KAPL - 1239

REFERENCES

HEX 2	G. J. Habetler	Knolls Atomic Power Laboratory
TITLE	COG. ENGR.	LABORATORY
UNIVAC	Debugged	Reactor Engineering
COMPUTER	PROBLEM STATUS	CATEGORY
Two-space-dimension; one dimensional fluid flow GEOMETRY & DIMENSIONS	20 hours per 1 hour thermal time TYPICAL COMPUTING TIME	Temperatures, weight-rates as functions of time. PARAMETERS SOLVED

Calculation of transient temperature distribution in a specific thermal blanket. The system consists of an assembly of metallic slabs separated by gaps which are flow channels for a liquid metal. An orifice meters flow into this system from a reservoir at the bottom of the system. In parallel with this system is another channel for which the flow-rate and the temperature distribution are known functions of time. Temperatures on the boundary of the system are known functions of time.

PROBLEM DESCRIPTION

1. The code, as such, is applicable only to this system.

LIMITATIONS OF THE CODE

Memos SKH - 1, 2, 3, 4,
ASME Paper 54-SA-53

REFERENCES

Heat Transfer	Melba Nead	North American Aviation, Inc. Nuclear Engineering and Manufacturing Division
TITLE	COG. ENGR.	LABORATORY
701	Debugged	Reactor Engineering
COMPUTER	PROBLEM STATUS	CATEGORY
Slab Geo Various Dimensions	7 min	Temperature at 17 pts.
GEOMETRY & DIMENSIONS	TYPICAL COMPUTING TIME	PARAMETERS SOLVED

Transient temperature in the corner of a reactor shield.

PROBLEM DESCRIPTION

LIMITATIONS OF THE CODE

REFERENCES

Runaway

D. B. MacMillan

Knolls Atomic Power Laboratory

TITLE

COG. ENGR.

LABORATORY

IBM 650

In Use

Reactor Engineering

COMPUTER

PROBLEM STATUS

CATEGORY

See below

of the order of 60 min.
computing time per sec. of real
time.

Various excess reactivities
and coolant inlet
temperature transients
for proposed SAR cores.
PARAMETERS SOLVED

GEOMETRY &
DIMENSIONS

TYPICAL COMPUTING TIME

This program was intended to compute the reaction of a SAR-type reactor to being put on a very short period, either by a control malfunction or by an influx of cold coolant. Thermally, the reactor is represented by an "average" coolant temperature and two temperatures in the nearby metal. The computation of the rate of power generation takes account of temperature coefficients, and represents delayed neutron effects by one group of delayed neutron emitters.

PROBLEM DESCRIPTION

None

LIMITATIONS OF THE CODE

None

REFERENCES

PUFF	D. B. MacMillan	Knolls Atomic Power Laboratory
TITLE	COG. ENGR.	LABORATORY
IEM 650	In Use	Reactor Engineering
COMPUTER	PROBLEM STATUS	CATEGORY
See below	Of the order of 30 min. computing time per min. real time.	Various flow transients for proposed SAR cores.
GEOMETRY & DIMENSIONS	TYPICAL COMPUTING TIME	PARAMETERS SOLVED

This program was intended to compute the short-run effects of failures in the coolant pumping system in a SAR-type reactor core. Given a coolant flow transient, it computes the transient metal and water temperatures at each of ten equally-spaced points along a typical coolant channel; it also computes the transient of rate power generation by formulas which take account of temperature coefficient and of three groups of delayed neutrons.

PROBLEM DESCRIPTION

None

LIMITATIONS OF THE CODE

None

REFERENCES

Program E - Special Functions	T. L. Glahn (Bessel Functions) E. Isenberg (exponential integrals)	Aircraft Gas Turbine Dev. Dept. General Electric Company Cincinnati 15, Ohio
TITLE	COG. ENGR.	LABORATORY
IBM-701	In Use	Miscellaneous
COMPUTER	PROBLEM STATUS	CATEGORY
GEOMETRY & DIMENSIONS	0.1 to 2 seconds TYPICAL COMPUTING TIME	PARAMETERS SOLVED

- (1) Subroutines for calculating $J_n(x)$, $Y_n(x)$, $I_n(x)$ and $K_n(x)$, for real positive values of x and integral values of $n \geq 0$, using either power series or asymptotic expansion depending on predetermined value of x .
- (2) Subroutine for the exponential integral $E_n(x)$ using a partial fraction expansion.

PROBLEM DESCRIPTION

- (1) Accuracy of $K_n(x)$ is low in intermediate range of x .
- (2) For $E_n(x)$, $10^{-5} < x < 10$.

LIMITATIONS OF THE CODE

Unpublished notes.

REFERENCES

FOUSER	H. F. Hunter	Knolls Atomic Power Laboratory
TITLE	COG. ENGR.	LABORATORY

IBM 650	In Use	Miscellaneous
COMPUTER	PROBLEM STATUS	CATEGORY
One-dimensional	3+n minutes for n Harmonics with 20 given points (Y_i, X_i)	Fourier coefficients
GEOMETRY AND DIMENSIONS	TYPICAL COMPUTING TIME	PARAMETERS SOLVED

Fouser computes $B_0 = \frac{1}{2\pi} \int_0^{2\pi} Y d\bar{x} = \frac{1}{\pi} \int_0^{2\pi} Y \sin n \bar{x} d\bar{x}$, $B_n = \frac{1}{\pi} \int_0^{2\pi} Y \cos n \bar{x} d\bar{x}$
and $F = \frac{1}{\pi} \int_0^{2\pi} Y^2 d\bar{x} = 2\theta_0^2 \sum (A_n^2 + B_n^2)$ by exact integration for any

empirical periodic function given as an odd number of points ($\bar{Y}_i, \bar{X}_i$) at not necessarily equispaced points $0 = \bar{X}_0 < \bar{X}_1 < \dots < \bar{X}_i = 2\pi$. Input consists of points (Y_i, X_i) taken from oscillograph tracing, or computed, etc. Transformations take these into ($\bar{Y}_i, \bar{X}_i$).

PROBLEM DESCRIPTION

1. The function $Y(x)$ is approximated by piecewise quadratic functions. Determined by the given points.

LIMITATIONS OF THE CODE

REFERENCES