A BIBLIOGRAPHY OF AVAILABLE DIGITAL COMPUTER CODES FOR NUCLEAR REACTOR PROBLEMS

By
A. Radkowsky
R. Brodsky

October 14, 1955

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

Technical Information Service, Oak Ridge, Tennessee
DISCLAIMER

This report was prepared as an account of work sponsored by an agency of the United States Government. Neither the United States Government nor any agency Thereof, nor any of their employees, makes any warranty, express or implied, or assumes any legal liability or responsibility for the accuracy, completeness, or usefulness of any information, apparatus, product, or process disclosed, or represents that its use would not infringe privately owned rights. Reference herein to any specific commercial product, process, or service by trade name, trademark, manufacturer, or otherwise does not necessarily constitute or imply its endorsement, recommendation, or favoring by the United States Government or any agency thereof. The views and opinions of authors expressed herein do not necessarily state or reflect those of the United States Government or any agency thereof.
DISCLAIMER

Portions of this document may be illegible in electronic image products. Images are produced from the best available original document.
A BIBLIOGRAPHY

of

AVAILABLE DIGITAL COMPUTER CODES

for

NUCLEAR REACTOR PROBLEMS

By A. Radkowsky and R. Brodsky

October 14, 1955

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

<table>
<thead>
<tr>
<th>1</th>
<th>2</th>
<th>3</th>
</tr>
</thead>
<tbody>
<tr>
<td><strong>TITLE</strong></td>
<td><strong>C.O.G. ENGR.</strong></td>
<td><strong>LABORATORY</strong></td>
</tr>
<tr>
<td>4</td>
<td>5</td>
<td>6</td>
</tr>
<tr>
<td><strong>COMPUTER</strong></td>
<td><strong>PROBLEM STATUS</strong></td>
<td><strong>CATEGORY</strong></td>
</tr>
<tr>
<td>7</td>
<td>8</td>
<td>9</td>
</tr>
<tr>
<td><strong>GEOMETRY &amp; DIMENSIONS</strong></td>
<td><strong>TYPICAL COMPUTING TIME</strong></td>
<td><strong>PARAMETERS SOLVED</strong></td>
</tr>
</tbody>
</table>

<table>
<thead>
<tr>
<th>10</th>
</tr>
</thead>
<tbody>
<tr>
<td><strong>PROBLEM DESCRIPTION</strong></td>
</tr>
</tbody>
</table>

<table>
<thead>
<tr>
<th>11</th>
</tr>
</thead>
<tbody>
<tr>
<td><strong>LIMITATIONS OF THE CODE</strong></td>
</tr>
</tbody>
</table>

<table>
<thead>
<tr>
<th>12</th>
</tr>
</thead>
<tbody>
<tr>
<td><strong>REFERENCES</strong></td>
</tr>
</tbody>
</table>
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.
## INDEX

<table>
<thead>
<tr>
<th>TITLE</th>
<th>COMPUTER</th>
<th>PAGE</th>
</tr>
</thead>
<tbody>
<tr>
<td>Reactor and Nuclear Physics:</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Resonance Absorption Integrals</td>
<td>CPC</td>
<td>P-1</td>
</tr>
<tr>
<td>REP</td>
<td></td>
<td></td>
</tr>
<tr>
<td>(Resonance Escape Probability)</td>
<td>UNIVAC</td>
<td>P-2</td>
</tr>
<tr>
<td>REP MOD. 1</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Monte Carlo Fast Fission Effect</td>
<td>UNIVAC</td>
<td>P-3</td>
</tr>
<tr>
<td>WREP</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Monte Carlo Resonance Escape</td>
<td>NORC</td>
<td>P-5</td>
</tr>
<tr>
<td>MUFT-II</td>
<td></td>
<td></td>
</tr>
<tr>
<td>S-Approximation-II</td>
<td>CPC</td>
<td>P-6</td>
</tr>
<tr>
<td>MUFT-III</td>
<td></td>
<td></td>
</tr>
<tr>
<td>IBM-650</td>
<td>P-7</td>
<td></td>
</tr>
<tr>
<td>SPH - HI</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Temperature Coefficients of Heterogenous Reactors (I-24184) 702 EDPM</td>
<td>IBM-701</td>
<td>P-9</td>
</tr>
<tr>
<td>Program H - Cylinder Transport Equation</td>
<td></td>
<td></td>
</tr>
<tr>
<td>SLAV-I</td>
<td></td>
<td></td>
</tr>
<tr>
<td>UNIVAC</td>
<td>P-11</td>
<td></td>
</tr>
<tr>
<td>SPOT I</td>
<td></td>
<td></td>
</tr>
<tr>
<td>UNIVAC</td>
<td>P-12</td>
<td></td>
</tr>
<tr>
<td>RE-18</td>
<td></td>
<td></td>
</tr>
<tr>
<td>AVIDAC</td>
<td>P-13</td>
<td></td>
</tr>
<tr>
<td>Program I - Analytic P₃ Calculation</td>
<td>IBM-701</td>
<td>P-14</td>
</tr>
<tr>
<td>Program J - Numerical P₃ Calculations</td>
<td>IBM-701</td>
<td>P-15</td>
</tr>
<tr>
<td>RE-24</td>
<td></td>
<td></td>
</tr>
<tr>
<td>AVIDAC</td>
<td>P-16</td>
<td></td>
</tr>
<tr>
<td>Wigner-Wilkins Calculation</td>
<td>IBM-650</td>
<td>P-17</td>
</tr>
</tbody>
</table>
## INDEX

<table>
<thead>
<tr>
<th>TITLE</th>
<th>COMPUTER</th>
<th>PAGE</th>
</tr>
</thead>
<tbody>
<tr>
<td>Reactor and Nuclear Physics (continued):</td>
<td></td>
<td></td>
</tr>
<tr>
<td>TEMP-I</td>
<td>UNIVAC</td>
<td>P-18</td>
</tr>
<tr>
<td>COM-CEN-CH-I</td>
<td>UNIVAC</td>
<td>P-19</td>
</tr>
<tr>
<td>NED</td>
<td>IBM-650</td>
<td>P-20</td>
</tr>
<tr>
<td>- Monte Carlo Reactor Calculation</td>
<td>NAREC</td>
<td>P-21</td>
</tr>
<tr>
<td>- Integral Network Method</td>
<td>UNIVAC</td>
<td>P-22</td>
</tr>
<tr>
<td>- Flexible Integral Network</td>
<td>NORC (NRL)</td>
<td>P-23</td>
</tr>
<tr>
<td>RE-23</td>
<td>AVIDAC</td>
<td>P-24</td>
</tr>
<tr>
<td>- Multiple-Level Decay Chains</td>
<td>CPC</td>
<td>P-25</td>
</tr>
<tr>
<td>- Non-Linear Isotope Buildup and Decay (I-23021)</td>
<td>702</td>
<td>P-26</td>
</tr>
<tr>
<td>- Xenon &amp; Iodine Concentration Calculation for Hanford Reactors (I-24119, 24179)</td>
<td>702</td>
<td>P-27</td>
</tr>
<tr>
<td>PHY-8</td>
<td>AVIDAC</td>
<td>P-28</td>
</tr>
<tr>
<td>- Cloudy Crystal Ball Neutron Scattering</td>
<td>CPC</td>
<td>P-29</td>
</tr>
<tr>
<td>- Cloudy Crystal Ball Neutron Scattering</td>
<td>UNIVAC</td>
<td>P-30</td>
</tr>
<tr>
<td>- Surface-Modified Optical Model</td>
<td>NORC</td>
<td>P-31</td>
</tr>
<tr>
<td>- End and Fast Source Corrections to Neutron Measurement in an exponential pile (I-24176)</td>
<td>702</td>
<td>P-32</td>
</tr>
</tbody>
</table>
## INDEX

<table>
<thead>
<tr>
<th>TITLE</th>
<th>COMPUTER</th>
<th>PAGE</th>
</tr>
</thead>
<tbody>
<tr>
<td>Reactor Survey (One Dimension - One Group):</td>
<td></td>
<td></td>
</tr>
<tr>
<td>- Cell Flux Calculations</td>
<td>IBM-701 (Duet Code) (Floating Point Arithmetic)</td>
<td>RS-1</td>
</tr>
<tr>
<td>Reactor Survey (One Dimension - Two Group):</td>
<td></td>
<td></td>
</tr>
<tr>
<td>RE-2 - Group Spherical Reactor Problem</td>
<td>IBM CPC</td>
<td>RS-2</td>
</tr>
<tr>
<td>RE-1</td>
<td></td>
<td>RS-3</td>
</tr>
<tr>
<td>Program B - Two Region Reflector Savings</td>
<td>IBM-701</td>
<td>RS-4</td>
</tr>
<tr>
<td>Program F - Digital Space Simulator</td>
<td>IBM-701</td>
<td>RS-5</td>
</tr>
<tr>
<td>LMFP-I</td>
<td>UNIVAC</td>
<td>RS-6</td>
</tr>
<tr>
<td>Two Group - Two Region Axial Problem</td>
<td>IBM 701 (Duet Code) (Floating Point Arithmetic)</td>
<td>RS-7</td>
</tr>
<tr>
<td>Two Group - Two Region Radial Problem</td>
<td>IBM 701 (Duet Code) (Floating Point Arithmetic)</td>
<td>RS-8</td>
</tr>
<tr>
<td>Two Group - Critical Mass Reactivity Calculations</td>
<td>CPC</td>
<td>RS-9</td>
</tr>
<tr>
<td>Two-Group Code - One Dimensional</td>
<td>UNIVAC</td>
<td>RS-10</td>
</tr>
<tr>
<td>Program A - Bare Reactor Criticality</td>
<td>IBM-701</td>
<td>RS-11</td>
</tr>
<tr>
<td>Program C - Bare Reactor Criticality</td>
<td>IBM-701</td>
<td>RS-12</td>
</tr>
<tr>
<td>Reactor Survey (One Dimension - Three Group):</td>
<td></td>
<td></td>
</tr>
<tr>
<td>IDA</td>
<td>UNIVAC</td>
<td>RS-13</td>
</tr>
<tr>
<td>PHY-4</td>
<td>AVIDAC</td>
<td>RS-14</td>
</tr>
</tbody>
</table>
# INDEX

<table>
<thead>
<tr>
<th>TITLE</th>
<th>COMPUTER</th>
<th>PAGE</th>
</tr>
</thead>
<tbody>
<tr>
<td>Reactor Survey (One Dimension - Multigroup):</td>
<td></td>
<td></td>
</tr>
<tr>
<td>RE-2</td>
<td>IBM CPC</td>
<td>RS-15</td>
</tr>
<tr>
<td>RE-2A</td>
<td>IBM CPC</td>
<td>RS-16</td>
</tr>
<tr>
<td>RE-4</td>
<td>AVIDAC</td>
<td>RS-17</td>
</tr>
<tr>
<td>RE-6</td>
<td>UNIVAC</td>
<td>RS-18</td>
</tr>
<tr>
<td>RE-6X</td>
<td>UNIVAC</td>
<td>RS-19</td>
</tr>
<tr>
<td>RE-7</td>
<td>UNIVAC</td>
<td>RS-20</td>
</tr>
<tr>
<td>RE-7X</td>
<td>UNIVAC</td>
<td>RS-21</td>
</tr>
<tr>
<td>RE-8</td>
<td>UNIVAC</td>
<td>RS-22</td>
</tr>
<tr>
<td>RE-8X</td>
<td>UNIVAC</td>
<td>RS-23</td>
</tr>
<tr>
<td>RE-26</td>
<td>UNIVAC</td>
<td>RS-24</td>
</tr>
<tr>
<td>RE-27</td>
<td>UNIVAC</td>
<td>RS-25</td>
</tr>
<tr>
<td>RE-28</td>
<td>UNIVAC</td>
<td>RS-26</td>
</tr>
<tr>
<td>Program D</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Program G</td>
<td></td>
<td></td>
</tr>
<tr>
<td>OMAR</td>
<td></td>
<td></td>
</tr>
<tr>
<td>- 650 Multigroup</td>
<td>IBM-650</td>
<td>RS-30</td>
</tr>
<tr>
<td>- Multigroup Reactor Analysis (Speed Co.)</td>
<td>IBM-701</td>
<td>RS-31</td>
</tr>
<tr>
<td>- Multigroup Reactor Analysis (Duet)</td>
<td>IBM-701</td>
<td>RS-32</td>
</tr>
<tr>
<td>Eyewash</td>
<td></td>
<td></td>
</tr>
<tr>
<td>BUNNY</td>
<td></td>
<td></td>
</tr>
<tr>
<td>- One-Dimensional Multi-Group Code</td>
<td>UNIVAC</td>
<td>RS-34</td>
</tr>
</tbody>
</table>
# INDEX

<table>
<thead>
<tr>
<th>TITLE</th>
<th>COMPUTER</th>
<th>PAGE</th>
</tr>
</thead>
<tbody>
<tr>
<td>Reactor Survey (Two Dimension - Few Group):</td>
<td></td>
<td></td>
</tr>
<tr>
<td>- Two Dimensional Few Group Code</td>
<td>UNIVAC</td>
<td>RS-35</td>
</tr>
<tr>
<td>Reactor Survey (Two Dimension - Multigroup):</td>
<td></td>
<td></td>
</tr>
<tr>
<td>MUG II</td>
<td>-</td>
<td>UNIVAC</td>
</tr>
<tr>
<td>TOSPY</td>
<td>-</td>
<td>NORC or UNIVAC</td>
</tr>
<tr>
<td>Reactor Survey (Three Dimension - Two Group):</td>
<td></td>
<td></td>
</tr>
<tr>
<td>- Three Dimensional Two Group Reactor Problem</td>
<td>SEAC</td>
<td>RS-38</td>
</tr>
</tbody>
</table>

## Shielding:

<table>
<thead>
<tr>
<th>Program</th>
<th>COMPUTER</th>
<th>PAGE</th>
</tr>
</thead>
<tbody>
<tr>
<td>ACHILLES</td>
<td></td>
<td></td>
</tr>
<tr>
<td>- Neutron Shielding of a Plane Symmetric Source by Hydrogen</td>
<td>UNIVAC</td>
<td>S-1</td>
</tr>
<tr>
<td>- Neutron Diffusion</td>
<td>NAREC</td>
<td>S-2</td>
</tr>
<tr>
<td>Program 0-G</td>
<td>-</td>
<td>IBM-701</td>
</tr>
<tr>
<td>Program 1-G</td>
<td>-</td>
<td>IBM-701</td>
</tr>
<tr>
<td>Program 2-F</td>
<td>-</td>
<td>IBM-701</td>
</tr>
<tr>
<td>Program 2-G</td>
<td>-</td>
<td>IBM-701</td>
</tr>
<tr>
<td>- Gamma Ray Diffusion, Shallow</td>
<td>NAREC</td>
<td>S-7</td>
</tr>
<tr>
<td>- Gamma Ray Diffusion, Deep</td>
<td>NAREC</td>
<td>S-8</td>
</tr>
<tr>
<td>WAPD NEPTUNE</td>
<td>-</td>
<td>UNIVAC</td>
</tr>
<tr>
<td>- Influence function for gamma ray emitting fission products</td>
<td>IBM-701</td>
<td>S-10</td>
</tr>
<tr>
<td>- Saturated activities of foils 702 EDFM placed in Pile Shielding (I-24115)</td>
<td></td>
<td>S-11</td>
</tr>
</tbody>
</table>
### INDEX

<table>
<thead>
<tr>
<th>TITLE</th>
<th>COMPUTER</th>
<th>PAGE</th>
</tr>
</thead>
<tbody>
<tr>
<td><strong>Reactor Kinetics:</strong></td>
<td></td>
<td></td>
</tr>
<tr>
<td>RE-9</td>
<td>AVIDAC</td>
<td>K-1</td>
</tr>
<tr>
<td>RE-12</td>
<td>AVIDAC</td>
<td>K-2</td>
</tr>
<tr>
<td>RE-13</td>
<td>AVIDAC</td>
<td>K-3</td>
</tr>
<tr>
<td>RE-17</td>
<td>AVIDAC</td>
<td>K-4</td>
</tr>
<tr>
<td>RE-21</td>
<td>AVIDAC</td>
<td>K-5</td>
</tr>
<tr>
<td>RE-22</td>
<td>AVIDAC</td>
<td>K-6</td>
</tr>
<tr>
<td>RE-19</td>
<td>AVIDAC</td>
<td>K-7</td>
</tr>
<tr>
<td>CHAIM</td>
<td>UNIVAC</td>
<td>K-8</td>
</tr>
<tr>
<td>TEMP</td>
<td>UNIVAC</td>
<td>K-9</td>
</tr>
</tbody>
</table>

| **Reactor Burnout:**         |           |      |
| Long Term Irradiation of     | CPC & IBM | RB-1 |
| Nuclear Fuels (A)             | 701       |      |
| Long term Irradiation of     | IBM 701   | RB-2 |
| Nuclear Fuels (B)             | (Duel Code)(Floating Point Arithmetic) |
| BURPP                        | UNIVAC    | RB-3 |
| BURPFE                       | UNIVAC    | RB-4 |
| Burnout Code (Tentative)      | UNIVAC    | RB-5 |

<p>| <strong>Reactor Engineering:</strong>     |           |      |
| HEX-3                        | UNIVAC    | E-1  |
| HEX-2                        | UNIVAC    | E-2  |</p>
<table>
<thead>
<tr>
<th>TITLE</th>
<th>COMPUTER</th>
<th>PAGE</th>
</tr>
</thead>
<tbody>
<tr>
<td>Reactor Engineering (Continued):</td>
<td></td>
<td></td>
</tr>
<tr>
<td>- Heat Transfer</td>
<td>IBM 701</td>
<td>F-3</td>
</tr>
<tr>
<td>- Runaway</td>
<td>IBM 650</td>
<td>F-4</td>
</tr>
<tr>
<td>PUFF</td>
<td>IBM 650</td>
<td>E-5</td>
</tr>
<tr>
<td>Miscellaneous:</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Program E - Special Functions</td>
<td>IBM-701</td>
<td>M-1</td>
</tr>
<tr>
<td>FOUSER</td>
<td>IBM-650</td>
<td>M-2</td>
</tr>
</tbody>
</table>
The resonance absorption integral \( \int \sigma/E \, dE \)

\[
\left\{ \sigma = \sigma(E) = \frac{\sigma_0 \sqrt{\sigma(E/E_o)^{1/2}}}{4 \left( E - E_o \right)^2 + (\sqrt{7}/2)^2} \right\}
\]

is evaluated directly

**Problem Description**

Limited only by applicability of the physical assumptions

**Limitations of the Code**


**References**

P-1
Rep  
(Resonance Escape Probability)  
Dr. R. Richtmyer  
R. L. Hellens (Phys.)  
G. Crane (Math.)  
AEC Computing Facility  
New York University  
Westinghouse Atomic Power Division

<table>
<thead>
<tr>
<th>TITLE</th>
<th>COG. ENGR.</th>
<th>LABORATORY</th>
</tr>
</thead>
<tbody>
<tr>
<td>UNIVAC</td>
<td>In Use</td>
<td>Reactor and Nuclear Physics</td>
</tr>
<tr>
<td>COMPUTER</td>
<td>PROBLEM STATUS</td>
<td></td>
</tr>
</tbody>
</table>
| Hexagonal lattice of round fuel rods with (vacuum) cladding in moderator (water) | 3 hours for 10,000 histories | 1) Resonance Escape  
2) Pu Production distribution  
3) Capture Spectrum |
| GEOMETRY & DIMENSIONS | TYPICAL COMPUTING TIME | 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**

P-2
REP MOD. 1

Dr. R. Richtmyer

ABC Computing Facility
New York University

TITILE

UNIVAC

Dr. R. Richtmyer

ABC Computing Facility
New York University

IN USE

Reactors and Nuclear Physics

PROBLEM STATUS

Computers

PROGRAMMENTAL PROBLEMS

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

P-3
<table>
<thead>
<tr>
<th>MONTE CARLO FAST FISSION EFFECT</th>
<th>R. Hellens (Phys.)</th>
<th>Westinghouse Atomic Power Division</th>
</tr>
</thead>
<tbody>
<tr>
<td>TITLE</td>
<td>COG. ENGR</td>
<td>LABORATORY</td>
</tr>
<tr>
<td>UNIVAC</td>
<td>Analyzed</td>
<td>Reactor and Nuclear Physics</td>
</tr>
<tr>
<td>COMPUTER</td>
<td>PROBLEM STATUS</td>
<td>CATEGORY</td>
</tr>
<tr>
<td>Rods in hexagonal lattice</td>
<td>- - - -</td>
<td>1) Fast fission factor</td>
</tr>
<tr>
<td></td>
<td></td>
<td>2) Average fast fission cross section</td>
</tr>
<tr>
<td>GEOMETRY &amp; DIMENSIONS</td>
<td>TYPICAL COMPUTING TIME</td>
<td>3) Average fast capture cross section</td>
</tr>
</tbody>
</table>

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**

P-4
MONTE CARLO RESONANCE ESCAPE - WREP

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

Westinghouse Atomic Power Division

TITLE

COMMUTER

NORC

COMPUTER

PROBLEM STATUS

Rods in hexagonal or square lattice

12 minutes per single resonance

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

CATEGORY

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

P-5
S-APPROXIMATION-II
MUFT-II

R. L. Hellens (Phys)
B. H. Mount (Math)
Westinghouse Atomic Power Division

<table>
<thead>
<tr>
<th>TITLE</th>
<th>CPC</th>
<th>LABORATORY</th>
</tr>
</thead>
<tbody>
<tr>
<td>COG. ENGR.</td>
<td>In Use</td>
<td>Reactor and Nuclear Physics</td>
</tr>
</tbody>
</table>

<table>
<thead>
<tr>
<th>COMPUTER</th>
<th>PROBLEM STATUS</th>
<th>CATEGORY</th>
</tr>
</thead>
<tbody>
<tr>
<td>Fourier transform one dimension</td>
<td>1-1/2 hours</td>
<td>See below</td>
</tr>
</tbody>
</table>

<table>
<thead>
<tr>
<th>GEOMETRY &amp; DIMENSIONS</th>
<th>TYPICAL COMPUTING TIME</th>
<th>PARAMETERS SOLVED</th>
</tr>
</thead>
<tbody>
<tr>
<td>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) $\xi, p$, two-group age, fast diffusion constant. 2) Moments of the slowing down distribution.</td>
<td></td>
<td></td>
</tr>
</tbody>
</table>

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

P-6
MUFT-III

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

Westinghouse Atomic Power Division

<table>
<thead>
<tr>
<th>TITLE</th>
<th>COD. ENGR.</th>
<th>LABORATORY</th>
</tr>
</thead>
<tbody>
<tr>
<td>IBM-650</td>
<td>Coded</td>
<td>Reactor and Nuclear Physics</td>
</tr>
</tbody>
</table>

<table>
<thead>
<tr>
<th>COMPUTER</th>
<th>PROBLEM STATUS</th>
<th>CATEGORY</th>
</tr>
</thead>
<tbody>
<tr>
<td>Fourier-transform-one-dimension</td>
<td>Approximately five minutes</td>
<td>&quot;See Below&quot;</td>
</tr>
</tbody>
</table>

<table>
<thead>
<tr>
<th>GEOMETRY &amp; DIMENSIONS</th>
<th>TYPICAL COMPUTING TIME</th>
<th>PARAMETERS SOLVED</th>
</tr>
</thead>
</table>

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

P-7
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 equations 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**

P-8
Temperature Coefficients of Heterogeneous Reactors (I-24184)

TITLE

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

Numerical Analysis Unit
HAPO
GE
Richland

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

\[ \Delta \frac{K_{eff}}{K_{eff}} = \Delta \frac{f}{f} + \Delta \frac{M}{M} - B^2 H^2 (\Delta M^2) \]

Where \( \Delta \) 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.

Up to 300 energy points.

LIMITATIONS OF THE CODE

HW-36825 (Confidential)

REFERENCES

P-9
Program H - Cylinder Transport Equation

<table>
<thead>
<tr>
<th>TITLE</th>
<th>COMPUTER</th>
<th>PROBLEM STATUS</th>
<th>CATEGORY</th>
</tr>
</thead>
<tbody>
<tr>
<td>E. D. Nix</td>
<td>IBM-701</td>
<td>Programmed</td>
<td>Reactor and Nuclear Physics</td>
</tr>
</tbody>
</table>

<table>
<thead>
<tr>
<th>GEOMETRY &amp; DIMENSIONS</th>
<th>TYPICAL COMPUTING TIME</th>
<th>PARAMETERS SOLVED</th>
</tr>
</thead>
<tbody>
<tr>
<td>Cylinder</td>
<td>30 Minutes (Estimated)</td>
<td>Disadvantage factors, flux, angular distribution of neutrons.</td>
</tr>
</tbody>
</table>

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**

F-10
<table>
<thead>
<tr>
<th>SLAV-I</th>
<th>Brookhaven National Laboratory</th>
</tr>
</thead>
<tbody>
<tr>
<td>TITLE</td>
<td>COG. ENGR.</td>
</tr>
<tr>
<td>UNIVAC</td>
<td>In Use</td>
</tr>
<tr>
<td>COMPUTEER</td>
<td>PROBLEM STATUS</td>
</tr>
<tr>
<td>Slab</td>
<td></td>
</tr>
<tr>
<td>GEOMETRY &amp; DIMENSIONS</td>
<td>TYPICAL COMPUTING TIME</td>
</tr>
<tr>
<td>Slab</td>
<td></td>
</tr>
</tbody>
</table>

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**

P-11
**APPLICATION**

**UNIVAC** | Coded | **Problem Status** | Reactor and Nuclear Physics  
---|---|---|---  
**Bare Homogeneous Reactors** | **Category** | Reactor Bucklings and Fluxes  
**Geometry & Dimensions** | **Typical Computing Time** | **Parameters Solved**  

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

**Problem Description**

**Limitations of the Code**

**References**

P-12
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

P-13
| Program I - Analytic P₃ Calculation | R. R. McCready | Aircraft Nuclear Propulsion Department  
|                                       |               | General Electric Company  
|                                       |               | Cincinnati 15, Ohio |

<table>
<thead>
<tr>
<th>TITLE</th>
<th>COG. ENGR.</th>
<th>LABORATORY</th>
</tr>
</thead>
<tbody>
<tr>
<td>IBM-701</td>
<td>Programmed</td>
<td>Reactor and Nuclear Physics</td>
</tr>
<tr>
<td>COMPUTER</td>
<td>PROBLEM STATUS</td>
<td>CATEGORY</td>
</tr>
<tr>
<td>Cylinder</td>
<td></td>
<td>Disadvantage factor, flux angular distribution</td>
</tr>
</tbody>
</table>

<table>
<thead>
<tr>
<th>GEOMETRY &amp; DIMENSIONS</th>
<th>TYPICAL COMPUTING TIME</th>
<th>PARAMETERS SOLVED</th>
</tr>
</thead>
</table>

Evaluates the analytic solution of the P₃ spherical harmonics equations for cylindrical geometry.

**PROBLEM DESCRIPTION**

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

**LIMITATIONS OF THE CODE**

**REFERENCES**

P-14
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**

P-15
This program is the $P_3$ spherical harmonics calculation for 2-region slab. Given: 

\[ \Sigma_s (\text{cm}^{-1}) = \text{scattering cross section}, \]
\[ \Sigma_a (\text{cm}^{-1}) = \text{total cross section, } (= \Sigma_s + \Sigma_a), \]
\[ \mu = \text{average cosine of scattering angle} \]
\[ X(\text{cm}) = \text{half thickness of region} \]
\[ Q/\Sigma_a (\text{cm}) = \text{source/absorption cross section for each region} \]

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

**LIMITATIONS OF THE CODE**

\[ 0 \leq \frac{Q}{\Sigma_a} \leq 1 \]
\[ 0 \leq \frac{Q}{\Sigma_a} \leq 1 \]

\[ 0 \leq \Sigma_s \leq 32 \]
\[ 0 \leq X \leq 64 \]
\[ 10^{-8} \leq X \leq 10 \]
\[ 0 \leq Q/\Sigma_a \leq 1 \]

**REFERENCES**

RED Memorandum - M. Butler to J. A. Thie
SUBJECT: $P_3$ Calculation for a 2-region Slab (RE-24)
7/28/55

P-16
**WIGNER-WILKINS CALCULATION**

<table>
<thead>
<tr>
<th>TITLE</th>
<th>COG. ENGR.</th>
<th>LABORATORY</th>
</tr>
</thead>
<tbody>
<tr>
<td>IBM-650</td>
<td>Coded</td>
<td>Reactor and Nuclear Physics</td>
</tr>
<tr>
<td>COMPUTER</td>
<td>PROBLEM STATUS</td>
<td>CATEGORY</td>
</tr>
<tr>
<td></td>
<td>Five hours</td>
<td>Flux energy distribution Ave. cross sections</td>
</tr>
<tr>
<td>GEOMETRY &amp; DIMENSIONS</td>
<td>TYPICAL COMPUTING TIME</td>
<td>PARAMETERS SOLVED</td>
</tr>
</tbody>
</table>

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**

P-17
| TEMP - I | J. A. Fleck | Origin. Brookhaven National Laboratory
| TITLE | COG. ENGR. | Prog. & Code - New York University
| 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 initial 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

P-18
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**

P-19
The equation is derived on the assumption of an infinite moderator.

**PARAMETERS SOLVED**

This program is in two parts. The first part computes the values of the kernel function in the Vigner-Wilkins integral equation (see AECI-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.

**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., AECI-2275.

**REFERENCES**

P-20
### MONTE CARLO REACTOR CALCULATION

<table>
<thead>
<tr>
<th>TITLE</th>
<th>S. Podgor</th>
<th>Naval Research Laboratory</th>
</tr>
</thead>
</table>

<table>
<thead>
<tr>
<th>COMPUTER</th>
<th>PROBLEM STATUS</th>
<th>CATEGORY</th>
</tr>
</thead>
<tbody>
<tr>
<td>Spherical Homogeneous; 25 cm. radius</td>
<td>2 hours per 1500 histories</td>
<td>Spatial distribution of thermal neutrons</td>
</tr>
</tbody>
</table>

**PROBLEM DESCRIPTION**

Spherical homogeneous mixture of U 235 and H 2O 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.

**LIMITATIONS**

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

**WRITTEN UP IN**

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

**REFERENCES**

P-21
INTEGRAL NETWORK METHOD
Daniel Schiff
E. H. Bareiss (DTMB)
Westinghouse Atomic Power Division
David Taylor Model Basin

TITIE
COG. ENGR.

UNIVAC
Obsolete

COMPUTER
PROBLEM STATUS

One-dimensional
10.5 cm thick
16 hours

GEOMETRY & DIMENSIONS
TYPICAL COMPUTING TIME

PARAMETERS SOLVED

Numberical 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

P-22
**FLEXIBLE INTEGRAL NETWORK**

**Daniel Schiff**  
**E. H. Bareiss**

**Westinghouse Atomic Power Division**  
**David Taylor Model Basin**

**TITLE**  
**COG. ENGR.**

**NORC (NRL)**  
**Programmed**

**Reactor and Nuclear Physics**

**COMPUTER**  
**PROBLEM STATUS**

**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, which can assume a fixed value for a problem in the range 0.05 ≤ ≤ 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**

P-23
This program calculates a table of $\tau$, for specified values of $k_{ex} = \text{excess reactivity}$, using the equation:

$$k_{ex} = \frac{\ell}{\ell - \rho} \left\{ \frac{\sum_{i=1}^{G} a_i}{1 + \lambda_i \ell} \right\}$$

- $\ell = \text{neutron lifetime}$
- $\rho = \text{fraction of neutrons delayed}$
- $a_i = \text{fraction of delayed neutrons emitted in } i\text{th group}$
- $\lambda_i = \text{decay constant for delayed neutron precursors}$

**LIMITATIONS OF THE CODE**

**REFERENCES**

P-24
Radioactive decay chain quantities $N_i$ described by the differential equations

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

are computed directly.

**PROBLEM DESCRIPTION**

Limited R 5th level

**LIMITATIONS OF THE CODE**


**REFERENCES**

P-25
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, or upon neutron temperature.

**LIMITATIONS OF THE CODE**

HW-33912 (Secret)

HW-36825 (Confidential)

Hanford Work Order I-24184

**REFERENCES**

F-26
Xenon & Iodine Concentration Calculation for Hanford Reactors (I-24119, 24179)  

R. E. McGrath  
Numerical Analysis Unit  
HAPO  
GE  
Richland  

TITLE  

702  
COMPUTER  

Debugged  
PROBLEM STATUS  

Reactor and Nuclear Physics  
CATEGORY  

GEOMETRY & DIMENSIONS  
TYPICAL COMPUTING TIME  

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

P-27
This program fits the flux distributions \( f(x) = \alpha \left[ e^{-\beta x} - \gamma e^{\beta x} \right] \) 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, \( \alpha, \beta, \gamma \) and an estimate of their probable errors.

**Problem Description**

\[ N \leq 68, \text{ where } N \text{ is the number of data points} \]

**Limitations of the Code**


**Subject**: Curve Fitting by Least Squares

**References**

P-28
<table>
<thead>
<tr>
<th>CLOUDY CRYSTAL BALL NEUTRON SCATTERING</th>
<th>B. H. Mount</th>
<th>Westinghouse Atomic Power Division</th>
</tr>
</thead>
</table>

<table>
<thead>
<tr>
<th>TITLE</th>
<th>COMPUTER</th>
<th>CPC</th>
<th>PROBLEM STATUS</th>
<th>TYPICAL COMPUTING TIME</th>
<th>CATEGORY</th>
<th>PARAMETERS SOLVED</th>
</tr>
</thead>
</table>
| COG. ENGR. | In Use | In Use | For moments, 7 minutes | Reactor and Nuclear Physics | 1) Shape elastic and compound elastic scattering at given angle. | 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. |
| LABORATORY | | | Moments and complete angular distribution, 25 minutes | | 2) Partial elastic cross sections |
| | | | | | 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**

P-29
<table>
<thead>
<tr>
<th>CLOUDY CRYSTAL BALL NEUTRON SCATTERING</th>
<th>UNIVAC</th>
<th>COMPUTER</th>
</tr>
</thead>
<tbody>
<tr>
<td>H. Amster (Phys)</td>
<td>In Use</td>
<td>PROBLEM STATUS</td>
</tr>
<tr>
<td>B. H. Mount (Math)</td>
<td></td>
<td>Roughly 6 sec per point of variables mentioned in (1) below</td>
</tr>
<tr>
<td>Westinghouse Atomic Power Division</td>
<td>Reactor and Nuclear Physics</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
</tr>
</tbody>
</table>

<table>
<thead>
<tr>
<th>TITLE</th>
<th>ING. ENGR.</th>
<th>LABORATORY</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td></td>
<td></td>
</tr>
</tbody>
</table>

<table>
<thead>
<tr>
<th>UNIVAC</th>
<th>PROBLEM STATUS</th>
<th>CATEGORY</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Roughly 6 sec per point of variables mentioned in (1) below</td>
<td>1) Shape elastic and compound elastic scattering at given angle</td>
</tr>
<tr>
<td></td>
<td></td>
<td>2) Partial elastic cross sections</td>
</tr>
</tbody>
</table>

<table>
<thead>
<tr>
<th>GEOMETRY &amp; DIMENSIONS</th>
<th>TYPICAL COMPUTING TIME</th>
<th>PARAMETERS SOLVED</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td></td>
<td>1) Computes angular distribution of elastic scattering as a function of angle, initial energy, well depth, well radius, and complex potential component.</td>
</tr>
<tr>
<td></td>
<td></td>
<td>2) Computes differential cross sections and their first four Legendre polynomial expansion coefficients.</td>
</tr>
<tr>
<td></td>
<td></td>
<td>3) Computes penetrabilities to be used in inelastic scattering theory of Hauser and Feshback.</td>
</tr>
</tbody>
</table>

**PROBLEM DESCRIPTION**

1) Treats angular momentum up to L = 12.

**LIMITATIONS OF THE CODE**

**REFERENCES**

P-30
<table>
<thead>
<tr>
<th>SURFACE-MODIFIED OPTICAL MODEL</th>
<th>H. Amster (Phys)</th>
<th>Westinghouse Atomic Power Laboratory</th>
</tr>
</thead>
<tbody>
<tr>
<td>TITLE</td>
<td>COG. ENGR.</td>
<td>LABORATORY</td>
</tr>
<tr>
<td>NORC</td>
<td>Analyzed</td>
<td>Reactor and Nuclear Physics</td>
</tr>
<tr>
<td>COMPUTER</td>
<td>PROBLEM STATUS</td>
<td>CATEGORY</td>
</tr>
<tr>
<td></td>
<td>No estimate</td>
<td>Neutron cross sections</td>
</tr>
<tr>
<td>GEOMETRY &amp; DIMENSIONS</td>
<td>TYPICAL COMPUTING TIME</td>
<td>PARAMETERS SOLVED</td>
</tr>
</tbody>
</table>

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**

P-31
End and Fast Source Corrections to Neutron Measurement in an exponential pile (I-24176)

D. E. Davenport

Numerical Analysis Unit
HAPO
CE
Richland

**TITLE**

**COG. ENGR.**

**LABORATORY**

<table>
<thead>
<tr>
<th>COMPUTER</th>
<th>PROBLEM STATUS</th>
<th>CATEGORY</th>
</tr>
</thead>
<tbody>
<tr>
<td>Parallelopiped</td>
<td>Debugged</td>
<td>Reactor and Nuclear Physics</td>
</tr>
</tbody>
</table>

<table>
<thead>
<tr>
<th>GEOMETRY &amp; DIMENSIONS</th>
<th>TYPICAL COMPUTING TIME</th>
<th>PARAMETERS SOLVED</th>
</tr>
</thead>
</table>

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**

P-32
<table>
<thead>
<tr>
<th>Cell Flux Calculations</th>
<th>Helen Rutgers</th>
<th>N.A.A., INC. Nuclear Engineering and Manufacturing Division</th>
</tr>
</thead>
<tbody>
<tr>
<td><strong>TITLE</strong></td>
<td><strong>COG. ENGR.</strong></td>
<td><strong>LABORATORY</strong></td>
</tr>
<tr>
<td>I.B.M. 701 (Duet Code) Floating Point Arithmetic</td>
<td>In Use</td>
<td>Reactor Survey (One Dimension) (One Group)</td>
</tr>
<tr>
<td><strong>COMPUTER</strong></td>
<td><strong>PROBLEM STATUS</strong></td>
<td><strong>CATEGORY</strong></td>
</tr>
<tr>
<td>Infinite Co-axial Cylinder</td>
<td>One case is computed in approx. 2.00 minutes</td>
<td>Flux distribution &amp; average flux in each region.</td>
</tr>
<tr>
<td><strong>GEOMETRY &amp; DIMENSIONS</strong></td>
<td><strong>TYPICAL COMPUTING TIME</strong></td>
<td><strong>PARAMETERS SOLVED</strong></td>
</tr>
</tbody>
</table>

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
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.
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 \( \gamma \), 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 \( \gamma \) must be less than the critical \( \gamma \). 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**

RS-3
Program B - Two-Region Reflector Savings

REFLECTOR SAVINGS

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

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** |
<table>
<thead>
<tr>
<th></th>
<th></th>
<th></th>
</tr>
</thead>
<tbody>
<tr>
<td><strong>TITLE</strong></td>
<td><strong>COG. ENGR.</strong></td>
<td><strong>LABORATORY</strong></td>
</tr>
</tbody>
</table>
| 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**

RS-5
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 & 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 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

RS-7
Two group - Two region radial problem

Title: IBI 701 Duet Code. (Floating point arithmetic)

Computer: Finite Cylinder with a finite reflector on side

Geometry & Dimensions: Typical computing time

Problem Status: One case is computed in 4.00 min.

Category: Radius of core which is required for criticality

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)

<table>
<thead>
<tr>
<th>TITLE</th>
<th>CPC</th>
<th>In Use</th>
<th>LABORATORY</th>
<th>Reactor Survey (One Dimension) (Two Group)</th>
</tr>
</thead>
<tbody>
<tr>
<td>GEOMETRY &amp; DIMENSIONS</td>
<td>TYPICAL COMPUTING TIME</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>N-Regions</td>
<td>2-Reg.</td>
<td>Cylinder</td>
<td>Other</td>
<td>Critical radius</td>
</tr>
<tr>
<td>Spherical; Cylindrical; slab</td>
<td>3-Reg.</td>
<td>45 min.</td>
<td>15 min.</td>
<td>Reactivity, k</td>
</tr>
<tr>
<td>PARAMETEBS SOLVED</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
</tbody>
</table>

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

RS-9
### Problem Description

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

### 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  

<table>
<thead>
<tr>
<th>TITLE</th>
<th>COG. ENGR.</th>
<th>LABORATORY</th>
</tr>
</thead>
<tbody>
<tr>
<td>IBM-701</td>
<td>Obsolete</td>
<td>Reactor Survey (One Dimension) (Two Group)</td>
</tr>
<tr>
<td>COMPUTE</td>
<td>Replaced by: Program C</td>
<td>PARAMETERS SOLVED</td>
</tr>
<tr>
<td>Unreflected cylinder</td>
<td>1-2 minutes</td>
<td>Reactivity, flux, slowing down density, two-group constants</td>
</tr>
<tr>
<td>GEOMETRY &amp; DIMENSIONS</td>
<td>TYPICAL COMPUTING TIME</td>
<td>PARAMETERS SOLVED</td>
</tr>
</tbody>
</table>

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

IBM-701
In Use

Reactor Survey
(One Dimension)
(Two Group)

Unreflected Sphere, slab or cylinder
1-1/2 - 6 minutes

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

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

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.

REFERENCES

RS-12
**Title**: Reactor Survey (one dimension) (three group)

**Category**: \( \int \) fluxes

\((\text{power curve, neutron balance quantities})\)

**Computer**: UNIVAC

**Problem Status**: in Use

**Geometry & Dimensions**: One-space-dimensional

\(\text{slab, cylinder, sphere};\) three groups

**Typical Computing Time**: 20 seconds per source iteration for approximately 40 space points.

**Parameters Solved**: Finite-difference solution of the three-group problem

\[-D_i \left( \frac{\partial^2 \phi_i}{\partial r^2} + \frac{\partial \phi_i}{\partial r} \right) + A_i \phi_i = \chi_1 \Phi_1 - \sum_{j=1}^{3} \phi_j - 1 \quad i=1,2,3\]

\[\phi_i (r) = 0; \quad \phi_0 (r) = 0; \quad \lambda = \sqrt{\sum F_i \phi_i}\]

Where the range of integration can be subdivided into at most 6 regions in each of which \(D_i, A_i, \chi_i, \Phi_i, \) 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 \(\leq 6\), only the first five of which may be cores.

3. Number of mesh points plus number of regions \(\leq 58\).

4. \(\Delta r\) constant within a region; \(\Delta r > \sqrt{\lambda}\)

5. At the present time, the coefficients \(D_i, A_i, \) etc. are input.

**Limitations of the Code**

**References**

RS-13

45
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

---

<table>
<thead>
<tr>
<th>AVIDAC</th>
<th>In Use</th>
<th>Reactor Survey (one dimension) (three group)</th>
</tr>
</thead>
</table>

<table>
<thead>
<tr>
<th>Sphere</th>
<th>2 hours</th>
<th>Solves for the critical radius</th>
<th>PARAMETERS SOLVED</th>
</tr>
</thead>
</table>

<table>
<thead>
<tr>
<th>GEOMETRY &amp; DIMENSIONS</th>
<th>TYPICAL COMPUTING TIME</th>
<th>GEOMETRY &amp; DIMENSIONS</th>
<th>TYPICAL COMPUTING TIME</th>
</tr>
</thead>
</table>

<table>
<thead>
<tr>
<th>PROBLEM DESCRIPTION</th>
</tr>
</thead>
<tbody>
<tr>
<td>Core radius &lt; 20 cm</td>
</tr>
</tbody>
</table>

<table>
<thead>
<tr>
<th>LIMITATIONS OF THE CODE</th>
</tr>
</thead>
<tbody>
<tr>
<td>ANL-5440 - AVIDAC Setup for Multigroup Serber-Wilson Calculations - H. Hummel</td>
</tr>
</tbody>
</table>

<table>
<thead>
<tr>
<th>REFERENCES</th>
</tr>
</thead>
<tbody>
<tr>
<td>RS-14</td>
</tr>
</tbody>
</table>
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, \( (= \frac{1}{3} \sigma_{tr}) \), \( \sigma_a = \sigma_s + \sigma_{in} + \sigma_f \phi \) = 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 \( \leq 10 \), however, this may be changed without too much difficulty.

\[
\Delta \ell = \Delta \ell - 1, \quad \Delta \ell = \text{interval size of region } \ell
\]

### Limitations of the Code

Memoranda by J. Alexander and D. Okrent

**SUBJECT:** - Part I Theory for IBM Setup of Multigroup
- Part II Instructions for IBM Setup of Multigroup in Spherical Coordinates for Uniform Regions

**REFERENCES**

1-23-53

RS-15
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, ($= 1/3 \sigma_{tr}$), $\sigma_a = \sigma_s + \sigma_{in} + \sigma_{c} + \sigma_{f} \phi$ 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 $\leq 10$, however, this may be changed without too much difficulty.

**LIMITATIONS OF THE CODE**

**REFERENCES**

RS-16
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, ($= 1/3 \sigma_{tr}$), $\sigma_a = \sigma_s + \sigma_{in} + \sigma_c + \sigma_f$ = 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 |
| $\Delta L \leq \Delta L^+$ | $\Delta L^+$ | $1 \leq Z \leq 10$ | $\Delta L^+$ = interval size in region $L$ |
| $1 \leq J \leq 15$ | $J$ = 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 - M. Butler to F. W. Thalgott
SUBJECT: Programming of the Multigroup Spherical Geometry Reactor Calculation for the AVIDAC - 2-19-54

**References**

RS-17
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, \( ( = 1/3 \sigma_{tr}) \), \( \sigma_a = \sigma_s + \sigma_{in} + \sigma_c + \sigma_f \), \( \phi \) = 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

\[
(4+G)N \times (4+2G)L \leq 553 \quad 1 \leq G \leq 6 \\
\Delta l \leq 1 \quad 2 \leq J \leq 20 \\
1 \leq Z \leq 10
\]

\( N \) = number of points \( L \) = number of regions

\( \Delta l \) = interval size in region \( \epsilon \)

\( J \) = number of energy groups

\( G \) = number of energy groups into which inelastic scattering is considered.

### LIMITATIONS OF THE CODE

\( Z \) = number of materials

### REFERENCES

- 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
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**

\[(4+G)N + (4+2G)L \leq 545\]

\[2 \leq J \leq 20\]

\[1 \leq G \leq 6\]

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

N = number of points, L = number of regions

\[\Delta r = \text{interval size in region}\]

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

RS-19
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, ($= 1/3 \sigma_{tr}$), $\sigma_a = \sigma_s + \sigma_i + \sigma_c + \sigma_r \phi$ = 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

\[(4+G)N + (4+2G)L \leq 521\]

$\Delta r < 10 \Delta_{r+1} r$  $\Delta r$ = interval size in region $r$

$2 \leq J \leq 20$  $J$ = number of energy groups

$1 \leq G \leq 6$  $G$ = number of energy groups into which inelastic scattering is considered

$1 \leq Z \leq 10$  $Z$ = 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

REFERENCES Assemblies - D. Okrent

RS-20
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**

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

\[\frac{\Delta l}{r} \leq 10 \frac{\Delta \rho}{r} \quad 1^r \]

\[2 \leq J \leq 20 \]

\[1 \leq G \leq 6 \]

**PARAMETERS SOLVED**

\[N = \text{number of points} \]

\[L = \text{number of regions} \]

\[\Delta \rho = \text{interval size in region} \]

\[J = \text{number of energy groups} \]

**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
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 \((= 1/3 \Sigma_{tr})\), \( \Sigma_a = \Sigma_s + \Sigma_{in} + \Sigma_{sc} + \Sigma_f \phi \) - 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**

\[
(4+G)N + (4+2G)L \leq 553 \quad 1 \leq Z \leq 10 \quad N = \text{number of points} \\
\Delta l < 10 \Delta L + 1 \quad \Delta l = \text{interval size in region} \\
2 \leq J \leq 20 \quad Z = \text{number of materials} \\
1 \leq G \leq 6 \quad G = \text{number of energy groups into which inelastic scattering is considered}
\]

**Limitations of the Code**

- ANL-5437 - UNIVAC Programs for the Solution of One-Dimensional Multigroup Reactor Equations - M. Butler and J. Cooko.
- 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
**TITLE**  
UNIVAC Slab ZC300

**COMPUTER**  
Argonne National Laboratory

**UNIVAC**  
In Use

**PROBLEM STATUS**  
15-20 minutes

**CATEGORY**  
Reactor Survey  
(One Dimension)  
(Multigroup)

**GEOMETRY & DIMENSIONS**  
Solves for critical composition

**TYPICAL COMPUTING TIME**  

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.

**PARAMETERS SOLVED**  

(4+G)N + (4+2G)L ≤ 544  
N = number of points  
L = number of regions  

Δlr < 10 δθ + 1r  
Δlr = interval size in region θ  

2 ≤ J ≤ 20  
J = number of energy groups  

1 ≤ G ≤ 6  
G = number of energy groups into which inelastic scattering is considered  

**LIMITATIONS OF THE CODE**  
0 ≤ disadvantage factor ≤ 1

**REFERENCES**

RS-23

RED Memorandum - M. Butler to F. W. Thalgott  
SUBJECT: UNIVAC Programs RE-6X, 7X, 8X - 5/31/55
This program solves the one-dimensional multigroup adjoint equations. This is the adjoint equivalent of RE-6.

**Problem Description**

\[(4G+N) + (4G+2G)L \leq 546\]

\[\Delta \leq \Delta_{(r)} \leq 1^r\]

\[2 \leq J \leq 20\]

\[1 \leq G \leq 6\]

\[1 \leq Z \leq 10\]

**Limitations of the Code**

- \(N\) = number of points
- \(L\) = number of regions
- \(\Delta\) = 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

**References**

- Reactor Equations - M. Butler and J. Cook

**RED Memorandum**

- D. Okrent to PBR File
  
  **Subject:** Programming of Multigroup Adjoint Solution - 1/11/55

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

**PROBLEM DESCRIPTION**

\[(4+G)N + (4+2G)L \leq 514\]

\[\Delta r \leq 10\Delta r^1\]

\[2 \leq J \leq 20\]

\[1 \leq G \leq 6\]

\[1 \leq Z \leq 10\]

N = number of points  
L = number of regions  
\(\Delta r\) = 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

RS-25
This program solves the one-dimensional multigroup adjoint equations. This is the adjoint equivalent of RE-8.

### Problem Description

\[(4+G)L = N \leq 546\]  
\[\Delta r < 10 \Delta L + 1\]  
\[2 \leq J \leq 20\]  
\[1 \leq G \leq 6\]  
\[1 \leq Z \leq 10\]

- **N** = number of points  
- **L** = number of regions  
- **Δr** = 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

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

**RS-26**
<table>
<thead>
<tr>
<th>Program D - Spherical Multigroup</th>
<th>E. D. Nix</th>
<th>Aircraft Nuclear Propulsion Department General Electric Company Cincinnati 15, Ohio</th>
</tr>
</thead>
<tbody>
<tr>
<td><strong>TITLE</strong></td>
<td>COG, ENGR.</td>
<td><strong>LABORATORY</strong></td>
</tr>
<tr>
<td><strong>IBM-701</strong></td>
<td>In Use</td>
<td><strong>PROBLEM STATUS</strong></td>
</tr>
<tr>
<td><strong>COMPUTER</strong></td>
<td></td>
<td><strong>CATEGORY</strong></td>
</tr>
<tr>
<td><strong>Sphere (One dimensional)</strong></td>
<td>20 Minutes</td>
<td><strong>PARAMETERS SOLVED</strong></td>
</tr>
<tr>
<td><strong>GEOMETRY &amp; DIMENSIONS</strong></td>
<td></td>
<td>Reactivity, flux, and power distribution</td>
</tr>
<tr>
<td><strong>TYPICAL COMPUTING TIME</strong></td>
<td></td>
<td></td>
</tr>
</tbody>
</table>

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
<table>
<thead>
<tr>
<th>Title</th>
<th>Author</th>
<th>Laboratory</th>
</tr>
</thead>
<tbody>
<tr>
<td>D. S. Selengut</td>
<td></td>
<td>Aircraft Nuclear Propulsion</td>
</tr>
<tr>
<td>General Nuclear</td>
<td></td>
<td>Department</td>
</tr>
<tr>
<td>Propulsion</td>
<td></td>
<td>General Electric Company</td>
</tr>
<tr>
<td>Department</td>
<td></td>
<td>Cincinnati, Ohio</td>
</tr>
</tbody>
</table>

<table>
<thead>
<tr>
<th>Computer</th>
<th>Status</th>
<th>Category</th>
</tr>
</thead>
<tbody>
<tr>
<td>IBM 701</td>
<td>Coded</td>
<td>Reactor Survey (One Dimension) (Multigroup)</td>
</tr>
</tbody>
</table>

<table>
<thead>
<tr>
<th>Geometry &amp; Dimensions</th>
<th>Typical Computing Time</th>
<th>Parameters Solved</th>
</tr>
</thead>
<tbody>
<tr>
<td>Sphere, slab, or</td>
<td></td>
<td>Reactivity, flux,</td>
</tr>
<tr>
<td>cylinder (one</td>
<td></td>
<td>and power</td>
</tr>
<tr>
<td>dimensional)</td>
<td></td>
<td>distribution</td>
</tr>
</tbody>
</table>

Will be an 18 group version of Program F.

**Problem Description**

Same as for Program F.

**Limitations of the Code**

**References**

RS-28
One-space dimensional (slab, cylinder, sphere); fifteen groups

**GEOMETRY & DIMENSIONS**

<table>
<thead>
<tr>
<th>UNIVAC</th>
<th>COMPUTER</th>
<th>REACTOR SURVEY (ONE DIMENSION) (MULTIGROUP)</th>
</tr>
</thead>
<tbody>
<tr>
<td>In Use</td>
<td>Problem Status</td>
<td>TYPICAL COMPUTING TIME</td>
</tr>
<tr>
<td>80 seconds per source iteration for 40 space points</td>
<td>( \nu, \phi, \text{fluxes, (power curve, neutron balance quantities)} )</td>
<td></td>
</tr>
</tbody>
</table>

**PARAMETERS SOLVED**

Finite-difference solution of

\[
-D \left( \frac{d^2 \Phi}{dr^2} + \frac{\Phi}{r^2} \right) + A \Phi = - \left( \frac{\partial \Phi}{\partial u} + \frac{\Phi}{u} \right) + \chi \Phi^0 \leq r \leq R \quad 0 \leq u \leq \mu
\]

\[
\frac{\partial \Phi}{\partial u} + q = A \cdot \Phi \quad \Phi(0, u) = 0
\]

\[
R = \nu \int_0^\mu F \phi du
\]

where the range of integration can be subdivided into at most 6 regions in each of which \( D, A, S, F, \) and \( \Delta \) 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 \( \leq 6 \), only the first 5 of which can be cores.
3. Number of mesh points plus number of regions \( \leq 58 \).
4. \( \Delta r \) constant within a region; \( \Delta r > \sqrt{A} \)
5. Group-averaged microscopic cross sections on tape.

**LIMITATIONS OF THE CODE**

**REFERENCES**

RS-29
<table>
<thead>
<tr>
<th>650 MULTIGROUP</th>
<th>C. J. Habetler</th>
<th>Knolls Atomic Power Laboratory</th>
</tr>
</thead>
<tbody>
<tr>
<td>TITLE</td>
<td>COG. ENGR.</td>
<td>LABORATORY</td>
</tr>
<tr>
<td>IBM 650</td>
<td>In Use</td>
<td>Reactor Survey (One Dimension)</td>
</tr>
<tr>
<td></td>
<td></td>
<td>(Multigroup)</td>
</tr>
<tr>
<td>COMPUTER</td>
<td>PROBLEM STATUS</td>
<td>CATEGORY</td>
</tr>
<tr>
<td>One-Space-Dimension (slab, cylinder, sphere) lethargy</td>
<td>1 group of approximately 40 space points in 1 minute.</td>
<td>( \gamma; \text{ neutron flux (per space point and group)}; \text{ power curve.} )</td>
</tr>
<tr>
<td>GEOMETRY &amp; DIMENSIONS</td>
<td>TYPICAL COMPUTING TIME</td>
<td>PARAMETERS SOLVED</td>
</tr>
</tbody>
</table>
| Finite - Difference Solution of \[-D (\frac{d\Phi}{d \xi} + \frac{\Phi}{r}) + A \Phi = - (\frac{\delta E}{\delta u} + \frac{\delta N}{3u}) + X \Phi \quad 0 < r < R \]
| \[0 < u < u_t\] | \[\frac{\delta F}{\delta u} = \lambda \cdot \Phi \quad \Phi(r, 0) = 0\] | \[\lambda = \int F \cdot \Phi \, du\] |
| Where the range of integration can be subdivided into regions in each of which \(D, A, S, F, \) and \(\lambda\) 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

RS-30
<table>
<thead>
<tr>
<th>Title</th>
<th>CoG. Engr.</th>
<th>Laboratory</th>
</tr>
</thead>
<tbody>
<tr>
<td>Multigroup Reactor Analysis (Speed Co.)</td>
<td>Melba Nead</td>
<td>North American Aviation, Inc. Nuclear Engineering &amp; Manufacturing Division</td>
</tr>
<tr>
<td>Computer</td>
<td>Problem Status</td>
<td>Reactor Survey (One Dimension) (Multigroup)</td>
</tr>
<tr>
<td>Spherical Geometry &amp; Dimensions</td>
<td>Typical Computing Time</td>
<td>Parameters Solved</td>
</tr>
</tbody>
</table>

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
<table>
<thead>
<tr>
<th>Multigroup Reactor Analysis (Duet)</th>
<th>Melba Nead</th>
<th>North American Aviation Inc. Nuclear Engineering &amp; Mfg. Division</th>
</tr>
</thead>
<tbody>
<tr>
<td>TITIE</td>
<td>COG. ENGR.</td>
<td>LABORATORY</td>
</tr>
<tr>
<td>701</td>
<td>Coded</td>
<td>Reactor Survey (One Dimension) (Multigroup)</td>
</tr>
<tr>
<td>COMPUTER</td>
<td>PROBLEM STATUS</td>
<td>CATEGORY</td>
</tr>
<tr>
<td>Spherical</td>
<td>20 minutes</td>
<td></td>
</tr>
<tr>
<td>GEOMETRY &amp; DIMENSIONS</td>
<td>TYPICAL COMPUTING TIME</td>
<td>PARAMETERS SOLVED</td>
</tr>
</tbody>
</table>

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 $V_0$ 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
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
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.

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 x 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
<table>
<thead>
<tr>
<th>MUG II</th>
<th>UNIVAC</th>
<th>COMPUTE</th>
<th>CATEGORY</th>
</tr>
</thead>
<tbody>
<tr>
<td>Knolls Atomic Power Laboratory</td>
<td>Reactor Survey (Two Dimension) (Multigroup)</td>
<td>In Use</td>
<td></td>
</tr>
</tbody>
</table>

<table>
<thead>
<tr>
<th>TITLE</th>
<th>PROBLEM STATUS</th>
<th>GEOMETRY &amp; DIMENSIONS</th>
<th>PARAMETERS SOLVED</th>
</tr>
</thead>
<tbody>
<tr>
<td>COG. ENGR.</td>
<td>(Two Dimension) (Multigroup)</td>
<td>Power surface, Flux Distribution Reactivity</td>
<td></td>
</tr>
</tbody>
</table>

**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. \((r, \theta)\) often gives convergence troubles.
5. No good edit exists.

**LIMITATIONS OF THE CODE**

<table>
<thead>
<tr>
<th>REFERENCES</th>
</tr>
</thead>
<tbody>
<tr>
<td>KAPL 950</td>
</tr>
<tr>
<td>KAPL 1068</td>
</tr>
<tr>
<td>KAPL 1096</td>
</tr>
<tr>
<td>KAPL 1172</td>
</tr>
<tr>
<td>KAPL 1245</td>
</tr>
<tr>
<td>KAPL 1321</td>
</tr>
</tbody>
</table>

\[
-D \nabla^2 \Phi + \frac{A}{\beta} \left( \frac{\partial \Phi}{\partial u} + \frac{\partial \Phi}{\partial u} \right) + \chi \| \Phi = \text{slowing down by heavy atoms} \\
\eta \| \Phi = \text{slowing down by hydrogen atoms} \\
\Phi = 0 \quad \text{or} \quad -D \nabla \Phi \| \Phi \text{ at boundaries}
\]

68
<table>
<thead>
<tr>
<th>TOSFY</th>
<th>E. Wachspress</th>
<th>Knolls Atomic Power Laboratory</th>
</tr>
</thead>
<tbody>
<tr>
<td>TITLE</td>
<td>COG. ENGR.</td>
<td>LABORATORY</td>
</tr>
<tr>
<td>NORC or UNIVAC</td>
<td>Analyzed</td>
<td>Reactor Survey</td>
</tr>
<tr>
<td></td>
<td></td>
<td>(Two Dimension)</td>
</tr>
<tr>
<td>COMPUTER</td>
<td>PROBLEM STATUS</td>
<td>(Multigroup)</td>
</tr>
<tr>
<td>X-Y r-z r-θ</td>
<td>Unknown</td>
<td>CATEGOR\</td>
</tr>
<tr>
<td>GEOMETRY &amp; DIMENSIONS</td>
<td>TYPICAL COMPUTING TIME</td>
<td>PARAMETERS SOLVED</td>
</tr>
</tbody>
</table>

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**

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

**REFERENCES**

RS-37
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 symmetric source by hydrogen.)

H. B. Keller
Jack Heller

AEC Computing Facility
New York University

<table>
<thead>
<tr>
<th>TITLE</th>
<th>COG. ENGR.</th>
<th>LABORATORY</th>
</tr>
</thead>
<tbody>
<tr>
<td>UNIVAC COMPUTER</td>
<td>Analyzed:</td>
<td>Coded:</td>
</tr>
<tr>
<td></td>
<td>In Use: X</td>
<td>Programmed:</td>
</tr>
<tr>
<td></td>
<td>Debugged:</td>
<td>Obsolete:</td>
</tr>
<tr>
<td>PROBLEM STATUS</td>
<td>Shielding</td>
<td></td>
</tr>
<tr>
<td>Replaced by:</td>
<td>CATEGORY</td>
<td></td>
</tr>
<tr>
<td>Plane Geometry (Symmetric)</td>
<td>50 hours for 100 Cm. shield and 50 energy steps.</td>
<td></td>
</tr>
</tbody>
</table>

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

S-1
**Title:** Neutron Diffusion  
**Authors:** R. B. Theus  
**Institute:** Naval Research Laboratory

<table>
<thead>
<tr>
<th>TITLE</th>
<th>COG. ENGR.</th>
<th>LABORATORY</th>
</tr>
</thead>
<tbody>
<tr>
<td>Neutron Diffusion</td>
<td>R. B. Theus</td>
<td>Naval Research Laboratory</td>
</tr>
</tbody>
</table>

<table>
<thead>
<tr>
<th>COMPUTER</th>
<th>PROBLEM STATUS</th>
<th>Replaced by</th>
<th>CATEGORY</th>
</tr>
</thead>
<tbody>
<tr>
<td>Infinite Medium</td>
<td>0.5 Sec/Hist.</td>
<td>Shielding</td>
<td></td>
</tr>
</tbody>
</table>

**Problem Description:**

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**

S-2
## PROGRAM O-G

**W. E. Edwards**  
**J. E. MacDonald**  
**Aircraft Nuclear Propulsion Department**  
**General Electric Company**  
**Cincinnati 15, Ohio**

<table>
<thead>
<tr>
<th>TITLE</th>
<th>COG. ENGR.</th>
<th>LABORATORY</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td></td>
<td></td>
</tr>
</tbody>
</table>

<table>
<thead>
<tr>
<th>COMPUTER</th>
<th>IBM-701</th>
<th></th>
</tr>
</thead>
<tbody>
<tr>
<td>In Use:</td>
<td>×</td>
<td>Programmed:</td>
</tr>
<tr>
<td></td>
<td></td>
<td>Shielding</td>
</tr>
<tr>
<td>Debugged:</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Obsolete:</td>
<td></td>
<td></td>
</tr>
<tr>
<td>PROBLEM STATUS</td>
<td>Replaced by:</td>
<td>CATEGORY</td>
</tr>
<tr>
<td></td>
<td></td>
<td>Neutron dose rate in and around reactor shield</td>
</tr>
</tbody>
</table>

<table>
<thead>
<tr>
<th>GEOMETRY &amp; DIMENSIONS</th>
<th>TYPICAL COMPUTING TIME</th>
<th>PARAMETERS SOLVED</th>
</tr>
</thead>
<tbody>
<tr>
<td>Cylindrical</td>
<td>5 sec per source-receiver path</td>
<td>Neutron dose rate in and around reactor shield</td>
</tr>
</tbody>
</table>

**Problem Description**

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.

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**

S-3
<table>
<thead>
<tr>
<th>Program 1-G</th>
<th>W. E. Edwards</th>
<th>Aircraft Nuclear Propulsion Department General Electric Company Cincinnati 15, Ohio</th>
</tr>
</thead>
<tbody>
<tr>
<td>TITLE</td>
<td>IBM-701</td>
<td>COMPUTER</td>
</tr>
<tr>
<td>TITLE</td>
<td>Cylindrical</td>
<td>Analyzed: Coded:</td>
</tr>
<tr>
<td></td>
<td></td>
<td>In Use: X Programmed:</td>
</tr>
<tr>
<td></td>
<td></td>
<td>Debugged: Obsolete:</td>
</tr>
<tr>
<td>LABORATORY</td>
<td></td>
<td>Shielding</td>
</tr>
<tr>
<td>GEOMETRY &amp; DIMENSIONS</td>
<td></td>
<td></td>
</tr>
<tr>
<td>TYPICAL COMPUTING TIME</td>
<td></td>
<td></td>
</tr>
<tr>
<td>PARAMETERS SOLVED</td>
<td></td>
<td></td>
</tr>
</tbody>
</table>

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

<table>
<thead>
<tr>
<th>TITLE</th>
<th>COG. ENGR.</th>
<th>LABORATORY</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>W. E. Edwards</td>
<td>Aircraft Nuclear Propulsion</td>
</tr>
<tr>
<td></td>
<td>J. E. MacDonald</td>
<td>Department</td>
</tr>
<tr>
<td></td>
<td></td>
<td>General Electric Company</td>
</tr>
<tr>
<td></td>
<td></td>
<td>Cincinnati 15, Ohio</td>
</tr>
</tbody>
</table>

**COMPUTER**

<table>
<thead>
<tr>
<th></th>
<th>PROBLEM STATUS</th>
<th>Replaced by:</th>
<th>CATEGORY</th>
</tr>
</thead>
<tbody>
<tr>
<td>In Use:</td>
<td>×</td>
<td>Programmed:</td>
<td>Shielding</td>
</tr>
<tr>
<td>Debugged:</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td>Obsolete:</td>
<td></td>
</tr>
</tbody>
</table>

**GEOMETRY & DIMENSIONS**

<table>
<thead>
<tr>
<th>TYPICAL COMPUTING TIME</th>
<th>PARAMETERS SOLVED</th>
</tr>
</thead>
<tbody>
<tr>
<td>4 seconds</td>
<td>Neutron dose rates in and around reactor shield.</td>
</tr>
</tbody>
</table>

**PROBLEM DESCRIPTION**

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

**LIMITATIONS OF THE CODE**

XDC 55-6-109

**REFERENCES**

S-5
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
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 $\gamma$-Ray Penetration.

NRL Report #4412
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 $\gamma$-Ray Penetration.

NRL Report #4412

**REFERENCES**
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**

S-9
Influence function for gamma ray emitting fission products

<table>
<thead>
<tr>
<th>TITLE</th>
<th>COG. ENGR.</th>
<th>LABORATORY</th>
</tr>
</thead>
<tbody>
<tr>
<td>701</td>
<td>Analyzed:</td>
<td>Coded:</td>
</tr>
<tr>
<td></td>
<td>In Use:</td>
<td>Programmed:</td>
</tr>
<tr>
<td></td>
<td>Debugged:</td>
<td>Obsolete</td>
</tr>
<tr>
<td>COMPUTER</td>
<td>PROBLEM STATUS</td>
<td>Replaced by:</td>
</tr>
<tr>
<td></td>
<td>2 min.</td>
<td>Shielding</td>
</tr>
<tr>
<td>GEOMETRY &amp; DIMENSIONS</td>
<td>TYPICAL COMPUTING TIME</td>
<td>PARAMETERS SOLVED</td>
</tr>
</tbody>
</table>

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

S-10
Saturated activities of foils placed in Pile Shielding (1-2415)

**TITL e:**
702 EDPM

**COMPUTER**

Analyzed: Coded: 
In Use: Programmed:
Debugged: Obsolete:

PROBLEM STATUS Replaced by:

**GEOMETRY & DIMENSIONS**

TYPICAL COMPUTING TIME

PARAMETERS SOLVED

250 measurements/min

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**

Reference:

**REFERENCES**
<table>
<thead>
<tr>
<th>TITLE</th>
<th>COG. ENGR.</th>
<th>LABORATORY</th>
</tr>
</thead>
<tbody>
<tr>
<td>AVIDAC</td>
<td>Obsolete</td>
<td>Argonne National Laboratory</td>
</tr>
<tr>
<td>COMPUTER</td>
<td>Replaced by: 12, 13, or 17</td>
<td>Reactor Kinetics</td>
</tr>
<tr>
<td>PROBLEM STATUS</td>
<td>15 minutes</td>
<td>CATEGORY</td>
</tr>
<tr>
<td>GEOMETRY &amp; DIMENSIONS</td>
<td>TYPICAL COMPUTING TIME</td>
<td>PARAMETERS SOLVED</td>
</tr>
</tbody>
</table>

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

\[
\frac{dn}{dt} = \frac{1}{\lambda} n - \frac{dc}{dt} \quad \frac{dk}{dt} = -\frac{k_{ex}}{\lambda} Dn + E
\]

\[
\frac{dc}{dt} = \left(1 + k_{ex} \right) \frac{\beta}{\lambda} n - \lambda_c
\]

\[\beta = \text{delayed neutron decay rate} \]
\[\lambda_c = \text{delayed neutron fraction} \]

**LIMITATIONS OF THE CODE**

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

**REFERENCES**

K-1
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:

\[
\frac{dx_1}{dt} = \frac{x_{n+1} (1 - \beta)}{\rho} x_1 + \sum_{i=2}^{n} b_{1,i} x_i = f_1 (x_1, \ldots, x_{n+1})
\]

\[
\frac{dx_i}{dt} = \frac{(x_{n+1})^i - 1}{\rho} x_i + \sum_{i=2}^{n} b_{j,i} x_i = f_j (x_1, \ldots, x_{n+1})
\]

\[i = 2, 3, \ldots, n\]

\[
\frac{dx_{n+1}}{dt} = A + Bx_1 + Cx_2 + Dx_3 + Et + Ft^2 + Gt^3 = f_{n+1}(x_1, t)
\]

and \(b_j = \lambda_j\) = delayed neutron decay constants \(\lambda =\) neutron lifetime,

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

\[
\frac{d n}{d t} = \frac{k_{\text{ex}} (1 - \beta)}{\mathcal{E}} n + \sum_{i=1}^{6} \lambda_i C_i + S
\]

\[
\frac{d C_i}{d t} = \frac{(k_{\text{ex}} + 1) \beta_i}{\mathcal{E}} n - \lambda_i C_i i = 1, 2 \ldots 6
\]

**Problem Description**

- \( n \) = neutron flux
- \( k_{\text{ex}} \) = excess reactivity
- \( t_{\text{ndt}} \) = integral of the flux
- and if desired, \( n/dn/dt \) = pile period as a function of time
- PARAMETERS SOLVED

**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**

K-3
This program solves the one energy group reactor kinetics equations with 6 delay groups by a Taylor series approximation. The equations solved are:

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

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

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

**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
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.
| RE-22 | L. Sommerfeld  
<table>
<thead>
<tr>
<th></th>
<th>M. Butler</th>
<th>Argonne National Laboratory</th>
</tr>
</thead>
<tbody>
<tr>
<td>TITLE</td>
<td>COG. ENGR.</td>
<td>LABORATORY</td>
</tr>
<tr>
<td>AVIDAC</td>
<td>In Use</td>
<td>Reactor Kinetics</td>
</tr>
<tr>
<td>COMPUTER</td>
<td>PROBLEM STATUS</td>
<td></td>
</tr>
<tr>
<td></td>
<td>15 Minutes</td>
<td>k_ex \int_0^t ndt, \frac{dk_{ex}}{dt} and \frac{d^2 k_{ex}}{dt^2} as a function of time.</td>
</tr>
<tr>
<td>GEOMETRY &amp; DIMENSIONS</td>
<td>TYPICAL COMPUTING TIME</td>
<td>PARAMETERS SOLVED</td>
</tr>
</tbody>
</table>

This program is similar to RE-21. It uses the one energy group reactor kinetics equations with up to 10 delay groups to compute \( k_{ex} \), \( \frac{dk_{ex}}{dt} \) and \( \frac{d^2 k_{ex}}{dt^2} \) 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

K-6
This program computes $k_{\text{ex}}$ and $\rho$ for specified $\lambda =$ neutron lifetime, $\rho = \text{fraction of neutrons delayed, } a_i = \text{fraction of delayed neutrons emitted in } i\text{th group, } \lambda_i = \text{decay constant for delayed neutrons precursors, and } m_0,$ from the equation:

$$
\begin{align*}
k_{\text{ex}} &= \frac{\lambda}{2m} + \rho \sum_{i=1}^{6} \frac{a_i}{1+2^m \lambda_i} \\
\rho &= \frac{1}{1 + k_{\text{ex}}} \\
\end{align*}
$$

where $m$ goes from $-m_0(1)$ to $m_0$ and $2^m = L_i \times \chi_j$.

- $10^{-9} \leq \lambda \leq 10^{-3}$
- $10^{-3} \leq \rho \leq 10^{-1}$
- $0 \leq a_i \leq 1$
- $0 \leq \lambda_i \leq 10^2$
- $1 \leq m_0 \leq 20$
- $0 \leq k_{\text{ex}} \leq 0.2$

**LIMITATIONS OF THE CODE**

**REFERENCES**

K-7
<table>
<thead>
<tr>
<th>CHAIM</th>
<th>TITLE</th>
<th>UNIVAC</th>
<th>PROBLEM STATUS</th>
<th>COMPUTER</th>
<th>PROBLEM STATUS</th>
<th>GEOMETRY &amp; DIMENSIONS</th>
<th>TYPICAL COMPUTING TIME</th>
<th>PARAMETERS SOLVED</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Origin - Brookhaven National Laboratory</td>
<td>In Use</td>
<td>Reactor Kinetics</td>
<td>Cylindrical</td>
<td>Reactor Periods</td>
<td>Typical Computing Time</td>
<td>Parameters Solved</td>
<td></td>
</tr>
</tbody>
</table>

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
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**


J. Fleck - Temperature dependent kinetics of circulating fuel reactors, Brookhaven National Laboratory forthcoming report.
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

RB-1
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
Endurance study of typical SIR-type reactor takes 2 to 3 hours.

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 $\Delta t$.

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 $\leq 57$.
4. Number of burnable poisons present in a given problem $\leq 2$.
5. Number of fission products allowed $\leq 1$.

LIMITATIONS OF THE CODE

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

REFERENCES

RB-3
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 $\Delta t$.

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 $\leq 58$.
4. $\kappa_{\text{eff}}$ constant within a region; $\Delta \nu > \nu \Delta T$
5. Number of burnable poisons present in a given problem $\leq 3$.
6. Number of fission products $\leq 2$.

**Limitations of the Code**

**References**

RB-4
**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.
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.

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

LIMITATIONS OF THE CODE

KAPL - 1239

REFERENCES

E-1
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

**Title**: COMPTER Slab Geo Various Dimensions

<table>
<thead>
<tr>
<th>Title</th>
<th>COG. ENGR.</th>
<th>Laboratory</th>
</tr>
</thead>
<tbody>
<tr>
<td>701</td>
<td>Debugged</td>
<td>Reactor Engineering</td>
</tr>
</tbody>
</table>

**Computer**

<table>
<thead>
<tr>
<th>Slab Geo Various Dimensions</th>
<th>Typical Computing Time</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td></td>
</tr>
</tbody>
</table>

**Problem Description**

Transient temperature in the corner of a reactor shield.

**Limitations of the Code**

**References**

E-3
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**

E-4
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**

E-5
<table>
<thead>
<tr>
<th>TITLE</th>
<th>COG. ENGR.</th>
<th>LABORATORY</th>
</tr>
</thead>
<tbody>
<tr>
<td>Program E - Special Functions</td>
<td>T. L. Glahn (Bessel Functions)</td>
<td>Aircraft Gas Turbine Dev. Dept. General Electric Company Cincinnati 15, Ohio</td>
</tr>
<tr>
<td></td>
<td>E. Isenberg (exponential integrals)</td>
<td></td>
</tr>
</tbody>
</table>

<table>
<thead>
<tr>
<th>COMPUTER</th>
<th>PROBLEM STATUS</th>
<th>CATEGORY</th>
</tr>
</thead>
<tbody>
<tr>
<td>IBM-701</td>
<td>In Use</td>
<td>Miscellaneous</td>
</tr>
</tbody>
</table>

<table>
<thead>
<tr>
<th>GEOMETRY &amp; DIMENSIONS</th>
<th>TYPICAL COMPUTING TIME</th>
<th>PARAMETERS SOLVED</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>0.1 to 2 seconds</td>
<td></td>
</tr>
</tbody>
</table>

(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 $n > 0$, 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**

M-1
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**

<table>
<thead>
<tr>
<th>TYPICAL COMPUTING TIME</th>
<th>PARAMETERS SOLVED</th>
</tr>
</thead>
</table>
| Fouser computes \[ B_n = \frac{1}{2\pi} \sum_{0}^{2\pi} \sin n x d \bar{x}, \quad R_n = \frac{1}{2\pi} \sum_{0}^{2\pi} \cos n x d \bar{x} \] and \[ F = \frac{1}{2\pi} \sum_{0}^{2\pi} x^2 (A_n^2 + B_n^2) \text{ by exact integration for any} \]
| \[ \text{empirical periodic function given as an odd number of points } \left( \bar{Y}_i, \bar{X}_i \right) \]
| \[ \text{at not necessarily equispaced points } 0 = \bar{X}_0 < \bar{X}_1 < \cdots < \bar{X}_n = 2\pi, \]
| \[ \text{Input consists of points } \left( \bar{Y}_i, \bar{X}_i \right) \text{ taken from oscillograph tracing,} \]
| \[ \text{or computed, etc. Transformations take these into } \left( \bar{Y}_i, \bar{X}_i \right). \]

**PROBLEM DESCRIPTION**

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

**LIMITATIONS OF THE CODE**

**REFERENCES**

M-2