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ARGONNE CODE CENTER:
COMPILATION OF PROGRAM ABSTRACTS

by

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Applied Mathematics Division

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<table>
<thead>
<tr>
<th>PAGES TO BE INSERTED</th>
<th>PAGES TO BE REMOVED</th>
<th>COMMENT</th>
</tr>
</thead>
<tbody>
<tr>
<td>1-8.2</td>
<td>1-8.2</td>
<td>Revised cover, title page and table of contents</td>
</tr>
<tr>
<td>76.1-76.2</td>
<td>76.1-76.2</td>
<td>Reactor Code Abstract 80 revised</td>
</tr>
<tr>
<td>87-90</td>
<td>87-90</td>
<td>Reactor Code Abstract 103 revised</td>
</tr>
<tr>
<td>277-278</td>
<td>277-278</td>
<td>Reactor Code Abstract 212 revised</td>
</tr>
<tr>
<td>291-292</td>
<td>291-292</td>
<td>Reactor Code Abstract 219 revised</td>
</tr>
<tr>
<td>377-378.2</td>
<td>377-378</td>
<td>Reactor Code Abstract 262 revised</td>
</tr>
<tr>
<td>419-420</td>
<td>419-420</td>
<td>Reactor Code Abstract 282 revised</td>
</tr>
<tr>
<td>451-456</td>
<td>451-456</td>
<td>Reactor Code Abstract 298 revised</td>
</tr>
<tr>
<td>509-510</td>
<td>509-510</td>
<td>Reactor Code Abstract 325 revised</td>
</tr>
<tr>
<td>579-751</td>
<td>579-683</td>
<td>Reactor Code Abstracts 358 through 384 added; Section VI Library Programs by Classification revised; Section VII KWIC Index revised; References replaced</td>
</tr>
</tbody>
</table>
# TABLE OF CONTENTS

<table>
<thead>
<tr>
<th>NO.</th>
<th>KWIC TITLE</th>
<th>INSTALLATION MACH</th>
<th>LANG</th>
<th>PACKAGE</th>
</tr>
</thead>
<tbody>
<tr>
<td>7</td>
<td>FIRN, 2-C FEW-GP S4 APPROXIMATION RZ GEOMETRY</td>
<td>LRL 709 F2</td>
<td>RS</td>
<td></td>
</tr>
<tr>
<td>9</td>
<td>FIRE5, 1-D AGE-DIFFUSION SLAB CYLINDER SPHERE</td>
<td>RDA 360 F4</td>
<td>RS</td>
<td>PLT</td>
</tr>
<tr>
<td>14</td>
<td>HAFFER, HAUSER-FESHBACH INELASTIC SCATTERING</td>
<td>CDC 1604 F63</td>
<td>RS</td>
<td>P</td>
</tr>
<tr>
<td>18</td>
<td>Z0XY, 2-C MULTI-GP SN APPROXIMATION X Y GEOM</td>
<td>AGC 7090 FLOCO</td>
<td>RS</td>
<td>BSBP</td>
</tr>
<tr>
<td>28</td>
<td>FOG, 1-C FEW-GP DIFFUSION SLAB CYLINDER SPHERE</td>
<td>AI 7090 F2</td>
<td>RS</td>
<td>P T</td>
</tr>
<tr>
<td>28</td>
<td>FOG, 1-C FEW-GP DIFFUSION SLAB CYLINDER SPHERE</td>
<td>CDC 1604 F63</td>
<td>RS</td>
<td>P T</td>
</tr>
<tr>
<td>28</td>
<td>FOG, 1-C FEW-GP DIFFUSION SLAB CYLINDER SPHERE</td>
<td>BC 625 F4</td>
<td>RS</td>
<td>T</td>
</tr>
<tr>
<td>29</td>
<td>AIM6, 1-C MULTI-GP DIFFUSION SLAB CYL SPHERE</td>
<td>AI 7090 F+FAP</td>
<td>RS</td>
<td>PLT</td>
</tr>
<tr>
<td>29</td>
<td>AIM6, 1-C MULTI-GP DIFFUSION CYL SPHERE</td>
<td>CDC 1604 F63</td>
<td>RS</td>
<td>PLT</td>
</tr>
<tr>
<td>30</td>
<td>PERT, 1-C PERTURBATION FOR AIM AND FOG CODES</td>
<td>AI 7090 F2</td>
<td>RS</td>
<td></td>
</tr>
<tr>
<td>30</td>
<td>PERT, 1-C PERTURBATION FOR AIM AND FOG CODES</td>
<td>CDC 1604 F63</td>
<td>RS</td>
<td></td>
</tr>
<tr>
<td>30</td>
<td>PERT, 1-C PERTURBATION FOR AIM AND FOG CODES</td>
<td>BHSC 360 F4</td>
<td>RS</td>
<td>P</td>
</tr>
<tr>
<td>32</td>
<td>WHIRLAWAY, 3-D 2-GROUP DIFFUSION XYZ GEOMETRY</td>
<td>ORNL 7090 F2</td>
<td>RS</td>
<td>P T</td>
</tr>
<tr>
<td>32</td>
<td>WHIRLAWAY, 3-D 2-GROUP DIFFUSION XYZ GEOMETRY</td>
<td>CDC 1604 F63</td>
<td>RS</td>
<td>P T</td>
</tr>
<tr>
<td>33</td>
<td>GAM1, FAST NEUTRON SPECTRUM X-SECTION CALC</td>
<td>GGA 7090 F2</td>
<td>RS</td>
<td>PLT</td>
</tr>
<tr>
<td>33</td>
<td>GAM1, FAST NEUTRON SPECTRUM X-SECTION CALC</td>
<td>CDC 1604 F63</td>
<td>RS</td>
<td>PLT</td>
</tr>
<tr>
<td>33</td>
<td>GAM1, FAST NEUTRON SPECTRUM X-SECTION CALC</td>
<td>ANL 3600 F63</td>
<td>RS</td>
<td>LT</td>
</tr>
<tr>
<td>39</td>
<td>EQUIPOIES3, 2-D 2-GROUP DIFFUSION SLAB CYL</td>
<td>ORNL 7090 F2</td>
<td>RS</td>
<td>P T</td>
</tr>
<tr>
<td>40</td>
<td>20GRAND, 2-C FEW-GROUP DIFFUSION SLAB CYL</td>
<td>ORNL 7090 F2</td>
<td>RS</td>
<td>P T</td>
</tr>
<tr>
<td>40</td>
<td>20GRAND, 2-C FEW-GROUP DIFFUSION SLAB CYL</td>
<td>CDC 1604 F63</td>
<td>RS</td>
<td>P T</td>
</tr>
<tr>
<td>41</td>
<td>ZUT, RESOLVED REGION RESONANCE INTEGRAL CALC</td>
<td>GGA 7090 F+FAP</td>
<td>RS</td>
<td>BSB</td>
</tr>
<tr>
<td>42</td>
<td>TUI2, UNRESOLVED REGION RESONANCE INTEGRAL CALC</td>
<td>GGA 7090 F+FAP</td>
<td>RS</td>
<td>T</td>
</tr>
<tr>
<td>43</td>
<td>CURFIT, CURVE FITTING EXPERIMENTAL DATA POINTS KAPL</td>
<td>KAPL 6600 F+FAP</td>
<td>RS</td>
<td>P T</td>
</tr>
<tr>
<td>45</td>
<td>GRACE1, GAMMA-RAY ATTENUATION SLAB GEOMETRY</td>
<td>AI 7090 F2</td>
<td>RS</td>
<td>P</td>
</tr>
<tr>
<td>45</td>
<td>GRACE1, GAMMA-RAY ATTENUATION SLAB GEOMETRY</td>
<td>CDC 1604 F63</td>
<td>RS</td>
<td>P</td>
</tr>
<tr>
<td>46</td>
<td>GRACE2, GAMMA-RAY ATTENUATION CYL SPHERE GEOM</td>
<td>AI 7090 F2</td>
<td>RS</td>
<td>P</td>
</tr>
<tr>
<td>46</td>
<td>GRACE2, GAMMA-RAY ATTENUATION CYL SPHERE GEOM</td>
<td>CDC 1604 F63</td>
<td>RS</td>
<td>P</td>
</tr>
<tr>
<td>47</td>
<td>CLOUD, GAMMA-RAY DOSE RATE FROM A CLOUD</td>
<td>AI 7090 F2</td>
<td>RS</td>
<td>P</td>
</tr>
<tr>
<td>47</td>
<td>CLOUD, GAMMA-RAY DOSE RATE FROM A CLOUD</td>
<td>CDC 1604 F63</td>
<td>RS</td>
<td>P</td>
</tr>
<tr>
<td>47</td>
<td>CLOUD, GAMMA-RAY DOSE RATE FROM A CLOUD</td>
<td>DP 360 F4</td>
<td>RS</td>
<td>P</td>
</tr>
<tr>
<td>48</td>
<td>FUGUE, STEADY-STATE TEMPERATURE VOID FRACTION</td>
<td>AI 7090 F2</td>
<td>RS</td>
<td>P</td>
</tr>
<tr>
<td>50</td>
<td>TEMPEST2, THERMAL NEUTRON SPECTRUM X-SECTION CALC</td>
<td>CDC 1604 F63</td>
<td>RS</td>
<td>P T</td>
</tr>
<tr>
<td>50</td>
<td>TEMPEST2, THERMAL NEUTRON SPECTRUM X-SECTION CALC</td>
<td>BHSC 360 F4</td>
<td>RS</td>
<td>P T</td>
</tr>
<tr>
<td>51</td>
<td>FORM, FAST NEUTRON SPECTRUM X-SECTION CALC</td>
<td>AI 7090 F2</td>
<td>RS</td>
<td>PLT</td>
</tr>
<tr>
<td>51</td>
<td>FORM, FAST NEUTRON SPECTRUM X-SECTION CALC</td>
<td>CDC 1604 F63</td>
<td>RS</td>
<td>PLT</td>
</tr>
<tr>
<td>52</td>
<td>SAIL, 1-D 1-GP SN APPROXIMATION SLAB GEOMETRY</td>
<td>AI 7090 F2</td>
<td>RS</td>
<td>P</td>
</tr>
<tr>
<td>52</td>
<td>SAIL, 1-D 1-GP SN APPROXIMATION SLAB GEOMETRY</td>
<td>CDC 1604 F63</td>
<td>RS</td>
<td>P</td>
</tr>
<tr>
<td>53</td>
<td>S4 CYL CELL CODE, 1-D 1-GP S4 APPROXIMATION</td>
<td>AI 7090 F2</td>
<td>RS</td>
<td></td>
</tr>
<tr>
<td>53</td>
<td>S4 CYL CELL CODE, 1-D 1-GP S4 APPROXIMATION</td>
<td>CDC 1604 F63</td>
<td>RS</td>
<td></td>
</tr>
</tbody>
</table>

**PREFACE** ........................................................................................................... 9

**I. HISTORY AND ACKNOWLEDGMENTS** ................................................................. 10

**II. ABSTRACT FORMAT** ......................................................................................... 11

**III. RECOMMENDED PROGRAM PACKAGE CONTENTS.** ............................................ 14

A. Card and/or Tape Material ............................................................................... 14
B. Program Report(s) ............................................................................................. 15
C. Installation Environment Report ........................................................................ 16

**IV. PROGRAM CLASSIFICATION GUIDE AND THESAURUS** ..................................... 17

**V. ABSTRACT COLLECTION** .................................................................................. 21
## TABLE OF CONTENTS

<table>
<thead>
<tr>
<th>NO.</th>
<th>KWIC TITLE</th>
<th>INSTALLATION MACH</th>
<th>LANG</th>
<th>PACKAGE</th>
</tr>
</thead>
<tbody>
<tr>
<td>55</td>
<td>AIMFIRE, URANIUM FUEL CYCLE COST ANALYSIS</td>
<td>AI 7090</td>
<td>F2</td>
<td>RS</td>
</tr>
<tr>
<td>56</td>
<td>SUMMIT, CRYSTALLINE SCATTERING KERNEL CALC</td>
<td>GGA 7090</td>
<td>F2</td>
<td>RS</td>
</tr>
<tr>
<td>57</td>
<td>SIZZLE, 1-D MULTIGROUP DIFFUSION DEPLETION</td>
<td>AI 7090</td>
<td>F2</td>
<td>RS</td>
</tr>
<tr>
<td>58</td>
<td>SIZZLE, 1-D MULTIGROUP DIFFUSION DEPLETION</td>
<td>CDC 1604</td>
<td>F63</td>
<td>RS</td>
</tr>
<tr>
<td>59</td>
<td>MIST, 1-C FEW-GP SN DOUBLE SN APPROX SLAB GEOM</td>
<td>PPCO 7090</td>
<td>F2</td>
<td>RS</td>
</tr>
<tr>
<td>60</td>
<td>LASL LEAST SQUARES, GENERAL CURVE FITTING</td>
<td>LASL 7094</td>
<td>F4</td>
<td></td>
</tr>
<tr>
<td>61</td>
<td>GE-HAPO-S13, 1-D MULTIGROUP DOUBLE SN APPROX</td>
<td>RNW 7090</td>
<td>FLOCO</td>
<td>RS</td>
</tr>
</tbody>
</table>

R 80 SOR2, STRESS ANALYSIS SHELLS OF REVOLUTION  
B 80 SOR2, STRESS ANALYSIS SHELLS OF REVOLUTION  

91 FASERIA, DOSE RATE FROM SNAP SHIELD LEAKAGE                                    | AI 7090          | F2   | RS      |
<p>| 92  | 99 DBB, 2-C FEW-GP DIFFUSION BURNUP RZ GEOMETRY                              | GGA 7090         | F+FAP| RS      |
| 102 | AXI, COUPLED NEUTRONICS-HYDRODYNAMICS SPHERE                                | CDC 3600         | F63  | RS      |
| 103 | CRAM, 1-C AND 2-D MULTIGROUP DIFFUSION PROGRAM                              | UK-R 7090        | F+FAP| RS      |
| 104 | CRAM, 1-C AND 2-D MULTIGROUP DIFFUSION PROGRAM                              | AEC 3600         | F+BAL| RS      |
| 105 | BAN, S4 CYL CELL CCDE AND TEPSST COMBINATION                                | AI 7090          | F2   | RS      |
| 106 | 4RESTRAINT PIPE STRESS, MAXIMUM MOMENT CALC                                 | AI 7090          | F+FAP| RS      |
| 107 | SCARF2, SCATTER FROM RADIATOR FINS SN APPROX                                | AI 7090          | F2   | RS      |
| 108 | 111 SCAR1, SCATTER FROM A RING SN APPROX                                    | AI 7090          | F2   | RS      |
| 112 | CROCK, SPACE POWER PLANT DESIGN OPTIMIZATION                                | AI 7090          | F2   | RS      |
| 113 | ZOT, GROUP-COLLAPSING OF MULTI-GP X-SECTIONS                                | LASL 7090        | FLOCO| RS      |
| 114 | 115 SHOCK, SPACE POWER PLANT DESIGN OPTIMIZATION                             | AI 7090          | F2   | RS      |
| 116 | 117 FEVER, 1-D FEW-GP DIFFUSION CELEPTON PROGRAM                             | GGA 7090         | F2   | RS      |
| 118 | 119 UCER, 1-D MULTIGROUP DIFFUSION CYL SPHERE                               | AI 7090          | F+FAP| RS      |
| 120 | QUICKIE, INFINITE MEDIUM SPECTRUM X-SECTIONS                                | AI 7090          | F+FAP| RS      |
| 121 | 122 FAIM, 1-C MULTIGROUP DIFFUSION CYL SPHERE                               | AI 7090          | F+FAP| RS      |
| 123 | S0R2, STRESS ANALYSIS SHELLS OF REVOLUTION                                 | AI 7090          | F2   | RS      |
| 124 | SNAPKIN5/5A, 1-REGION KINETICS W/FEEDBACK                                   | AI 7090          | F2   | RS      |
| 125 | LIPRECSCRI, MC NEUTRON PENETRATION CALCULATION                              | DAC 7090         | F2   | RS      |
| 126 | CONEC, COUPLED NEUTRONICS-HYDRODYNAMICS SPHERE                              | LRL 7090         | F2   | RS      |
| 127 | W-DSN, 1-D MULTIGROUP DIFFUSION CYL SPHERE                                  | UK-W 7090        | F+FAP| RS      |
| 128 | W-DSN, 1-D MULTIGROUP DIFFUSION CYL SPHERE                                  | UK-W 7090        | F+FAP| RS      |
| 129 | 132 NUCY, SOLUTION OF NUCLEON CHAIN EQUATIONS                               | ORNL 7090        | F2   | RS      |
| 130 | 133 NUCY, SOLUTION OF NUCLEON CHAIN EQUATIONS                               | ORNL 7090        | F2   | RS      |
| 131 | 134 TRAFFIC CORPORATION, TRANSFER FUNCTION SYNTHESIS                         | ANL 3600         | F63  | RS      |
| 132 | HERESYI, LATTICE PARAMETERS HETEROGENOUS CALC                                | FMA 7090         | F2   | RS      |
| 133 | RATER, COSE RATE CALCULATION SNAPSHOT GEOMETRY                              | AI 7090          | F2   | RS      |
| 134 | 135 MORTIMER, DSCE RATE CALCULATION SNAPSHOT GEOMETRY                       | AI 7090          | F2   | RS      |
| 135 | MAC, SHIELD DESIGN MULTIGROUP SLAB GEOMETRY                                 | BNM 7090         | F2   | RS      |
| 136 | MAC, SHIELD DESIGN MULTIGROUP SLAB GEOMETRY                                 | ANC 3600         | F63  | RS      |
| 137 | 142 DTF, 1-C MULTIGROUP DISCRETE ORDINATE CALC                              | UNC 1604         | F63  | RS      |
| 138 | 143 DTF, 1-C MULTIGROUP DISCRETE ORDINATE CALC                              | ANC 3600         | F63  | RS      |
| 139 | 144 NPRFCCP, FUEL CYCLE COSTS PERFORMANCE DATA                               | KE 7090          | F2   | RS      |
| 140 | AILMOE, X-SECTION CALC ELASTIC SCAT RESONANCES                              | AI 7090          | F+FAP| RS      |
| 141 | 142 AILMOE, X-SECTION CALC ELASTIC SCAT RESONANCES                           | AI 7090          | F+FAP| RS      |
| 142 | 143 TOPIC, 1-C FEW-GP SN APPROXIMATION CYLINDER                             | PPPO 7090        | F4   | RS      |
| 143 | 144 TOPIC, 1-C FEW-GP SN APPROXIMATION CYLINDER                             | PPPO 7090        | F4   | RS      |
| 144 | 145 TYCHE3, MCNE CARLO SLOWING-DOWN DENSITY CALC                             | AI 7090          | F+FAP| RS      |
| 145 | 146 DANCOFF JR, MODERATOR SPACE CHORD DIST FUNCT                             | AEG 7090         | F2   | RS      |
| 146 | 147 DTFC, 1-C MULTIGROUP DISCRETE ORDINATE CALC                             | AI 7090          | F2   | RS      |
| 147 | 148 ARGUS, TRANSIENT TEMPERATURE CYLINDER                                   | ANC 3600         | F63  | RS      |
| 148 | 149 HATCHET, COUPLED NEUTRONICS-HYDRODYNAMICS CODE                          | AGC 7090         | F2   | RS      |
| 149 | 150 CROC90, ML-1 FLUID FLOW EXPERIMENT ANALYSIS                             | AGC 7090         | F2   | RS      |
| 150 | 151 PSI, BLOOCDW PRESSURE TEMPERATURE HISTORY                                | KE 7094          | F2   | RS      |
| 151 | 152 EXTERMINATOR, 2-D MULTIGROUP DIFFUSION PROGRAM                          | ORNL 7090        | F+FAP| RS      |
| 152 | 153 EXTERMINATOR2, 2-D MULTIGROUP DIFFUSION PROGRAM                          | ORNL 3600        | F4   | RS      |
| 153 | 154 EXTERMINATOR2, 2-D MULTIGROUP DIFFUSION PROGRAM                          | BC 625           | F4   | RS      |</p>
<table>
<thead>
<tr>
<th>NO.</th>
<th>KMC TITLE</th>
<th>INSTALLATION MACH</th>
<th>LANG</th>
<th>PACKAGE</th>
<th>PAGE</th>
</tr>
</thead>
<tbody>
<tr>
<td>159</td>
<td>MOMUS, X-SECTION LIBRARY UTILIZATION PROGRAM</td>
<td>AI</td>
<td>7094</td>
<td>F+FAP</td>
<td>165</td>
</tr>
<tr>
<td>160</td>
<td>SOPHIST/2/25, MULTI-GP TRANSFER COEFFICIENTS</td>
<td>LRL</td>
<td>7094</td>
<td>F+FAP</td>
<td>167</td>
</tr>
<tr>
<td>161</td>
<td>FORTRAN TDC, 2-D MULTI-GP SN APPROXIMATION</td>
<td>PW</td>
<td>1604</td>
<td>F+F63</td>
<td>167</td>
</tr>
<tr>
<td>162</td>
<td>GRAVE, GROUP-AVERAGING X-SECTIONS PARAMETERS</td>
<td>AI</td>
<td>7094</td>
<td>F+FAP</td>
<td>172</td>
</tr>
<tr>
<td>163</td>
<td>AIROS, SPACE-DEPENDENT KINETICS W/FEEDBACK</td>
<td>AI</td>
<td>7094</td>
<td>F+MAP</td>
<td>174</td>
</tr>
<tr>
<td>164</td>
<td>BURP1, DETECTOR EFFICIENCY POINT SOURCE</td>
<td>UM</td>
<td>7090</td>
<td>MAD</td>
<td>177</td>
</tr>
<tr>
<td>165</td>
<td>BURP2, DETECTOR EFFICIENCY DISK SOURCE</td>
<td>UM</td>
<td>7090</td>
<td>MAD</td>
<td>179</td>
</tr>
<tr>
<td>166</td>
<td>BURP3, DETECTOR EFFICIENCY POINT SOURCE</td>
<td>UM</td>
<td>7090</td>
<td>MAD</td>
<td>182</td>
</tr>
<tr>
<td>167</td>
<td>FLARE, 3-D REACTIVITY AND POWER DISTRIBUTION</td>
<td>NED</td>
<td>635</td>
<td>F4</td>
<td>185</td>
</tr>
<tr>
<td>168</td>
<td>FLARE, 3-D REACTIVITY AND POWER DISTRIBUTION</td>
<td>CDC</td>
<td>3600</td>
<td>F63</td>
<td>185</td>
</tr>
<tr>
<td>169</td>
<td>RI02, SPACE-INDEPENDENT INVERSE KINETICS CALC</td>
<td>ANL</td>
<td>3600</td>
<td>F63</td>
<td>187</td>
</tr>
<tr>
<td>170</td>
<td>RI02, SPACE-INDEPENDENT INVERSE KINETICS CALC</td>
<td>WANL</td>
<td>7094</td>
<td>F4</td>
<td>187</td>
</tr>
<tr>
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<td>DIPOL, RESOLVED RESONANCE X-SECTION CALC</td>
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<td>SPARTA, SPATIALLY-AVERAGED Doppler EFFECTS</td>
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<td>ISOTOPES, MAXIMUM YIELD FROM REACTION OR DECAY</td>
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<td>ISOCRUNCH, REACTION DECAY CHAIN ANALYSIS</td>
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<td>THERMOS(ANL), THERMAL SPECTRUM X-SECTION CALC</td>
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<td>LAG, ASSEMBLER FOR FLOCO2 INSTRUCTION SET</td>
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<td>CMPWXMT, TRANSFER FUNCTION EQUATION</td>
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<td>SNC, CALCULATION OF SN CONSTANTS FOR DSN TDC</td>
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<td>2DXYL, 3-D MULTI-GP FLUX SYNTHESIS PROGRAM CYL</td>
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<td>CSP2A, SN X-SECTION LIBRARY TAPE PREPARATION</td>
<td>PW</td>
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<td>FMC-N/FMC-G, MC NEUTRON, GAMMA-RAY HISTORIES</td>
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<td>CURIE/DSM/THUNDERHEAD, EXTERNAL+INTERNAL DOSE</td>
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<td>HEATING2, TRANSIENT STEADY-STATE HEAT TRANSFER</td>
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<td>TOP, 2-D PERTURBATION TDC OR 2DX FLUX INPUT</td>
<td>PW</td>
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<td>SATURATEC BLOWDOWN, BLOWDOWN ANALYSIS LOFT</td>
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<td>EPITHERMCS, SPECTRUM AND X-SECTION CALCULATION</td>
<td>GEV</td>
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<td>MICS, MONTE CARLO NEUTRON PENETRATION STUDY</td>
<td>LASL</td>
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<td>COMBO, COMBINED B-W MULTI-LEVEL CORRELATION</td>
<td>PPCO</td>
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<td>AGN-GAM, FAST SPECTRUM MULTI-GP CONSTANT CALC</td>
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<td>SHOE, SHIELD WEIGHT OPTIMIZATION DOSE CALC</td>
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<td>UNPACK, RETRIEVAL FROM CSRS X-SECTION TAPE</td>
<td>GCA</td>
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<td>CROSSPLT, SC4020 PLOTS FROM X-SECTION TAPES</td>
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<td>VARI-QUIR, TIME-DEP 2-D MULTI-GP DIFFUSION</td>
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<td>RAZZ, HTGR FISSION PRODUCT ACTIVITY DIST STUDY</td>
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<td>DPC, DATA PREPARATION FOR 2-C DESIGN PROGRAMS</td>
<td>LASL</td>
<td>7090</td>
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<td>RS</td>
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<td>GAMMA-P, PRODUCTION X-SECTIONS FOR GAMMA-RAYS</td>
<td>GEC</td>
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<td>DFSR, DATA FILE SERVICE ROUTINES ENDF TAPES</td>
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<td>EXT, X-SECTIONS FROM B-W RESONANCE PARAMETERS</td>
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<td>RATH, 2- OR 3-D HEAT CONDUCTION LUMPED MASS</td>
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<td>AGN-SIGMA, CALC OF MULTI-GP TRANSFER MATRICES</td>
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<td>FLANGE1, SCATTERING LAW X-SECTION CALCULATION</td>
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<td>GGA</td>
<td>7044</td>
<td>F4</td>
<td>RS</td>
</tr>
<tr>
<td>254</td>
<td>SAFE-SHELL, STRESS ANALYSIS THIN SHELLS</td>
<td>GGA</td>
<td>1108</td>
<td>F4</td>
<td>RS</td>
</tr>
<tr>
<td>255</td>
<td>2PLUS, ACS-SHPEICAL OPTICAL MODEL X-SECTION</td>
<td>AI</td>
<td>7094</td>
<td>F2</td>
<td>RS</td>
</tr>
<tr>
<td>256</td>
<td>2PLUS, ACS-SHPEICAL OPTICAL MODEL X-SECTION</td>
<td>GGA</td>
<td>3600</td>
<td>F63</td>
<td>RSBP</td>
</tr>
<tr>
<td>257</td>
<td>R101, SPACE-INDEPENDENT KINETICS KEX OPTIONS</td>
<td>ANL</td>
<td>3600</td>
<td>F63</td>
<td>RSBP</td>
</tr>
<tr>
<td>258</td>
<td>R101, SPACE-INDEPENDENT KINETICS KEX OPTIONS</td>
<td>WNL</td>
<td>7094</td>
<td>F4</td>
<td>RS</td>
</tr>
<tr>
<td>259</td>
<td>MANTA, STEADY-STATE THERMAL-HYDRAULIC ANALYSIS</td>
<td>NED</td>
<td>635</td>
<td>F4</td>
<td>RS</td>
</tr>
<tr>
<td>260</td>
<td>REX, RESOLVE RESONANCE EPITHERMAL X-SECTIONS</td>
<td>NED</td>
<td>635</td>
<td>F4</td>
<td>RS</td>
</tr>
<tr>
<td>261</td>
<td>EXP, ANALYSIS OF PULSED NEUTRON SOURCE DATA</td>
<td>NED</td>
<td>635</td>
<td>F4</td>
<td>RS</td>
</tr>
<tr>
<td>262</td>
<td>MUSCAT, VIEW FACTOR SHIELDING CODE CAVIITY GEOM</td>
<td>GGA</td>
<td>1108</td>
<td>F4</td>
<td>RS</td>
</tr>
<tr>
<td>263</td>
<td>GAROYFE, FUEL CYCLE ANALYSIS PARTIAL REFUEL</td>
<td>GGA</td>
<td>7044</td>
<td>F4</td>
<td>RS</td>
</tr>
<tr>
<td>264</td>
<td>GADDOSE/DSSSET, HTGR ACCIDENT ANALYSIS DOSE CALC</td>
<td>GGA</td>
<td>1108</td>
<td>F4</td>
<td>RS</td>
</tr>
<tr>
<td>265</td>
<td>MACII, 1-D MULTI-GP DIFFUSION SLAB CYL SPHERE</td>
<td>ANL</td>
<td>3600</td>
<td>F63</td>
<td>RS P</td>
</tr>
<tr>
<td>266</td>
<td>GASKET, THERMAL SCATTERING LAW CALCULATION</td>
<td>GGA</td>
<td>1108</td>
<td>F4</td>
<td>RS</td>
</tr>
<tr>
<td>267</td>
<td>VARI-QUIR3, 2-D MULTI-GP DIFFUSION XY RZ RTH</td>
<td>WNL</td>
<td>7094</td>
<td>F4</td>
<td>RS</td>
</tr>
<tr>
<td>268</td>
<td>R5AG, RADIATION PROTECTION ANALYSIS PROGRAM</td>
<td>PPCO</td>
<td>7040</td>
<td>F+FAK</td>
<td>RS</td>
</tr>
<tr>
<td>269</td>
<td>CYGRO2, STRESS ANALYSIS CYL FUEL ELEMENT</td>
<td>BAPL</td>
<td>6600</td>
<td>F4</td>
<td>RS</td>
</tr>
<tr>
<td>270</td>
<td>CYGRO2, STRESS ANALYSIS CYL FUEL ELEMENT</td>
<td>RNL</td>
<td>145</td>
<td>6720</td>
<td>REFLS</td>
</tr>
<tr>
<td>271</td>
<td>TRANS-FUGUE, TRANSIENT FLOW AND HEAT TRANSFER</td>
<td>AI</td>
<td>7094</td>
<td>F2</td>
<td>RS</td>
</tr>
<tr>
<td>272</td>
<td>DTF-BURN, 1-D MULTI-GP DTF4 WITH DEPLETION</td>
<td>LASL</td>
<td>7030</td>
<td>F4</td>
<td>RS</td>
</tr>
<tr>
<td>NO.</td>
<td>KWG TITLE</td>
<td>INSTALLATION</td>
<td>MACH</td>
<td>LANG</td>
<td>PACKAGE</td>
</tr>
<tr>
<td>-----</td>
<td>---------------------------------------------------------------------------</td>
<td>--------------</td>
<td>------</td>
<td>------</td>
<td>---------</td>
</tr>
<tr>
<td>270</td>
<td>CAESAR4/LIBLST, 1-C MULTI-GP DIFFUSION + LIB</td>
<td>AI</td>
<td>360</td>
<td>F4</td>
<td>RSBLT</td>
</tr>
<tr>
<td>271</td>
<td>CLS, PCRM OR THRESES LIBRARY UTILITY ROUTINE</td>
<td>AI</td>
<td>360</td>
<td>F4</td>
<td>RSB LT</td>
</tr>
<tr>
<td>272</td>
<td>FIGRO, 1-SB FUEL SWELLING TEMPERATURE STUDY</td>
<td>BAPL</td>
<td>6600</td>
<td>F4</td>
<td>RS P T</td>
</tr>
<tr>
<td>273</td>
<td>THRESES, 1-D FEW-GP DIFFUSION DESIGN SYSTEM</td>
<td>AI</td>
<td>360</td>
<td>F4</td>
<td>RSBP T</td>
</tr>
<tr>
<td>274</td>
<td>WIGL2, 1-C 2-GP SPACE-TIME DIFFUSION 3-GEOM</td>
<td>BAPL</td>
<td>6600</td>
<td>F4</td>
<td>RS P T</td>
</tr>
<tr>
<td>275</td>
<td>PDQ7, 1,2 OR 3-D FEW-GP DIFFUSION DEPLETION</td>
<td>BAPL</td>
<td>6600</td>
<td>F+ASC</td>
<td>RS T</td>
</tr>
<tr>
<td>276</td>
<td>AVOID, ANNUALAR VOIC X-SECTION CALCULATION</td>
<td>GEC</td>
<td>625</td>
<td>F4</td>
<td>RS P T</td>
</tr>
<tr>
<td>277</td>
<td>HAMMER, CRITICAL EXPERIMENT ANALYSIS SYSTEM</td>
<td>BNL-DP</td>
<td>7090</td>
<td>F2</td>
<td>RS PLT</td>
</tr>
<tr>
<td>278</td>
<td>WATER-HAMMER, LIQUID BLOWDOWN ANALYSIS FOUT</td>
<td>UGA</td>
<td>36c</td>
<td>RS</td>
<td>PL T</td>
</tr>
<tr>
<td>279</td>
<td>LEOPARD, SPECTRA CALCULATION WITH DEPLETION</td>
<td>WAPD</td>
<td>360</td>
<td>F4</td>
<td>RS PLT</td>
</tr>
<tr>
<td>280</td>
<td>MOBC7, 2-D DIFFUSION ABSORPTION REMOVAL X-SECS</td>
<td>BAPL</td>
<td>6600</td>
<td>F4</td>
<td>RS T</td>
</tr>
<tr>
<td>281</td>
<td>RABBLE/MLIB/FLAT. RESONANCE ABSORPTION, CELL</td>
<td>ANL</td>
<td>3600</td>
<td>F36</td>
<td>RSBP T</td>
</tr>
<tr>
<td>282</td>
<td>SEALSHELL2, SHELL STRESS ANALYSIS AXISYM LOAD</td>
<td>BAPL</td>
<td>6600</td>
<td>F4</td>
<td>RS P T</td>
</tr>
<tr>
<td>283</td>
<td>MO552, DYNAMIC ANALYSIS LINEAR ELASTIC SYSTEMS</td>
<td>BAPL</td>
<td>660c</td>
<td>F4</td>
<td>RS P T</td>
</tr>
<tr>
<td>284</td>
<td>MO555 ACT1, LOSS-OF-COOLANT ACCIDENT ANALYSIS</td>
<td>BAPL</td>
<td>6600</td>
<td>F4</td>
<td>RS P T</td>
</tr>
<tr>
<td>285</td>
<td>RESQ2/RESQ/DBF1, RESONANCE INTEGRAL HEX CELL</td>
<td>BAPL</td>
<td>6600</td>
<td>F4</td>
<td>RS P T</td>
</tr>
<tr>
<td>286</td>
<td>HOT2, 1-2 OR 3-D TRANSIENT HEAT CONDUCTION PROGRAM</td>
<td>BAPL</td>
<td>6600</td>
<td>F4</td>
<td>RS P T</td>
</tr>
<tr>
<td>287</td>
<td>RISYN, 2-D MULTI-GP DIFFUSION SYNTHESIS CALC</td>
<td>NED</td>
<td>635</td>
<td>F4</td>
<td>RS PLT</td>
</tr>
<tr>
<td>288</td>
<td>SNARG-1C, 1-D MULTI-GP DISCRETE ORDINATE CALC</td>
<td>ANL</td>
<td>3600</td>
<td>F36</td>
<td>RS P T</td>
</tr>
<tr>
<td>289</td>
<td>GAKER, INELASTIC SCAT X-SECTION CALC MODERATOR</td>
<td>GGA</td>
<td>1108</td>
<td>F4</td>
<td>RS P</td>
</tr>
<tr>
<td>290</td>
<td>GASA, STABILITY ANALYSIS REACTOR KINETICS EQU</td>
<td>GGA</td>
<td>1108</td>
<td>F4</td>
<td>RS P</td>
</tr>
<tr>
<td>291</td>
<td>HEXSCAT, ELASTIC SCAT X-SECTIONS HEX LATTICE</td>
<td>GGA</td>
<td>1108</td>
<td>F4</td>
<td>RS P</td>
</tr>
<tr>
<td>292</td>
<td>PSEUDO, STATISTICAL RESONANCE PARAMETER CALC</td>
<td>GGA</td>
<td>1108</td>
<td>F4</td>
<td>RS P</td>
</tr>
<tr>
<td>293</td>
<td>MARS, 2-C EXCERSION ALUATION R-Z GEOMETRY</td>
<td>GGA</td>
<td>1108</td>
<td>F4</td>
<td>RS P</td>
</tr>
<tr>
<td>294</td>
<td>MO899/HC+, STEAM TABLES 14-5-2536 PSA</td>
<td>BAPL</td>
<td>6600</td>
<td>F4</td>
<td>RS P T</td>
</tr>
<tr>
<td>295</td>
<td>FLASH2, LOSS-OF-COOLANT ACCIDENT ANALYSIS</td>
<td>BAPL</td>
<td>6600</td>
<td>F4</td>
<td>RS P T</td>
</tr>
<tr>
<td>296</td>
<td>GRDWRK, GRID GENERATION FOR SAFE PROGRAMS</td>
<td>GGA</td>
<td>1108</td>
<td>F4</td>
<td>RS P</td>
</tr>
<tr>
<td>297</td>
<td>CONTEMPT, LOSS-OF-COOLANT ACCIDENT ANALYSIS</td>
<td>PPCO</td>
<td>7040</td>
<td>F+MAP</td>
<td>RS P T</td>
</tr>
<tr>
<td>298</td>
<td>GCC4, MULTI-GP X-SECTIONS FAST THERMAL SPECTRA</td>
<td>GGA</td>
<td>1108</td>
<td>F4</td>
<td>RS P</td>
</tr>
<tr>
<td>299</td>
<td>LICN, 3-C TEMPERATURE DISTRIBUTION PROGRAM</td>
<td>KAPL</td>
<td>6600</td>
<td>F4</td>
<td>RS P T</td>
</tr>
<tr>
<td>300</td>
<td>SAFE-CREEP, VISCOELASTIC ANALYSIS CONCRETE</td>
<td>GGA</td>
<td>1108</td>
<td>F4</td>
<td>RS P</td>
</tr>
<tr>
<td>301</td>
<td>FREAP6, HTGR METALLIC FISSION PRODUCT RELEASE</td>
<td>GGA</td>
<td>1108</td>
<td>F4</td>
<td>RS P</td>
</tr>
<tr>
<td>302</td>
<td>CAFFE, EQUILIBRUM FUEL CYCLE CALCULATION</td>
<td>GGA</td>
<td>1108</td>
<td>F4</td>
<td>S T</td>
</tr>
<tr>
<td>303</td>
<td>BLOOD6, COMBINED KINETICS 2-D HEAT TRANSFER</td>
<td>GGA</td>
<td>1108</td>
<td>F4</td>
<td>RS P T</td>
</tr>
<tr>
<td>304</td>
<td>PERT4, 2-D PERTURBATION XY RZ RTHEA GEOMETRY</td>
<td>BNW</td>
<td>7094</td>
<td>F4</td>
<td>RS P</td>
</tr>
<tr>
<td>305</td>
<td>STRIP, RESOLVED RESONANCE INTEGRAL CALCULATION</td>
<td>BAPL</td>
<td>6600</td>
<td>F4</td>
<td>RS P T</td>
</tr>
<tr>
<td>306</td>
<td>FCC4, FUNDAMENTAL MODE FAST REACTOR X-SEC CALC</td>
<td>BNP</td>
<td>1108</td>
<td>F4</td>
<td>RS P T</td>
</tr>
<tr>
<td>307</td>
<td>HWOCR-SAFE, 2-D MCNE CARLO CELL CALCULATION</td>
<td>AI</td>
<td>360</td>
<td>F4</td>
<td>RS P T</td>
</tr>
<tr>
<td>308</td>
<td>JUPITORI, COUPLED-CHANNEL X-SEC EVOLUTION</td>
<td>ORNL</td>
<td>1604</td>
<td>F63</td>
<td>RS P T</td>
</tr>
<tr>
<td>309</td>
<td>TSN, SPATIALLY-DEPENDENT REACTOR KINETICS</td>
<td>AI</td>
<td>7094</td>
<td>F4</td>
<td>RS P T</td>
</tr>
<tr>
<td>310</td>
<td>TSN, SPATIALLY-DEPENDENT REACTOR KINETICS</td>
<td>AI</td>
<td>360</td>
<td>F4</td>
<td>RS P T</td>
</tr>
<tr>
<td>311</td>
<td>GAAN9, 1-D MULTI-GROUP TIME-DEPENDENT DIFFUSION</td>
<td>GGA</td>
<td>1108</td>
<td>F4</td>
<td>RSBP T</td>
</tr>
<tr>
<td>312</td>
<td>BURNUP, HEAVY ELEMENT ISOTOPIC BURNUP ANALYSIS</td>
<td>GEA</td>
<td>635</td>
<td>F4</td>
<td>RS P T</td>
</tr>
<tr>
<td>313</td>
<td>TDSN, 2-C MULTIGROUP DISCRETE ORDINATE PROGRAM</td>
<td>LER</td>
<td>7090</td>
<td>F+MAP</td>
<td>RS P T</td>
</tr>
<tr>
<td>314</td>
<td>CINDER(M0102), POINT DEPLETION FISSION PRODUCT</td>
<td>BAPL</td>
<td>6600</td>
<td>F4</td>
<td>RS P</td>
</tr>
<tr>
<td>315</td>
<td>NAP, NEUTRON-INDUCED GAMMA-RAY RADIOACTIVITY</td>
<td>IITR</td>
<td>7094</td>
<td>F4</td>
<td>RSBPLT</td>
</tr>
<tr>
<td>316</td>
<td>WIREX, COMPUTER-PRODUCED WIRING LISTS UHTREX</td>
<td>LASL</td>
<td>7090</td>
<td>F2</td>
<td>RS P</td>
</tr>
<tr>
<td>317</td>
<td>GAF/GAR, SPECTRA AND GROUP-AVERAGED X-SEC CALC</td>
<td>GGA</td>
<td>1108</td>
<td>F4</td>
<td>RS P</td>
</tr>
<tr>
<td>318</td>
<td>GAPOTKIN, SPACE-DEPENDENT REACTOR KINETICS</td>
<td>GGA</td>
<td>1108</td>
<td>F4</td>
<td>RS P</td>
</tr>
<tr>
<td>319</td>
<td>FEVERT, 1-D MULTIGROUP DIFFUSION AND DEPLETION</td>
<td>GGA</td>
<td>1108</td>
<td>F4</td>
<td>RS P T</td>
</tr>
<tr>
<td>320</td>
<td>GASP7, 1-D BURNUP POWER DISTRIBUTION SEARCH</td>
<td>GGA</td>
<td>1108</td>
<td>F4</td>
<td>RS P T</td>
</tr>
<tr>
<td>321</td>
<td>EXPOT2, TEMPERATURE DEPENDENT DIFFUSION CALC</td>
<td>GGA</td>
<td>1108</td>
<td>F4</td>
<td>RS P T</td>
</tr>
<tr>
<td>322</td>
<td>EXPOT2, TEMPERATURE DEPENDENT DIFFUSION CALC</td>
<td>GGA</td>
<td>1108</td>
<td>F4</td>
<td>RS P T</td>
</tr>
<tr>
<td>323</td>
<td>ISOSCALC, ISOTOPE PRODUCTION FLUX, X-SEC CALC</td>
<td>ORNL</td>
<td>1604</td>
<td>F63</td>
<td>RS P T</td>
</tr>
<tr>
<td>324</td>
<td>ISOSCALC, ISOTOPE PRODUCTION FLUX, X-SEC CALC</td>
<td>ORNL</td>
<td>1604</td>
<td>F63</td>
<td>RS P T</td>
</tr>
<tr>
<td>325</td>
<td>FISU, 2-D CUMULIGROUP DIFFUSION AND DEPLETION</td>
<td>BNP</td>
<td>7090</td>
<td>F4</td>
<td>RS P T</td>
</tr>
<tr>
<td>326</td>
<td>AIROSZA, SIMULATION OF REACTOR DYNAMICS</td>
<td>AI</td>
<td>360</td>
<td>F+BAL</td>
<td>RS P T</td>
</tr>
</tbody>
</table>
TABLE OF CONTENTS

<table>
<thead>
<tr>
<th>NO.</th>
<th>KWIC TITLE</th>
<th>INSTALLATION MACH</th>
<th>LANG</th>
<th>PACKAGE</th>
</tr>
</thead>
<tbody>
<tr>
<td>R327</td>
<td>DAFT1, LEAST SQUARES FIT FISSILE NUCLIDE DATA</td>
<td>BAPL 6600</td>
<td>F4</td>
<td>RS P T</td>
</tr>
<tr>
<td>R328</td>
<td>NURLOC-1,0, LOSS-OF-COOLANT THERMAL ANALYSIS</td>
<td>BCL 6400</td>
<td>F4</td>
<td>RS P T</td>
</tr>
<tr>
<td>R329</td>
<td>M0457 (PIPE), ELASTIC STRESS OF PIPING SYSTEM</td>
<td>BAPL 6600</td>
<td>F4</td>
<td>RS P T</td>
</tr>
<tr>
<td>R330</td>
<td>ECCL1, LOSS-OF-COOLANT + EMERGENCY COOLING</td>
<td>BCL 6400</td>
<td>F4</td>
<td>RS P T</td>
</tr>
<tr>
<td>R331</td>
<td>FLOT1 (MC219), PWR FLOW TRANSIENT ANALYSIS</td>
<td>BAPL 6600</td>
<td>F4</td>
<td>RS P T</td>
</tr>
<tr>
<td>R332</td>
<td>SAFE-3D, 3-D COMPOSITE STRUCTURE STRESS STUDY</td>
<td>GGA 1108</td>
<td>F4</td>
<td>RS P T</td>
</tr>
<tr>
<td>R333</td>
<td>TOAD, PROCESSING OF ANALYZER GAMMA-RAY SPECTRA</td>
<td>GGA 1108</td>
<td>F4</td>
<td>RS P T</td>
</tr>
<tr>
<td>R334</td>
<td>PEGG, ELASTIC SCATTERING PHASE-SHIFT ANALYSIS</td>
<td>ORNL 1604</td>
<td>F63</td>
<td>RS P</td>
</tr>
<tr>
<td>R335</td>
<td>RAMES, PARTICLE WAVE FUNCTION RADIAL INTEGRALS</td>
<td>ORNL 1604</td>
<td>F63</td>
<td>RS P</td>
</tr>
<tr>
<td>R336</td>
<td>PDGS, 2-D FEW-GROUP DIFFUSION AND DEPLETION</td>
<td>IBM 360</td>
<td>F+BAL</td>
<td>RS P T</td>
</tr>
<tr>
<td>R337</td>
<td>STEM, MATRIX GENERATION FOR A SYSTEM OF BEAMS</td>
<td>KAPL 6600</td>
<td>F4</td>
<td>RS P</td>
</tr>
<tr>
<td>R338</td>
<td>TWIGLE, 2-D 2-GP SPACE-TIME DIFFUSION FEEDBACK</td>
<td>BAPL 6600</td>
<td>F4</td>
<td>RS P T</td>
</tr>
<tr>
<td>R339</td>
<td>GMS, 2-D 2-GP SPACE-TIME DIFFUSION</td>
<td>ANL 360</td>
<td>F4</td>
<td>RS P</td>
</tr>
<tr>
<td>R340</td>
<td>GAUGE, 2-D FEW-GP HEX GEOM DIFFUSION DEPLETION</td>
<td>GGA 1108</td>
<td>F4</td>
<td>RS P T</td>
</tr>
<tr>
<td>R341</td>
<td>GAUND, UNRESOLVED RESONANCE X-SECTION CALC</td>
<td>GGA 1108</td>
<td>F4</td>
<td>RS P</td>
</tr>
<tr>
<td>R342</td>
<td>M0648, 1-D SLAB TRANSPORT WITH SLOWING DOWN</td>
<td>BAPL 6600</td>
<td>F4</td>
<td>RS T</td>
</tr>
<tr>
<td>R343</td>
<td>M0756 (LET0), 1-0 SLAB GAMMA-RAY TRANSPORT</td>
<td>BAPL 6600</td>
<td>F4</td>
<td>RS P T</td>
</tr>
<tr>
<td>R344</td>
<td>GEM, EIGENVALUE PROBLEM FOR VIBRATING SYSTEMS</td>
<td>KAPL 6600</td>
<td>F4</td>
<td>RS T</td>
</tr>
<tr>
<td>R345</td>
<td>GAND, GAFGAR X-SECTION LIBRARY PREPARATION</td>
<td>GGA 1108</td>
<td>F4</td>
<td>RS P T</td>
</tr>
<tr>
<td>R346</td>
<td>THTF, 3-C TRANSIENT HEAT TRANSFER PROGRAM</td>
<td>GEC 635</td>
<td>F+GMP</td>
<td></td>
</tr>
<tr>
<td>R347</td>
<td>CODILLI, LEAST SQUARES ANALYSIS RESONANCE DATA</td>
<td>UILL 360</td>
<td>F4</td>
<td></td>
</tr>
<tr>
<td>R348</td>
<td>TOPS, TRANSIENT THERMODYNAMICS OF PRESSURIZERS</td>
<td>BAPL 6600</td>
<td>F4</td>
<td>RS P T</td>
</tr>
<tr>
<td>R349</td>
<td>GATT, 3-C FEW-GP DIFFUSION CALC HEX-Z MESH</td>
<td>GGA 1108</td>
<td>F4</td>
<td>RS P T</td>
</tr>
<tr>
<td>R350</td>
<td>M0266, LINEAR ELASTIC STRUCTURAL DYNAMICS</td>
<td>BAPL 6600</td>
<td>F4</td>
<td>RS P</td>
</tr>
<tr>
<td>R351</td>
<td>ECSIL, EXPERIMENTAL NEUTRON DATA LIBRARY</td>
<td>LRLL 7094</td>
<td>F2</td>
<td></td>
</tr>
<tr>
<td>R352</td>
<td>RAUNZEIT, 1-D TIME-DEPENDENT DIFFUSION CALC</td>
<td>KAPL 6600</td>
<td>F4</td>
<td>RS P</td>
</tr>
<tr>
<td>R353</td>
<td>SWEL2, FUEL ELEMENT LIFETIME ANALYSIS</td>
<td>ANL 360</td>
<td>F36</td>
<td>RSBP T</td>
</tr>
<tr>
<td>R354</td>
<td>CINCA, NUCLEAR FUEL CYCLE CCST AND ECONOMICS</td>
<td>COMM 360</td>
<td>F4</td>
<td>RS P T</td>
</tr>
<tr>
<td>R355</td>
<td>M0752, ENDF MULTIGROUP X-SECTION CALCULATION</td>
<td>ANL 360</td>
<td>F36</td>
<td>RSBP T</td>
</tr>
<tr>
<td>R356</td>
<td>ZPR-III ASSEMBLY 48 GAFGAR ENDF/B DATA TAPES</td>
<td>GGA 1108</td>
<td>F4</td>
<td></td>
</tr>
<tr>
<td>R357</td>
<td>SUPORAN, REACTOR CORE SUPPORT STRESS ANALYSIS</td>
<td>APDA 3600</td>
<td>ASAFA4</td>
<td></td>
</tr>
<tr>
<td>R358</td>
<td>GANDY, UNRESOLVED RESONANCE X-SECTION CALC</td>
<td>GGA 1108</td>
<td>F4</td>
<td>RS P</td>
</tr>
<tr>
<td>R359</td>
<td>M0756 (LET0), 1-0 SLAB GAMMA-RAY TRANSPORT</td>
<td>BAPL 6600</td>
<td>F4</td>
<td>RS P T</td>
</tr>
<tr>
<td>R360</td>
<td>TUBE, U-TUBE HEAT EXCHANGE STRESS ANALYSIS</td>
<td>KAPL 6600</td>
<td>F4</td>
<td>RS P</td>
</tr>
<tr>
<td>R361</td>
<td>IDX, 1-D DIFFUSION FAST X-SECTION GENERATION</td>
<td>BCL 6400</td>
<td>F4</td>
<td>RS P</td>
</tr>
<tr>
<td>R362</td>
<td>WELWING, MATERIAL BUCKLING CYL FUEL ELEMENTS</td>
<td>AEB 360</td>
<td>F4</td>
<td></td>
</tr>
<tr>
<td>R363</td>
<td>GLEN, GROUP CONSTANT CALC FROM TOR OUTPUT DATA</td>
<td>LASL 6600</td>
<td>F4</td>
<td>RS P</td>
</tr>
<tr>
<td>R364</td>
<td>EVOLUTION, PARTICLE WAVE FUNCTION RADIAL INTEGRALS</td>
<td>ORNL 1604</td>
<td>F63</td>
<td></td>
</tr>
<tr>
<td>R365</td>
<td>SCHL2, FUEL ELEMENT LIFETIME ANALYSIS</td>
<td>ANL 360</td>
<td>F36</td>
<td>RSBP T</td>
</tr>
<tr>
<td>R366</td>
<td>CHEML0C2, CORE HEATING CLADDING-STEAM REACTION</td>
<td>ANL 360</td>
<td>F36</td>
<td>RSBP</td>
</tr>
<tr>
<td>R367</td>
<td>IGGCN, RADIONUCLIDE GENERATION AND DECAY</td>
<td>BNW 1108</td>
<td>F5</td>
<td>RSBPT</td>
</tr>
<tr>
<td>R368</td>
<td>FLANGE2, ENDF/B THERMAL SCATTERING DATA PROC</td>
<td>DP 360</td>
<td>F4</td>
<td>RS P T</td>
</tr>
<tr>
<td>R369</td>
<td>RELAP2, REACTOR BLOWDOWN - EXCURSION ANALYSIS</td>
<td>INC 7094</td>
<td>F4</td>
<td>RS P T</td>
</tr>
<tr>
<td>R370</td>
<td>GAKIT, 1-D MULTIGP KINETICS WITH TEMP FEEDBACK</td>
<td>GGA 1108</td>
<td>F4</td>
<td>RS P T</td>
</tr>
<tr>
<td>R371</td>
<td>NOWIG, 1-D 2-GP KINETICS TEMPERATURE FEEDBACK</td>
<td>BAPL 6600</td>
<td>F4</td>
<td>RS T</td>
</tr>
<tr>
<td>R372</td>
<td>RAPFU, FUEL CYCLE PARAMETERS FAST BREEDERS</td>
<td>APD 635</td>
<td>F4</td>
<td>RS P</td>
</tr>
<tr>
<td>R373</td>
<td>BRAY, DRAFTING TOOL TO PLOT PLANE STRUCTURES</td>
<td>BAPL 6600</td>
<td>F4</td>
<td>RS P T</td>
</tr>
<tr>
<td>R374</td>
<td>1DX, 1-D DIFFUSION FAST X-SECTION GENERATION</td>
<td>BNLW 1108</td>
<td>F4</td>
<td>RS P T</td>
</tr>
<tr>
<td>R375</td>
<td>SCORE2, SCALARS ENDF/B GRAPHIC X-SEC EVALUATION</td>
<td>AI 360</td>
<td>F+BAL</td>
<td>RS LT</td>
</tr>
<tr>
<td>R376</td>
<td>UPGRADE, UNRESOLVED REGION AVERAGE X-SEC CALC</td>
<td>BNL 7094</td>
<td>F4</td>
<td>RS P T</td>
</tr>
<tr>
<td>R377</td>
<td>SIGPLOT, RESOLVED MULTILEVEL B-W X-SEC CALC</td>
<td>BNL 7094</td>
<td>F4</td>
<td>RS P T</td>
</tr>
<tr>
<td>R378</td>
<td>TUB2, U-TUBE HEAT EXCHANGE STRESS ANALYSIS</td>
<td>KAPL 6600</td>
<td>F4</td>
<td>RS P</td>
</tr>
<tr>
<td>R379</td>
<td>SAFE-2D, PLANE + AXISYMMETRIC STRESS ANALYSIS</td>
<td>GGA 1108</td>
<td>F4</td>
<td>RS P T</td>
</tr>
<tr>
<td>R380</td>
<td>GATT, 3-D FEW-GP DIFFUSION CALC HEX-Z MESH</td>
<td>GGA 1108</td>
<td>F4</td>
<td>RS P T</td>
</tr>
<tr>
<td>R381</td>
<td>RAPP, HIGH-VELOCITY FLOW STUDY STEAM-WATER MIX</td>
<td>KAPL 6600</td>
<td>F4</td>
<td>RS P T</td>
</tr>
<tr>
<td>R382</td>
<td>M0266, LINEAR ELASTIC STRUCTURAL DYNAMICS</td>
<td>BAPL 6600</td>
<td>F4</td>
<td>RS T</td>
</tr>
<tr>
<td>R383</td>
<td>CHECKER/CRECT/DAMMET/PL0TFB/SLAVE3 ENDF/B PROC</td>
<td>NCSC 6600</td>
<td>F4</td>
<td>RS T</td>
</tr>
</tbody>
</table>

Page 514
# TABLE OF CONTENTS

## VI. LIBRARY PROGRAMS BY CLASSIFICATION ................................. 635

A. Cross-section and Resonance-integral Calculations .................. 637

B. Spectrum Calculations, Generation of Group Constants, Lattice and Cell Problems .................................................. 644

C. Static Design Studies .......................................................... 652

D. Depletion, Fuel Management, Cost Analysis, and Reactor Economics ................................................................. 661

E. Space-independent Kinetics ..................................................... 668

F. Space-Time Kinetics, Coupled Neutronics-Hydrodynamics-Thermodynamics, and Excursion Simulations ..................... 671

G. Radiological Safety, Hazard and Accident Analysis ................. 675

H. Steady-state and Transient Heat Transfer ................................. 679

I. Deformation and Stress Distribution Computations, Structural Analysis and Engineering Design Studies ..................... 683

J. Gamma Heating and Shield Design Problems .............................. 689

K. Total Systems Analysis .......................................................... 692

L. Data Preparation ..................................................................... 693

M. Data Management ..................................................................... 695

N. Subsidiary Calculations ............................................................. 698

O. Experimental Data Processing .................................................. 699

P. General Mathematical and Computing System Routines .............. 702

Z. Nuclear Data ............................................................................ 704

## VII. KWIC INDEX ........................................................................... 705

REFERENCES ................................................................................... 751
1. NAME OR DESIGNATION OF PROGRAM - SOR2

2. COMPUTER FOR WHICH PROGRAM IS DESIGNED AND OTHERS UPON WHICH IT IS OPERABLE - CDC6600


5. RESTRICTIONS ON THE COMPLEXITY OF THE PROBLEM - SOR2 WILL SOLVE AND JOIN TOGETHER UP TO 30 SHELLS. THE INTEGRATION ERROR BOUNDS, WHICH MAY BE ALTERED BY INPUT, ARE PRESET AT 5 SIGNIFICANT FIGURES. TO INSURE OPTIMUM RESULTS, IT IS RECOMMENDED THAT THE MAXIMUM LENGTH OF ANY ONE SHELL NOT EXCEED 6 ATTENUATION LENGTHS.

6. TYPICAL RUNNING TIME - AN AVERAGE TIME OF 10 SECONDS PER CASE IS REQUIRED.

7. UNUSUAL FEATURES OF THE PROGRAM - THE METHOD USED PERMITS SOLUTION OF PROBLEMS IN WHICH THE SHELL INCLUDES THE AXIS OF REVOLUTION. IT ALSO ALLOWS STEP CHANGES IN LOADINGS AND SOME SHELL PROPERTIES.


   CDC6600 VERSION SUBMITTED JUNE 1968, REPLACED BY REVISED VERSION JULY 1969.

    R. L. FAGAN, SOR-II INPUT AND MODIFICATIONS, KAPL NOTE, FEBRUARY 9, 1968.
11. MACHINE REQUIREMENTS -

12. PROGRAMMING LANGUAGE USED - FORTRAN IV

13. OPERATING SYSTEM OR MONITOR UNDER WHICH PROGRAM IS EXECUTED - SCOPE 2.0.

14. ANY OTHER PROGRAMMING OR OPERATING INFORMATION OR RESTRICTIONS -

15. NAME AND ESTABLISHMENT OF AUTHOR -
   R. L. FAGAN
   KNOLLS ATOMIC POWER LABORATORY
   GENERAL ELECTRIC COMPANY
   SCHENECTADY, NEW YORK

16. MATERIAL AVAILABLE - RESTRICTED DISTRIBUTION
   SOURCE DECK (1481 CARDS)
   SAMPLE PROBLEM (79 CARDS)
   REFERENCE REPORT AND NOTES

17. CATEGORY - I
   KEYWORDS - STRESSES, SHELLS, SPHERES, CYLINDERS, TEMPERATURE DISTRIBUTION
1. NAME OR DESIGNATION OF PROGRAM - CRAM

2. COMPUTER FOR WHICH PROGRAM IS DESIGNED AND OTHERS UPON WHICH IT IS OPERABLE - IBM7090, 360

3. NATURE OF PHYSICAL PROBLEM SOLVED - CRAM IS A PROGRAM TO SOLVE THE MULTIGROUP DIFFUSION EQUATIONS IN TWO-DIMENSIONS (R-Z, X-Y, OR R-THETA GEOMETRY), OR IN ONE-DIMENSION (SLAB, CYLINDRICAL, OR SPHERICAL GEOMETRY). NEUTRONS MAY SCATTER FROM ANY GROUP TO ANY OTHER. REAL, ADJOINT, AND SOURCE-TYPE PROBLEMS ARE ALL SOLVABLE. THE PROGRAM WILL COMPUTE THE K-EFFECTIVE OF THE SYSTEM OR ALTERNATIVELY SEARCH FOR CRITICALITY BY MOVING SPATIAL BOUNDARIES, VARYING MATERIAL COMPOSITIONS, OR VARYING TRANSVERSE BUCKLING.

4. METHOD OF SOLUTION - THE EQUATIONS ARE SOLVED BY FINITE DIFFERENCE METHODS.

5. RESTRICTIONS ON THE COMPLEXITY OF THE PROBLEM - MAXIMA ARE -
   - 100 POINTS IN EACH DIRECTION
   - 100 POINTS IN CHANGE VECTOR
   - 100 ENERGY GROUPS
   - 1000 ISOTOPES
   - 1000 MATERIALS
   - 1000 ZONES
   - 1000 (POINTS) * (NUMBER OF SPECTRA IN LIBRARY DATA) - IBM7090 VERSION ONLY

6. TYPICAL RUNNING TIME - 30 SECONDS PER 100 POINTS IN EACH ENERGY GROUP ARE REQUIRED ON THE IBM7090. 40 SECONDS PER 100 POINTS IN EACH ENERGY GROUP ARE REQUIRED ON AN IBM360/50.

7. UNUSUAL FEATURES OF THE PROGRAM - A GENERAL OUTPUT COMPILER IS PROVIDED WHICH CAN BE GIVEN OUTPUT INSTRUCTIONS IN ALGEBRAIC FORM AS PART OF THE PROBLEM INPUT DATA. THE USER CAN THEREBY DEVELOP HIS OWN ROUTINES FOR PROCESSING RESULTS. CROSS SECTIONS MAY BE PUT IN DIRECTLY OR MAY BE CHOSEN FROM THE PROGRAM LIBRARIES -- A GROUP CONDENSATION ROUTINE IS PROVIDED. IN BOTH THE IBM7090 AND IBM360 VERSIONS THERE ARE FACILITIES FOR RUNNING PROBLEMS IN SUCCESSION -- PASSING FLUXES AND/OR DATA FROM PROBLEM TO PROBLEM. OPTIONAL MESH DOUBLING IS PROVIDED TO CHECK FINITE DIFFERENCE ERRORS OR TO SPEED UP CRITICALITY SEARCHES IN THE EARLY STAGES. SOME STANDARD OUTPUT ROUTINES ARE PROVIDED IN THE IBM360 VERSION. OTHER FORTRAN ROUTINES CAN BE TEMPORARILY INSERTED USING THE LINKAGE EDITOR.

8. RELATED AND AUXILIARY PROGRAMS -

   IBM360 VERSION SUBMITTED NOVEMBER 1968.
10. REFERENCES - A. Hassitt, A COMPUTER PROGRAM TO SOLVE THE MULTI-
GROUP DIFFUSION EQUATIONS, TRG REPORT 229(R), MARCH 1962.
A. Hassitt, SUPPLEMENT TO TRG REPORT 229 (R), JANUARY
1963.
B. M. Segal, CRAM USERS GUIDE - REVISION NO. 1,
B. M. Segal, INFORMATION FOR USERS OF APDA CRAM COM-
PILER OUTPUT ROUTINES, MAY 24, 1963.
W. H. Hannum, CRAMX CONTROL PARAMETERS, ACCELERATION
W. H. Hannum, CRAM COMPILER, LOS ALAMOS MEMO
B. McGregor, A. G. Richards, AND R. G. J. Wood, AN
I.B.M. 360 VERSION OF THE NEUTRON DIFFUSION CODE CRAM, AAEC/TM429,
FEBRUARY 1968.

11. MACHINE REQUIREMENTS - ON THE IBM360 256K BYTES - THE CODE CAN
EASILY BE CHANGED TO PROFITABLY USE MORE STORAGE, 4 TAPE UNITS
(OR EQUIVALENT DISK STORAGE).

12. PROGRAMMING LANGUAGES USED - FORTRAN II AND FAP (IBM7090),
FORTRAN IV AND BAL (IBM360).

13. OPERATING SYSTEM OR MONITOR UNDER WHICH PROGRAM IS EXECUTED -
STANDARD IBM FORTRAN II MONITOR (IBM7090), OS/360 RELEASE 16
(IBM360).

14. ANY OTHER PROGRAMMING OR OPERATING INFORMATION OR RESTRICTIONS -
FILE 1 OF THE 7090 VERSION TAPE IS THE SOURCE DECK, BINARY DECK,
AND SAMPLE PROBLEM FOR THE STANDARD LASL VERSION OF CRAM WHICH
USES MAGNETIC TAPE FOR STORAGE. THE CROSS SECTION LIBRARIES, YOM,
HANSEN, AND ANL-11, ARE CONTAINED IN CHAINS 1, 5, AND 6, RESpec-
TIVELY, OF BOTH THE SOURCE AND OBJECT DECKS. FILE 2 IS THE SOURCE
DECKS FOR THOSE SUBROUTINES WHICH HAVE BEEN ALTERED BY LASL TO USE
DISKS AND DRUMS RATHER THAN MAGNETIC TAPE FOR INTERMEDIATE 'STOR-
AGE, THE OBJECT DECK FOR THE LASL DISK VERSION, AND A SAMPLE PROB-
LEM. FILE 3 IS HASSITT'S CPYCHN ROUTINE USED TO SUBSTITUTE ALTERN-
ATE CHAINS (E.G., COMPILERS) WITHOUT HAVING TO START FROM A FULL
BINARY OBJECT DECK. THE ID CARD, CONTROL CARDS, AND IOUS MAY
REQUIRE CHANGING TO CONFORM TO THE SYSTEM BY WHICH THE CODE IS TO
BE OPERATED. IOU IS AS FOLLOWS -

<table>
<thead>
<tr>
<th>Decks</th>
<th>Contents</th>
</tr>
</thead>
<tbody>
<tr>
<td>MONITOR</td>
<td>A1</td>
</tr>
<tr>
<td>INPUT</td>
<td>A2</td>
</tr>
<tr>
<td>OUTPUT</td>
<td>A3</td>
</tr>
<tr>
<td>F.D. COEFFICIENTS</td>
<td>A4</td>
</tr>
<tr>
<td>DUMP 1</td>
<td>A5</td>
</tr>
<tr>
<td>SPARE COEFFICIENTS</td>
<td>A6</td>
</tr>
<tr>
<td>FLUX 1</td>
<td>B1</td>
</tr>
<tr>
<td>FLUX 2</td>
<td>B2</td>
</tr>
<tr>
<td>CHAINS</td>
<td>B3</td>
</tr>
<tr>
<td>PUNCH</td>
<td>B4</td>
</tr>
<tr>
<td>DUMP 2</td>
<td>B6</td>
</tr>
<tr>
<td>IOU</td>
<td>B5</td>
</tr>
</tbody>
</table>

SCRATCH |
SCRATCH |
SCRATCH (LARGE PROBLEMS ONLY) |
SCRATCH |
15. NAME AND ESTABLISHMENT OF AUTHORS -

7090  A. HASSITT
      UNIVERSITY OF CALIFORNIA
      SAN DIEGO, CALIFORNIA

REVISION  W. H. HANNUM
       LOS ALAMOS SCIENTIFIC LABORATORY
       P. O. BOX 1663
       LOS ALAMOS, NEW MEXICO 87544

360  B. MCGREGOR
      AUSTRALIAN ATOMIC ENERGY COMMISSION
      SUTHERLAND, N.S.W., AUSTRALIA

16. MATERIAL AVAILABLE - MAGNETIC TAPE TRANSMITTAL (7090-22,270 CARDS, 360-9129 CARDS)

   SOURCE DECKS (7090 AND 360)
   BINARY DECKS (7090)
   SAMPLE PROBLEMS (7090 AND 360)
   YOM, HANSEN, AND ANL-11 LIBRARIES (7090)
   REFERENCE REPORTS, SUPPLEMENT, AND MEMORANDA

17. CATEGORY - C

KEYWORDS - DIFFUSION EQUATIONS, 1-DIMENSIONAL, SLABS, CYLINDERS, SPHERES, 2-DIMENSIONAL, R-Z, X-Y, R-THETA, MULTIGROUP, CRITICALITY SEARCHES, FLUX DISTRIBUTION, PROGRAMMING LANGUAGES
1. NAME OR DESIGNATION OF PROGRAM - BAM

2. COMPUTER FOR WHICH PROGRAM IS DESIGNED AND OTHERS UPON WHICH IT IS OPERABLE - IBM7090

3. NATURE OF PHYSICAL PROBLEM SOLVED - BAM COMPUTES THERMAL CONSTANTS, SPATIAL AND ENERGY DISTRIBUTIONS IN HETEROGENEOUS CYLINDRICAL CELLS BY ASSUMING SEPARABILITY OF SPACE AND ENERGY IN THE BOLTZMANN EQUATION.

4. METHOD OF SOLUTION - THE SPATIAL DISTRIBUTION IS FOUND USING THE S4 CYLINDRICAL GEOMETRY CODE AND THE SPECTRUM BY MEANS OF TEMPEST2. ALTHOUGH THIS METHOD IS NOT AS ACCURATE AS MORE SOPHISTICATED METHODS, THE METHOD IS COMMONLY USED AS AN ENGINEERING APPROXIMATION AND LEADS TO REASONABLY ACCURATE RESULTS IN SHORT TIMES.

5. RESTRICTIONS ON THE COMPLEXITY OF THE PROBLEM - CYLINDRICAL GEOMETRY
   MAXIMUM NUMBER OF SPACE POINTS = 100
   MAXIMUM NUMBER OF REGIONS = 10

6. TYPICAL RUNNING TIME - .5 TO 1.5 MINUTES ARE REQUIRED ON THE IBM7090.

7. UNUSUAL FEATURES OF THE PROGRAM -

8. RELATED AND AUXILIARY PROGRAMS - S4 CYLINDRICAL GEOMETRY CELL CODE (ACC ABSTRACT 53) AND TEMPEST2 (ACC ABSTRACT 50)

   IBM7090 VERSION SUBMITTED MAY 1963.


11. MACHINE REQUIREMENTS -

12. PROGRAMMING LANGUAGE USED - FORTRAN II

13. OPERATING SYSTEM OR MONITOR UNDER WHICH PROGRAM IS EXECUTED -

14. ANY OTHER PROGRAMMING OR OPERATING INFORMATION OR RESTRICTIONS - THE TEMPEST2 CODE PACKAGE MATERIAL SHOULD BE REQUESTED TOGETHER WITH THE BAM MATERIAL.
1. NAME OR DESIGNATION OF PROGRAM - VARI-QUIR

2. COMPUTER FOR WHICH THE PROGRAM IS DESIGNED AND OTHERS UPON WHICH IT IS OPERABLE - CDC6600, IBM360

3. NATURE OF PHYSICAL PROBLEM SOLVED - THE TIME-DEPENDENT, MULTI-GROUP, TWO-DIMENSIONAL NEUTRON DIFFUSION EQUATIONS ARE SOLVED IN X-Y OR R-Z GEOMETRY.

4. METHOD OF SOLUTION - MODIFIED-EULER WITH A SINGLE ITERATION. THE ICE SUBROUTINE IS USED. (AAICE3 - INTEGRATION WITH CONTROLLED ERROR).

5. RESTRICTIONS ON THE COMPLEXITY OF THE PROBLEM - MAXIMA OF -
   4 GROUPS
   6 PRECURSOR GROUPS
   36 REGIONS
   20 DIFFERENT MATERIALS


7. UNUSUAL FEATURES OF THE PROGRAM - FEEDBACK IS INTRODUCED INTO THE CODE THROUGH THREE SUBROUTINES...TO INITIALIZE THE FEEDBACK, TO CARRY IT THROUGH, AND TO PRINT OUT ANY DESIRED FEEDBACK VARIABLES AT VARIOUS TIMES. THESE SUBROUTINES ARE WRITTEN BY THE USER, SO THAT ANY DESIRED FEEDBACK MAY BE INTRODUCED.

8. RELATED AND AUXILIARY PROGRAMS - VARI-QUIR REQUIRES THE ICE SUBROUTINE.


11. MACHINE REQUIREMENTS -

12. PROGRAMMING LANGUAGE USED - FORTRAN IV
13. OPERATING SYSTEM OR MONITOR UNDER WHICH PROGRAM IS EXECUTED - 
SCOPE (CDC6600) AND OS/360 (IBM360).

14. ANY OTHER PROGRAMMING OR OPERATING INFORMATION OR RESTRICTIONS -

15. NAME AND ESTABLISHMENT OF AUTHORS -
   6600 GEORGE COLLIER
   INDUSTRY AND DEFENSE PRODUCTS
   WESTINGHOUSE ELECTRIC CORPORATION
   ASTRONUCLEAR LABORATORY
   P. O. BOX 10864
   PITTSBURGH, PENNSYLVANIA 15236

   360 EDWIN L. COX
   RESEARCH REACTOR FACILITY
   UNIVERSITY OF MISSOURI
   RESEARCH PARK
   COLUMBIA, MISSOURI 65201

16. MATERIAL AVAILABLE - MAGNETIC TAPE TRANSMITTAL
   SOURCE DECKS (6600-2012 CARDS, 360-1945 CARDS)
   SAMPLE PROBLEMS (6600-74 CARDS, 360-85 CARDS)
   REFERENCE REPORT AND NOTE

17. CATEGORY - C
   KEYWORDS - SPACE-TIME, 2-DIMENSIONAL, DIFFUSION EQUATIONS, X-Y,
   R-Z, FLUX DISTRIBUTION, FEEDBACK, ICE CODES
1. NAME OR DESIGNATION OF PROGRAM - GAROL

2. COMPUTER FOR WHICH PROGRAM IS DESIGNED AND OTHERS UPON WHICH
IT IS OPERABLE - IBM7044,7094

3. NATURE OF PHYSICAL PROBLEM SOLVED - GAROL COMPUTES EFFECTIVE GROUP
CROSS SECTIONS FOR THE RESOLVED RESONANCES OF A MIXTURE OF
ISOTOPES IN A TWO-REGION CELL. BASIC CROSS SECTIONS INCLUDE
TEMPERATURE DEPENDENCE. THE PROGRAM ALLOWS A CHOICE OF GEOMETRIES
AND CAN ACCEPT AN ARBITRARY TABLE OF ESCAPE PROBABILITIES. A
DANCOFF CORRECTION MAY BE USED TO ACCOUNT FOR SHADOWING EFFECTS IN
A TIGHT LATTICE, AND CROSS SECTIONS MAY BE 1/V, CONSTANT, COMPUTED
FROM BREIT-WIGNER RESONANCE PARAMETERS, OR GIVEN IN TABULAR
FORM. THE MESH MAY BE CHOSEN AT EQUAL ENERGY OR LETHARGY
INTERVALS, PROPORTIONAL TO THE NEUTRON VELOCITY, OR AS AN ARBITRARY
TABLE OF VALUES.

4. METHOD OF SOLUTION - THE COLLISION PROBABILITY METHOD YIELDS TWO
COUPLED INTEGRAL EQUATIONS WHICH ARISE FROM A NEUTRON BALANCE IN
EACH REGION. THESE ARE SOLVED NUMERICALLY TO OBTAIN THE FLUX
SPECTRUM IN EACH REGION, AS WELL AS GROUP CROSS SECTIONS FOR EACH
REGION AND FOR THE CELL.

5. RESTRICTIONS ON THE COMPLEXITY OF THE PROBLEM -
   NUMBER OF NUCLIDES EQUAL TO OR LESS THAN 10.
   NUMBER OF RESONANCES PER ISOTOPE EQUAL TO OR LESS THAN 500.
   TOTAL NUMBER OF RESONANCES FOR ALL ISOTOPES EQUAL TO OR LESS
   THAN 600.
   NUMBER OF REGIONS EQUAL TO OR LESS THAN 2.

6. TYPICAL RUNNING TIME - VARIES CONSIDERABLY FROM PROBLEM TO
PROBLEM. WITH 2 ISOTOPES, U235 AND H, AND 6600 ENERGY POINTS
COVERING THE ENERGY RANGE, 0.5 EV TO 950 EV, THE RUNNING TIME WAS
3 MINUTES ON THE IBM7044.

7. UNUSUAL FEATURES OF THE PROGRAM - OVERLAP OF RESONANCES OF AN
INDIVIDUAL RESONANCE ABSORBER AND OF MIXTURES OF DIFFERENT
RESONANCE ABSORBERS IS TREATED EXACTLY. SLOWING DOWN IN ALL
ISOTOPES IS COMPUTED EXACTLY, SO THAT THE NR APPROXIMATION NEED
NOT BE MADE FOR A HEAVY MODERATOR. OPTIONS PERMIT THE AUTOMATIC
PLOTTING ON AN SC-4020 RECORDER OF FLUXES, CROSS SECTIONS,
COLLISION DENSITY AND REACTION RATES.

8. RELATED AND AUXILIARY PROGRAMS -

   IBM7044 VERSION SUBMITTED FEBRUARY 1966.
   IBM7094 VERSION SUBMITTED JULY 1969.

10. REFERENCES - C. A. STEVENS AND C. V. SMITH, GAROL, A COMPUTER
    PROGRAM FOR EVALUATING RESONANCE ABSORPTION INCLUDING RESONANCE
10. REFERENCES (CONTINUED)

11. MACHINE REQUIREMENTS - 32K IBM7044. IF PLOTTING IS DESIRED, AN SC-4020 RECORDER IS NEEDED.

12. PROGRAMMING LANGUAGES USED - FORTRAN IV (IBM7044 AND IBM7094) AND MAP (IBM7044)

13. OPERATING SYSTEM OR MONITOR UNDER WHICH PROGRAM IS EXECUTED - IBM7040/7044 OPERATING SYSTEM (16/32K), VERSION IX, STANDARD IBM FORTRAN MONITOR (IBM7094).

14. ANY OTHER PROGRAMMING OR OPERATING INFORMATION OR RESTRICTIONS - A CROSS SECTION TAPE GENERATED IN A GAROL RUN CAN BE USED IN SUBSEQUENT RUNS. WITH THE EXCEPTION OF THE NASA SYSTEMS ROUTINE, TIME1, THE 7094 VERSION OF GAROL WITH OVERLAYS SHOULD BE OPERABLE ON ANY 7090/94.

15. NAME AND ESTABLISHMENT OF AUTHORS -
7044 C. A. STEVENS AND C. V. SMITH
GULF GENERAL ATOMIC INCORPORATED
P. O. BOX 608
SAN DIEGO, CALIFORNIA 92112

7094 ROBERT M. WESTFALL
NATIONAL AERONAUTICS AND SPACE ADMINISTRATION
LEWIS RESEARCH CENTER
21000 BROOKPARK ROAD
CLEVELAND, OHIO 44135

16. MATERIAL AVAILABLE - MAGNETIC TAPE TRANSMITTAL
SOURCE DECKS (7044-2727 CARDS, 7094-2234 CARDS)
SAMPLE PROBLEMS (7044-32 CARDS, 7094-33 CARDS)
REFERENCE REPORT AND TECH BRIEFS

17. CATEGORY - B
KEYWORDS - RESOLVED REGION, BREIT-WIGNER FORMULA, RESONANCE PARAMETERS, CELL CALCULATION, CROSS SECTIONS, DANCOFF CORRECTION, TEMPERATURE DISTRIBUTION, GRAPHS
1. NAME OR DESIGNATION OF PROGRAM - MACH1

2. COMPUTER FOR WHICH PROGRAM IS DESIGNED AND OTHERS UPON WHICH IT IS OPERABLE - CDC3600,6500

3. NATURE OF PHYSICAL PROBLEM SOLVED - MACH1 PERFORMS ONE-DIMENSIONAL MULTIGROUP DIFFUSION SOLUTIONS AND ASSOCIATED CALCULATIONS, INCLUDING CRITICALITY SEARCHES, PERTURBATION, REACTION SUMMARY, BETA EFFECTIVE, GROUP COLLAPSING, AND POINTWISE REACTION RATES AND RATIOS. SEVERAL CARD DUMPS OF COMPUTED DATA ARE AVAILABLE ON OPTION.

4. METHOD OF SOLUTION - THE DIFFUSION EQUATION SOLUTION METHOD IS THE SAME AS THAT IN AIM6 (ACC ABSTRACT 29). PERTURBATION AND EFFECTIVE BETA ROUTINES HAVE BEEN ADAPTED FROM DEL AND 1188/RE. REMAINING COMPUTATIONS CONSIST OF STRAIGHTFORWARD DATA MANIPULATION.

5. RESTRICTIONS ON THE COMPLEXITY OF THE PROBLEM - MAXIMA OF:
   - 20 SPATIAL REGIONS
   - 150 MESH POINTS
   - 30 ENERGY GROUPS (3600 VERSION ONLY)
   - 28 ENERGY GROUPS (6500 VERSION ONLY)
   - 15 DOWNSCATTER GROUPS
   - 20 ISOTOPES PER PROBLEM
   - 20 HOMOGENIZED MATERIALS
   - 20 ISOTOPES PER HOMOGENIZED MATERIAL

   A SPECIAL SCATTER TREATMENT IS USED WHICH ALLOWS FULL DOWNSCATTER FOR HYDROGEN ISOTOPES.

6. TYPICAL RUNNING TIME - RUNNING TIME IS STRONGLY DEPENDENT ON THE TYPE OF PROBLEM AND QUALITY OF INITIAL GUESS. A K-CALCULATION OF A WATER-MODERATED SYSTEM WITH 16 GROUPS, 5 DOWNSCATTER GROUPS, 20 REGIONS, AND 130 MESH POINTS, WITH A CONSTANT INITIAL SOURCE GUESS CONVERGED IN 11 INNER ITERATIONS TO WITHIN 10**-5(DELTAK/K) IN 46.5 SECONDS. A COMPOSITION SEARCH ON A FAST REACTOR CORE USING 22 GROUPS, 12 DOWNSCATTERS, 9 REGIONS, 150 MESH POINTS, AND REGION-DEPENDENT INITIAL SOURCE GUESS, CONVERGED WITH A TOTAL OF 19 INNER ITERATIONS AND 3 TRIALS TO 10**-6(DELTAK/K), FOUND THE ADJOINT SOLUTION TO THE SAME ACCURACY AND RAN A SINGLE PERTURBATION PASS IN 144.9 SECONDS.

7. UNUSUAL FEATURES OF THE PROGRAM - THE PROGRAM WAS DESIGNED FOR USE ON FAST REACTOR SURVEY COMPUTATIONS AND AS A SUPPORTING UTILITY FOR MORE COMPLEX PROGRAMS. PROBLEMS CAN BE RUN IN SEQUENCE WITH ONLY REQUIRED DATA CHANGES AS INPUT FOR PROBLEMS AFTER THE FIRST. THE PROGRAM CONTAINS A CALL TO A NON-EXISTENT OVERLAY. THIS CAN PROVIDE A LINK TO ANY ROUTINE WRITTEN BY THE USER FOR MODIFICATION OR MANIPULATION OF MACH1 OUTPUT DATA. THE ROUTINE HAS ACCESS TO INPUT DATA AND DATA GENERATED BY THE CODE, AND CAN CONTROL LATER PROBLEMS THROUGH OPTION SWITCHES. MICROSCOPIC CROSS SECTIONS MAY BE CALLED FROM A LIBRARY OR INPUT AT RUN TIME. INDIVIDUAL CROSS SECTION QUANTITIES MAY BE MODIFIED TEMPORARILY FOR A PROBLEM.
8. RELATED AND AUXILIARY PROGRAMS - AIM6, DEL, AND 1188/RE WERE ADAPTED AND MODIFIED TO MAKE UP THE MAJOR PORTIONS OF THE PROGRAM.

   CDC3600 VERSION SUBMITTED DECEMBER 1966. 
   CDC6500 VERSION SUBMITTED MARCH 1969.

    L. C. KVITEK, DEL, A PERTURBATION PROGRAM WRITTEN IN FORTRAN, ANL-7052, JUNE 1965.

11. MACHINE REQUIREMENTS - CDC 3600 WITH 64K CORE STORAGE AND 9 TAPE DRIVES (INCLUDING 5 SYSTEM TAPES)

12. PROGRAMMING LANGUAGES USED - 3600 FORTRAN (CDC3600) AND FORTRAN IV (CDC6500)

13. OPERATING SYSTEM OR MONITOR UNDER WHICH PROGRAM IS EXECUTED - CDC 3600 SCOPE-SATCOPS II.

14. ANY OTHER PROGRAMMING OR OPERATING INFORMATION OR RESTRICTIONS - THE PROGRAM IS DIVIDED INTO SIX OVERLAYS WITH A TOTAL OF ELEVEN SEGMENTS. EXTENSIVE USE HAS BEEN MADE OF 3600 FORTRAN CAPABILITIES. APPROXIMATELY 51000 CORE CELLS ARE REQUIRED BY THE PROGRAM.

15. NAME AND ESTABLISHMENT OF AUTHORS -
    3600 D. A. MENELEY
       REACTOR PHYSICS DIVISION
       ARGONNE NATIONAL LABORATORY
       9700 SOUTH CASS AVENUE
       ARGONNE, ILLINOIS 60439

    6500 K. O. OTT
       NUCLEAR ENGINEERING DEPARTMENT
       PURDUE UNIVERSITY
       WEST LAFAYETTE, INDIANA 47906

16. MATERIAL AVAILABLE - MAGNETIC TAPE TRANSMITTAL
    SOURCE DECKS (MACH1 3600-6980 CARDS, 6500-7105 CARDS, RXMACH 3600-137 CARDS, MACHLIB3 6500-479 CARDS)
    SAMPLE PROBLEMS (MACH1 3600-95 CARDS, 6500-40 CARDS)
    LIBRARY (MACHLIB3 6500-2145 CARDS)
    MEMORANDA - COMMON BLOCK DICTIONARY, MACH1 GUIDE
    REFERENCE REPORT, ANL-7223
17. CATEGORY - C
KEYWORDS - DIFFUSION EQUATIONS, 1-DIMENSIONAL, MULTIGROUP, SLABS, CYLINDERS, SPHERES, CRITICALITY SEARCHES, REACTION RATES, AVERAGES, PERTURBATION THEORY
1. **NAME OR DESIGNATION OF PROGRAM** - SEAL SHELL 2 (M0110)

2. **COMPUTER FOR WHICH PROGRAM IS DESIGNED AND OTHERS UPON WHICH IT IS OPERABLE** - CDC 6600

3. **NATURE OF PHYSICAL PROBLEM SOLVED** - The SEAL SHELL 2 program determines stresses, strains, deflections, and reactions in a thick shell of revolution with axisymmetric loading. The loading consists of a temperature distribution, inside and outside pressure distributions, and circumferential forces and moments applied to the middle surface. The shell is linear-elastic with tensile, bending, and shear strains.

4. **METHOD OF SOLUTION** - The shell is divided into 2 to 100 segments. Each segment is subjected to 11 deformation shapes, whose magnitudes are calculated by the principle of virtual work. That is, for small virtual changes in magnitude, the changes in strain energy and external work are determined. Then equating the energy to the work gives a set of simultaneous equations which may be solved for the magnitudes.

5. **RESTRICTIONS ON THE COMPLEXITY OF THE PROBLEM** - Maxima of:
   - 100 segments
   - 11 deformation shapes per segment

6. **TYPICAL RUNNING TIME** -

7. **UNUSUAL FEATURES OF THE PROGRAM** - The program will handle thick shells of revolution, as well as the usual thin shell structures. As discussed in Addendum I, an option allows the requestor to input influence coefficients of the flexibility matrix of any segment. The coefficients may be for any axisymmetric structure, such as a tube sheet in a heat exchanger. In this way, different branches of shells may be connected together. Another application is to condense a group of segments into a single element which may be used in a series of problems and so cut down running time on the computer. Also, using the condensed groups of segments as a single segment allows the user to specify more elements in the analysis of a critical region in the shell.

8. **RELATED AND AUXILIARY PROGRAMS** -


11. MACHINE REQUIREMENTS - 32K MEMORY
12. PROGRAMMING LANGUAGE USED - FORTRAN IV
13. OPERATING SYSTEM OR MONITOR UNDER WHICH PROGRAM IS EXECUTED - SCOPE 2.0.
14. ANY OTHER PROGRAMMING OR OPERATING INFORMATION OR RESTRICTIONS - AN OPTION IS INCLUDED TO PLOT THE SHELL GEOMETRY ON THE CDC280MICROFILM RECORDER.
15. NAME AND ESTABLISHMENT OF AUTHOR - C. M. FRIEDRICH WESTINGHOUSE ELECTRIC CORPORATION BETTIS ATOMIC POWER LABORATORY P. O. BOX 79 WEST MIFFLIN, PENNSYLVANIA 15122
17. CATEGORY - I KEYWORDS - STRESSES, SHELLS, PRESSURE DISTRIBUTION, TEMPERATURE DISTRIBUTION, DEFORMATION, ELASTICITY
6. **Typical Running Time** - On the IBM7040 approximately 0.32 second per time advancement is required for a problem using 90 mesh points for heat structures.

7. **Unusual Features of the Program** -

8. **Related and Auxiliary Programs** -


11. **Machine Requirements** - 32K memory

12. **Programming Languages Used** - Fortran IV and MAP (Decimal input, date, and clocking routines)

13. **Operating System or Monitor Under Which Program Is Executed** - IBSYS.

14. **Any Other Programming or Operating Information or Restrictions** -

15. **Name and Establishment of Authors** -
   L. C. Richardson, L. J. Finnegar, R. J. Wagner, and J. M. Waage
   Atomic Energy Division
   Phillips Petroleum Company
   P. O. Box 2067
   Idaho Falls, Idaho 83401

16. **Material Available** - Magnetic tape transmittal
   Source deck (7040-2154 cards)
   Sample problem (7040-58 cards)
   Reference report

17. **Category** - G
    **Keywords** - Accidents, temperature distribution, pressure distribution, containment, coolants, water, liquids, vapors, leakage, heat transfer
1. NAME OR DESIGNATION OF PROGRAM - GGC4

2. COMPUTER FOR WHICH PROGRAM IS DESIGNED AND OTHERS UPON WHICH IT IS OPERABLE - UNIVAC1108

3. NATURE OF PHYSICAL PROBLEM SOLVED - THE GGC4 PROGRAM SOLVES THE MULTIGROUP SPECTRUM EQUATIONS WITH SPATIAL DEPENDENCE REPRESENTED BY A SINGLE POSITIVE INPUT BUCKLING. BROAD GROUP CROSS SECTIONS (SHIELDED OR UNSHIELDED) ARE PREPARED FOR DIFFUSION AND TRANSPORT CODES BY AVERAGING WITH THE CALCULATED SPECTRA OVER INPUT-DESIGNATED ENERGY LIMITS. THE CODE IS DIVided INTO THREE MAIN PARTS: A FAST (GAM) SECTION WHICH COVERS THE ENERGY RANGE FROM 14.9 MEV TO 0.414 EV, A THERMAL (GATHER) SECTION WHICH COVERS THE ENERGY RANGE FROM 0 TO 2.38 EV, AND A COMBINING (COMBO) SECTION WHICH COMBINES FAST AND THERMAL CROSS SECTIONS INTO SINGLE SETS. BASIC NUCLEAR DATA FOR THE FAST SECTION WHICH CONSISTS OF FINE GROUP-AVERAGED CROSS SECTIONS AND RESONANCE PARAMETERS IS READ OFF A DATA TAPE. THE FINE GROUP ABSORPTION AND FISSION CROSS SECTIONS MAY BE ADJUSTED BY PERFORMING A RESONANCE INTEGRAL CALCULATION. UTILIZING A FISSION SOURCE AND AN INPUT BUCKLING, THE CODE SOLVES THE P1, B1, B2, OR B3 APPROXIMATION TO OBTAIN THE ENERGY-DEPENDENT FAST SPECTRUM. TWO OR SIX SPATIAL MOMENTS OF THE SPECTRUM (DUE TO A PLANE SOURCE) MAY ALSO BE EVALUATED. INSTEAD OF PERFORMING A SPECTRUM CALCULATION, THE USER MAY ENTER THE LEGENDRE COMPONENTS OF THE ANGULAR FLUX DIRECTLY. AS MANY INPUT-DESIGNATED BROAD GROUP STRUCTURES AS DESIRED, THE CODE CALCULATES AND SAVES (FOR THE COMBINING SECTION) SPECTRUM-WEIGHTED AVERAGES OF MICROSCOPIC AND MACROSCOPIC CROSS SECTIONS AND TRANSFER ARRAYS. SLOWING DOWN SOURCES ARE CALCULATED AND SAVED FOR USE IN THE LOWER ENERGY RANGE. GIVEN BASIC NUCLEAR DATA, THE THERMAL SECTION OF GGC4 DETERMINES A THERMAL SPECTRUM BY EITHER READING IT AS INPUT, BY CALCULATING A MAXWELLIAN SPECTRUM FOR A GIVEN TEMPERATURE, OR BY AN ITERATIVE SOLUTION OF THE P3, B0, P1, OR B1 EQUATIONS FOR AN INPUT BUCKLING. TIME MOMENTS OF THE TIME AND ENERGY-DEPENDENT DIFFUSION EQUATIONS ARE CALCULATED (AS AN OPTION) USING THE INPUT BUCKLING TO REPRESENT LEAKAGE. BROAD GROUP CROSS SECTIONS ARE PREPARED BY AVERAGING FINE GROUP CROSS SECTIONS OVER THE CALCULATED SPECTRA. BROAD GROUP STRUCTURES ARE READ AS INPUT. THE COMBINING SECTION OF GGC4 TAKES THE BROAD GROUP-AVERAGED CROSS SECTIONS FROM THE FAST AND THERMAL PORTIONS OF GGC4 AND FORMS MULTIGROUP CROSS SECTION TABLES. THESE TABLES ARE PREPARED IN STANDARD FORMATS FOR TRANSFER OR DIFFUSION THEORY CALCULATIONS. IN ADDITION, IT IS POSSIBLE TO USE THE COMBINING SECTION TO PRODUCE MIXTURES NOT USED IN THE SPECTRUM CALCULATION OR TO COMBINE THE RESULTS OF DIFFERENT FAST AND THERMAL SECTION CALCULATIONS AND SO ON. THESE OPTIONS ARE DESCRIBED IN REFERENCE 2.

4. METHOD OF SOLUTION - IN THE FAST SECTION EITHER THE P1 OR THE B1, B2, OR B3 APPROXIMATION IS MADE TO THE TRANSPORT EQUATION USING THE POSITIVE, ENERGY-INDEPENDENT BUCKLING. IN EACH APPROXIMATION LEGENDRE MOMENTS OF THE ANGULAR FLUX ARE COMPUTED BY DIRECT NUMERICAL INTEGRATION OF THE SLOWING DOWN EQUATIONS. IN THE RESO-
4. Method of Solution (Continued)

Nance calculations, Doppler broadened (at an input temperature) absorption and scattering cross sections are used. The resonance treatment allows up to two admixed moderators in an absorber lump imbedded in a surrounding moderator. The absorber in the lump is treated by using either the narrow resonance approximation, the narrow resonance infinite mass approximation, or a solution of the slowing down integral equations to determine the collision density through the resonance. The admixed moderators are treated by using either an asymptotic form of, or an integral equation solution for, the collision density. In the resonance calculation either standard geometry collision probabilities are used or tables of collision probabilities are entered. Dancoff corrections can also be made. In the region of unresolved resonances, resonance absorption is calculated by using Porter-Thomas distributions, but only S-wave neutrons are considered. In the thermal section either the $B_0$, $B_1$, $P_0$, or $P_1$ approximation to the transport equation is made, and in all options Legendre moments of the angular flux are computed. A trapezoidal energy integration mesh is used, and the resulting equations are solved iteratively by using a source-normalized, over-relaxed, Gaussian technique. Averages over broad groups are performed by simple numerical integration. The results obtained in the fast and thermal sections are stored on special tapes. These tapes may contain results for a number of problems, each problem including fine group cross section data for a number of nuclides. If the problem number is specified on these tapes, and a desired list of nuclides is given, the combining code will punch microscopic cross sections for the requested list of nuclides. The program also treats mixtures. Given the atomic densities of the nuclides in a mixture, the code will punch macroscopic cross sections.

5. Restrictions on the Complexity of the Problem - Maxima of -

- 99 fast groups
- 101 thermal fine groups
- 99 fast broad groups
- 50 thermal broad groups
- 50 broad groups in the combining section
- 100 resonances per nuclide
- 2 moderators admixed with a resonance absorber
- 305 entries in the escape probability table for cylindrical geometries
- 505 entries in the escape probability table for slab geometries

A single and positive value for the buckling ($B^2$) must be supplied.

6. Typical Running Time - A B1 calculation in the fast section for 3 nuclides and 6 broad groups takes approximately 4 minutes on the UNIVAC1108 if a resonance calculation (1/2 minute) is performed for one nuclide. The thermal calculation for 3 broad groups requires approximately 2 minutes, which include about 7 seconds for the iterative procedure. To punch standard diffusion and standard transport cross sections for this problem requires 2 seconds.
7. **UNUSUAL FEATURES OF THE PROGRAM** - THERE IS AN OPTION IN GGC4 WHICH MAKES IT POSSIBLE TO SHORTEN THE PUNCHING PROCESS FOR LARGE TWO-DIMENSIONAL TRANSFER ARRAYS. THIS CAN BE DONE BY SPECIFYING A MAXIMUM NUMBER OF DESIRED UPSCATTERING AND DOWNSCATTERING TERMS.


   UNIVAC1108 VERSION OF GGC3 SUBMITTED SEPTEMBER 1967, DELETED JUNE 1968.
   UNIVAC1108 VERSION OF GGC4 SUBMITTED JUNE 1968, REPLACED BY LATER VERSION JULY 1969.

    BCDCON, GA NOTE.

11. **MACHINE REQUIREMENTS** - 64K MEMORY WITH 11 TAPE UNITS (SOME OF WHICH MAY BE DRUM AREAS).

12. **PROGRAMMING LANGUAGE USED** - FORTRAN IV

13. **OPERATING SYSTEM OR MONITOR UNDER WHICH PROGRAM IS EXECUTED** - UNIVAC EXEC II, GAX29.

14. **ANY OTHER PROGRAMMING OR OPERATING INFORMATION OR RESTRICTIONS** - THERE IS NO RESTRICTION ON THE NUMBER OF PROBLEMS THAT CAN BE RUN CONSECUTIVELY IN EACH SECTION, NOR IS THERE A RESTRICTION ON THE NUMBER OF NUCLIDES PER PROBLEM. WITHOUT USING THE NEW OPTIONS, GGC4 CAN ALSO BE RUN WITH THE GGC3 INPUT INSTRUCTIONS.

15. **NAME AND ESTABLISHMENT OF AUTHORS** -

   **PROGRAM AUTHOR**
   J. ADIR
   GULF GENERAL ATOMIC INCORPORATED
   P. O. BOX 608
   SAN DIEGO, CALIFORNIA 92112

   **PRESENT CONTACT**
   D. R. MATHEWS
   GULF GENERAL ATOMIC INCORPORATED
   P. O. BOX 608
   SAN DIEGO, CALIFORNIA 92112
16. MATERIAL AVAILABLE - MAGNETIC TAPE TRANSMITTAL (3 TAPES)
SOURCE DECKS (GGC4 11,567 CARDS, WTFG 1646 CARDS, MAKE 1033 CARDS, MST 463 CARDS, PRINT 769 CARDS, MIXER 500 CARDS, MGT3 251 CARDS, SPRINT 334 CARDS, COMBIN 479 CARDS, DOP 927 CARDS)
SAMPLE PROBLEM (GGC4 38 CARDS)
FAST DATA TAPE (151 NUCLIDES - TAPE 2 - 7362 BINARY RECORDS)
THERMAL DATA TAPE (222 NUCLIDES - TAPE 3 - 2854 BINARY RECORDS)
REFERENCE REPORTS AND NOTE

17. CATEGORY - B
KEYWORDS - MULTIGROUP CROSS SECTIONS, AVERAGES, FAST, THERMAL, SPECTRA, DOPPLER BROADENING, DANCOFF CORRECTION, ANGULAR DISTRIBUTION, RESONANCE PARAMETERS, MAKE CODES, MST CODES, PRINT CODES, MIXER CODES, WTFG CODES, MGT3 CODES, SPRINT CODES, COMBIN CODES, DOP CODES
1. **NAME OR DESIGNATION OF PROGRAM** - LION

2. **COMPUTER FOR WHICH PROGRAM IS DESIGNED AND OTHERS UPON WHICH IT IS OPERABLE** - CDC6600

3. **NATURE OF PHYSICAL PROBLEM SOLVED** - LION IS A DIGITAL COMPUTER PROGRAM WHICH WILL SOLVE THREE-DIMENSIONAL TRANSIENT AND STEADY-STATE TEMPERATURE DISTRIBUTION PROBLEMS. THE INPUT CONSISTS OF GEOMETRY, PHYSICAL PROPERTIES, BOUNDARY CONDITIONS, INTERNAL HEAT GENERATION RATES, AND COOLANT FLOW RATES AS A FUNCTION OF TIME. IN ADDITION TO SOLVING PROBLEMS OF HEAT CONDUCTION IN A STRUCTURE, LION CAN HANDLE FORCED CONVECTION, FREE CONVECTION, AND RADIATION OR A COMBINATION OF THESE AT THE SURFACE OF THE STRUCTURE. THE OUTPUT CONSISTS OF COMPLETE NODAL TEMPERATURE DISTRIBUTIONS ALONG WITH SURFACE FLUXES AND HEAT TRANSFER COEFFICIENTS. AN OPTION IS INCLUDED IN THE PROGRAM FOR DETERMINING THE MEAN TEMPERATURE IN ANY SPECIFIED SECTION OF THE STRUCTURE.

4. **METHOD OF SOLUTION** - THE MODELING IS BASED ON THE CONCEPT OF NODAL POINTS CONNECTED BY ONE-DIMENSIONAL THERMAL CONDUCTANCE EQUATIONS IN AS MANY AS SIX DIRECTIONS SIMULTANEOUSLY. THE EXPLICIT, OR FIRST FORWARD DIFFERENCE METHOD, IS THEN USED TO OBTAIN THE SOLUTIONS TO THESE EQUATIONS FOR THE THREE-DIMENSIONAL FIELD. THE TEMPERATURE CHANGE FOR A SURFACE NODE IS CALCULATED BY APPLYING THE OHMS LAW ANALOGY FOR HEAT FLOW TO A NODE WITH NO CAPACITANCE.

5. **RESTRICTIONS ON THE COMPLEXITY OF THE PROBLEM** -
   \[ 11 \times D_1 + 22 \times D_2 + 20 \times D_3 + 61 \times D_4 + 15 \times D_5 \leq 21000 \]
   - VARIABLE NUMBER OF SURFACE CONNECTIONS (D1)
   - VARIABLE NUMBER OF INTERNAL NODES (D2)
   - VARIABLE NUMBER OF SURFACE NODES (D3)
   - VARIABLE NUMBER OF BOUNDARY NODES (D4)
   - VARIABLE NUMBER OF COOLANT NODES (D5)
   - 15 MATERIALS
   - 7 \times D_2 INTERNAL CONNECTIONS
   - 12 COOLANT CHANNELS (EACH WITH 4 SIDES MAXIMUM)
   - 75 PRINTOUTS
   - 7 INTERNAL CONNECTIONS PER INTERNAL NODE
   - 6 SURFACE CONNECTIONS PER SURFACE NODE
   - 30 MEANS AND DIFFERENCES SETS
   - 100 NODES PER SET (WITHOUT EQUIVALENT LINEAR)
   - 50 NODES PER SET (WITH EQUIVALENT LINEAR)
   - 6 TYPES OF MEANS AND DIFFERENCES SETS
   - 5 GRAPHS
   - 6 CURVES PER GRAPH
   - 13 TYPES OF QUANTITIES GRAPHABLE
   - 4 SUB-COOLED NUCLEATE BOILING REGIONS

6. **TYPICAL RUNNING TIME** - THE APPROXIMATE RUNNING TIME ON THE CDC6600 IS 1 MINUTE PER 10 NODES FOR A STEADY-STATE PROBLEM AND 1.5 MINUTES PER 10 NODES FOR A TRANSIENT PROBLEM.
1. NAME OR DESIGNATION OF PROGRAM - 2DB

2. COMPUTER FOR WHICH PROGRAM IS DESIGNED AND OTHERS UPON WHICH IT IS OPERABLE - UNIVAC1108, CDC6600

3. NATURE OF PHYSICAL PROBLEM SOLVED - 2DB IS A FLEXIBLE, TWO-DIMENSIONAL (X-Y, R-Z, R-THETA, HEX GEOMETRY) DIFFUSION CODE FOR USE IN FAST REACTOR ANALYSES. THE CODE CAN BE USED TO -
   (A) COMPUTE FUEL BURNUP USING FLEXIBLE MATERIAL SHUFFLING SCHEME,
   (B) PERFORM CRITICALITY SEARCHES ON TIME ABSORPTION (ALPHA), MATERIAL CONCENTRATIONS, AND REGION DIMENSIONS USING A REGULAR OR ADJOINT MODEL. CRITICALITY SEARCHES CAN BE PERFORMED DURING BURNUP TO COMPENSATE FOR FUEL DEPLETION,
   (C) COMPUTE FLUX DISTRIBUTIONS FOR AN ARBITRARY EXTRANEOUS SOURCE.

4. METHOD OF SOLUTION - STANDARD SOURCE-ITERATION TECHNIQUES ARE USED. GROUP REBALANCING AND SUCCESSIVE OVERRELAXATION WITH LINE INVERSION ARE USED TO ACCELERATE CONVERGENCE. MATERIAL BURNUP IS BY REACTOR ZONE. THE BURNUP RATE IS DETERMINED BY THE ZONE AND ENERGY (GROUP) AVERAGED CROSS SECTIONS WHICH ARE RECOMPUTED AFTER EACH TIME-STEP. THE ISOTOPIC CHAINS, WHICH CAN CONTAIN ANY NUMBER OF ISOTOPES, ARE FORMED BY THE USER. THE CODE DOES NOT CONTAIN BUILT-IN OR INTERNAL CHAINS.

5. RESTRICTIONS ON THE COMPLEXITY OF THE PROBLEM - SINCE VARIABLE DIMENSIONING IS EMPLOYED, NO SIMPLE BOUNDS CAN BE STATED. THE CURRENT 1108 VERSION, HOWEVER, IS NOMINALLY RESTRICTED TO 50 ENERGY GROUPS IN A 65K MEMORY. IN THE 6600 VERSION THE POWER FRACTION, AVERAGE BURNUP RATE, AND BREEDING RATIO CALCULATIONS ARE LIMITED TO REACTORS WITH A MAXIMUM OF 50 ZONES.

6. TYPICAL RUNNING TIME - A KEFF CALCULATION WITH A 30 X 30 MESH AND FOUR ENERGY GROUPS REQUIRES APPROXIMATELY ONE MINUTE. A TYPICAL BURNUP CALCULATION REQUIRES ROUGHLY TWICE THE MACHINE TIME OF A KEFF CALCULATION.

7. UNUSUAL FEATURES OF THE PROGRAM - THE INPUT DATA ARE ARRANGED SO THE CODE CAN BE EASILY USED FOR KEFF AND SEARCH CALCULATIONS WITHOUT BURDENING THE USER WITH BURNUP PARAMETERS.


9. STATUS (CONTINUED)

1969.
CDC6600 VERSION SUBMITTED FEBRUARY 1969.

   2DF, A TWO-DIMENSIONAL TRANSPORT CODE FROM THE LOS ALAMOS SCIENTIFIC LABORATORY (UNPUBLISHED).

11. MACHINE REQUIREMENTS - 65K MEMORY AND THREE PERIPHERAL STORAGE DEVICES

12. PROGRAMMING LANGUAGE USED - FORTRAN IV

13. OPERATING SYSTEM OR MONITOR UNDER WHICH PROGRAM IS EXECUTED -

14. ANY OTHER PROGRAMMING OR OPERATING INFORMATION OR RESTRICTIONS -

15. NAME AND ESTABLISHMENT OF AUTHORS -

   1108    W. W. LITTLE, JR. AND R. W. HARDIE
            BATTELLE-NORTHWEST LABORATORY
            P. O. BOX 999
            RICHLAND, WASHINGTON 99352

   6600    T. J. HIRONS AND R. D. ODELL
            UNIVERSITY OF CALIFORNIA
            LOS ALAMOS SCIENTIFIC LABORATORY
            P. O. BOX 1663
            LOS ALAMOS, NEW MEXICO 87544

16. MATERIAL AVAILABLE - MAGNETIC TAPE TRANSMITTAL
   SOURCE DECKS (1108-2425 CARDS, 6600-3110 CARDS)
   SAMPLE PROBLEMS (1108-63 CARDS, 6600-169 CARDS)
   REFERENCE REPORTS, BNWL-640 AND BNWL-831, AND LASL NOTE

17. CATEGORY - D
   KEYWORDS - 2-DIMENSIONAL, MULTIGROUP, REACTIVITY, CRITICALITY
              SEARCHES, DEPLETION, DIFFUSION EQUATIONS, X-Y, R-Z,
              R-THETA GEOMETRIES, DTF4 CODES, PERT4 CODES
15. NAME AND ESTABLISHMENT OF AUTHOR -
   A. H. MARCHERTAS
   REACTOR ENGINEERING DIVISION
   ARGONNE NATIONAL LABORATORY
   9700 SOUTH CASS AVENUE
   ARGONNE, ILLINOIS 60439

16. MATERIAL AVAILABLE - MAGNETIC TAPE TRANSMITTAL
   SOURCE DECK (2230 CARDS)
   SAMPLE PROBLEM (25 CARDS)
   REFERENCE REPORT AND SUPPLEMENT

17. CATEGORY - I
    KEYWORDS - STRUCTURAL ANALYSIS, STRESSES, DEFORMATION
1. NAME OR DESIGNATION OF PROGRAM - TWOTRAN

2. COMPUTER FOR WHICH PROGRAM IS DESIGNED AND OTHERS UPON WHICH IT IS OPERABLE - UNIVAC1108, CDC6600, IBM360

3. NATURE OF PHYSICAL PROBLEM SOLVED - TWOTRAN SOLVES TWO-DIMENSIONAL PARTICLE TRANSPORT PROBLEMS IN X-Y GEOMETRY. BOTH DIRECT AND ADJOINT, HOMOGENEOUS (KEFF OR PARAMETRIC EIGENVALUE SEARCHES) OR INHOMOGENEOUS TIME-INDEPENDENT PROBLEMS ARE SOLVED SUBJECT TO VACUUM, REFLECTIVE, OR INPUT SPECIFICATION OF BOUNDARY FLUX CONDITIONS. BOTH ANISOTROPIC INHOMOGENEOUS PROBLEMS AND GENERAL ANISOTROPIC SCATTERING PROBLEMS ARE TREATED.

4. METHOD OF SOLUTION - ENERGY DEPENDENCE IS TREATED BY THE MULTI-GROUP APPROXIMATION AND THE ANGULAR DEPENDENCE BY A DISCRETE ORDINATES APPROXIMATION. SPACE DEPENDENCE IS APPROXIMATED BY A POSITIVE, WEIGHTED, DIAMOND DIFFERENCE SCHEME OR (AT THE USERS OPTION) BY THE SOMewhat MORE ACCURATE (BUT NOT POSITIVE) EQUAL WEIGHT DIAMOND DIFFERENCE SCHEME. ANISOTROPIC SCATTERING AND ANISOTROPIC INHOMOGENEOUS SOURCES ARE REPRESENTED BY FINITE SPHERICAL HARMONICS EXPANSIONS. WITHIN-GROUP ITERATIONS, UPSCATTERING ITERATIONS, KEFF ITERATIONS, AND EIGENVALUE SEARCH ITERATIONS ARE ACCELERATED BY A COARSE-MESH PARTICLE REBALANCING ALGORITHM, AND, AT THE OPTION OF THE USER, WITHIN-GROUP ITERATIONS INCLUDING IMPLICIT BOUNDARY CONDITIONS ARE ACCELERATED BY AN AUTOMATIC CHEBYCHEV ACCELERATION.

5. RESTRICTIONS ON THE COMPLEXITY OF THE PROBLEM - THE VARIABLE DIMENSIONING CAPABILITY OF FORTRAN IV IS USED SO THAT ANY COMBINATION OF PROBLEM PARAMETERS LEADING TO A BLANK COMMON VECTOR LENGTH LESS THAN MAX CAN BE USED. FOR 65,536 WORD MACHINES MAX IS SLIGHTLY GREATER THAN 30,000 WORDS. WITH A FEW EXCEPTIONS, ONLY WITHIN-GROUP PROBLEM DATA ARE STORED IN FAST MEMORY AND DATA FOR ALL OTHER GROUPS ARE STORED IN AUXILIARY STORAGE. ARBITRARY NUMBERS OF GROUPS OF UP OR DOWN SCATTERING ARE ALLOWED.

6. TYPICAL RUNNING TIME - ON THE CDC6600 A SEVEN-GROUP, LINEAR ANISOTROPIC, UPSCATTERING, 40 X 40 S4 KEFF CALCULATION Requires 57.23 MINUTES. A ONE-GROUP, ISOTROPIC SCATTERING, 32 X 32 S12 INHOMOGENEOUS SOURCE CALCULATION Requires 0.54 MINUTE (PURE ABSORBER) OR 1.81 MINUTES (C=0.99). 1.26 MINUTES ARE REQUIRED BY A TWO-GROUP, LINEAR ANISOTROPIC, 10 X 10 S6 CRITICAL DIMENSION CALCULATION WITH BOTH X AND Y DIMENSIONS MODIFIED.

7. UNUSUAL FEATURES OF THE PROGRAM - UNUSUAL FEATURES INCLUDE COARSE MESH AND CHEBYCHEV CONVERGENCE ACCELERATIONS, COARSE MESH SPATIAL AND ANGULAR ORGANIZATION TO PERMIT LARGER PROBLEMS, GENERAL ANISOTROPIC SCATTERING AND INHOMOGENEOUS SOURCE OPTION, INPUT SPECIFICATION OF TOP OR RIGHT BOUNDARY FLUXES, BUILT-IN DISCRETE ORDINATES CONSTANTS (S2, S4, ...S16), AND POSITIVE SPATIAL DIFFERENCE SCHEME.
8. RELATED AND AUXILIARY PROGRAMS - TWOTRAN REPLACES THE (X,Y) GEOMETRY OPTION OF THE 2DF PROGRAM (ACC ABSTRACT 173).

   - UNIVAC1108 VERSION SUBMITTED DECEMBER 1968.
   - IBM360 VERSION SUBMITTED APRIL 1969.


11. MACHINE REQUIREMENTS - FOUR OUTPUT UNITS (DISKS, DRUMS, OR TAPES) IN ADDITION TO INPUT AND OUTPUT UNITS. A VERSION IS AVAILABLE FOR THE CDC6600 WHICH USES EXTENDED CORE STORAGE INSTEAD OF THESE FOUR UNITS AND ALSO ALLOWS BIGGER PROBLEMS BY ELIMINATING MUCH OF THE FAST MEMORY REQUIREMENT.

12. PROGRAMMING LANGUAGE USED - FORTRAN IV WITH A SMALL AMOUNT OF MIXED INTEGER-FLOATING ARITHMETIC AND GENERALIZED SUBSCRIPTING.

13. OPERATING SYSTEM OR MONITOR UNDER WHICH PROGRAM IS EXECUTED - EXEC II (UNIVAC1108), SCOPE 3.1 (CDC6600), AND OS/360 (IBM360).

14. ANY OTHER PROGRAMMING OR OPERATING INFORMATION OR RESTRICTIONS -

15. NAME AND ESTABLISHMENT OF AUTHORS -
   - UNIVAC1108  K. D. LATHROP
     LOS ALAMOS SCIENTIFIC LABORATORY
     P. O. BOX 1663
     LOS ALAMOS, NEW MEXICO 87544

   - IBM360  C. N. KELBER
     REACTOR PHYSICS DIVISION
     ARGONNE NATIONAL LABORATORY
     9700 SOUTH CASS AVENUE
     ARGONNE, ILLINOIS 60439

16. MATERIAL AVAILABLE - MAGNETIC TAPE TRANSMITTAL
   SOURCE DECKS (1108-3602 CARDS, 360-3643 CARDS)
   SAMPLE PROBLEMS (1108-290 CARDS, 360-290 CARDS)
   REFERENCE REPORTS, GA-8747 AND LA-4058

17. CATEGORY - C
    KEYWORDS - 2-DIMENSIONAL, TRANSPORT THEORY, SN METHOD, X-Y, CRITICALITY SEARCHES, FLUX DISTRIBUTION, ANISOTROPIC SCATTERING, 2DF CODES
1. NAME OR DESIGNATION OF PROGRAM - PUNI

2. COMPUTER FOR WHICH PROGRAM IS DESIGNED AND OTHERS UPON WHICH IT IS OPERABLE - CDC6600

3. NATURE OF PHYSICAL PROBLEM SOLVED - PUNI EVALUATES UNRESOLVED RADIATIVE CAPTURE INTEGRALS AND RELATED MULTIGROUP CROSS SECTIONS. THE UNRESOLVED DISTRIBUTIONS MAY HAVE VARIOUS ORBITAL ANGULAR MOMENTUM QUANTUM NUMBERS AND THE EFFECTS OF DOPPLER BROADENING AND SELF-SHIELDING ARE INCLUDED.

4. METHOD OF SOLUTION - THE PSI(THETA,X) FUNCTION IS EVALUATED ON THE INTERVAL (0,20/THETA) BY THE DFB ROUTINE DEVELOPED BY GELBARD (SEE REFERENCE 2). THE J(THETA,BETA) FUNCTION IS OBTAINED FROM THE PSI(THETA,X) FUNCTION BY TWO TRAPEZOIDAL INTEGRATIONS, FOLLOWED BY RICHARDSON EXTRAPOLATION PLUS A WING CORRECTION FOR X GREATER THAN 20/THETA. THE INTEGRAL INVOLVING CHI-SQUARE FLUCTUATION OF THE REDUCED NEUTRON WIDTH, GAMMA(NO), IS EVALUATED BY GAUSSIAN QUADRATURE. THE INTEGRATION OVER THE NEUTRON WEIGHTING SPECTRUM IS APPROXIMATED BY GAUSS QUADRATURE FOR THE 1/E CASE AND TRAPEZOIDAL INTEGRATION FOR OTHER WEIGHTING SPECTRA.

5. RESTRICTIONS ON THE COMPLEXITY OF THE PROBLEM - NO MORE THAN 20 SPIN STATES PER CASE AND 54 ENERGY GROUPS ARE ALLOWED. AN ARBITRARY ENERGY WEIGHTING SPECTRUM OF UP TO 498 POINTS MAY BE USED, BUT ALL POINTS MUST BE POSITIVE (W(E) GREATER THAN 0).

6. TYPICAL RUNNING TIME - LESS THAN 1/4 SECOND PER SPIN STATE PER GROUP IS REQUIRED.

7. UNUSUAL FEATURES OF THE PROGRAM -
(A) ANGULAR MOMENTUM QUANTUM NUMBER L = 0, 1, 2, 3, 4 PERMITTED.
(B) GAMMA(NO) PERMITTED TO FLUCTUATE WITH A CHI-SQUARE DISTRIBUTION OF UP TO 8 DEGREES OF FREEDOM FOR EACH SPIN STATE.
(C) EVALUATION OF THE INTEGRAL CONTRIBUTION FROM LEVELS WITH GAMMA(NO) LESS THAN SOME FRACTION OF THE AVERAGE PERMITTED.
(D) AN ARBITRARY GROSS ENERGY WEIGHTING SPECTRUM PERMITTED.
(E) DANCOFF CORRECTION IS COMPUTED INTERNALLY FOR EITHER HEXAGONAL OR SQUARE LATTICE ARRAYS OF CYLINDRICAL CLAD FUEL RODS.

8. RELATED AND AUXILIARY PROGRAMS -

CDC6600 VERSION SUBMITTED APRIL 1969.

10. REFERENCES - N. M. STEEN, PUN-1 - A FORTRAN-IV PROGRAM FOR THE EVALUATION OF UNRESOLVED RESONANCE INTEGRALS AND RELATED MULTIGROUP CROSS SECTIONS (LWBR DEVELOPMENT PROGRAM), WAPD-TM-768,
10. REFERENCES (CONTINUED)
   MARCH 1969, AND ERRATA.

11. MACHINE REQUIREMENTS - 40,000 (OCTAL) LOCATIONS OF CENTRAL MEMORY STORAGE ARE REQUIRED.

12. PROGRAMMING LANGUAGE USED - FORTRAN IV

13. OPERATING SYSTEM OR MONITOR UNDER WHICH PROGRAM IS EXECUTED - SCOPE 3.

14. ANY OTHER PROGRAMMING OR OPERATING INFORMATION OR RESTRICTIONS -

15. NAME AND ESTABLISHMENT OF AUTHOR -
   N. M. STEEN
   WESTINGHOUSE ELECTRIC CORPORATION
   BETTIS ATOMIC POWER LABORATORY
   P. O. BOX 79
   WEST MIFFLIN, PENNSYLVANIA 15122

16. MATERIAL AVAILABLE - RESTRICTED DISTRIBUTION
   MAGNETIC TAPE TRANSMITTAL (2 TAPES)
   SOURCE DECK (1648 CARDS)
   SAMPLE PROBLEM (38 CARDS)
   BETTIS ENVIRONMENTAL ROUTINES (21,123 CARDS)
   REFERENCE REPORTS, WAPD-TM-668, WAPD-TM-768, AND ERRATA

17. CATEGORY - A
   KEYWORDS - UNRESOLVED REGION, RESONANCE INTEGRALS, MULTIGROUP, CROSS SECTIONS, DOPPLER BROADENING, CAPTURE
1. NAME OR DESIGNATION OF PROGRAM - GLEN

2. COMPUTER FOR WHICH PROGRAM IS DESIGNED AND OTHERS UPON WHICH IT IS OPERABLE - CDC6600

3. NATURE OF PHYSICAL PROBLEM SOLVED - THE GLEN PROGRAM INTERPOLATES VALUES OF A FACTOR PROPORTIONAL TO THE SCATTERING LAW FROM THE PUNCHED OUTPUT OF THE TOR CODE (ACC ABSTRACT 360). THE DIFFERENTIAL CROSS SECTION DETERMINED FROM THESE IS INTEGRATED OVER THE SCATTERING ANGLE TO OBTAIN COEFFICIENTS OF AN EXPANSION IN LEGENDRE POLYNOMIALS OF THIS ANGLE FOR L = 0, 1, 2, 3. INTEGRATION OVER FINAL ENERGIES YIELDS VALUES OF THE TOTAL SCATTERING CROSS SECTION AND TRANSPORT CROSS SECTION. FOR EACH OF A SERIES OF ISOTOPIC COMPOSITIONS (UP TO 10 COMPOSITIONS) THE GLEN CODE CALCULATES THE DIFFUSION LENGTH AND VALUES OF THE FLUX-WEIGHTED GROUP AVERAGE MACROSCOPIC SCATTERING, ABSORPTION, FISSION, AND TRANSFER CROSS SECTIONS.

4. METHOD OF SOLUTION - THE TABLE OF VALUES ON WHICH THE DIFFERENTIAL CROSS SECTION IS BASED HAS A MESH WHICH VARIES IN INTERVAL WITH THE TWO INDEPENDENT VARIABLES. A QUADRATIC INTERPOLATION FORMULA HAS BEEN DEVELOPED FOR THIS MESH. ANGULAR INTEGRATIONS ARE BASED ON A TOTAL OF FORTY POINTS, USED IN GAUSS INTEGRATIONS OVER SUBINTERVALS OF THE ANGULAR RANGE. THE DIFFUSION LENGTH IS DETERMINED IN AN EIGENVALUE CALCULATION USING GAUSS-SEIDEL ITERATION. ENERGY SPECTRA CORRESPONDING TO THE COMPOSITION AND A SPECIFIED BUCKLING ARE CALCULATED IN THE P3 APPROXIMATION FOR DETERMINATION OF GROUP CROSS SECTIONS. ALTERNATIVELY, SPECTRA CORRESPONDING TO A MAXWELL DISTRIBUTION SCALAR FLUX, OR AN INPUT SCALAR FLUX MAY BE USED.

5. RESTRICTIONS ON THE COMPLEXITY OF THE PROBLEM - A MAXIMUM OF 87 ENERGY POINTS, AND 20 GROUPS IN THE COLLAPSED STRUCTURE ARE ALLOWED. EACH COMPOSITION IS LIMITED TO 4 HEAVY, NON-MODERATING ISOTOPES BESIDES THE MODERATOR.

6. TYPICAL RUNNING TIME - RUNNING TIME IS 3 MINUTES ON THE CDC6600 PLUS ABOUT 12 SECONDS PER COMPOSITION.

7. UNUSUAL FEATURES OF THE PROGRAM - THE DIFFUSION LENGTH IS CALCULATED IN A P11 APPROXIMATION TO THE TRANSPORT EQUATION, IN A FORMULATION WHICH MAKES POSSIBLE AN ACCURATE DETERMINATION OF THE EXISTENCE OR NON-EXISTENCE OF THE DIFFUSION LENGTH.

8. RELATED AND AUXILIARY PROGRAMS - THE TOR PROGRAM CARRIES OUT THE COMPUTATION OF THE SCATTERING LAW ON WHICH THE CROSS SECTIONS OF GLEN ARE BASED, GENERATING THE TABLE REFERRED TO ABOVE.

   CDC6600 VERSION SUBMITTED JANUARY 1969.
    W. W. CLENDENIN, SAMPLE PROBLEM FOR PROGRAM GLEN, LOS ALAMOS MEMORANDUM.

11. MACHINE REQUIREMENTS - 64K MEMORY

12. PROGRAMMING LANGUAGE USED - FORTRAN IV

13. OPERATING SYSTEM OR MONITOR UNDER WHICH PROGRAM IS EXECUTED - SCOPE.

14. ANY OTHER PROGRAMMING OR OPERATING INFORMATION OR RESTRICTIONS - LOGICAL TAPE 10 IS INPUT, LOGICAL TAPE 9 IS OUTPUT, LOGICAL TAPE 11 IS PUNCH.

15. NAME AND ESTABLISHMENT OF AUTHOR -
    PROGRAM AUTHOR
    W. W. CLENDENIN, GROUP T-1
    LOS ALAMOS SCIENTIFIC LABORATORY
    P. O. BOX 1663
    LOS ALAMOS, NEW MEXICO 87544

    PRESENT CONTACT
    RAPHAEL LABAUVE, GROUP K-1
    LOS ALAMOS SCIENTIFIC LABORATORY
    P. O. BOX 1663
    LOS ALAMOS, NEW MEXICO 87544

16. MATERIAL AVAILABLE - MAGNETIC TAPE TRANSMITTAL
    SOURCE DECK (6600-1912 CARDS)
    SAMPLE PROBLEM (6600-477 CARDS)
    REFERENCE REPORT, LA-3893, AND MEMORANDUM

17. CATEGORY - B
    KEYWORDS - GROUP CONSTANTS, SCATTERING CROSS SECTIONS, ABSORPTION, FISSION, ANGULAR DISTRIBUTION, TOR CODES
1. NAME OR DESIGNATION OF PROGRAM - WELWING

2. COMPUTER FOR WHICH PROGRAM IS DESIGNED AND OTHERS UPON WHICH IT IS OPERABLE - IBM360

3. NATURE OF PHYSICAL PROBLEM SOLVED - WELWING WAS DEVELOPED TO CALCULATE THE MATERIAL BUCKLING OF REACTOR SYSTEMS CONSISTING OF ANNULAR FUEL ELEMENTS IN HEAVY WATER AS MODERATOR FOR VARIOUS MODERATOR TO FUEL RATIOS. THE MODERATOR TO FUEL RATIO FOR THE MAXIMUM MATERIAL BUCKLING FOR THE PARTICULAR SYSTEM IS SELECTED AUTOMATICALLY AND THE CORRESPONDING MATERIAL BUCKLING IS CALCULATED.

4. METHOD OF SOLUTION - THE METHOD USED IS AN ANALYTICAL SOLUTION OF THE ONE-GROUP DIFFUSION EQUATIONS WITH VARIOUS CORRECTIONS AND APPROXIMATIONS.

5. RESTRICTIONS ON THE COMPLEXITY OF THE PROBLEM - UP TO 32 DIFFERENT MATERIALS IN THE FUEL ELEMENT MAY BE USED.

6. TYPICAL RUNNING TIME -

7. UNUSUAL FEATURES OF THE PROGRAM -

8. RELATED AND AUXILIARY PROGRAMS -


11. MACHINE REQUIREMENTS -

12. PROGRAMMING LANGUAGE USED - FORTRAN IV

13. OPERATING SYSTEM OR MONITOR UNDER WHICH PROGRAM IS EXECUTED - OS/360.

14. ANY OTHER PROGRAMMING OR OPERATING INFORMATION OR RESTRICTIONS -

15. NAME AND ESTABLISHMENT OF AUTHOR -  
   O. G. P. GROSSKOPF  
   SOUTH AFRICAN ATOMIC ENERGY BOARD  
   PRIVATE BAG 256  
   PELINDABA, PRETORIA, SOUTH AFRICA

16. MATERIAL AVAILABLE -  
   SOURCE DECK (813 CARDS)  
   SAMPLE PROBLEM (82 CARDS)
17. CATEGORY - B
KEYWORDS - BUCKLING, 1-GROUP, DIFFUSION EQUATIONS, FUEL ELEMENTS, HEAVY WATER
1. **NAME OR DESIGNATION OF PROGRAM** - BLAST

2. **COMPUTER FOR WHICH PROGRAM IS DESIGNED AND OTHERS UPON WHICH IT IS OPERABLE** - IBM360

3. **NATURE OF PHYSICAL PROBLEM SOLVED** - BLAST has been developed to study accident conditions in critical and subcritical thermal multiplying systems. The programme computes the time behaviour of the thermal neutron density and the system temperature following a step change in reactivity. The integrated thermal neutron density is also computed, from which the total number of fissions during an excursion may be obtained.

4. **METHOD OF SOLUTION** - The computational method consists of computing the neutron flux over short time intervals during which the reactivity is considered to be constant. The fact that the reactivity does change during such time intervals is duly corrected for. The temperature rise during this time interval is calculated from the average neutron flux during the time interval, and the reactivity change due to the temperature coefficients of the system is calculated. The process is then repeated.

5. **RESTRICTIONS ON THE COMPLEXITY OF THE PROBLEM** -

6. **TYPICAL RUNNING TIME** - The average time per step is about 1.25 seconds for the sample problem.

7. **UNUSUAL FEATURES OF THE PROGRAM** -

8. **RELATED AND AUXILIARY PROGRAMS** -


11. **MACHINE REQUIREMENTS** - 128K bytes of storage

12. **PROGRAMMING LANGUAGE USED** - FORTRAN IV

13. **OPERATING SYSTEM OR MONITOR UNDER WHICH PROGRAM IS EXECUTED** - OS/360.

14. **ANY OTHER PROGRAMMING OR OPERATING INFORMATION OR RESTRICTIONS** -

15. **NAME AND ESTABLISHMENT OF AUTHOR** -

   O. G. P. Grosskopf
   South African Atomic Energy Board
   Private Bag 256
   Pelindaba, Pretoria, South Africa
16. MATERIAL AVAILABLE -
    SOURCE DECK (303 CARDS)
    SAMPLE PROBLEM (15 CARDS)
    REFERENCE REPORT AND ERRATA

17. CATEGORY - E
    KEYWORDS - THERMAL, KINETICS, REACTIVITY, TEMPERATURE COEFFICIENT, EXCURSIONS
1. NAME OR DESIGNATION OF PROGRAM - SNEQ

2. COMPUTER FOR WHICH PROGRAM IS DESIGNED AND OTHERS UPON WHICH IT IS OPERABLE - CDC6600

3. NATURE OF PHYSICAL PROBLEM SOLVED - SNEQ CONSISTS OF THE TWO CODES, SNAP AND EQPLT, WHICH HAVE BEEN MERGED DUE TO THEIR COMMON USE OF THE SLIP COMPILER. SNAP INTERPRETS AND SOLVES PSEUDO-FORTRAN INPUT EQUATIONS REPRESENTING NONLINEAR ALGEBRAIC SYSTEMS. EQPLT INTERPRETS PSEUDO-FORTRAN INPUT EQUATIONS AND CALCULATES AND PLOTS MULTIPLE CURVES ON A SINGLE GRAPH. EQPLT IS USEFUL FOR PARAMETER STUDIES.

4. METHOD OF SOLUTION - SNAP USES (1) DIRECT ITERATION, (2) NEWTON-RAPHSON, (3) SECANT, AND (4) STEEPEST DESCENT METHODS TO SOLVE NONLINEAR, ALGEBRAIC EQUATIONS.

5. RESTRICTIONS ON THE COMPLEXITY OF THE PROBLEM - IN SNAP THE MAXIMUM NUMBER OF UNKNOWNS IS 50. IN EQPLT THE PSEUDO-FORTRAN EQUATION IS TWO-DIMENSIONAL, HAVING AN INDEPENDENT VARIABLE, A DEPENDENT VARIABLE AND A PARAMETER. THE PARAMETER MAY ASSUME UP TO 6 VALUES FOR ANY ONE GRAPH, 100 CONSTANT VALUES, AND UP TO 200 VALUES FOR THE INDEPENDENT VARIABLES.

6. TYPICAL RUNNING TIME - APPROXIMATELY 1 SECOND PER PROBLEM IS REQUIRED.

7. UNUSUAL FEATURES OF THE PROGRAM -

8. RELATED AND AUXILIARY PROGRAMS - SNAP USES CODEGEN TO COMPILE INPUT EQUATIONS. EQPLT USES CODEGEN TO COMPILE INPUT EQUATIONS AND THE CDC 6600 PLOTTER ROUTINES FOR PLOTTING THE CALCULATED DATA.


11. MACHINE REQUIREMENTS -

12. PROGRAMMING LANGUAGES USED - FORTRAN IV AND ASCENT

13. OPERATING SYSTEM OR MONITOR UNDER WHICH PROGRAM IS EXECUTED -

14. ANY OTHER PROGRAMMING OR OPERATING INFORMATION OR RESTRICTIONS -
15. NAME AND ESTABLISHMENT OF AUTHOR -
   G. L. LECHLITER
   GENERAL ELECTRIC COMPANY
   KNOLLS ATOMIC POWER LABORATORY
   SCHENECTADY, NEW YORK 12301

16. MATERIAL AVAILABLE - RESTRICTED DISTRIBUTION
    MAGNETIC TAPE TRANSMITTAL
    SOURCE DECK (2018 CARDS)
    SAMPLE PROBLEM (66 CARDS)
    REFERENCE REPORT AND NOTE

17. CATEGORY - P
    KEYWORDS - NONLINEAR EQUATIONS, GRAPHS, DATA PROCESSING, SNAP
                CODES, EQPLT CODES
1. NAME OR DESIGNATION OF PROGRAM - B0W2

2. COMPUTER FOR WHICH PROGRAM IS DESIGNED AND OTHERS UPON WHICH IT IS OPERABLE - CDC3600

3. NATURE OF PHYSICAL PROBLEM SOLVED - B0W2 IS USED TO CALCULATE DEFLECTIONS OF CLOSELY-SPACED PARALLEL BEAMS, EACH WITH LIMITED-PIVOT SUPPORT AT ONE END, POSSIBLE BEAM INTERACTIONS AT THE OTHER END AND AT ONE INTERMEDIATE POSITION, ASSUMING AN ARBITRARY TEMPERATURE DISTRIBUTION.

4. METHOD OF SOLUTION - B0W2 CALCULATES THE UNRESTRAINED, THERMALLY-INDUCED DEFLECTIONS OF EACH ROW OF EBR2 REACTOR SUBASSEMBLIES USING THE ASSUMED TEMPERATURE DIFFERENCES ACROSS EACH BEAM GIVEN AS INPUT. THE RESULTING DEFLECTION CURVES ARE THEN DISTORTED BY SUPERPOSITION OF STRESS-INDUCED ELASTIC DEFLECTIONS AT END AND INTERMEDIATE POSITIONS. IF THE DISTORTION-INDUCED LOAD ON ANY BEAM IS NOT COMPATIBLE WITH THE CORRESPONDING LOAD ON THE ADJACENT BEAM, THE POSITIONS OF THE ASSUMED RESTRAINT LOCATIONS ARE CHANGED AND THIS ITERATIVE PROCEDURE CONTINUES UNTIL A REASONABLE LOAD COMPATIBILITY IS ACHIEVED. TRANSVERSE ELASTIC COMPRESSION OF EACH BEAM IS INCLUDED AS REQUIRED.

5. RESTRICTIONS ON THE COMPLEXITY OF THE PROBLEM - THE INTERMEDIATE RESTRAINT POSITION MUST BE THE SAME FOR ALL BEAMS.

6. TYPICAL RUNNING TIME - THE SAMPLE PROBLEM ACCOMPANYING THE PROGRAM Requires about 1.5 MINUTES.

7. UNUSUAL FEATURES OF THE PROGRAM -

8. RELATED AND AUXILIARY PROGRAMS - THE B0W3 AND B0W4 PROGRAMS ARE MODIFICATIONS OF B0W2 TO CORRECT DEFICIENCIES IN THE ORIGINAL PROGRAM. B0W3 INCORPORATES THE EXACT RELATIONS FOR THE VARIOUS BENDING MODES AND INCLUDES AN IMPORTANT MODE IGNORED IN B0W2. B0W4 TREATS THE SUBASSEMBLY ARRAY AS A LARGE GROUP OF COUPLED SPRINGS.

   CDC3600 VERSION SUBMITTED FEBRUARY 1969.


11. MACHINE REQUIREMENTS -

12. PROGRAMMING LANGUAGE USED - 3600 FORTRAN
13. OPERATING SYSTEM OR MONITOR UNDER WHICH PROGRAM IS EXECUTED -
SCOPE.

14. ANY OTHER PROGRAMMING OR OPERATING INFORMATION OR RESTRICTIONS -

15. NAME AND ESTABLISHMENT OF AUTHOR -
   T. R. BUMP
   REACTOR ENGINEERING DIVISION
   ARGONNE NATIONAL LABORATORY
   9700 SOUTH CASS AVENUE
   ARGONNE, ILLINOIS 60439

16. MATERIAL AVAILABLE -
   SOURCE DECK (712 CARDS)
   BINARY DECK (212 CARDS)
   SAMPLE PROBLEM (84 CARDS)

17. CATEGORY - I
   KEYWORDS - TEMPERATURE, DEFORMATION, BEAMS, ELASTIC, EBR2 REACTORS
1. **NAME OR DESIGNATION OF PROGRAM** - CHEMLOC2

2. **COMPUTER FOR WHICH PROGRAM IS DESIGNED AND OTHERS UPON WHICH IT IS OPERABLE** - CDC3600

3. **NATURE OF PHYSICAL PROBLEM SOLVED** - CHEMLOC2 is used to compute the extent of core heating and metal-steam reaction following a hypothetical loss-of-coolant accident, including emergency core-cooling failure, in a water-cooled reactor. The program includes the effects of heat production by decay heating in the fuel as well as by steam-metal chemical reaction, heat transfer from the core to the bottom and top grid-support plates and to the wall surrounding the core, and methods of calculating the effects of core movement and material transfer on reaching slumping temperature.

4. **METHOD OF SOLUTION** - By means of mass and heat balances on an elemental section of a fuel rod, simultaneous iteration equations were developed for the temperature of the fuel, cladding, and gas, as well as for the thickness of cladding oxidized and the fraction of steam in the gas throughout the core. Iteration equations are included also for the temperatures of the bottom grid plate, top grid plate, and core wall. The core-slumping model is governed by input parameters including the slump factor and temperature, the fall factor, fuel-retained factor, drip and steam-block age factors.

5. **RESTRICTIONS ON THE COMPLEXITY OF THE PROBLEM** - Maxima of -
   - 25 annular axial zones, each consisting of an equal number of rods
   - 40 axial length increments

6. **TYPICAL RUNNING TIME** - Running time for the sample problem is 1 minute 57 seconds, including the time required to load the program into memory.

7. **UNUSUAL FEATURES OF THE PROGRAM** -

8. **RELATED AND AUXILIARY PROGRAMS** - NURLOC-1.0 (ACC ABSTRACT 328), ARC-II (PPC), TACT-V (GESJ), and LOCTA-R (WAPD).


11. **MACHINE REQUIREMENTS** - Minimum 3600
12. PROGRAMMING LANGUAGE USED - 3600 FORTRAN

13. OPERATING SYSTEM OR MONITOR UNDER WHICH PROGRAM IS EXECUTED - SCOPE.

14. ANY OTHER PROGRAMMING OR OPERATING INFORMATION OR RESTRICTIONS - AN ALLISON DIVISION GM COOP SUBROUTINE DISCOT IS USED TO PERFORM INTERPOLATION.

15. NAME AND ESTABLISHMENT OF AUTHORS - J. C. HESSON, J. L. ANDERSON, AND R. O. IVINS ARGONNE NATIONAL LABORATORY 9700 SOUTH CASS AVENUE ARGONNE, ILLINOIS 60439

16. MATERIAL AVAILABLE - SOURCE DECK (613 CARDS) BINARY DECK (237 CARDS) SAMPLE PROBLEM (31 CARDS) REFERENCE REPORT

17. CATEGORY - G
KEYWORDS - DECAY HEATING, WATER COOLANTS, ACCIDENTS, HEAT TRANSFER
1. NAME OR DESIGNATION OF PROGRAM - ISOGEN

2. COMPUTER FOR WHICH PROGRAM IS DESIGNED AND OTHERS UPON WHICH IT IS OPERABLE - UNIVAC1108

3. NATURE OF PHYSICAL PROBLEM SOLVED - ISOGEN CALCULATES RADIOISOTOPE GENERATION AND DECAY, USING TWO-GROUP NEUTRON CROSS SECTIONS.

4. METHOD OF SOLUTION - BATEMAN EQUATIONS MODIFIED TO PREVENT EXCESSIVE TRUNCATION ERROR AND ASSURE 0.1 PERCENT ACCURACY ARE USED AND A CHAIN TRACING TECHNIQUE WHICH ALLOWS FEEDBACK AND BRANCHING AT ANY NUCLIDE.

5. RESTRICTIONS ON THE COMPLEXITY OF THE PROBLEM - MAXIMA OF:
   - 30 TIME- STEPS INVOLVING EITHER IRRADIATION OR DECAY
   - 50 MEMBERS IN A SINGLE CHAIN
   - 500 NUCLIDES IN A SINGLE CASE

6. TYPICAL RUNNING TIME - THE TEST CASE REQUIRES ABOUT 3.5 MINUTES AND INCLUDES THORIUM IRRADIATION FOR SIX TIMES WITH 12 NUCLIDES, TWO CASES OF CALIFORNINIUM PRODUCTION BY IRRADIATING NATURAL UPA N IUM IN EIGHT TIME- STEPS WITH 39 NUCLIDES, CALCULATION OF THE FLUX REQUIRED TO ACHIEVE 30,000 MWD/TON AT 20 MW/TON IN SEVEN TIME- STEPS (FOUR IRRADIATION AND THREE REFUELING) WITH 29 NUCLIDES AND CALCULATION OF OVER 200 FISSION PRODUCT NUCLIDES AT REACTOR DISCHARGE AND FIVE COOLING TIMES.

7. UNUSUAL FEATURES OF THE PROGRAM - ACCURACY IS WITHIN 0.1 PERCENT FOR ANY TRANSMUTATION CHAIN INVOLVING DECAY AND/OR NEUTRON REACTIONS. DECAY CONSTANTS MAY BE SIMILAR OR EVEN IDENTICAL WITHOUT AFFECTING ACCURACY. NUCLIDES MAY DECAY THROUGH ANY NUMBER OF HALF LIVES WITHOUT LOSS OF ACCURACY. TWO-GROUP NEUTRON CROSS SECTIONS AND LIMITED CAPABILITY FOR HANDLING RESONANCE AND THERMAL FLUX DEPRESSION MAKE IT UNSUITED FOR DETAILED NEUTRON BALANCE CALCULATIONS.

8. RELATED AND AUXILIARY PROGRAMS -

   UNIVAC1108 VERSION SUBMITTED FEBRUARY 1969.

    H. H. VAN TUYL, SUPPLEMENTAL INFORMATION ON PROGRAM ISOGEN, BNWL NOTE.

11. MACHINE REQUIREMENTS - 65K MEMORY AND ONE TAPE VERY HELPFUL, BUT NOT ESSENTIAL.

12. PROGRAMMING LANGUAGE USED - FORTRAN V
13. OPERATING SYSTEM OR MONITOR UNDER WHICH PROGRAM IS EXECUTED - EXEC II.

14. ANY OTHER PROGRAMMING OR OPERATING INFORMATION OR RESTRICTIONS - A CALL TO SETEOF IS COMPLETELY UNNECESSARY BUT ELIMINATES AN UNNECESSARY DIAGNOSTIC ON THE 1108.

15. NAME AND ESTABLISHMENT OF AUTHOR -
H. H. VAN TUYL
BATTELLE-NORTHWEST LABORATORY
P. O. BOX 999
RICHLAND, PENNSYLVANIA 99352

16. MATERIAL AVAILABLE - MAGNETIC TAPE TRANSMITTAL
SOURCE DECK (1175 CARDS)
BINARY DECK (505 CARDS)
SAMPLE PROBLEM (71 CARDS)
LIBRARY (1512 CARDS)
REFERENCE REPORT AND NOTE

17. CATEGORY - D
KEYWORDS - 2-GROUP, ISOTOPES, PRODUCTION, DECAY, FISSION PRODUCTS, REACTION RATES
1. NAME OR DESIGNATION OF PROGRAM - FLANGE2

2. COMPUTER FOR WHICH PROGRAM IS DESIGNED AND OTHERS UPON WHICH IT IS OPERABLE - IBM360, UNIVAC1108, CDC6600

3. NATURE OF PHYSICAL PROBLEM SOLVED - FLANGE2 TAKES CROSS SECTIONS, ANGULAR DISTRIBUTION, RESONANCE PARAMETER, AND SCATTERING LAW DATA FROM ENDF/B DATA TAPES AND PREPARES THERMAL MULTIGROUP CROSS SECTIONS AND SCATTERING MATRICES.

4. METHOD OF SOLUTION - DIRECT INTEGRATION OF THE SCATTERING LAW IS USED TO OBTAIN LEGENDRE MOMENTS.

5. RESTRICTIONS ON THE COMPLEXITY OF THE PROBLEM -
   MAXIMUM ENERGY GROUPS = 200
   MAXIMUM LEGENDRE ORDER = 5

6. TYPICAL RUNNING TIME - A CROSS SECTION CALCULATION ONLY REQUIRES APPROXIMATELY 1 MINUTE, WHILE A FULL SCATTERING MATRIX PROBLEM (L=5) TAKES ABOUT 10 MINUTES.

7. UNUSUAL FEATURES OF THE PROGRAM -

8. RELATED AND AUXILIARY PROGRAMS - FLANGE1 (ACC ABSTRACT 247)

   IBM360 VERSION SUBMITTED JANUARY 1969.

10. REFERENCE - H. C. HONECK, Y. O. NALIBOFF, FLANGE-II, A CODE TO PROCESS THERMAL NEUTRON SCATTERING DATA FROM AN ENDF/B TAPE, PRELIMINARY REPORT (SECTIONS 1-5 ONLY), DECEMBER 1968.

11. MACHINE REQUIREMENTS - 32K WORDS

12. PROGRAMMING LANGUAGE USED - FORTRAN IV

13. OPERATING SYSTEM OR MONITOR UNDER WHICH PROGRAM IS EXECUTED - OS/360.

14. ANY OTHER PROGRAMMING OR OPERATING INFORMATION OR RESTRICTIONS - FLANGE2 REQUIRES FOR INPUT THE ENDF/B LIBRARY WHICH IS AVAILABLE FROM THE NATIONAL NEUTRON CROSS SECTION CENTER AT BROOKHAVEN NATIONAL LABORATORY.

15. NAME AND ESTABLISHMENT OF AUTHOR -
   H. C. HONECK
   COMPUTER APPLICATIONS DIVISION
   SAVANNAH RIVER LABORATORY
   AIKEN, SOUTH CAROLINA 29801

16. MATERIAL AVAILABLE - MAGNETIC TAPE TRANSMITTAL SOURCE DECK (360-2839 CARDS)
16. MATERIAL AVAILABLE (CONTINUED)
   SAMPLE PROBLEM (360-28 CARDS)
   REFERENCE REPORT

17. CATEGORY - A
   KEYWORDS - MULTIGROUP CROSS SECTIONS, SCATTERING LAW, ANGULAR DISTRIBUTION, RESONANCE PARAMETERS, THERMAL, FLANGE CODES
1. NAME OR DESIGNATION OF PROGRAM - RELAP2

2. COMPUTER FOR WHICH PROGRAM IS DESIGNED AND OTHERS UPON WHICH IT IS OPERABLE - CDC6600, IBM7044, UNIVAC1108


4. METHOD OF SOLUTION - THE TABULAR VALUES OF PRESSURE ARE INVESTIGATED SUCCESSIVELY, STARTING AT THE PREVIOUS POINT IN THE STEAM TABLES, UNTIL BOTH THE KNOWN DENSITY AND ENTHALPY VALUES ARE BRACKETED. WITHIN THESE LIMITS THE CALCULATED PRESSURE IS CHANGED ITERATIVELY UNTIL THE DENSITY, CALCULATED FROM MULTIPPOINT LINEAR INTERPOLATION FORMULAS, MATCHES THE KNOWN DENSITY WITHIN THE COMPUTER ACCURACY. A SUBROUTINE USING THE NEWTON-RAPHSON METHOD WAS WRITTEN WHICH CONVERGES THE FLOW EQUATIONS TO WITHIN COMPUTER ACCURACY LIMITS. NO INPUT INCREMENT IS REQUIRED AND NO FAILURES HAVE BEEN NOTED. THE MASS AND ENERGY DIFFERENTIAL EQUATIONS ARE SOLVED BY FORWARD FINITE DIFFERENCE TECHNIQUES.

5. RESTRICTIONS ON THE COMPLEXITY OF THE PROBLEM - THE PRESSURIZED WATER REACTOR (PWR) SYSTEM IS DIVIDED INTO THREE BASIC VOLUMES - (A) THE COLD LEG WHICH INCLUDES THE FLUID FROM THE HEAT EXCHANGER TO THE REACTOR INLET, (B) THE HOT LEG FROM THE REACTOR OUTLET TO THE HEAT EXCHANGER, AND (C) A SYSTEM PRESSURIZER CONNECTED TO THE HOT VOLUME. EACH VOLUME IS DEFINED AS A SIMPLE CYLINDRICAL TANK WHERE RELATIVE ENTRANCE AND EXIT JUNCTIONS ARE SPECIFIED BY THE USER. SYSTEM BREAKS INVOLVING LEAKS ARE ALLOWED IN ANY OF THE THREE VOLUMES. THE TIME-STEMS MUST BE EMPIRICALLY DETERMINED BY THE USER.

6. TYPICAL RUNNING TIME - THE APPROXIMATE SPEED ON THE IBM7044 IS 200 TIME-STEPS PER MACHINE MINUTE. THE CDC6600 IS ABOUT TEN TIMES FASTER.

7. UNUSUAL FEATURES OF THE PROGRAM - RELAP2 RETAINS MOST OF THE CALCULATIONAL FEATURES OF ITS PREDECESSORS, BUT DIFFERS MAINLY IN THE REACTOR KINETICS, REACTOR CONTROL OPTIONS, TWO-PHASE SEPARATION MODELS, PRESSURE AND FLOW SEARCH TECHNIQUES, AND INPUT/OUTPUT FORM.
8. RELATED AND AUXILIARY PROGRAMS - RELAP, FLASH (FLASH2 ACC ABSTRACT 295), AND RELAPSE


11. MACHINE REQUIREMENTS - 32K MEMORY

12. PROGRAMMING LANGUAGE USED - FORTRAN IV

13. OPERATING SYSTEM OR MONITOR UNDER WHICH PROGRAM IS EXECUTED - SCOPE (CDC6600), IBSYS (IBM7044), AND EXEC II (UNIVAC1108).

14. ANY OTHER PROGRAMMING OR OPERATING INFORMATION OR RESTRICTIONS -

15. NAME AND ESTABLISHMENT OF AUTHORS -
    K. V. MOORE AND W. H. RETTIG
    IDAHO NUCLEAR CORPORATION
    P. O. BOX 1845
    IDAHO FALLS, IDAHO 83401

16. MATERIAL AVAILABLE - MAGNETIC TAPE TRANSMITTAL
    SOURCE DECK (7044-4171 CARDS)
    SAMPLE PROBLEM (7044-106 CARDS)
    REFERENCE REPORT, IDO-17263

17. CATEGORY - G
    KEYWORDS - FLUID FLOW, TEMPERATURE DISTRIBUTION, PRESSURE DISTRIBUTION, WATER REACTORS, BLOWDOWN, REACTIVITY EXCURSIONS, RELAP CODES, RELAPSE CODES, FLASH CODES
1. NAME OR DESIGNATION OF PROGRAM - GAKIT

2. COMPUTER FOR WHICH PROGRAM IS DESIGNED AND OTHERS UPON WHICH IT IS OPERABLE - UNIVAC1108


4. METHOD OF SOLUTION - THE MULTIGROUP, ONE-DIMENSIONAL, TIME-DEPENDENT DIFFUSION THEORY KINETICS EQUATIONS ARE SOLVED BY A SEMI-IMPLICIT TIME INTEGRATION METHOD (SEE REFERENCES 1, 2, AND 3). THE HEAT TRANSFER EQUATIONS ARE SOLVED BY A QUASI TWO-DIMENSIONAL METHOD, ASSUMING THE AXIAL HEAT CONDUCTION IS NEGLIGIBLE COMPARED WITH THE HEAT TRANSPORT DUE TO COOLANT FLOW. FOR THE NUMERICAL SOLUTION OF THE KINETICS EQUATIONS AND ALSO FOR THE SOLUTION OF THE HEAT TRANSFER EQUATIONS FINITE DIFFERENCE METHODS IN SPACE AND TIME ARE EMPLOYED. THE MATERIAL PROPERTIES ARE ASSUMED TO BE REGIONWISE CONSTANT AND THE TIME-STEP SIZE IS HELD CONSTANT IN EACH TIME ZONE.

5. RESTRICTIONS ON THE COMPLEXITY OF THE PROBLEM -
   ONE-DIMENSIONAL KINETICS CALCULATION - MAXIMA OF -
   10 PROMPT NEUTRON ENERGY GROUPS
   6 DELAYED NEUTRON GROUPS
   10 HOMOGENEOUS REGIONS (CHANNELS)
   100 MESH POINTS

   FUEL ELEMENT GEOMETRY FOR HEAT TRANSFER CALCULATION - IN THE AXIAL DIRECTION THE FUEL ELEMENT MAY CONSIST OF A BOTTOM REFLECTOR, A CORE SECTION, AND A TOP REFLECTOR. THE BOTTOM AND TOP REFLECTORS MUST HAVE UNIFORM DENSITIES AND UNIFORM THERMAL PROPERTIES, WHILE WITHIN THE CORE SECTION OF THE FUEL ELEMENT UP TO 5 RADIAL REGIONS MAY BE USED. MAXIMA OF -
   15 RADIAL MESH POINTS
   20 AXIAL MESH POINTS
5. RESTRICTIONS ON THE COMPLEXITY OF THE PROBLEM (CONTINUED)  
   THE FUEL ELEMENTS IN ALL THE CHANNELS MUST HAVE THE SAME GEOMETRY, THE SAME THERMAL PROPERTIES, AND THE SAME INLET TEMPERATURE, BUT THE FUEL ELEMENTS OF DIFFERENT CHANNELS MAY HAVE DIFFERENT COOLANT FLOW RATES AND DIFFERENT AXIAL POWER SHAPES.

6. TYPICAL RUNNING TIME - FOR A TYPICAL HTGR ROD WITHDRAWAL ACCIDENT PROBLEM (2 ENERGY GROUPS, 6 DELAYED GROUPS, 49 KINETICS MESH POINTS, $180 = 18\times10$ HEAT TRANSFER MESH POINTS) THE RUNNING TIME PER TIME-STEP ON THE UNIVAC1108 IS APPROXIMATELY 0.3 SECOND. THE ROD REMOVAL WAS COMPLETED IN 180 SECONDS AND THE SOLUTION WAS CALCULATED FOR 300 SECONDS TAKING 20 MINUTES OF UNIVAC1108 COMPUTER TIME.

7. UNUSUAL FEATURES OF THE PROGRAM -  
   (A) THE TIME INTEGRATION PROCEDURE IS NUMERICALLY STABLE FOR RELATIVELY LARGE TIME-STEPS. THE PROCEDURE IS ASYMPTOTICALLY STABLE FOR TIME-STEPS OF ARBITRARY LENGTH.  
   (B) THE METHOD IS ESPECIALLY SUITED FOR MULTIGROUP CALCULATIONS BECAUSE THE RUNNING TIME INCREASES ROUGHLY LINEARLY WITH THE NUMBER OF ENERGY GROUPS.  
   (C) THE CODE HAS THE CAPABILITY OF CALCULATING STEADY-STATE CONDITIONS FOR THE COMBINED DIFFUSION AND HEAT TRANSFER EQUATIONS ASSUMING A CERTAIN POWER LEVEL.  
   (D) THE LIMITS FOR THE NUMBER OF ENERGY GROUPS, MESH POINTS, ETC. CAN EASILY BE CHANGED BY RECOMPILING THE FORTRAN PROGRAM.  
   (E) THE INPUT OF THE CODE IS FLEXIBLE. FOR EXAMPLE, THE USER CAN SUPPLY HIS TEMPERATURE-DEPENDENT CROSS SECTIONS IN TABLE-FORM, OR HE CAN GENERATE TRANSIENTS BY PIECEWISE LINEAR TIME-DEPENDENT CROSS SECTIONS.

8. RELATED AND AUXILIARY PROGRAMS - GAKIN, A ONE-DIMENSIONAL KINETICS CODE WITHOUT TEMPERATURE FEEDBACK (ACC ABSTRACT 310)


11. MACHINE REQUIREMENTS - 65,536 WORDS OF CORE STORAGE AND THE FACILITY FOR SEGMENTING PROGRAMS
12. PROGRAMMING LANGUAGE USED - FORTRAN IV

13. OPERATING SYSTEM OR MONITOR UNDER WHICH PROGRAM IS EXECUTED - EXEC-II, GAX 23, BUT ANY OTHER SYSTEM WITH A SEGMENTING FACILITY CAN BE USED.

14. ANY OTHER PROGRAMMING OR OPERATING INFORMATION OR RESTRICTIONS -

15. NAME AND ESTABLISHMENT OF AUTHORS -
   R. FROEHLICH, S. R. JOHNSON, AND M. H. MERRILL
   GULF GENERAL ATOMIC INCORPORATED
   P. O. BOX 608
   SAN DIEGO, CALIFORNIA 92112

16. MATERIAL AVAILABLE - MAGNETIC TAPE TRANSMITTAL
   SOURCE DECK (7455 CARDS)
   SAMPLE PROBLEM (95 CARDS)
   REFERENCE REPORT, GA-8576

17. CATEGORY - F
   KEYWORDS - 1-DIMENSIONAL, MULTIGROUP, DIFFUSION, KINETICS, TEMPERATURE FEEDBACK, 2-DIMENSIONAL, R-Z, HEAT TRANSFER, FUEL ELEMENTS, GAKIN CODES
1. NAME OR DESIGNATION OF PROGRAM - NOWIG

2. COMPUTER FOR WHICH PROGRAM IS DESIGNED AND OTHERS UPON WHICH IT IS OPERABLE - CDC6600

3. NATURE OF PHYSICAL PROBLEM SOLVED - NOWIG IS A PROGRAM FOR SOLVING THE ONE-DIMENSIONAL TWO-GROUP NEUTRON DIFFUSION AND DELAYED PRECURSOR EQUATIONS USING A SHAPE-SPECIFIED POINT KINETICS APPROXIMATION. FEEDBACK DUE TO CHANGES IN THE FUEL METAL TEMPERATURE AND COOLANT DENSITY IS ACCOUNTED FOR BY USING A MODEL WHICH IS IDENTICAL WITH THAT USED IN THE WIGL2 (ACC ABSTRACT 274) PROGRAM.

4. METHOD OF SOLUTION - THE APPROPRIATE KINETICS AND TEMPERATURE EQUATIONS ARE SOLVED BY A SPECIFIED THETA TIME DIFFERENCING TECHNIQUE.

5. RESTRICTIONS ON THE COMPLEXITY OF THE PROBLEM - 2 ENERGY GROUPS MUST BE USED AND UP TO 6 GROUPS OF DELAYED NEUTRON PRECURSORS MAY BE USED. THE MAXIMUM NUMBER OF SPATIAL MESH POINTS IS 250 AND UP TO 20 MATERIAL COMPOSITIONS AND 50 THERMALHYDRAULIC CHANNELS MAY BE SPECIFIED.

6. TYPICAL RUNNING TIME - A TRANSIENT PROBLEM WITH 31 SPATIAL MESH POINTS, 7 THERMALHYDRAULIC CHANNELS, 1 DELAYED GROUP, AND 2000 TIME-SECONDS REQUIRES 39 SECONDS ON A CDC6600.


8. RELATED AND AUXILIARY PROGRAMS - THE INITIAL CONDITIONS REQUIRED FOR A NOWIG PROBLEM MAY BE GENERATED BY USE OF THE STEADY-STATE OPTION OF THE WIGL3 PROGRAM. THE WANDA PROGRAM MAY ALSO BE USED TO GENERATE THE INITIAL NOWIG SHAPE FUNCTIONS AND ADJOINT WEIGHT FUNCTIONS.

CDC6600 VERSION SUBMITTED MARCH 1969.

A. F. HENRY, N. J. CURLEE, JR., A. V. VOTA, WIGL3 -
10. REFERENCES (CONTINUED)
A PROGRAM FOR THE STEADY-STATE AND TRANSIENT SOLUTION OF THE ONE-
DIMENSIONAL, TWO-GROUP, SPACE-TIME DIFFUSION EQUATIONS ACCOUNTING
FOR TEMPERATURE, XENON, AND CONTROL FEEDBACK, WAPD-TM-788,
FEBRUARY 1969.
O. J. MARLOWE, M. G. SUGGS, WANDA-5, A ONE-DIMENSIONAL
NEUTRON DIFFUSION PROGRAM FOR THE PHILCO-2000 COMPUTER,
C. J. PFEIFER, CDC-6600 FORTRAN PROGRAMMING - BETTIS

11. MACHINE REQUIREMENTS -

12. PROGRAMMING LANGUAGE USED - FORTRAN IV

13. OPERATING SYSTEM OR MONITOR UNDER WHICH PROGRAM IS EXECUTED -
SCOPE 3.1.

14. ANY OTHER PROGRAMMING OR OPERATING INFORMATION OR RESTRICTIONS -

15. NAME AND ESTABLISHMENT OF AUTHOR -
J. B. YASINSKY
WESTINGHOUSE ELECTRIC CORPORATION
BETTIS ATOMIC POWER LABORATORY
P. O. BOX 79
WEST MIFFLIN, PENNSYLVANIA 15122

16. MATERIAL AVAILABLE - RESTRICTED DISTRIBUTION
MAGNETIC TAPE TRANSMITTAL
SOURCE DECK (923 CARDS)
SAMPLE PROBLEM (35 CARDS)
BETTIS ENVIRONMENTAL ROUTINES (21,123 CARDS)
REFERENCE REPORTS, WAPD-TM-806 AND WAPD-TM-668

17. CATEGORY - F
KEYWORDS - 1-DIMENSIONAL, 2-GROUP, DIFFUSION, KINETICS, TEMPERATURE FEEDBACK, WIGL2 CODES
1. NAME OR DESIGNATION OF PROGRAM - RAPFU

2. COMPUTER FOR WHICH PROGRAM IS DESIGNED AND OTHERS UPON WHICH
   IT IS OPERABLE - GE635

3. NATURE OF PHYSICAL PROBLEM SOLVED - RAPFU CALCULATES EQUILIBRIUM
   FUEL CYCLE ISOTOPICS IN FAST BREEDER REACTORS. THE RECYCLED PLU-
   TONIUM IS PERMITTED TO HAVE DIFFERENT ISOTOPIC COMPOSITIONS IN
   TWO DIFFERENT CORE ZONES, AND SEVERAL RECYCLE SCHEMES ARE AVAIL-
   ABLE AS OPTIONS. OUTPUT DATA INCLUDES THE INITIAL, AVERAGE, AND
   DISCHARGED FUEL ISOTOPIC CONCENTRATIONS FOR EACH REGION OF THE
   CORE ZONES AND THE BLANKETS, BREEDING RATIO, DOUBLING TIME, AND
   (OPTIONALLY) FUEL COSTS CALCULATED USING SIMPLIFIED RELATIONSHIPS.

4. METHOD OF SOLUTION -

5. RESTRICTIONS ON THE COMPLEXITY OF THE PROBLEM - FOUR PLUTONIUM
   RECYCLE SCHEMES ARE AVAILABLE. THE MOST COMPLICATED OF THESE USES
   TWO CORE ISOTOPIC ZONES PLUS A ZONE CONTAINING THE BLANKET
   ZONES. THE OTHER THREE USE ONLY A SINGLE CORE ISOTOPIC ZONE AND
   ALLOW A SELECTION TO BE MADE FROM THREE ALTERNATE MIXTURES IN THE
   PLUTONIUM DISCHARGED FROM THE CORE AND BLANKET REGIONS AND RECY-
   CLED BACK INTO THE CORE REGIONS. UP TO 100 REGIONS OF DIFFERENT
   COMPOSITION MAY BE USED, AND THE REGIONS WITHIN ANY ZONE OF FIXED
   PLUTONIUM INPUT ISOTOPIC COMPOSITION DO NOT HAVE TO BE CONNECTED.

6. TYPICAL RUNNING TIME - FOR A TYPICAL CASE USING FIVE CORE AND
   BLANKET REGIONS AND THE MOST COMPLICATED RECYCLE SCHEME, THE COM-
   PUTATION CONVERGED AFTER 12 ITERATIONS AND TOTAL RUNNING TIME WAS
   LESS THAN ONE MINUTE.

7. UNUSUAL FEATURES OF THE PROGRAM -

8. RELATED AND AUXILIARY PROGRAMS - RAPFU IS INTENDED TO BE USED WITH
   A SEPARATE MULTIGROUP DIFFUSION THEORY CODE SUCH AS CRAM (ACC
   ABSTRACT 103) OR BISYN (ACC ABSTRACT 287).

   GE635 VERSION SUBMITTED MARCH 1969.

10. REFERENCES - P. GREEBLER AND C. L. COWAN, RAPFU, A COMPUTER CODE
    FOR RAPID CALCULATION OF FUEL CYCLE PHYSICS PARAMETERS IN FAST
    BREEDER REACTORS WITH VARIABLE FISSILE RECYCLE SCHEMES, GEAP-5494,
    MAY 1967.
    A. HASSITT, A COMPUTER PROGRAM TO SOLVE THE MULTI-
    GROUP DIFFUSION EQUATIONS, TRG REPORT 229(R), MARCH 1962.
    P. GREEBLER, M. D. KELLEY, R. A. DAVIS, C. A. KECK,
    AND W. A. DUNCAN, BISYN - A TWO DIMENSIONAL SYNTHESIS PROGRAM,
    GEAP-4922, JULY 1965.

11. MACHINE REQUIREMENTS -
12. PROGRAMMING LANGUAGE USED - FORTRAN IV

13. OPERATING SYSTEM OR MONITOR UNDER WHICH PROGRAM IS EXECUTED -

14. ANY OTHER PROGRAMMING OR OPERATING INFORMATION OR RESTRICTIONS -

15. NAME AND ESTABLISHMENT OF AUTHORS -
    P. GREEBLER AND C. COWAN
    GENERAL ELECTRIC COMPANY
    ADVANCED PRODUCTS OPERATION
    SUNNYVALE, CALIFORNIA

16. MATERIAL AVAILABLE -
    SOURCE DECK (1091 CARDS)
    SAMPLE PROBLEM (60 CARDS)
    REFERENCE REPORT, GEAP-5494

17. CATEGORY - D
    KEYWORDS - FUEL CYCLE, FAST REACTORS, BREEDING, ISOTOPES, ECONOMICS, CRAM CODES, BISYN CODES
1. NAME OR DESIGNATION OF PROGRAM - BL47

2. COMPUTER FOR WHICH PROGRAM IS DESIGNED AND OTHERS UPON WHICH IT IS OPERABLE - CDC6600

3. NATURE OF PHYSICAL PROBLEM SOLVED - BL47 IS A PLOTTING ROUTINE DESIGNED FOR PLANE STRUCTURES THAT ARE TO UNDERGO STRESS ANALYSIS. POINTS AND LINES ARE INPUT IN VARIOUS PARAMETRIC FORMS, AND CURVED SEGMENTS ARE DRAWN BETWEEN GIVEN POINTS ALONG THE GIVEN LINES. THE PROGRAM MAY BE USED AS A DRAFTING TOOL TO CONSTRUCT ENGINEERING DRAWINGS. BL47 USES THREE POINTS ON A STRAIGHT LINE SEGMENT TO OBTAIN DIMENSIONS FOR SEAL-SHELL2 (ACC ABSTRACT 282) INPUT DATA.

4. METHOD OF SOLUTION - FROM GIVEN DATA, BL47 SETS UP PARAMETRIC DATA FOR POINTS, STRAIGHT LINES, PARABOLAS, AND ELLIPSSES.

5. RESTRICTIONS ON THE COMPLEXITY OF THE PROBLEM - MAXIMA OF -
   1000 POINTS
   1000 LINES
   1000 WORDS

6. TYPICAL RUNNING TIME - 15 SECONDS ARE REQUIRED.

7. UNUSUAL FEATURES OF THE PROGRAM - A POINT IS LOCATED BY INTERSECTION OF TWO LINES, BY THE NORMAL FROM A POINT TO A LINE, OR BY ROTATED COORDINATES FROM A GIVEN POINT. A LINE IS LOCATED BY PARAMETERS IN A ROTATED COORDINATE SYSTEM.

8. RELATED AND AUXILIARY PROGRAMS - SEAL-SHELL2 (ACC ABSTRACT 282) AND A PLOT ROUTINE DISCUSSED IN REFERENCE 2.


11. MACHINE REQUIREMENTS - 20,000 WORDS OF HIGH-SPEED MEMORY

12. PROGRAMMING LANGUAGE USED - FORTRAN IV

13. OPERATING SYSTEM OR MONITOR UNDER WHICH PROGRAM IS EXECUTED - SCOPE.

14. ANY OTHER PROGRAMMING OR OPERATING INFORMATION OR RESTRICTIONS -
15. NAME AND ESTABLISHMENT OF AUTHORS -
C. M. FRIEDRICH AND J. J. CULLENS
WESTINGHOUSE ELECTRIC CORPORATION
BETTIS ATOMIC POWER LABORATORY
P. O. BOX 79
WEST MIFFLIN, PENNSYLVANIA 15122

16. MATERIAL AVAILABLE - RESTRICTED DISTRIBUTION
MAGNETIC TAPE TRANSMITTAL
SOURCE DECK (938 CARDS)
SAMPLE PROBLEM (189 CARDS)
BETTIS ENVIRONMENTAL ROUTINES (21,123 CARDS)
REFERENCE REPORTS, WAPD-TM-783 AND WAPD-TM-668

17. CATEGORY - L
KEYWORDS - GRAPHS, STRESSES, DESIGN, GEOMETRIES, INPUT DATA,
PARAMETERS, SEAL SHELL CODES
1. NAME OR DESIGNATION OF PROGRAM - IDX

2. COMPUTER FOR WHICH PROGRAM IS DESIGNED AND OTHERS UPON WHICH IT IS OPERABLE - UNIVAC1108

3. NATURE OF PHYSICAL PROBLEM SOLVED - IDX IS A MULTIPURPOSE, ONE-DIMENSIONAL DIFFUSION CODE FOR GENERATING CROSS SECTIONS TO BE USED IN FAST REACTOR ANALYSES. THE CODE IS DESIGNED TO -
   (A) COMPUTE AND PUNCH RESONANCE SHIELDED CROSS SECTIONS USING DATA IN THE RUSSIAN (SEE REFERENCE 2) FORMAT,
   (B) COMPUTE AND PUNCH GROUP-COLLAPSED MICROSCOPIC AND/OR MACROSCOPIC CROSS SECTIONS AVERAGED OVER THE SPECTRUM IN ANY SPECIFIED ZONE, AND
   (C) COMPUTE KEFF AND PERFORM CRITICALITY SEARCHES ON TIME ABSORPTION, MATERIAL CONCENTRATIONS, ZONE DIMENSIONS, AND BUCKLING USING EITHER A FLUX OR AN ADJOINT MODEL.

4. METHOD OF SOLUTION - RESONANCE SHIELDED CROSS SECTIONS ARE CALCULATED USING DATA (INFINITE DILUTION CROSS SECTIONS AND RESONANCE SHIELDING FACTORS) IN THE RUSSIAN FORMAT. INTERPOLATION SCHEMES ARE USED TO COMPUTE SHIELDING FACTORS APPLICABLE TO SPECIFIC COMPOSITIONS. A FLUX ITERATION OPTION FOR COMPUTING THE ELASTIC REMOVAL CROSS SECTION IS ALSO INCLUDED. GROUP-COLLAPSED CROSS SECTIONS BY REACTOR ZONE ARE CALCULATED USING FLUX WEIGHTING. THE EIGENVALUE AND SPATIAL FLUX PROFILES ARE COMPUTED USING STANDARD SOURCE-ITERATION TECHNIQUES. CONVERGENCE IS ACCELERATED USING FISSION SOURCE OVERRELAXATION. IDX WILL ALSO ACCEPT INPUT CROSS SECTIONS IN THE DTF FORMAT (I.E., RESONANCE SHIELDED CROSS SECTIONS).

5. RESTRICTIONS ON THE COMPLEXITY OF THE PROBLEM - IDX USES VARIABLE DIMENSIONING TO MAKE MAXIMUM USE OF EXISTING CORE STORAGE. ALL SUBSCRIPTED VARIABLES ARE STORED IN ONE ARRAY DIMENSIONED TO 35,000.

6. TYPICAL RUNNING TIME - A REPRESENTATIVE 26-GROUP PROBLEM WITH 30 SPATIAL INTERVALS USING DATA IN THE RUSSIAN FORMAT REQUIRES ABOUT 50 SECONDS ON A UNIVAC1108. IF THE CROSS SECTION DATA IS ALREADY IN THE DTF FORMAT, THE SAME PROBLEM WOULD REQUIRE ABOUT 5 SECONDS.

7. UNUSUAL FEATURES OF THE PROGRAM -

UNIVAC 1108 VERSION SUBMITTED MAY 1969.

10. REFERENCES -
I. I. BONDARENKO ET AL., GROUP CONSTANTS FOR NUCLEAR REACTOR CALCULATIONS, CONSULTANTS BUREAU, NEW YORK, 1964.
2DF, A TWO-DIMENSIONAL TRANSPORT CODE FROM THE LOS ALAMOS SCIENTIFIC LABORATORY (UNPUBLISHED).

11. MACHINE REQUIREMENTS - 65K MEMORY AND 7 PERIPHERAL STORAGE DEVICES

12. PROGRAMMING LANGUAGE USED - FORTRAN IV

13. OPERATING SYSTEM OR MONITOR UNDER WHICH PROGRAM IS EXECUTED -

14. ANY OTHER PROGRAMMING OR OPERATING INFORMATION OR RESTRICTIONS -

15. NAME AND ESTABLISHMENT OF AUTHOR -
R. W. HARDIE
BATT E LLE-NORTHWEST LABORATORY
P. O. BOX 999
RICHLAND, WASHINGTON 99352

16. MATERIAL AVAILABLE - MAGNETIC TAPE TRANSMITTAL
SOURCE DECKS (IDX 2373 CARDS, PUPX 212 CARDS)
SAMPLE PROBLEM (IDX 435 CARDS)
REFERENCE REPORT, BNWL-954

17. CATEGORY - B
KEYWORDS - 1-DIMENSIONAL, DIFFUSION, CROSS SECTIONS, FAST REACTORS, CRITICALITY SEARCHES, SHIELDING FACTORS
1. NAME OR DESIGNATION OF PROGRAM - SCORE2

2. COMPUTER FOR WHICH PROGRAM IS DESIGNED AND OTHERS UPON WHICH IT IS OPERABLE - IBM360

3. NATURE OF PHYSICAL PROBLEM SOLVED - SCORE IS AN INTERACTIVE NEUTRON CROSS SECTION EVALUATION SYSTEM.

4. METHOD OF SOLUTION - EXPERIMENTAL NEUTRON CROSS SECTION DATA CAN BE RETRIEVED AND DISPLAYED ON AN ACTIVE GRAPHICS CONSOLE. MANY BOOKKEEPING OPERATIONS MAY BE INITIATED BY OPTION SELECTION OF THE CONSOLE. EVALUATED DATA ANALYSIS MODULES, PERMITTING THE PRODUCTION OF EVALUATED DATA CURVES OR RESONANCE PARAMETERS, ARE AVAILABLE. THE EVALUATED DATA CURVE MODULE INCLUDES LINEAR, CUBIC SPLINE, OR LEAST SQUARES CUBIC SPLINE (SEE REFERENCE ?) CURVE GENERATION. THE RESONANCE ANALYSIS MODULE PERMITS SINGLE-LEVEL OR REICH-MOORE MULTILEVEL ANALYSIS.

5. RESTRICTIONS ON THE COMPLEXITY OF THE PROBLEM - SCORE CAN DISPLAY UP TO 500 EXPERIMENTAL DATA POINTS WITH THEIR ASSOCIATED ERROR BARS. UP TO 2 SMOOTH CURVES MAY BE OVERLAYED ON A DISPLAY OF EXPERIMENTAL DATA POINTS. EACH CURVE IS RESTRICTED TO 150 POINTS.

6. TYPICAL RUNNING TIME - APPROXIMATELY 6 TO 12 MINUTES PER HOUR AT THE CONSOLE ARE REQUIRED ON AN IBM360/50.

7. UNUSUAL FEATURES OF THE PROGRAM - SCORE EMPLOYS INTERACTIVE GRAPHICS FOR CONTROL OF EXECUTION PATHS.

8. RELATED AND AUXILIARY PROGRAMS - TWO PROGRAMS ARE NEEDED TO PRODUCE THE REQUIRED DATA LIBRARIES. SAP IS USED TO CONSTRUCT EXPERIMENTAL DATA LIBRARIES FROM SCISRS. SCORF CONSTRUCTS EVALUATED DATA LIBRARIES FROM ENDF/B.


    H. C. HONECK, SPECIFICATIONS FOR AN EVALUATED NUCLEAR DATA FILE FOR REACTOR APPLICATIONS, ENDF/B, BNL-50066 (T-467), JULY 1967.
    PROGRAM TAPE CONTENT, AI NOTE.
11. MACHINE REQUIREMENTS - IBM360 MODEL 50 OR HIGHER WITH 180,000 BYTES OF FAST MEMORY, AN IBM2250 GRAPHICS CONSOLE, MODELS 1, 2, OR 3, 2 9-TRACK TAPE DRIVES, AND 1 DISK

12. PROGRAMMING LANGUAGES USED - FORTRAN IV AND ASSEMBLY LANGUAGE

13. OPERATING SYSTEM OR MONITOR UNDER WHICH PROGRAM IS EXECUTED - OS/360, VERSION 13 OR HIGHER.

14. ANY OTHER PROGRAMMING OR OPERATING INFORMATION OR RESTRICTIONS -

15. NAME AND ESTABLISHMENT OF AUTHOR -
   C. L. DUNFORD
   ATOMICS INTERNATIONAL
   P. O. BOX 309
   CANOGA PARK, CALIFORNIA 91304

16. MATERIAL AVAILABLE - MAGNETIC TAPE TRANSMITTAL
   SOURCE DECKS
   LIBRARIES
   REFERENCE REPORT, AI-AEC-12757, AND AI NOTE

17. CATEGORY - M
   KEYWORDS - CROSS SECTIONS, RESONANCE PARAMETERS, GRAPHS, ENDF/B, SCISRS, LIBRARIES, RETRIEVAL, MEASUREMENTS
1. NAME OR DESIGNATION OF PROGRAM - AVERAGE

2. COMPUTER FOR WHICH PROGRAM IS DESIGNED AND OTHERS UPON WHICH IT IS OPERABLE - CDC6600, IBM7094

3. NATURE OF PHYSICAL PROBLEM SOLVED - AVERAGE CALCULATES AVERAGE SCATTERING, CAPTURE, AND FISSION CROSS SECTIONS FROM S- AND P-WAVE DATA OF THE UNRESOLVED PARAMETERS OF FILE 2 OF ENDF/B.

4. METHOD OF SOLUTION - THE THEORY OF AVERAGE CROSS SECTION DUE TO LANE AND LYNN IS USED.

5. RESTRICTIONS ON THE COMPLEXITY OF THE PROBLEM - AVERAGE CALCULATES AVERAGE CROSS SECTIONS FOR UP TO 100 ENERGY VALUES IN THE UNRESOLVED REGION. IT DOES NOT ALLOW FOR INELASTIC SCATTERING AND CALCULATES ONLY S- AND P-WAVE CONTRIBUTIONS. WIDTH-FLUCTUATION CORRECTIONS CAN VARY FROM 1 TO 4.

6. TYPICAL RUNNING TIME - A TEST RUN CALCULATING S- AND P-WAVE CONTRIBUTIONS TO AVERAGE SCATTERING, CAPTURE, AND FISSION CROSS SECTIONS OF Pu-239 AT 16 ENERGY POINTS AND S- AND P-WAVE CONTRIBUTIONS TO AVERAGE SCATTERING AT CAPTURE CROSS SECTION OF U-238 AT 100 ENERGY POINTS REQUIRES 5 SECONDS ON THE CDC6600.

7. UNUSUAL FEATURES OF THE PROGRAM - AVERAGE CAN ALLOW FOR DIFFERENT DEGREES OF FREEDOM FOR FISSION WIDTH DISTRIBUTION OF RESONANCES OF DIFFERENT SPINS.

8. RELATED AND AUXILIARY PROGRAMS -

   CDC6600 VERSION SUBMITTED MAY 1969.
   IBM7094 VERSION SUBMITTED MAY 1969.


11. MACHINE REQUIREMENTS - 18K OCTAL MEMORY

12. PROGRAMMING LANGUAGE USED - FORTRAN IV

13. OPERATING SYSTEM OR MONITOR UNDER WHICH PROGRAM IS EXECUTED - SCOPE 2.0 (CDC6600) AND IBSYS (IBM7094).

14. ANY OTHER PROGRAMMING OR OPERATING INFORMATION OR RESTRICTIONS -

15. NAME AND ESTABLISHMENT OF AUTHOR -
   M. R. BHAT
   NATIONAL NEUTRON CROSS SECTION CENTER
15. NAME AND ESTABLISHMENT OF AUTHOR(S) (CONTINUED)
BROOKHAVEN NATIONAL LABORATORY
UPTON, LONG ISLAND, NEW YORK 11073

16. MATERIAL AVAILABLE -
SOURCE DECKS (6600-441 CARDS, 7094-536 CARDS)
SAMPLE PROBLEMS (6600-94 CARDS, 7094-42 CARDS)
REFERENCE NOTE

17. CATEGORY - A
KEYWORDS - AVERAGES, UNRESOLVED REGION, RESONANCE PARAMETERS,
SCATTERING, CAPTURE, FISSION, CROSS SECTIONS
1. NAME OR DESIGNATION OF PROGRAM - SIGPLOT

2. COMPUTER FOR WHICH PROGRAM IS DESIGNED AND OTHERS UPON WHICH IT IS OPERABLE - CDC6600, IBM7094

3. NATURE OF PHYSICAL PROBLEM SOLVED - SIGPLOT CALCULATES THE SCATTERING, CAPTURE, FISSION, AND TOTAL CROSS SECTIONS FROM RESOLVED RESONANCE PARAMETER DATA. SCATTERING CROSS SECTIONS MAY BE CALCULATED WITH OR WITHOUT LEVEL-LEVEL INTERFERENCE. PROVISION IS ALSO MADE TO NUMERICALLY DOPPLER-BROADEN ANY OF THE CROSS SECTIONS.

4. METHOD OF SOLUTION - THE MULTILEVEL BREIT-WIGNER FORMULA IS USED (SEE REFERENCE 2).

5. RESTRICTIONS ON THE COMPLEXITY OF THE PROBLEM - SIGPLOT CAN HANDLE RESONANCE DATA UP TO A MAXIMUM OF 10 DIFFERENT ISOTOPES WITH A TOTAL NUMBER OF 500 RESONANCES AND AN L VALUE NOT EXCEEDING 5. IT FURTHER ASSUMES THAT THE RESOLVED RESONANCE PARAMETERS ARE GIVEN FOR ONE ENERGY RANGE WHICH IS THE SAME FOR ALL THE ISOTOPES OF AN ELEMENT. THE MESH POINTS AT WHICH THE CROSS SECTIONS ARE CALCULATED CAN BE VARIED. SINCE THE CALCULATED DATA ARE NOT STORED, AN INCREASE IN THE NUMBER OF MESH POINTS DOES NOT CONFLICT WITH ANY STORAGE REQUIREMENTS.


7. UNUSUAL FEATURES OF THE PROGRAM -
   (A) IN CALCULATING SCATTERING CROSS SECTIONS WITH LEVEL-LEVEL INTERFERENCE, RESONANCES ARE GROUPED ACCORDING TO THEIR SPINS FOR THE SAME L VALUE. FOR EXAMPLE, THE SPINS ARE 3 AND 4. IT IS POSSIBLE THAT THERE ARE A NUMBER OF RESONANCES WHOSE SPINS HAVE NOT BEEN MEASURED, AND THESE ARE GIVEN AN AVERAGE SPIN OF 3.5. THE PROGRAM NORMALLY CALCULATES SCATTERING CROSS SECTIONS WITH LEVEL-LEVEL INTERFERENCE AMONG THE SPIN-3 GROUP AND THE SPIN-4 GROUP. CONTRIBUTIONS OF THE SPIN-3.5 RESONANCES ARE CALCULATED AS A SUM OF SINGLE-LEVEL BREIT-WIGNER TERMS. IF, HOWEVER, IT IS DESIRED TO INCLUDE THE LEVEL-LEVEL INTERFERENCE TERMS AMONG THESE RESONANCES, THIS MAY BE INDICATED ON THE CONTROL CARD.
   (B) THE METHOD OF NUMERICAL INTEGRATION USED TO DOPPLER-BROADEN CROSS SECTIONS IS VERY GENERAL AND IS DESIGNED TO TAKE CARE OF SITUATIONS WHERE THE FINE STRUCTURE OF THE CROSS SECTION IS RAPIDLY VARYING.

8. RELATED AND AUXILIARY PROGRAMS - SIGPLOT HAS BEEN PATTERNED AFTER THE MLBW PROGRAM (ENE A ABSTRACT 076) WITH CERTAIN CORRECTIONS AND CHANGES IN THE SUBROUTINES ORDER, FACTS, AND SIGMA.
   CDC6600 VERSION SUBMITTED MAY 1969.
   IBM7094 VERSION SUBMITTED MAY 1969.

10. REFERENCES - PROGRAM SIGPLOT, BNL NOTE.
    K. GREGSON, M. F. JAMES, AND D. S. NORTON, MLBW - A

11. MACHINE REQUIREMENTS - 37K OCTAL MEMORY

12. PROGRAMMING LANGUAGE USED - FORTRAN IV

13. OPERATING SYSTEM OR MONITOR UNDER WHICH PROGRAM IS EXECUTED -
    SCOPE 2.0 (CDC6600) AND IBSYS (IBM7094).

14. ANY OTHER PROGRAMMING OR OPERATING INFORMATION OR RESTRICTIONS -

15. NAME AND ESTABLISHMENT OF AUTHORS -
    M. R. BHAT AND D. CULLEN
    NATIONAL NEUTRON CROSS SECTION CENTER
    BROOKHAVEN NATIONAL LABORATORY
    UPTON, LONG ISLAND, NEW YORK 11073

16. MATERIAL AVAILABLE -
    SOURCE DECKS (6600-1021 CARDS, 7094-1087 CARDS)
    SAMPLE PROBLEMS (6600-48 CARDS, 7094-49 CARDS)
    REFERENCE NOTE

17. CATEGORY - A
    KEYWORDS - SCATTERING, CAPTURE, FISSION, CROSS SECTIONS, RESOLVED
    REGION, RESONANCE PARAMETERS, DOPPLER BROADENING, MULTILEVEL, BREIT-WIGNER FORMULA
1. NAME OR DESIGNATION OF PROGRAM - TUBE

2. COMPUTER FOR WHICH PROGRAM IS DESIGNED AND OTHERS UPON WHICH IT IS OPERABLE - CDC6600

3. NATURE OF PHYSICAL PROBLEM SOLVED - TUBE SOLVES FOR THE STRESSES DUE TO PRESSURE AND TEMPERATURE IN A U-TUBE TYPE HEAT EXCHANGER. SPECIFICALLY, IT HANDLES A CONFIGURATION CONSISTING OF A SPHERICAL HEAD, PRIMARY TRANSITION CYLINDER, AND SECONDARY CYLINDER. THE TRANSITION CYLINDERS MAY BE CONICAL AND TAPERED IN THICKNESS, BUT THE REMAINING SHELLS ARE OF UNIFORM THICKNESS. THE SPHERICAL HEAD AND THE TRANSITION CYLINDERS MAY BE OMITTED FROM A PROBLEM.


5. RESTRICTIONS ON THE COMPLEXITY OF THE PROBLEM -

6. TYPICAL RUNNING TIME - THE AVERAGE RUNNING TIME ON THE CDC6600 IS APPROXIMATELY 5 SECONDS PER CASE.

7. UNUSUAL FEATURES OF THE PROGRAM -

8. RELATED AND AUXILIARY PROGRAMS -


11. MACHINE REQUIREMENTS -

12. PROGRAMMING LANGUAGE USED - FORTRAN IV

13. OPERATING SYSTEM OR MONITOR UNDER WHICH PROGRAM IS EXECUTED -

14. ANY OTHER PROGRAMMING OR OPERATING INFORMATION OR RESTRICTIONS -

15. NAME AND ESTABLISHMENT OF AUTHORS -
A. E. SPIVAK AND G. L. LECHLITER
GENERAL ELECTRIC COMPANY
15. NAME AND ESTABLISHMENT OF AUTHOR(S) (CONTINUED)
KNOLLS ATOMIC POWER LABORATORY
SCHENECTADY, NEW YORK 12301

16. MATERIAL AVAILABLE - RESTRICTED DISTRIBUTION
SOURCE DECK (1011 CARDS)
SAMPLE PROBLEM (20 CARDS)
REFERENCE REPORT

17. CATEGORY - I
KEYWORDS - STRESSES, STRUCTURAL ANALYSIS, DESIGN, PRESSURE, TEMPERATURE, CYLINDERS
1. NAME OR DESIGNATION OF PROGRAM - SAFE-2D

2. COMPUTER FOR WHICH PROGRAM IS DESIGNED AND OTHERS UPON WHICH IT IS OPERABLE - UNIVAC1108

3. NATURE OF PHYSICAL PROBLEM SOLVED - SAFE-2D PERFORMS THE ELASTIC STRESS ANALYSIS OF GENERAL AXISYMMETRIC, PLANE, AND COMBINED AXISYMMETRIC AND PLANE COMPOSITE STRUCTURES.

4. METHOD OF SOLUTION - THE FINITE ELEMENT VARIATIONAL METHOD IS USED. EQUILIBRIUM EQUATIONS ARE SOLVED BY BLOCK OR TRI-DIAGONALIZATION OF THE STIFFNESS MATRICES.

5. RESTRICTIONS ON THE COMPLEXITY OF THE PROBLEM - MAXIMA OF
   2 DEGREES OF FREEDOM (RADIAL OR HORIZONTAL TRANSLATION AND AXIAL OR VERTICAL TRANSLATION)
   1200 NODAL POINTS
   2400 FINITE ELEMENTS

6. TYPICAL RUNNING TIME - 8 MINUTES ARE REQUIRED FOR A FULL CAPACITY PROBLEM.

7. UNUSUAL FEATURES OF THE PROGRAM -

8. RELATED AND AUXILIARY PROGRAMS - SAFE-PCRS (ACC ABSTRACT 250), SAFE-3D (ACC ABSTRACT 332), AND SAFE-PLANE (ACC ABSTRACT 252). IN MOST CASES SAFE-PLANE IS TO BE REPLACED BY SAFE-2D. SAFE-2D HAS LARGER CAPACITIES AND OFFERS A MORE DESIRABLE TECHNIQUE FOR SOLUTION OF THE EQUATIONS. HOWEVER, SAFE-2D DOES IMPOSE A FEW MORE RESTRICTIONS ON THE INPUT DATA AND DOES NOT HANDLE POLAR COORDINATE SYSTEMS.

   UNIVAC1108 VERSION SUBMITTED MAY 1969.


11. MACHINE REQUIREMENTS - SAFE-2D REQUIRES 50K, FAST MEMORY STORAGE, AND 500,000 WORDS OF AUXILIARY STORAGE.

12. PROGRAMMING LANGUAGE USED - FORTRAN IV

13. OPERATING SYSTEM OR MONITOR UNDER WHICH PROGRAM IS EXECUTED - EXEC II.

14. ANY OTHER PROGRAMMING OR OPERATING INFORMATION OR RESTRICTIONS - SAFE-2D CONSISTS OF TWO OVERLAYS.
15. NAME AND ESTABLISHMENT OF AUTHOR -
D. C. CORNELL
GULF GENERAL ATOMIC INCORPORATED
P. O. BOX 608
SAN DIEGO, CALIFORNIA 92112

16. MATERIAL AVAILABLE -
SOURCE DECK (1902 CARDS)
SAMPLE PROBLEM (48 CARDS)
REFERENCE REPORT, GA-9076

17. CATEGORY - I
KEYWORDS - ELASTIC, STRESS, FINITE-ELEMENT, STRUCTURAL ANALYSIS, SAFE CODES
NAME OR DESIGNATION OF PROGRAM - GATT

COMPUTER FOR WHICH PROGRAM IS DESIGNED AND OTHERS UPON WHICH IT IS OPERABLE - UNIVAC1108

NATURE OF PHYSICAL PROBLEM SOLVED - GATT IS A THREE-DIMENSIONAL FEW-GROUP NEUTRON DIFFUSION THEORY PROGRAM FOR CALCULATING THE DETAILED SPATIAL FLUX AND POWER DISTRIBUTION FOR REACTORS WITH HEXAGONAL CORE CONFIGURATION. THE PROGRAM USES A UNIFORM TRIANGULAR MESH IN THE HORIZONTAL MESH PLANES AND ASSUMES A RELATIVELY SIMPLE REGION STRUCTURE IN THE AXIAL DIRECTION. IT WAS DESIGNED TO REPRESENT THE SPECIAL PATCH-TYPE CORE STRUCTURE OF THE HTGR REACTOR AS CLOSELY AS POSSIBLE.

METHOD OF SOLUTION - GATT MAKES USE OF THE DIRECT BLOCK INVERSION TECHNIQUE FOR MULTIDIMENSIONAL REACTOR CALCULATIONS (SEE REFERENCE 3). THIS METHOD LEADS TO AN IMPROVED CONVERGENCE OF THE INNER SPATIAL FLUX ITERATION. THE OUTER FISSION SOURCE ITERATION IS ACCELERATED BY AN ASYMPTOTIC SOURCE EXTRAPOLATION WHICH IS APPLIED PERIODICALLY TO DAMP OUT THE NEXT HIGHER EIGENMODE.

RESTRICTIONS ON THE COMPLEXITY OF THE PROBLEM - GATT ALLOWS DOWNSCATTERING FROM ONE GROUP INTO THE NEXT AND A MAXIMA OF -
- 81 Z-PLANES
- 104,487 MESH POINTS
- 4 ENERGY GROUPS
- 40 MACROSCOPIC CROSS SECTION SETS (MIXTURES)
IT IS ASSUMED THAT ALL FISSION NEUTRONS ARE BORN IN GROUP 1.

TYPICAL RUNNING TIME - A FOUR-GROUP PROBLEM WITH 42,000 MESH POINTS, REPRESENTING A 330 MW(E) HTGR REACTOR, WITH FOUR FULLY AND THREE PARTIALLY-INSERTED CONTROL RODS REQUIRES 90 MINUTES.

UNUSUAL FEATURES OF THE PROGRAM -

RELATED AND AUXILIARY PROGRAMS - GAUGE (ACC ABSTRACT 339) IS THE TWO-DIMENSIONAL COUNTERPART OF GATT.

UNIVAC1108 VERSION SUBMITTED MAY 1969.

11. MACHINE REQUIREMENTS - 65K MEMORY AND 10 FILES OF DRUM STORAGE WITH A TOTAL OF 500,000 WORDS CAPACITY

12. PROGRAMMING LANGUAGE USED - FORTRAN IV

13. OPERATING SYSTEM OR MONITOR UNDER WHICH PROGRAM IS EXECUTED - EXEC II.

14. ANY OTHER PROGRAMMING OR OPERATING INFORMATION OR RESTRICTIONS - IT IS NOT POSSIBLE TO REPLACE THE DRUMS BY TAPE UNITS.

15. NAME AND ESTABLISHMENT OF AUTHORS -
   H. KRAETSCH AND M. R. WAGNER
   GULF GENERAL ATOMIC INCORPORATED
   P. O. BOX 608
   SAN DIEGO, CALIFORNIA 92112

16. MATERIAL AVAILABLE - MAGNETIC TAPE TRANSMITTAL
   SOURCE DECK (3203 CARDS)
   SAMPLE PROBLEM (39 CARDS)
   REFERENCE REPORT, GA-8547

17. CATEGORY - C
   KEYWORDS - 3-DIMENSIONAL, FEW-GROUP, DIFFUSION, FLUX DISTRIBUTION, POWER DISTRIBUTION, HEXAGONAL, HTGR REACTORS
1. NAME OR DESIGNATION OF PROGRAM - LYNNE

2. COMPUTER FOR WHICH PROGRAM IS DESIGNED AND OTHERS UPON WHICH IT IS OPERABLE - IBM7090


4. METHOD OF SOLUTION - THE ROMBERG METHOD OF INTEGRATION IS USED IN LYNNE.

5. RESTRICTIONS ON THE COMPLEXITY OF THE PROBLEM -

6. TYPICAL RUNNING TIME -

7. UNUSUAL FEATURES OF THE PROGRAM -

8. RELATED AND AUXILIARY PROGRAMS - LYNNE IS WRITTEN TO BE COMPATIBLE WITH THE CODE ATHENA WHICH CALCULATES MICROSCOPIC FORM FACTORS USING SINGLE PARTICLE WAVE FUNCTIONS.


11. MACHINE REQUIREMENTS -

12. PROGRAMMING LANGUAGE USED - FORTRAN II

13. OPERATING SYSTEM OR MONITOR UNDER WHICH PROGRAM IS EXECUTED -

14. ANY OTHER PROGRAMMING OR OPERATING INFORMATION OR RESTRICTIONS - ALTHOUGH LYNNE IS WRITTEN SPECIFICALLY FOR THE WOODS-SAXON POTENTIAL, THE CODE COULD BE MODIFIED TO ACCEPT OTHER POTENTIALS.

15. NAME AND ESTABLISHMENT OF AUTHOR - M. B. JOHNSON CENTER FOR THEORETICAL PHYSICS MASSACHUSETTS INSTITUTE OF TECHNOLOGY CAMBRIDGE, MASSACHUSETTS 02139

16. MATERIAL AVAILABLE - SOURCE DECK (452 CARDS) REFERENCE REPORT

17. CATEGORY - A
KEYWORDS - INELASTIC SCATTERING, OPTICAL MODEL, CROSS SECTIONS, ATHENA CODES
1. NAME OR DESIGNATION OF PROGRAM - RAPP

2. COMPUTER FOR WHICH PROGRAM IS DESIGNED AND OTHERS UPON WHICH IT IS OPERABLE - CDC6600

3. NATURE OF PHYSICAL PROBLEM SOLVED - RAPP COMPUTES THE RELATIONSHIP AMONG MASS FLOW, PRESSURE, AND PIPING RESISTANCE (K-FACTOR) FOR HIGH VELOCITY FLOW OF A TWO-PHASE MIXTURE OF STEAM AND WATER. THE SOURCE FLUID MAY BE SUBCOOLED OR SATURATED WATER, SATURATED STEAM, OR A MIXTURE OF STEAM AND WATER. THE DOWNSTREAM PRESSURE MUST BE BELOW THE SATURATION PRESSURE OF THE SOURCE FLUID. SPECIFIC APPLICATIONS INCLUDE PRESSURIZER SURGE LINE PRESSURE DROP AND PRESSURE DISTRIBUTION DOWNSTREAM OF A RELIEF VALVE.

4. METHOD OF SOLUTION - THE PROGRAM USES THE FIRST AND SECOND LAWS OF THERMODYNAMICS TO EITHER COMPUTE MASS FLOW WHEN PRESSURES AND RESISTANCES ARE SPECIFIED FOR A CONSTANT-AREA FLOW PATH OR COMPUTE PRESSURE DISTRIBUTION IN A PIPING RUN WHEN FLOW IS SPECIFIED. MAJOR ASSUMPTIONS ARE THAT THE DIFFERENTIAL LOSS OF AVAILABLE ENERGY IS PROPORTIONAL TO KINETIC ENERGY AND DIFFERENTIAL INCREASE IN A GEOMETRIC (K) FACTOR, THAT SLIP RATIO IS PROPORTIONAL TO SPECIFIC VOLUME RATIO RAISED TO A CONSTANT (INPUT) POWER, AND THAT THERMODYNAMIC EQUILIBRIUM IS PRESENT.

5. RESTRICTIONS ON THE COMPLEXITY OF THE PROBLEM - MAXIMA OF -
   6 FLOW AREAS FOR COMPUTING PRESSURE DISTRIBUTION WITH KNOWN FLOW
   2000 PRESSURE INCREMENTS

6. TYPICAL RUNNING TIME - .1 TO 5 SECONDS PER CASE ARE REQUIRED.

7. UNUSUAL FEATURES OF THE PROGRAM -

8. RELATED AND AUXILIARY PROGRAMS -

   CDC6600 VERSION SUBMITTED JUNE 1969.


11. MACHINE REQUIREMENTS - 45K TO COMPILE AND RUN

12. PROGRAMMING LANGUAGE USED - FORTRAN IV

13. OPERATING SYSTEM OR MONITOR UNDER WHICH PROGRAM IS EXECUTED -

14. ANY OTHER PROGRAMMING OR OPERATING INFORMATION OR RESTRICTIONS -
    RAPP CONSISTS OF THE MAIN PROGRAM WHICH PERFORMS ALL FLOW CALCULATIONS AND TWO SUBROUTINES, STEAMR, WHICH COMPUTES STEAM AND WATER PROPERTIES, AND LINTPW, A GENERAL INTERPOLATION ROUTINE.
15. NAME AND ESTABLISHMENT OF AUTHOR -
C. W. SORENSON
GENERAL ELECTRIC COMPANY
KNOLLS ATOMIC POWER LABORATORY
SCHENECTADY, NEW YORK 12301

16. MATERIAL AVAILABLE - RESTRICTED DISTRIBUTION
SOURCE DECK (480 CARDS)
SAMPLE PROBLEM (11 CARDS)
REFERENCE REPORT

17. CATEGORY - H
KEYWORDS - FLUID FLOW, PRESSURE, LIQUIDS, VAPORS, THERMODYNAMICS,
WATER, COOLANTS, PIPES
1. NAME OR DESIGNATION OF PROGRAM - M0266

2. COMPUTER FOR WHICH PROGRAM IS DESIGNED AND OTHERS UPON WHICH IT IS OPERABLE - CDC6600

3. NATURE OF PHYSICAL PROBLEM SOLVED - M0266 COMPUTES THE DYNAMIC SHOCK FORCES AND MODAL FREQUENCIES ACTING ON A LUMPED MASS, LINEAR ELASTIC MODEL OF A STRUCTURE SUBJECTED TO SHOCK SPECTRUM INPUTS. THE MODEL EMPLOYED IS A COLLECTION OF LUMPED MASSES CONNECTED BY WEIGHTLESS FLEXIBLE ELEMENTS. IF THE ORIGINAL STRUCTURE IS NOT STATICALLY DETERMINATE, REDUNDANT FORCES MUST BE INTRODUCED TO ENSURE A PRIMARY STRUCTURE THAT IS.

4. METHOD OF SOLUTION - THE CONNECTION AND FLEXIBILITY MATRICES WHICH ARE INPUT TO THE PROGRAM BY THE USER ARE CONVERTED INTO A FINAL FLEXIBILITY MATRIX USING CTAC MATRIX MANIPULATION. THE MODAL FREQUENCIES ARE THEN CALCULATED BY MEANS OF THE MATVEC SUBROUTINE. A DYNAMIC ANALYSIS MAY THEN BE PERFORMED GIVEN THE RESPONSE SPECTRUM USING THE NORMAL MODE-SPECTRUM METHOD.

5. RESTRICTIONS ON THE COMPLEXITY OF THE PROBLEM - M0266 IS LIMITED TO A 50 X 50 MASS-FLEXIBILITY MATRIX. OTHER MAXIMA INCLUDE -
   50 MASSES PLUS REDUNDANTS
   20 REDUNDANTS
   999 FLEXIBLE ELEMENTS
   20 EFFECTS PER ELEMENT

   THE DYNAMIC ANALYSIS IS LIMITED TO A SPECTRUM-TYPE ANALYSIS, WHICH IS LIMITED TO A ONE-DIRECTIONAL MOTION OF THE FOUNDATION.

6. TYPICAL RUNNING TIME - LARGE PROBLEMS MAY HAVE RUNNING TIMES AS LONG AS 1.5 MINUTES. HOWEVER, MOST PROBLEMS WILL RUN IN LESS THAN 1 MINUTE.

7. UNUSUAL FEATURES OF THE PROGRAM -

8. RELATED AND AUXILIARY PROGRAMS - M0552 (ACC ABSTRACT 283), M0753, MADAR, AND GEM (ACC ABSTRACT 344).


    L. W. EHRlich, EIGENVALUES AND EIGENVECTORS OF COMPLEX NON-HERMITIAN MATRICES USING THE DIRECT AND INVERSE POWER METHODS AND MATRIX DEFLATION, THE UNIVERSITY OF TEXAS COMPUTATION
10. REFERENCES (CONTINUED)
CENTER, UTF4-01, AUGUST 2, 1961.
C. M. FRIEDRICH, STRUCTURE DESIGN NOTES - CALCULATION
OF STRESSES, FORCES, AND DEFLECTIONS IN LINEAR-ELASTIC SKELETON
STRUCTURES WITH TEMPERATURE, PRESSURES, APPLIED LOADS, AND REDUN-
J. L. BITNER, DEVELOPING FLEXIBILITY MATRICES USING

11. MACHINE REQUIREMENTS - 105,000 OCTAL LOCATIONS, INPUT TAPE, OUTPUT
TAPE, ONE SCRATCH TAPE, AND PRINTER

12. PROGRAMMING LANGUAGE USED - FORTRAN IV

13. OPERATING SYSTEM OR MONITOR UNDER WHICH PROGRAM IS EXECUTED -
SCOPE 3.1.

14. ANY OTHER PROGRAMMING OR OPERATING INFORMATION OR RESTRICTIONS -

15. NAME AND ESTABLISHMENT OF AUTHOR -
W. A. WENZEL
WESTINGHOUSE ELECTRIC CORPORATION
BETTIS ATOMIC POWER LABORATORY
P. O. BOX 79
WEST MIFFLIN, PENNSYLVANIA 15122

16. MATERIAL AVAILABLE - RESTRICTED DISTRIBUTION
MAGNETIC TAPE TRANSMITTAL
SOURCE DECK (946 CARDS)
SAMPLE PROBLEM (30 CARDS)
BETTIS ENVIRONMENTAL ROUTINES (21,123 CARDS)
REFERENCE REPORTS, WAPD-TM-739(L) AND WAPD-TM-668

17. CATEGORY - I
KEYWORDS - STRUCTURAL ANALYSIS, FREQUENCY, STRESSES, BEAMS, ELAS-
TIC, DESIGN
1. NAME OR DESIGNATION OF PROGRAM - CHECKER/CRECT/DAMMET/PLTTFB/SLAVE3

2. COMPUTER FOR WHICH PROGRAM IS DESIGNED AND OTHERS UPON WHICH IT IS OPERABLE - CDC6600

3. NATURE OF PHYSICAL PROBLEM SOLVED - THIS PACKAGE OF FIVE PROGRAMS IS DESIGNED FOR PROCESSING ENDF/B (EVALUATED NUCLEAR DATA FILE VERSION B) TAPES.

   CHECKER CHECKS THAT THE ENDF/B TAPES ARE IN PROPER FORMAT AND ALL FIELDS ARE WITHIN SPECIFIED LIMITS, RATHER THAN THE PHYSICS OF THE DATA LIBRARY. ANGULAR DISTRIBUTIONS RECONSTRUCTED FROM LEGENDRE COEFFICIENTS ARE EVERYWHERE POSITIVE.

   CRECT PROVIDES A MEANS OF CORRECTING ASSEMBLED DATA ON A TAPE BY INSERTION AND DELETION OF DATA.

   DAMMET SELECTIVELY MERGES DATA FROM ONE OR TWO ENDF/B LIBRARY TAPES ONTO A FINAL TAPE. THE MODE (BCD OR BINARY) AND ARRANGEMENT (STANDARD OR ALTERNATE) MAY BE CHANGED DURING THIS PROCESS.

   PLTTFB PROCESSES ENDF/B LIBRARY TAPES WHICH CONTAIN DATA EMBEDDED WITHIN A NECESSARY LIBRARY STRUCTURE IN ORDER TO PRODUCE COMPREHENSIVE LISTINGS AND/OR PLOTS. THE LISTINGS AND/OR PLOTS CONTAIN AN EXTENSIVE AMOUNT OF INFORMATION RELATED TO THE DATA, SUCH AS TEMPERATURE DEPENDENCE, PHYSICAL UNITS OF THE DATA, INTERPOLATION LAWS FOR THE DATA, CRYPTIC TITLES DEFINING THE REACTION TYPE, ETC.

   SLAVE3 PROVIDES MODULAR SUBROUTINES WHICH CAN BE ASSEMBLED TO RETRIEVE AND PROCESS ENDF/B DATA FOR A SPECIFIC PROBLEM.

4. METHOD OF SOLUTION - IN CHECKER MOST DATA ARE SUBMITTED TO RANGE LIMIT CHECKS TO INSURE THE STRUCTURE AGREES WITH ENDF/B-DEFINED SPECIFICATIONS. THE DEVIANT POINT CHECK (DETECTION OF MISPIUNCHED VALUES) IS ACCOMPLISHED BY SEQUENTIALLY CHECKING FIVE DATA POINTS, TWO POINTS ON EITHER SIDE OF THE POINT UNDER SCRUTINY, FORWARD EXTRAPOLATION ON THE FIRST TWO POINTS, REVERSE EXTRAPOLATION ON THE FOURTH AND FIFTH, AND INTERPOLATIONS BETWEEN THE SECOND AND FOURTH POINTS. IF THE THIRD POINT DOES NOT FALL WITHIN THE RANGE OF THE VALUES CALCULATED BY EXTRAPOLATIONS AND INTERPOLATION, THE VALUE IS LISTED AS DEVIANT. IN ESSENCE, THE ABOVE PROCEDURE MERE-LY SIMULATES A CHECKING OF GRAPhICAL DATA BY EYE. CHECKING OF LEGENDRE COEFFICIENTS FOR FEASIBILITY AND POSITIVITY IS ACCOM- PLISHED BY CONVERTING THE COEFFICIENTS TO MOMENTS, CHECKING WHETHER MOMENTS ARE PHYSICALLY POSSIBLE AND TESTING WHETHER THE LEGENDRE EXPANSION IS NEGATIVE WITHIN THE INTERVAL -1 TO 1 BY THE STURM SEQUENCE METHOD. IF THE LEGENDRE EXPANSION IS NEGATIVE, IT IS DEFINED OVER THE NEGATIVE REGION(S) AND INTEGRATED, THUS DETER- MINING THE TOTAL NEGATIVE PROBABILITY.

   CRECT ACCEPTS A DATA TAPE TO BE CORRECTED AS INPUT AND LOCATES AREAS TO BE CORRECTED BY SEQUENCE AND MATERIAL NUMBERS. A CORRECTED TAPE IS THEN GENERATED REFLECTING THE INSERTIONS AND DELE- TIONS.

   WHEN Merging TWO TAPES AND CONVERTING TO A DATA FORMAT OTHER THAN THAT OF EITHER INPUT TAPE, DAMMET PERFORMS A TWO-PASS OPERA-
4. **METHOD OF SOLUTION (CONTINUED)**

The first pass consists of selecting only the data required and converting it to the format of the final tape, with storage of the data on an intermediate tape. Upon completion of the first pass on both tapes, resulting in two intermediate tapes, the second pass merges both intermediate tapes to the final data tape.

In PLOTFB the specified structure of the library allows data to be subclassified under nuclides, classes of data, and reaction types. The well-defined structure permits the associated information to be accessed by converting material numbers to Hollerith titles, etc. Plots are generated utilizing a plotting package which includes scaling, grid, and normalization routines. Plots can be generated in either linear-log axis or a mixture of both.

5. **RESTRICTIONS ON THE COMPLEXITY OF THE PROBLEM**

In Checker correct structure of the data tape is assumed and minor anomalies are noted. Common and major irregularities cause termination of execution accompanied by a core dump.

In Crect correct sequence and material fields of the data tape are assumed. In both Checker and Crect data must be in ENDF/B BCD standard arrangement card image format.

In DAMMET a specific reaction cannot be extracted from a tape unless it is within a fuel or material selected.

In PLOTFB data must be in either the ENDF/B BCD or binary standard arrangement card image format. PLOTFB can extract data only to the material and file level of data within the ENDF library.

6. **TYPICAL RUNNING TIME**

- **.007 second** of CPU time is required to process one record of BCD information in Checker. Typical CPU to PP time ratio for the short form output is 1 to 4, for long form output, 1 to 6.
- In Crect **2 seconds** of CPU time are required. Typical CPU to PP time ratio is 1 to 5.
- In DAMMET **3 to 7 minutes** of CPU time are required to process one nuclide. Typical CPU to PP time ratio is approximately 3 to 1.
- In PLOTFB 4 minutes of CPU time and 10 minutes of PP time are required to list and plot 35 materials (approximately 3500 data cards).

In Slave3 running time is dependent on the selection of subroutines and processing requirements. Typical CPU to PP time ratio is 1 to 3, and an average run is 4 minutes.

7. **UNUSUAL FEATURES OF THE PROGRAM**

In Checker there are two output options - (A) long form, which lists each card image as well as the anomalies encountered, and (B) short form, which lists only the anomalies encountered by the program (minimum output option).

8. **RELATED AND AUXILIARY PROGRAMS**

All five programs perform an independent operation on ENDF/B data. Crect may be used to correct errors detected by Checker.
   CDC6600 VERSION SUBMITTED APRIL 1969.

10. REFERENCES - NATIONAL NEUTRON CROSS SECTION CENTER, DESCRIPTION OF
    THE ENDF/B PROCESSING CODES CHECKER, CRECT, DAMMET, PLOTFB, AND
    RETRIEVAL SUBROUTINES, ENDF-110, BNL-13582, SEPTEMBER 1967,
    REVISED APRIL 1969.

11. MACHINE REQUIREMENTS - FOR CHECKER 56,300 OCTAL (APPROXIMATELY 24,000 DECIMAL) WORDS OF CORE STORAGE AND ONE TAPE (OR DISC) UNIT (BESIDES INPUT AND OUTPUT UNITS).
    FOR CRECT 13,000 OCTAL WORDS OF CORE STORAGE AND TWO DATA STORAGE UNITS.
    FOR DAMMET 61,000 OCTAL (ABOUT 25,000 DECIMAL) WORDS OF CORE STORAGE AND A MAXIMUM OF FIVE TAPE (OR DISC) UNITS.
    FOR PLOTFB 73,700 OCTAL (ABOUT 30,000 DECIMAL) WORDS OF CORE STORAGE AND THREE TAPE (OR DISC) UNITS.
    FOR SLAVE3 40,000 OCTAL (APPROXIMATELY 18,000 DECIMAL) WORDS OF CORE STORAGE AND ONE OR TWO TAPE (OR DISC) UNITS WILL BE REQUIRED FOR A TYPICAL JOB.

12. PROGRAMMING LANGUAGE USED - FORTRAN IV

13. OPERATING SYSTEM OR MONITOR UNDER WHICH PROGRAM IS EXECUTED - SCOPE3 (LOCALLY MODIFIED).

14. ANY OTHER PROGRAMMING OR OPERATING INFORMATION OR RESTRICTIONS - WHEN USING NON-CDC6600 VERSIONS OF FORTRAN IV THE IMPLEMENTATION OF AND/OR OPERATIONS AND EOF TESTING OPERATIONS SHOULD BE CHECKED.

15. NAME AND ESTABLISHMENT OF AUTHORS -
    D. E. CULLEN, WILLIAM KROPP, S. PEARLSTEIN
    NATIONAL NEUTRON CROSS SECTION CENTER
    BROOKHAVEN NATIONAL LABORATORY
    UPTON, LONG ISLAND, NEW YORK 11973

16. MATERIAL AVAILABLE - MAGNETIC TAPE TRANSMITTAL
    SOURCE DECKS (CHECKER 2355 CARDS, CRECT 260 CARDS, DAMMET 1857 CARDS, PLOTFB 4037 CARDS, SLAVE3 2245 CARDS)
    REFERENCE REPORT, BNL-13582

17. CATEGORY - M
    KEYWORDS - DATA PROCESSING, CROSS SECTIONS, GRAPHS, INPUT DATA, LIBRARIES, MAINTENANCE, RETRIEVAL
VI. LIBRARY PROGRAMS BY CLASSIFICATION
A. CROSS SECTION AND RESONANCE INTEGRAL CALCULATIONS

14 HAEEVER

HAEEVER CALCULATES THE ENERGY EXCHANGE INELASTIC SCATTERING CROSS SECTION ACCORDING TO THE HAUSER-FESHBACH THEORY MODIFIED TO INCLUDE THE EFFECT OF SPIN-ORBIT COUPLING. THE CROSS SECTION IS INTEGRATED OVER ANGLE.

41 ZUT

ZUT COMPUTES RESONANCE INTEGRALS FROM RESONANCE PARAMETERS FOR A WIDE VARIETY OF TEMPERATURES, COMPOSITIONS, AND GEOMETRIES FOR THE RESOLVED RESONANCES.

42 TYZ

TYZ COMPUTES RESONANCE INTEGRALS FOR A WIDE VARIETY OF TEMPERATURES, COMPOSITIONS, AND GEOMETRIES FOR THE UNRESOLVED RESONANCES.

56 SUMMIT

THE PROGRAM EVALUATES THE DIFFERENTIAL ENERGY-TRANSFER CROSS SECTION FOR SCATTERING BY A CRYSTALLINE MODERATOR, UTILIZING THE SO-CALLED PHONON EXPANSION. THE SCATTERING KERNEL FOR A 1-PHONON CHANGE IN ENERGY IS ADDED TO THAT FOR A 2-PHONON ENERGY EXCHANGE, AND SO ON. THIS PROGRAM HAS BEEN USED TO DETERMINE SCATTERING MATRICES FOR BERYLLIUM, GRAPHITE, AND OXYGEN.

\[
\frac{\sigma(E(0) \rightarrow E)}{\sigma(0)} = \left(\frac{M+1}{M}\right) \cdot 2 \cdot \sqrt{\frac{E}{E(0)}} \cdot \frac{1}{2}
\]

THE INTEGRAL FROM -1 TO 1 OF \( \sigma(E(0) \rightarrow E, \cos(\theta)) \) \( \cos(\theta) \)

WHERE E(0) AND E ARE THE INITIAL AND FINAL ENERGIES, Theta IS THE ANGLE OF SCATTERING, SIGMA(0) THE FREE-ATOM CROSS SECTION, AND M THE RATIO OF THE MASS OF THE SCATTERING NUCLEUS TO THAT OF THE NEUTRON.

89 ARES2

ARES2 IS USED TO CALCULATE EFFECTIVE RESONANCE INTEGRALS AND MULTIGROUP CROSS SECTIONS FOR LUMPS AND MIXTURES USING RESONANCE PARAMETERS. IT COMBINES IN A SINGLE CODE, THE RESOLVED, UNRESOLVED, AND 1/V PARTS OF THE CALCULATION WHICH WERE PREVIOUSLY IN SEPARATE CODES. THE MULTIGROUP CROSS SECTIONS ARE PRINTED IN A FORM FOR USE IN MULTIGROUP REACTOR CALCULATIONS. FOR EXAMPLE, THEY CAN BE INSERTED INTO THE AIM6 (ACC ABSTRACT 29) OR FAIM (ACC ABSTRACT 120) LIBRARIES. THE ENERGY BREAKPOINTS ARE INPUT DATA AND A RESONANCE PARAMETER LIBRARY IS INCLUDED.

171 NEARREX

NEARREX COMPUTES NEUTRON-INDUCED, AVERAGE FLUCTUATION (OR COMPOUND NUCLEUS) CROSS SECTIONS. PROVISION IS MADE FOR THE COMPUTATION OF COMPOUND ELASTIC AND INELASTIC NEUTRON CROSS SECTIONS, RADIATIVE CAPTURE AND FISSION CROSS SECTIONS, AS WELL AS OTHER PROCESSES, SUCH AS PROTON EMISSION. IT CAN ALSO BE USED TO COMPUTE PROTON-INDUCED AVERAGE CROSS SECTIONS.
This program computes resonance integrals, averages them over resonance parametric distributions, and computes fission and capture cross sections as a function of fuel temperature and of potential scattering cross section per absorber isotope.

Dopie was developed to study the effects of:

1. Overlap of resonances of unlike fuel isotopes such as U238 and the plutonium isotopes, and
2. The temperature dependence of the average flux over an energy interval containing resonances in the calculation of the Doppler coefficient.

Dopie computes flux-averaged cross sections over a given energy range under the following assumptions:

1. Interference scattering corrections will be small and can be ignored.
2. Scattered neutrons are distributed uniformly over the range of energy degradation constant for each material.
3. All resonances are resolved or can be treated as resolved.
4. Leakage is insignificant, making sigma R(J) = potential scattering.

The program permits computing and convoluting of a combination cross section curve composed of the cross section curves of a sample and up to four contaminants.

Trix1 calculates multi-group, Doppler-broadened effective resonance integrals and cross sections and their temperature coefficients for a heterogeneous resonance absorber. Using basic resonance parameters, geometry and concentrations, the code combines resolved resonance, unresolved resonance, and 1/V and negative energy resonance calculations. Fissile isotope calculation is available. L=1 unresolved resonance calculations are available for isotopes of all spins. Equivalence relationships are coded for cylinders, spheres, and planes. Interference between resonance and potential scattering and its temperature dependence are available. The narrow-resonance, infinite-mass, and intermediate representations of scatterings with absorber atoms are all available. The isolated single-level Breit-Wigner line shape is used.

The program computes resonance integrals for infinite homogeneous mixtures for above thermal energies. The major physical approximation is that resonances are represented by the single-level Breit-Wigner formulation. It is also assumed that scattering is elastic and that a nonresonant material has a constant scattering cross section.
215 CHAD
CHAD IS DESIGNED TO FACILITATE ANALYSIS AND HANDLING OF DIFFERENTIAL NEUTRON SCATTERING DATA. IT PRODUCES LEGENDRE SCATTERING COEFFICIENTS FROM ANGULAR DISTRIBUTION DATA TABULATED IN MANY DIFFERENT FORMATS. IT CAN TRANSFORM ANGULAR DATA INTO LEGENDRE SCATTERING COEFFICIENTS IN EITHER THE LABORATORY OR THE CENTER-OF-MASS FRAME OF REFERENCE. IT CALCULATES THE AVERAGE COSINE OF THE SCATTERING ANGLE IN THE LABORATORY SYSTEM AND THE AVERAGE LOGARITHMIC ENERGY DECREMENT PER ELASTIC COLLISION.

216 FASDOP
FASDOP EVALUATES POINTWISE, TEMPERATURE-DEPENDENT CROSS SECTIONS FROM SINGLE-LEVEL BREIT-WIGNER RESONANCE PARAMETERS.

217 LEGCOEF3
CALCULATES LEGENDRE EXPANSION COEFFICIENTS FOR THE ANGULAR DISTRIBUTION OF ELASTICALLY SCATTERED NEUTRONS.

219 GAROL
GAROL COMPUTES EFFECTIVE GROUP CROSS SECTIONS FOR THE RESOLVED RESONANCES OF A MIXTURE OF ISOTOPES IN A TWO-REGION CELL. BASIC CROSS SECTIONS INCLUDE TEMPERATURE DEPENDENCE. THE PROGRAM ALLOWS A CHOICE OF GEOMETRIES AND CAN ACCEPT AN ARBITRARY TABLE OF ESCAPE PROBABILITIES. A DANCOFF CORRECTION MAY BE USED TO ACCOUNT FOR SHADOWING EFFECTS IN A TIGHT LATTICE, AND CROSS SECTIONS MAY BE 1/V, CONSTANT, COMPUTED FROM BREIT-WIGNER RESONANCE PARAMETERS, OR GIVEN IN TABULAR FORM. THE MESH MAY BE CHOSEN AT EQUAL ENERGY OR LETHARGY INTERVALS, PROPORTIONAL TO THE NEUTRON VELOCITY, OR AS AN ARBITRARY TABLE OF VALUES.

238 EXT/XO
GIVEN A SET OF BREIT-WIGNER RESONANCE PARAMETERS (ZERO TEMPERATURE) EXT CALCULATES THE EFFECTIVE DOPPLER-BROADENED CROSS SECTIONS FOR ANY TEMPERATURE. THE EFFECTIVE CROSS SECTION WHEN MULTIPLIED BY THE TARGET DENSITY AND THE NEUTRON VELOCITY (LAB) GIVES THE REACTION RATE PER INCIDENT NEUTRON. A MAXWELLIAN VELOCITY DISTRIBUTION IS ASSUMED FOR THE TARGET NUCLEI. THE ANALYSIS IS CARRIED OUT FOR SUFFICIENTLY LOW ENERGIES SO THAT ONLY ZERO NEUTRON ANGULAR MOMENTA (L = 0) INTERACTIONS ARE IMPORTANT. INTERFERENCE BETWEEN LEVELS IS NEGLECTED, HOWEVER, INTERFERENCE BETWEEN RESONANCE SCATTERING AND POTENTIAL SCATTERING IS INCLUDED. THERE ARE PROVISIONS FOR ADDING A CORRECTION CROSS SECTION OF THE FORM 1.0/SQRT(E) FOR LEVELS NOT EXPLICITLY CONSIDERED. ALSO THE CROSS SECTION IN THE WINGS OF A NEGATIVE ENERGY RESONANCE, CONSTANT/ (SQRT(E)**(E-E0)**2) MAY BE ADDED.
FLANGE1 operates on a tabulated neutron scattering law, \( S(\alpha,\beta) \), to develop several types of cross section information. The various calculated cross sections now available are:

- \( \Sigma (E' \to E, \mu) \)
- \( \Sigma (E', \mu) \)
- \( \Sigma_{L} (E' \to E) \)
- \( \Sigma_{L} (E') \)
- \( M_{2} (E') \), second energy transfer moment
- \( \chi \Sigma_{L} \), average slowing down power
- \( \Sigma_{L} (\text{Maxwellian averaged}) \)

\( L \) refers to the Legendre moment, \( L = 0, 1, 2, 3 \) are available.

A tape of angular cross sections is prepared to be used as input to the multiple scattering code MUSE.

The 2PLUS code solves the problem of the scattering of charged or uncharged nucleons by a nucleus represented by a deformed nuclear potential. The model assumes that the target nucleus has a \( 0^+ \) ground state and a \( 2^+ \) first excited level, and the interaction potential has a quadrupole deformation. A Hauser-Feshbach compound nucleus calculation has been included. The output contains total, potential elastic, potential inelastic (\( 2^+ \)), reaction, and compound nucleus cross sections as well as elastic and inelastic angular distributions.

GASKET calculates the thermal neutron scattering law, \( S(\alpha,\beta) \), for a large class of moderators. Provision has been made in GASKET for the following dynamical modes of the scatterer:

1. Free translation (gas)
2. Diffusive or Brownian motion
3. Harmonic isotropic vibrations with continuous frequency spectrum
4. Harmonic anisotropic vibrations with continuous frequency spectrum (as applied for instance to graphite), and
5. Harmonic isotropic vibrations with discrete frequency spectrum.

The GAKER code evaluates the inelastic double-differential neutron scattering cross sections for moderators with phonon spectra which can be represented as sums of delta-functions. It is based on the original model for light water by Nelkin, which consisted of a translator, a hindered rotator (treated as an isotropic oscillator), and several vibrational oscillators. The code has been modified several times to include more oscillators and to treat anisotropic effects. Final energy-integrated cross sections are also calculated.
292 PSEUDO RESONANCE PARAMETERS ARE CONSTRUCTED FROM AVERAGE NUCLEAR PROPERTIES IN THE RESOLVED RESONANCE REGION.

305 STRIP CAPTURE AND FISSION RESONANCE INTEGRALS ARE CALCULATED BY A FAST METHOD IN THE RESOLVED RESONANCE RANGE, TAKING EXPLICIT ACCOUNT OF OVERLAP AND INTERFERENCE BETWEEN RESONANCES IN A MIXTURE OF RESONANCE ABSORBERS. THE RESONANCE INTEGRALS ARE CALCULATED OVER ARBITRARY ENERGY BANDS. OPTIONALLY, THE NEUTRON FLUX AS A FUNCTION OF ENERGY IN ONE OR TWO REGIONS MAY BE PRINTED OUT.

308 JUPITOR1 THIS IS A PROGRAM FOR PERFORMING COUPLED-CHANNEL CALCULATIONS TO EVALUATE THE CROSS SECTIONS FOR THE SCATTERING OF NUCLEAR PARTICLES BY VARIOUS COLLECTIVE NUCLEII.

323 MUFFLE THIS PROGRAM COMPUTES THE NEUTRON CROSS SECTIONS FOR A FISSILE NUCLIDE IN WHICH ONE TO THREE REACTION CHANNELS ARE OPEN FOR THE FISSION PROCESS. PROVISION IS MADE FOR TWO INDEPENDENT SPIN STATES WITH INTERFERING LEVELS AS WELL AS A SET OF NON-INTERFERING LEVELS. THE CROSS SECTIONS ARE INTEGRATED OVER A SPECIFIED GROUP STRUCTURE TO YIELD RESONANCE INTEGRALS.

334 PEGGY PEGGY IS A LEAST SQUARES SEARCH PROGRAM WHICH ANALYZES, IN TERMS OF PHASE SHIFTS, THE ELASTIC SCATTERING OF SPIN ZERO AND SPIN ONE-HALF PARTICLES BY SPIN ZERO NUCLEI. REAL OR COMPLEX PHASE SHIFTS MAY BE USED WITH OR WITHOUT SPIN-ORBIT COUPLING. DIFFERENTIAL CROSS SECTION AND POLARIZATION ANGULAR DISTRIBUTIONS MAY BE ANALYZED EITHER SEPARATELY OR SIMULTANEOUSLY.

335 RAMES RAMES COMPUTES BOTH LOCAL AND NONLOCAL RADIAL INTEGRALS OF A VARIETY OF RADIAL OPERATORS USING SINGLE-PARTICLE WAVE FUNCTIONS WHICH ARE EIGENSTATES OF MOTION IN A WOODS-SAXON POTENTIAL WELL. THE OPERATORS CURRENTLY AVAILABLE ARE R**N, N=0,1,2,3,4,5 AND THE DERIVATIVE WITH RESPECT TO X OF 1/(EXP(X)+1) WHERE X=(R-R(0))/A(0).

341 GANDY THE GANDY CODE EVALUATES TEMPERATURE-DEPENDENT EFFECTIVE NEUTRON CAPTURE, FISSION, AND SCATTERING CROSS SECTIONS IN THE UNRESOLVED RESONANCE REGION FROM AVERAGE RESONANCE PARAMETERS.
A LEAST SQUARES ANALYSIS OF NEUTRON RESONANCE DATA IS PERFORMED USING THE MULTI-LEVEL EXPANSION. THE PROGRAM CAN HANDLE ONLY ONE SET OF CROSS SECTIONS AT A TIME. OPTIONS ARE PROVIDED FOR THE ANALYSIS OF REACTION OR TOTAL CROSS SECTION DATA, AND FOR THE DIRECT HANDLING OF TRANSMISSION DATA. BY OPTION, ONE CAN INCLUDE THE MULTI-LEVEL INTERFERENCE OR PERFORM THE FIT IN TERMS OF SUPERIMPOSED SYMMETRIC BREIT-WIGNER LINES, WHILE THE POTENTIAL SCATTERING INTERFERENCE IS ALWAYS INCLUDED IN THE TRIAL FUNCTION FOR THE TOTAL CROSS SECTION. PROVISIONS ARE GIVEN FOR GAUSSIAN AND NON-GAUSSIAN RESOLUTION FUNCTIONS. IN THE LATTER CASE, MODIFICATION OF ONE SUBROUTINE ALLOWS FOR ADAPTATION TO ANY KIND OF EXPERIMENTAL CONDITIONS. THE RELEVANT CONVOLUTION INTEGRALS INVOLVING THE TRIAL FUNCTION ARE EVALUATED BY SIMPSON INTEGRATION WITH AN OPTIONAL NUMBER OF INTEGRATION STEPS. BESIDES THE RESONANCES TO BE FITTED, THE TRIAL FUNCTION CONTAINS AN OPTIONAL NUMBER OF RESONANCES HAVING KNOWN PARAMETERS, WHICH MAY REPRESENT RESONANCES EXTERNAL TO THE REGION BEING FITTED AS WELL AS RESONANCES WITHIN THE ENERGY INTERVAL OF INTEREST, THUS PERMITTING IMPURITY EFFECTS TO BE DESCRIBED, OR, IF NEEDED, SPIN STATES SEPARATED.

PUN1 EVALUATES UNRESOLVED RADIATIVE CAPTURE INTEGRALS AND RELATED MULTIGROUP CROSS SECTIONS. THE UNRESOLVED DISTRIBUTIONS MAY HAVE VARIOUS ORBITAL ANGULAR MOMENTUM QUANTUM NUMBERS AND THE EFFECTS OF DOPPLER BROADENING AND SELF-SHIELDING ARE INCLUDED.


FLANGE2 TAKES CROSS SECTIONS, ANGULAR DISTRIBUTION, RESONANCE PARAMETER, AND SCATTERING LAW DATA FROM ENDF/B DATA TAPES AND PREPARES THERMAL MULTIGROUP CROSS SECTIONS AND SCATTERING MATRICES.

AVERAGE CALCULATES AVERAGE SCATTERING, CAPTURE, AND FISSION CROSS SECTIONS FROM S- AND P-WAVE DATA OF THE UNRESOLVED PARAMETERS OF FILE 2 OF ENDF/B.

SIGPLOT CALCULATES THE SCATTERING, CAPTURE, FISSION, AND TOTAL CROSS SECTIONS FROM RESOLVED RESONANCE PARAMETER DATA. SCATTERING CROSS SECTIONS MAY BE CALCULATED WITH OR WITHOUT LEVEL-LEVEL INTERFERENCE. PROVISION IS ALSO MADE TO NUMERICALLY DOPPLER-BROADEN ANY OF THE CROSS SECTIONS.
LYNNE performs a multipole expansion of the Woods-Saxon potential. The numbers generated are suitable for microscopic calculations of inelastic scattering from nuclei which use a Woods-Saxon interaction between the projectile and the target nucleons.
B. SPECTRUM CALCULATIONS, GENERATION OF GROUP CONSTANTS, LATTICE AND CELL PROBLEMS

33 GAM1/REP/UPDATE

This program computes the slowing-down spectrum in either the P1 or the B1 approximation using 68 groups of neutrons with a constant group width DU=0.25. Multigroup constants are calculated for up to 32 fast groups.

50 TEMPEST2

TEMPEST2 is a neutron thermalization program based upon the Wigner-Wilkins approximation for light moderators and the Wilkins approximation for heavy moderators. A Maxwellian distribution may also be used. The model used may be selected as a function of energy. The second-order differential equations are integrated directly rather than transformed to the Riccati equation. The program provides microscopic and macroscopic cross section averages over the thermal neutron spectrum.

51 FORM

The FORM, or FORTRAN-MUFT, program is a Fourier transform slowing-down code. A library tape containing 54-group microscopic cross sections, resonance parameters, inelastic scattering matrices, and source spectra is used to generate a 54-group flux spectrum and few-group constants.

52 SAIL

The monoenergetic neutron transport equation is solved using the discrete SN method for a one-dimensional plane cell. Cell properties are computed.

53 S4 CYL GEOMETRY CELL CODE

This program solves the one-dimensional monoenergetic Boltzmann equation in cylindrical geometry, using the S4 approximation. In addition to the flux distribution, cell-averaged parameters are computed.

108 BAM

BAM computes thermal constants, spatial and energy distributions in heterogeneous cylindrical cells by assuming separability of space and energy in the Boltzmann equation.

113 ZOT

ZOT takes multigroup neutron cross section sets in the SN format (SNG, DSN, etc.) and reduces the number of groups (collapses) according to a given or computed multigroup flux spectrum. Average velocities for the few-group set may be generated on the basis of the fluxes and volumes for a given reactor configuration.
119 QUICKIE  QUICKIE SOLVES THE NEUTRON SLOWING DOWN AND THERMALIZATION PROBLEM IN INFINITE MEDIA BY INVERTING A SET OF SIMULTANEOUS MULTIGROUP EQUATIONS. THE CODE USES THE ULCER (ACC ABSTRACT 118) LIBRARY TAPE AND IS IN EXCELLENT AGREEMENT WITH ULCER FOR THOSE CASES WHERE BUCKLING IS KNOWN.

147 AILMOE  AILMOE IS A MODIFIED FORM OF THE ANL ELMCE PROGRAM. THE FOURIER TRANSFORM OF THE FAST NEUTRON FLUX IS FOUND FOR A MIXTURE OF MODERATORS HEAVIER THAN HYDROGEN WITH THE MODERATOR SCATTERING LAW RIGOROUSLY ACCOUNTED FOR.

149 TYCHE3  TYCHE3 IS A MONTE CARLO CODE DESIGNED TO FIND THE SECOND, FOURTH AND SIXTH MOMENTS OF THE NEUTRON SLOWING DOWN DENSITY DISTRIBUTION IN AN INFINITE HOMOGENEOUS MEDIUM. NEUTRON WEIGHTS ARE USED TO AVOID THE TERMINATION OF A HISTORY BY ABSORPTION AND MINIMIZE THE RUNNING TIME. PROVISIONS ARE MADE FOR RESTART OF NON-CONVERGED PROBLEMS, GRAPHICAL DISPLAYS OF THE MOMENTS AND AVERAGE FISSION ENERGY AS A FUNCTION OF THE NUMBER OF SETS OF HISTORIES AND CALCULATION OF THE CORRECTION TO FLUX MOMENTS.

150 DANCOFF JR.  THIS CODE EVALUATES MODERATOR SPACE CHORD DISTRIBUTION FUNCTIONS OF ZEROTh AND FIRST ORDER, PLUS THEIR LINEAR, SQUARE, LOGARITHMIC AND EXPONENTIAL MOMENTS, FOR REGULAR AND IRREGULAR LATTICES OF CYLINDRICAL FUEL RODS CLAD WITH MATERIAL OF NEGLIGIBLE TOTAL CROSS SECTION. OF PARTICULAR SIGNIFICANCE FOR REACTOR DESIGN CALCULATIONS IS THE EXPONENTIAL MOMENT, OR DANCOFF CORRECTION, WHICH CAN BE CALCULATED EXACTLY IN INFINITE SQUARE AND HEXAGONAL LATTICES, IN CLUMPED SQUARE LATTICES WITH STRAIGHT OR CRUCIFORM WATER GAPS, OR IN CLUSTERS OF TWO, THREE, SEVEN, AND NINETEEN FUEL RODS.

160 SOPHIST1/2/5  SOPHIST1 CALCULATES TEMPERATURE-DEPENDENT MULTIGROUP ENERGY TRANSFER COEFFICIENTS FOR A MAXWELL GAS MODERATOR WITH ELASTIC, ISOTROPIC SCATTERING IN THE CENTER OF MASS SYSTEM. SOPHIST2 CALCULATES TEMPERATURE-DEPENDENT MULTIGROUP CROSS SECTIONS FOR A MAXWELL GAS. SOPHIST5 CALCULATES MULTIGROUP ENERGY TRANSFER MATRICES FOR ANISOTROPIC ELASTIC SCATTERING.


THE SPECTRUM IS CONSTRUCTED FROM A COMBINATION OF FISSION, EN, POWER SERIES, MAXWELLIAN OR INPUT SPECTRA.
SPARTA COMPUTES SPATIALLY-AVERAGED DOPPLER COEFFICIENTS AND SPATIALLY-AVERAGED DOPPLER REACTIVITY CHANGES.


GAMTEC2 GENERATES MULTIGROUP CONSTANTS IN THE ENERGY RANGE FROM 0 TO 10 MEV FOR EITHER HOMOGENEOUS MIXTURES OR HETEROGENEOUS ARRAYS CONSisting OF CYLINDERIZED LATTICE CELLS. THE THERMAL GROUP CONSTANTS ARE AVERAGEC OVER EITHER (1) WIGNER-WILKINS LIGHT MODERATOR SPECTRUM, (2) WILKINS HEAVY MODERATOR SPECTRUM, OR (3) A MAXWELLIAN DISTRIBUTION. FOR HETEROGENEOUS ARRAYS THE SPATIAL THERMAL FLUX IS CALCULATED BY A MONOENERGETIC P3 APPROXIMATION. FOR EPITHERMAL ENERGIES, THE SLOWING-DOWN DISTRIBUTION IS DESCRIBED BY EITHER A B1 OR P1 APPROXIMATION TO THE BOLTZMANN EQUATION. RESONANCE ABSORPTION AND FISSION ARE TREATED BY THE ADLER-NORDHEIM METHOD. AN IMPROVED METHOD OVER THAT IN GAM-I FOR AVERAGING THE RESONANCE ABSORPTION CONTRIBUTION TO THE MULTIGROUP CONSTANTS IS INCLUDED. FUEL LUMPING EFFECTS ON THE FAST FISSION OF U238 AND TH232 ARE TREATED BY AN N-FLIGHT COLLISION PROBABILITY TECHNIQUE. GROUP CONSTANTS ARE PUNCHED ON CARDS IN HFN (DIFFUSION CODE) AND DTF (SN TRANSPORT CODE) FORMATS.

THE FLEXIBLE MONTE CARLO PROGRAMS FMC-N AND FMC-G WERE ORIGINALLY WRITTEN IN ASSEMBLY LANGUAGE FOR THE IBM704 BY GE-ANP. THESE PROGRAMS HAVE BEEN CONVERTED TO FORTRAN FOR THE CDC1604B BY PRATT AND WHITNEY. THE PROGRAMS APPLY MONTE CARLO METHODS TO SIMULATE NEUTRON AND GAMMA RAY LIFE HISTORIES, RESPECTIVELY, IN A SOURCE-SHIELD CONFIGURATION. AS THE NAMES IMPLY, THE CODES ARE DESIGNED FOR FLEXIBILITY IN THE GEOMETRICAL, MATERIAL, NUCLEAR, AND SOURCE DESCRIPTIONS OF SOURCE-SHIELD CONFIGURATIONS AND VARIANCE REDUCTION TECHNIQUES.
**201 EPITHERMOS**

The EPITHERMOS code is a modification of the THERMOS thermal transport theory code of Honeck. The essential difference between the codes is in the library preparation subroutine BASK. The EPITHERMOS code is designed to compute the neutron density above \(0.7849\) eV, the maximum energy of the usual THERMOS calculation. EPITHERMOS computes the scalar neutron density as a function of position and speed in a one-dimensional slab or cylindrical system. The scattering model is arbitrary in the thermal range and the built-in free gas model is used in the epithermal range.

**202 MCS**

The MCS code determines the spatial distribution of some nuclear reaction for a given neutron source in a given configuration of materials. The Monte Carlo estimate of the solution consists of picking a sample of neutrons from the given source and following each neutron through a sequence of surface crossings and collisions until the neutron either escapes or is no longer of interest to the solution. The desired flux or collision density is accumulated for every neutron of the sample, and the sample size increased until results of sufficient statistical significance are obtained.

**204 AGN-GAM**

AGN-GAM calculates the subgroup fluxes and current terms from a solution of the \(P_1\) or \(B_1\) equations. The \(P_1\) equations incorporate both volume and surface sources, allowing spectral calculations in reflector regions. Other spectral options are flux known, current term calculated and both flux and current terms known. The age for the material is obtained from a second moments calculation. The 75 subgroups used in the calculations are established as quarter lethargy groups from 10 MeV to 0.07 eV. Elastic scattering matrices maintain the energy-angle correlation by including both \(P_0\) and \(P_1\) terms. Isotropic scattering matrices are included for inelastic and \((N,2N)\) processes. The method of Adler, Hinman and Nordheim is used to calculate resonance absorption and fission cross sections. Multigroup constants are generated by spectral averaging over the subgroups.

**213 RIFF-RAFF**

The program computes resonance integrals for absorbers in a rod in a two-region circularized cell for above thermal energies. The flux distribution in the cell is also computed and can be printed. The method of solution is based upon the following assumptions — isotropic scattering in the laboratory system, constant total cross section in the moderator and isotropic neutron fluxes entering and leaving the rod.
235 GAMMA-P (NMP468) CROSS SECTIONS FOR THE PRODUCTION OF GAMMA RAYS BY NEUTRON RADIATIVE CAPTURE, BY NEUTRON INELASTIC SCATTERING, AND BY NEUTRON-INDUCED FISSIONING ARE COMPUTED AS A FUNCTION OF NEUTRON LETHARGY GROUPS AND PHOTON ENERGY GROUPS.

237 BOUNCE THE BOUNCE CODE DETERMINES THE ONE-GROUP THERMAL NEUTRON FLUX DISTRIBUTION WITHIN THE PIN BUNDLE OF A MULTIPLE-PIN FUEL ELEMENT. ALL SCATTERING WITHIN THE PINS IS CONSIDERED TO BE ISOTROPIC. ELEMENTS WITH 6, 7, 12, 13, 18, AND 19 FUEL PINS, WHERE EACH PIN IS COMPOSED OF A CENTRAL FUEL PEELLET SURROUNDED BY A CLADDING MATERIAL, MAY BE TREATED. THE COOLANT REGION AROUND THE PINS IS ASSUMED TO CONTAIN A VOID.

243 AGN-SIGMA AGN-SIGMA CALCULATES THE LEGENDRE COMPONENTS OF THE MULTIGROUP TRANSFER MATRICES \( \sigma_\ell \) FOR FAST NEUTRONS. REACTIONS CONSIDERED ARE ELASTIC SCATTERING, INELASTIC SCATTERING (LEVEL EXCITATION AND THE EVAPORATION MODEL), AND THE FOLLOWING FIVE DECAY MODES FOR THE \( (N,2N) \) REACTION -- \( A(N,N1)A* (N2) (A-1)* \), 3- AND 4- BODY PHASE SPACE MODEL, EVAPORATION MODEL, AND THE CLUSTER MODEL, WHERE \( A* \) IS THE RECOIL NUCLEUS. ALL NUCLEAR LEVELS INVOLVED IN THE TRANSITIONS ARE DISCRETE. THE CODE MAY ALSO BE USED TO CALCULATE GROUP-AVERAGED CROSS SECTIONS AS WELL AS TO MANIPULATE, E.G., ADD, MULTIPLY, ETC., THE OUTPUT MATRICES. THE NEUTRON SPECTRUM MAY BE A COMBINATION OF FISSION AND 1/E OR ARBITRARY INPUT DATA.


257 REAX REAX CALCULATES EPITHERMAL FLUX, ACTIVITIES AND CROSS SECTIONS AS A FUNCTION OF RADIUS AND ENERGY FOR A CONSTANT TEMPERATURE FUEL ROD IMMERSED IN A HOMOGENEOUS MEDIUM.
276 AVOID
AVOID COMPUTES THE EQUIVALENT DIFFUSION COEFFICIENT AND LOSS CROSS SECTION OF AN ANNULAR VOID IN A CYLINDRICAL REACTOR AND THE RADIAL FLUX DISTRIBUTION IN THE VOID.

277 HAMMER/LITHE/HELP/LIBCON
HAMMER PERFORMS INFINITE LATTICE, ONE-DIMENSIONAL CELL MULTIGROUP CALCULATIONS, FOLLOWED (OPTIONALLY) BY ONE-DIMENSIONAL, FEW-GROUP, MULTIREGION REACTOR CALCULATIONS WITH NEUTRON BALANCE EDITS.

279 LEOPARD/AITPI/AITP3/SPOTS
LEOPARD IS A UNIT CELL HOMOGENIZATION AND SPECTRUM GENERATION (MUFT-SOFOCATE) PROGRAM WITH A FUEL DEPLETION OPTION.

R280 M0807
M0807 SOLVES THE TWO-DIMENSIONAL FIXED SOURCE DIFFUSION EQUATION FOR THE ABSORPTION AND REMOVAL MACROSCOPIC CROSS SECTIONS REQUIRED TO YIELD A SPECIFIED REACTION RATE DISTRIBUTION.

281 RABBLE/WLIB/FLAT
RABBLE COMPUTES EFFECTIVE CROSS SECTIONS FOR ABOVE THERMAL ENERGIES BASED ON RESOLVED SINGLE-LEVEL RESONANCE PARAMETERS FOR INFINITE HOMOGENEOUS OR HETEROGENEOUS SYSTEMS.

R285 RESQ2/RESQ0/DBF1
RESQ2 CALCULATES THE RESONANCE INTEGRAL IN A TWO-DIMENSIONAL, HEXAGONAL SYSTEM CONSISTING OF FUEL, CLAD AND WATER WITH A REFLECTING BOUNDARY CONDITION.

291 HEXSCAT
HEXSCAT CALCULATES P0 THROUGH P3 COMPONENTS OF THE POLYCRYSTALLINE COHERENT ELASTIC NEUTRON SCATTERING CROSS SECTION PER NUCLEUS FOR A HEXAGONAL LATTICE. THE CODE AVERAGES POINT VALUES OVER INPUT GROUP BOUNDARIES TO GIVE SMOOTHED GROUP CROSS SECTIONS.
THE GGC4 PROGRAM SOLVES THE MULTIGROUP SPECTRUM EQUATIONS WITH SPATIAL DEPENDENCE REPRESENTED BY A SINGLE POSITIVE INPUT BUCKLING. BROAD GROUP CROSS SECTIONS (SHIELDED OR UNSHIELDED) ARE PREPARED FOR DIFFUSION AND TRANSPORT CODES BY AVERAGING WITH THE CALCULATED SPECTRA OVER INPUT-DESIGNATED ENERGY LIMITS. THE CODE IS DIVIDED INTO THREE MAIN PARTS. A FAST (GAM) SECTION WHICH COVERS THE ENERGY RANGE FROM 14.9 MEV TO 0.414 EV, A THERMAL (GATHER) SECTION WHICH COVERS THE ENERGY RANGE FROM 0 TO 2.38 EV, AND A COMBINING (COMBO) SECTION WHICH COMBINES FAST AND THERMAL CROSS SECTIONS INTO SINGLE SETS. BASIC NUCLEAR DATA FOR THE FAST SECTION WHICH CONSISTS OF FINE GROUP-AVERAGED CROSS SECTIONS AND RESONANCE PARAMETERS IS READ OFF A DATA TAPE. THE FINE GROUP ABSORPTION AND FISSION CROSS SECTIONS MAY BE ADJUSTED BY PERFORMING A RESONANCE INTEGRAL CALCULATION. UTILIZING A FISSION SOURCE AND AN INPUT BUCKLING, THE CODE SOLVES THE P1, B1, B2, OR B3 APPROXIMATION TO OBTAIN THE ENERGY-DEPENDENT FAST SPECTRUM. TWO OR SIX SPATIAL MOMENTS OF THE SPECTRUM (DUE TO A PLANE SOURCE) MAY ALSO BE EVALUATED. INSTEAD OF PERFORMING A SPECTRUM CALCULATION, THE USER MAY ENTER THE LEGENDRE COMPONENTS OF THE ANGULAR FLUX DIRECTLY. FOR AS MANY INPUT-DESIGNATED BROAD GROUP STRUCTURES AS DESIRED, THE CODE CALCULATES AND SAVES (FOR THE COMBINING SECTION) SPECTRUM-WEIGHTED AVERAGES OF MICROSCOPIC AND MACROSCOPIC CROSS SECTIONS AND TRANSFER ARRAYS. SLOWING DOWN SOURCES ARE CALCULATED AND SAVED FOR USE IN THE LOWER ENERGY RANGE. GIVEN BASIC NUCLEAR DATA, THE THERMAL SECTION OF GGC4 DETERMINES A THERMAL SPECTRUM BY EITHER READING IT AS INPUT, BY CALCULATING A MAXWELLIAN SPECTRUM FOR A GIVEN TEMPERATURE, OR BY AN ITERATIVE SOLUTION OF THE P3, BO, P1, OR B1 EQUATIONS FOR AN INPUT BUCKLING. TIME MOMENTS OF THE TIME AND ENERGY-DEPENDENT DIFFUSION EQUATIONS ARE CALCULATED (AS AN OPTION) USING THE INPUT BUCKLING TO REPRESENT LEAKAGE. BROAD GROUP CROSS SECTIONS ARE PREPARED BY AVERAGING FINE GROUP CROSS SECTIONS OVER THE CALCULATED SPECTRA. BROAD GROUP STRUCTURES ARE READ AS INPUT. THE COMBINING SECTION OF GGC4 TAKES THE BROAD GROUP-AVERAGED CROSS SECTIONS FROM THE FAST AND THERMAL PORTIONS OF GGC4 AND FORMS MULTIGROUP CROSS SECTION TABLES. THESE TABLES ARE PREPARED IN STANDARD FORMATS FOR TRANSPORT OR DIFFUSION THEORY CALCULATIONS. IN ADDITION, IT IS POSSIBLE TO USE THE COMBINING SECTION TO PRODUCE MIXTURES NOT USED IN THE SPECTRUM CALCULATION OR TO COMBINE THE RESULTS OF DIFFERENT FAST AND THERMAL SECTION CALCULATIONS AND SO ON.

FCC4 IS A MULTIPURPOSE DATA MANIPULATION CODE FOR USE IN FAST REACTOR ANALYSIS. THE CODE CAN BE USED TO - (A) COMPUTE RESONANCE-SHIELDED CROSS SECTIONS USING DATA IN THE RUSSIAN FORMAT (SHIELDING FACTORS AND INFINITE-DILUTION CROSS SECTIONS), (B) COMPUTE MULTIGROUP FUNDAMENTAL-MODE FLUX AND ADJOINT FLUX, (C) COMPUTE AND PUNCH GROUP-COLLAPSED MICROSCOPIC OR MACROSCOPIC CROSS SECTIONS IN THE DTF FORMAT, (D) COMPUTE FUEL BURNUP AT CONSTANT FLUX OR POWER DENSITY.
HWOCR-SAFE is a Monte Carlo thermal reactor analysis program designed for use with proposed HWOCR lattice configurations. It is a benchmark tool to check multigroup diffusion and transport calculations and to evaluate the effect of their use of geometric approximations.

The problem is to obtain very detailed neutron flux and current distributions as functions of energy considering explicitly the possible overlap effects between resonances of a resonance absorber and of mixtures of resonance absorbers and to use these distributions to prepare group-averaged cross sections and transfer arrays for use in fast reactor analyses.

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

The Glen program interpolates values of a factor proportional to the scattering law from the punched output of the TOR code (ACC abstract 360). The differential cross section determined from these is integrated over the scattering angle to obtain coefficients of an expansion in Legendre polynomials of this angle for \( L = 0, 1, 2, 3 \). Integration over final energies yields values of the total scattering cross section and transport cross section. For each of a series of isotopic compositions (up to 10 compositions) the Glen code calculates the diffusion length and values of the flux-weighted group average macroscopic scattering, absorption, fission, and transfer cross sections.

Welwing was developed to calculate the material buckling of reactor systems consisting of annular fuel elements in heavy water as moderator for various moderator to fuel ratios. The moderator to fuel ratio for the maximum material buckling for the particular system is selected automatically and the corresponding material buckling is calculated.

1DX is a multipurpose, one-dimensional diffusion code for generating cross sections to be used in fast reactor analyses. The code is designed to:
(A) Compute and punch resonance shielded cross sections using data in the Russian (see reference 2) format,
(B) Compute and punch group-collapsed microscopic and/or macroscopic cross sections averaged over the spectrum in any specified zone, and
(C) Compute KEFF and perform criticality searches on time absorption, material concentrations, zone dimensions, and buckling using either a flux or an adjoint model.
C. STATIC DESIGN STUDIES

7 FIRN
FIRN SOLVES THE TWO-DIMENSIONAL SN APPROXIMATION IN FINITE CYLINDRICAL GEOMETRY. IT IS A FORTRAN ADAPTATION OF THE LOS ALAMOS TDC PROGRAM WRITTEN BY BENGT CARLSON AND CLARENCE LEE. FIRN IS LIMITED TO A MAXIMUM OF SIX GROUPS AND THE S2, S4 OR S6 APPROXIMATION.

9 FIRES
FIRES SOLVES THE ONE-DIMENSIONAL MULTIGROUP AGE-DIFFUSION EQUATIONS FOR SLAB (SYMMETRIC AND ASYMMETRIC), CYLINDER, AND SPHERE GEOMETRIES. AN EXTENSION OF THE ORIGINAL LOS ALAMOS FIRE PROGRAM WRITTEN BY F. W. BRINKLEY AND C. B. MILLS, IT CONTAINS A 34-GROUP LIBRARY OF 10 ELEMENTS. FIRES CAN BE USED FOR WALL CALCULATIONS WITH THE BOUNDARY CONDITION THAT THE CURRENT BE ZERO ON THE OUTER BOUNDARY. THE EFFECTIVE CROSS SECTIONS AND DIFFUSION COEFFICIENTS FOR A HOMOGENISED REGION CONSISTING OF ANY NUMBER OF NEIGHBORING REGIONS CAN ALSO BE OBTAINED, BUT ONLY FOR CYLINDRICAL GEOMETRY. FIRES ALSO COMPUTES THE MACROSCOPIC FLUX WEIGHTED CROSS SECTIONS FOR COLLAPSED GROUPS IN EACH REGION.

18 2DXY
THE 2DXY PROGRAM SOLVES THE HOMOGENEOUS OR INHOMOGENEOUS MULTIGROUP EQUATIONS IN X-Y GEOMETRY, USING THE SN TRANSPORT EQUATION APPROXIMATION. VACUUM, SURFACE SOURCE, OR REFLECTING BOUNDARY CONDITIONS ARE AVAILABLE AS OPTIONS. IN THE HOMOGENEOUS CASE THE USER MAY REQUEST THE COMPUTATION OF REACTIVITY, REACTOR PERIOD, CRITICAL CONCENTRATIONS OF SOME COMPOSITION, OR THE CRITICAL THICKNESS OF A ZONE.

28 FCG

29 AIM6
AIM6 SOLVES THE ONE-DIMENSIONAL MULTIGROUP DIFFUSION EQUATIONS UTILIZING A MICROSCOPIC CROSS SECTION LIBRARY. ANY OF THREE GEOMETRIES ARE AVAILABLE SLAB, CYLINDER OR SPHERE. CRITICALITY SEARCHES ARE PROVIDED INCLUDING A CONCENTRATION SEARCH ON ONE OR TWO ELEMENTS. HOMOGENEOUS AND INHOMOGENEOUS PROBLEMS MAY BE SOLVED WITH A VARIETY OF BOUNDARY CONDITION OPTIONS.
PERT is a perturbation program designed for use with the AIM6 and FOG programs. Punched card output from these codes is used as input to PERT. Using cross section data, fluxes, and adjoint fluxes, the relative change in \( K_{eff} \) can be calculated. Cross sections may be weighted with the adjoint flux and/or direct flux. The neutron lifetime for the delay groups may also be determined.

Whirlaway solves the two-group, three-dimensional, neutron diffusion equations in \( X-Y-Z \) geometry.

Equipoise3 solves the two-group, two-dimensional, neutron diffusion equations in cylindrical or slab geometry.

20Grand solves the few-group, two-dimensional, neutron diffusion equations in cylindrical or slab geometry.

Mist obtains the solution to the one-dimensional Boltzmann equation in slab geometry. The numerical approximation used is a linear one which can be described as an extension and generalization of the SN approximation. The equations are formulated in terms of a double SN approximation. The boundary conditions for each group may be independently specified and permit very general specifications with respect to:

- (A) Perfect mirror reflection or symmetry, by input of mirror albedos,
- (B) Anisotropic diffuse sources, by input of Legendre polynomial coefficients up to \( \lambda = 9 \), or a short table describing a known angular distribution of the flux,
- (C) Isotropic (Lambert surface) reflection.

Isotropic volume sources in each group may also be independently specified. The scattering from one group to another is assumed to be isotropic but the scattering function within each group can be a second-order Legendre polynomial series.

Program 5 constructs bilinearly coupled time-variant multienergy neutron-and-photon transport and nuclide-transmutation fields having slab, cylindrical, or spherical symmetry. Assurance of unbiased convergence is provided by use of a dual adjoint-and-flux loop constructed in precise correspondence with the physics of successive free flights.
EQUIPOISE-3A solves the two-dimensional two-group diffusion equations in cylindrical or slab geometry. It is a slightly revised version of EQUIPOISE3 (ACC abstract 39). In addition to the standard output, a picture is printed of the material arrangement in the reactor. If the adjoint flux option is used, the prompt neutron lifetime is calculated and printed, with the reactivity per unit change in each group constant in each region of the reactor.

CRAM is a program to solve the multigroup diffusion equations in two-dimensions (R-Z, X-Y, or R-theta geometry), or in one-dimension (slab, cylindrical, or spherical geometry). Neutrons may scatter from any group to any other. Real, adjoint, and source-type problems are all solvable. The program will compute the k-effective of the system or alternatively search for criticality by moving spatial boundaries, varying material compositions, or varying transverse buckling.

ULCER is a multigroup, one-dimensional diffusion equation code with upscatter based on FAIM (ACC abstract 120). ULCER differs from FAIM in that:
(A) upscattering is included,
(B) downscatter to all lower groups,
(C) microscopic cross sections are on tape,
(D) provision for multiple fission spectra,
(E) restart dump,
(F) few-group reduction, and
(G) spectrum computation and graphical display.

FAIM is a multigroup, one-dimensional diffusion equation program based on AIM6 (ACC abstract 29). The principal features are:
(A) three geometries,
(B) calculation of fluxes and multiplication factor,
(C) one-iteration problems,
(D) choice of one of five sets of boundary conditions at both boundaries,
(E) criticality searches on transverse buckling, homogeneous poison, critical radius, one, two, or three element concentration, location of poison region boundary, location of a fuel region boundary,
(F) adjoint flux calculation, and
(G) extensive data edit.

FAIMOS is a modified version of FAIM. Three general modifications were made:
(A) the microscopic cross section library and its associated subroutines were removed. As a result, options requiring the use of microscopic cross sections are not available.
(B) FAIM ran as a chain job. Overlays are not used by FAIMOS.
(C) the program language was converted from FORTRAN II to FORTRAN IV(H). This modification made necessary a minor change in the data input format.
W-DSN solves the discrete SN equations in cylindrical geometry. The eigenvalue option is reactivity (KEFF) only. Volume distributed sources are allowed, but no surface sources.

HERESY1/KERNEL calculates the reactivity, thermal utilization, resonance escape probability, relative rod absorptions and power distribution in heterogeneous reactors having two spatial dimensions. HERESY1 can be used for non-uniform lattices, lattices with many types of fuel and control rods, and spiked and seeded reactors. Fissions are assumed to occur only at thermal energy. Resonance absorptions are lumped into one equivalent resonance. The moderator is assumed to be infinite in the radial direction, and rods are treated as line sources and sinks. The rod parameters are independent of the inter-rod separation distances. Slowing-down kernel functions may be of any type - age theory, transport theory, or empirical. A self-consistent procedure can be used which effectively cancels out any errors in the kernel functions.

DTF is the DTF program is a one-dimensional multigroup program for solving the neutron transport equation. The program can determine the regular or adjoint solution for slab, cylindrical, or spherical geometry. Isotropic or a form of linear anisotropic scattering may be considered. Various boundary conditions are allowed so that cells or time-dependent solutions may be obtained using finite as well as infinite configurations in the case of slabs or cylinders. The program also contains a number of search options whereby one can vary dimensions or concentrations to arrive at a predetermined eigenvalue. Distributed or shell sources may be specified at any position within the configuration. As output the program supplies the eigenvalue, angular fluxes, total fluxes, fission distributions, and other quantities. A library of cross sections is available on magnetic tape. Cross sections may be read from this library tape and/or from cards.

TOPIC solves the one-dimensional Boltzmann equation in cylindrical geometry with up to six energy groups, 240 space points, 40 regions, and anisotropic (P1) scattering.

The boundary conditions for each group can be independently specified and the flexibility of the specifications permit -

(A) perfect mirror reflection or symmetry,
(B) isotropic reflection (Lambert surface reflection), and
(C) anisotropic diffuse sources by means of either a P1 Legendre series or a short table of point values for the angular flux.

Independent specification of isotropic fixed volume sources for each group is also allowed. As implied, both homogeneous and inhomogeneous problems are solved, and fissions can occur in either type of problem.
151 DTF2/ANISN  THE MULTIGROUP, ONE-SPACE DIMENSION NEUTRON TRANSPORT EQUATION IS SOLVED. ISOTROPIC OR LINEAR ANISOTROPIC SCATTERING IS PERMITTED BETWEEN ALL GROUPS AND A DIFFUSION SOLUTION MAY BE OBTAINED FOR ANY OR ALL GROUPS. HIGH-ORDER ANISOTROPIC SCATTERING PROBLEMS (PL) CAN BE RUN USING ANISN, THE 360 VERSION OF DTF2. WHITE/GREY BOUNDARY CONDITIONS ARE AVAILABLE, AND AN ALBEDO CAN BE SPECIFIED FOR EACH GROUP. A VOID STREAMING CORRECTION IS INCLUDED. A COMPLETE SHELL SOURCE DESCRIPTION BY GROUP, POSITION, AND ANGLE IS AVAILABLE. GRAPHICAL DISPLAY (CRT) FEATURES ARE AVAILABLE WITH ANISN.

156 EXTERMINATOR/EXTERMINATOR2  THE MULTIGROUP, TWO-DIMENSIONAL NEUTRON DIFFUSION EQUATIONS ARE SOLVED IN X-Y, R-Z, OR R-THETA GEOMETRY.

161 FORTRAN TDC  TDC SOLVES THE BOLTZMANN EQUATION IN MULTIGROUP FORM FOR THE TRANSPORT OF NEUTRONS OR THE ADJOINT EQUATION IN FINITE (R,Z) CYLINDRICAL GEOMETRY BY THE DISCRETE SN METHOD. THE PROBLEM MAY BE HOMOGENEOUS (NO SOURCES INDEPENDENT OF FLUXES) OR INHOMOGENEOUS, BUT ALL SOURCES MUST BE ISOTROPIC. NEUTRON SCATTERING MUST ALSO BE ISOTROPIC. A HOMOGENEOUS PROBLEM MAY BE SOLVED FOR THE EIGENVALUE K-EFF OR THE EIGENVALUE ALPHA (TIME CONSTANT). ALTERNATIVELY, THE HOMOGENEOUS PROBLEM MAY BE SOLVED FOR THE SIZE (RADIUS, HEIGHT OR BOTH) OF THE SYSTEM CORRESPONDING TO A SPECIFIED K-EFF OR FOR THE ATOM CONCENTRATION OF SOME MATERIAL CORRESPONDING TO A SPECIFIED EIGENVALUE K-EFF OR ALPHA. FOR INHOMOGENEOUS PROBLEMS, THE IMPOSED SOURCE MAY BE AN ISOTROPIC VOLUME-DISTRIBUTED SOURCE OR AN ISOTROPIC SHELL SOURCE ON THE OUTER BOUNDARY.

167 FLARE  FLARE IS AN INEXPENSIVE CALCULATIONAL METHOD TO DETERMINE CORE REACTIVITY AND CORE POWER DISTRIBUTION. A SCOPING CALCULATION OF THIS TYPE IS VALUABLE IN APPRAISING THE PHYSICS CHARACTERISTICS OF PLANNED TEST MODES OF OPERATION SO THAT DETAILED ANALYSIS CAN BE RESERVED FOR THOSE CORE CALCULATIONS OF GREATER INTEREST FROM EITHER A TECHNICAL OR SAFETY STANDPOINT.

173 2DF  2DF IS A TWO-DIMENSIONAL MULTIGROUP PROGRAM WRITTEN IN FORTRAN FOR SOLVING THE NEUTRON TRANSPORT EQUATION USING THE SN METHOD. THE PROGRAM CAN DETERMINE THE REAL OR ADJOINT SOLUTION FOR X-Y, R-Z, OR R-THETA GEOMETRY. ISOTROPIC OR A FORM OF LINEAR ANISOTROPIC SCATTERING MAY BE CONSIDERED. VARIOUS BOUNDARY CONDITIONS ARE ALLOWED. THE PROGRAM ALSO CONTAINS A NUMBER OF SEARCH OPTIONS WHEREBY ONE CAN VARY DIMENSIONS OR CONCENTRATIONS TO ARRIVE AT A PREDETERMINED EIGENVALUE. A DISTRIBUTED SOURCE MAY BE SPECIFIED. A LIBRARY OF CROSS SECTIONS IS AVAILABLE ON MAGNETIC TAPE. CROSS SECTIONS MAY BE READ FROM THE LIBRARY TAPE AND/OR FROM CARDS.
THE 2DXY PROGRAM (ACC ABSTRACT 18) HAS BEEN CONVERTED FROM FLOCO TO FORTRAN 63 FOR USE ON THE CDC1604 WITH CHANGES TO PERMIT THE INCLUSION OF FIXED SOURCE TERMS FROM TDC - TERMS REPRESENTING THE EFFECTIVE NET LOSS PER UNIT VOLUME DUE TO AXIAL LEAKAGE. IN THIS MANNER, A THREE-DIMENSIONAL FLUX SYNTHESIS CODE IS ACHIEVED.

TDP IS A TWO-DIMENSIONAL LINEAR PERTURBATION THEORY CODE WHICH CALCULATES REACTIVITY COEFFICIENTS, PROMPT NEUTRON LIFETIMES, AND EFFECTIVE DELAYED FRACTIONS USING FLUXES FROM TDC CYLINDRICAL (R-Z) OR RECTANGULAR (X-Y) GEOMETRY.

THE LINEAR, TIME-INDEPENDENT, BOLTZMANN EQUATION FOR PARTICLE TRANSPORT IS SOLVED FOR THE ENERGY, SPACE, AND ANGULAR DEPENDENCE OF THE PARTICLE DISTRIBUTION IN ONE-DIMENSIONAL SLABS, CYLINDERS, AND SPHERES. INDEPENDENT SOURCE OR EIGENVALUE (MULTIPLICATION, TIME-ABSORPTION, ELEMENT CONCENTRATION, ZONE THICKNESS OR SYSTEM DIMENSION) PROBLEMS ARE SOLVED SUBJECT TO VACUUM, REFLECTIVE, OR PERIODIC BOUNDARY CONDITIONS. A COMPLETE ENERGY-TRANSFER SCATTERING MATRIX IS ALLOWED FOR EACH LEGENDRE COMPONENT OF THE SCATTERING CROSS SECTION MATRICES.

MGDSN IS A MODIFICATION OF DSN, THE ONE-DIMENSIONAL THEORY CODE DESIGNED TO ACCOMMODATE 100 GROUP ISOTROPIC MATERIAL CROSS SECTION DATA TAPES PREPARED BY CSP1 AND CSP2A.

THE TIME-DEPENDENT, MULTI-GROUP, TWO-DIMENSIONAL NEUTRON DIFFUSION EQUATIONS ARE SOLVED IN X-Y OR R-Z GEOMETRY.

SOLVES FOR THE ONE-DIMENSIONAL DISTRIBUTION OF FISSION AND FERTILE MATERIALS IN A NUCLEAR REACTOR WHICH WILL YIELD ANY DESIRED POWER DISTRIBUTION AND APPROXIMATELY RETAIN THIS DESIRED POWER DISTRIBUTION DURING THE BURN-UP HISTORY OF THE REACTOR CORE. A POISON AND POISON DISTRIBUTION SEARCH FOR A DESIRED MULTIPLICATION AND MINIMUM POWER DISTRIBUTION PERTURBATION CAN ALSO BE PERFORMED.

HFN SOLVES THE HOMOGENEOUS OR INHOMOGENEOUS ONE-DIMENSIONAL MULTIGROUP DIFFUSION EQUATION FOR ITS LOWEST EIGENVALUE AND THE CORRESPONDING DIRECT AND/OR ADJOINT EIGENVECTORS. INHOMOGENEOUS BOUNDARY CONDITIONS AND A FLEXIBLE SCATTER-TRANSFER MATRIX STRUCTURE ARE INCLUDED. OPTIONAL CALCULATIONS INCLUDE CRITICALITY SEARCHES, DETECTOR ACTIVATION TRAVERSING, AND INTEGRALS FOR PERTURBATION THEORY ANALYSIS.

MACH1 PERFORMS ONE-DIMENSIONAL MULTIGROUP DIFFUSION SOLUTIONS AND ASSOCIATED CALCULATIONS, INCLUDING CRITICALITY SEARCHES, PERTURBATION, REACTION SUMMARY, BETA EFFECTIVE, GROUP COLLAPSING, AND POINTWISE REACTION RATES AND RATIOS. SEVERAL CARD DUMPS OF COMPUTED DATA ARE AVAILABLE ON OPTION.

THE STEADY-STATE, MULTIGROUP, TWO-DIMENSIONAL NEUTRON DIFFUSION EQUATIONS ARE SOLVED IN X-Y, R-Z, AND R-THETA GEOMETRY.

CAESAR4 SOLVES THE ONE-DIMENSIONAL, MULTIGROUP DIFFUSION EQUATIONS IN ANY OF THREE GEOMETRIES AND PROVIDES A WIDE CHOICE OF BOUNDARY CONDITIONS, CRITICALITY SEARCHES, EDITS AND OTHER AUXILIARY COMPUTATIONS.

BISYN SOLVES THE TWO-DIMENSIONAL MULTIGROUP NEUTRON DIFFUSION EQUATIONS IN X-Y OR R-Z GEOMETRY USING A NONITERATIVE SYNTHESIS METHOD. THIS APPROACH IS DESIGNED TO GREATLY REDUCE THE COMPUTER COST OF RUNNING TWO-DIMENSIONAL MULTIGROUP PROBLEMS AT THE RISK OF SOME LOSS IN ACCURACY OF THE DETAILED FLUX DISTRIBUTION.
288 SNARG-1D THE PROGRAM IS DESIGNED TO
SOLVE THE ONE-DIMENSIONAL NEUTRON TRANSPORT EQUATIONS. SNARG-1D IS WRITTEN FOR THE SOLUTION OF ONE-DIMENSIONAL PROBLEMS USING THE ORDER N = 2, 4, 6, 8, 12, 16, OR 32, AND APPLICABLE TO PLANE, CYLINDRICAL AND SPHERICAL GEOMETRIES. THE REAL OR ADJOINT SOLUTION MAY BE CALCULATED AND HOMOGENEOUS OR INHOMOGENEOUS PROBLEMS MAY BE SOLVED. FOUR CRITICALITY SEARCH OPTIONS ARE PROVIDED FOR WHICH A FIXED KEFF OR ALPHA, INVERSE PERIOD, VALUE MAY BE SPECIFIED RATHER THAN THE CRITICALITY VALUE, KEFF = 1. EITHER OF TWO INHOMOGENEOUS SOLUTIONS MAY BE OBTAINED (SHELL OR DISTRIBUTED SOURCE CALCULATIONS). LINEAR ANISOTROPIC COMPONENTS OF BOTH THE SHELL SOURCE AND THE SCATTERING CROSS SECTIONS MAY BE INCLUDED. FISSION SPECTRUM MATRICES ARE ALLOWED AS BOTH MATERIAL-DEPENDENT AND INCIDENT NEUTRON ENERGY-DEPENDENT FUNCTIONS.

304 PERT4 PERT4 COMPUTES REACTIVITY COEFFICIENT TRAVERSSES IN X-Y, R-Z, OR R-THETA GEOMETRY USING THE FIRST-ORDER PERTURBATION EQUATIONS IN THE DIFFUSION APPROXIMATION. FLUX AND ADJOINT INPUT CAN BE TAKEN DIRECTLY FROM 2-D CALCULATIONS OR SYNTHESIZED FROM RADIAL AND AXIAL 1-D CALCULATIONS. THE CODE CAN ALSO BE USED TO COMPUTE ACTIVITY TRAVERSSES FOR ANY CROSS SECTION OF ANY MATERIAL, THE NEUTRON GENERATION TIME, AND THE EFFECTIVE DELAYED NEUTRON FRACTION.

312 TCSN THE LINEAR, TIME-INDEPENDENT, BOLTZMANN EQUATION IS SOLVED FOR THE ENERGY, SPACE, AND ANGULAR DEPENDENCE OF THE NEUTRON DISTRIBUTION IN ONE-DIMENSIONAL SLABS, CYLINDERS OR SPHERES OR IN TWO-DIMENSIONAL X-Y OR R-Z GEOMETRY. FIXED SOURCE OR MULTIPLICATION FACTOR (ADJOINT OR FLUX) PROBLEMS ARE SOLVED SUBJECT TO VACUUM, PLANE REFLECTIVE, ISOTROPIC REFLECTIVE OR 180 DEGREE ROTATIONALLY SYMMETRIC BOUNDARY CONDITIONS. A COMPLETE ENERGY TRANSFER SCATTERING MATRIX IS ALLOWED FOR EACH LEGENDRE COMPONENT OF SCATTERING CROSS SECTION MATRICES THROUGH P1.

319 GASP7 GASP7 CALCULATES THE ONE-DIMENSIONAL DISTRIBUTION OF FISSILE AND FERTILE MATERIALS IN A NUCLEAR REACTOR WHICH WILL YIELD ANY DESIRED POWER DISTRIBUTION DURING THE BURNUP HISTORY OF THE REACTOR CORE. A POISON AND POISON DISTRIBUTION SEARCH FOR A DESIRED MULTIPLICATION AND MINIMUM POWER DISTRIBUTION PERTURBATION CAN ALSO BE PERFORMED.

320 TEMC07 TEMC07 COMPUTES REACTOR TEMPERATURE COEFFICIENTS.

342 MC648 MC648 SOLVES THE ONE-DIMENSIONAL SLAB TRANSPORT PROBLEM WITH SLOWING DOWN FOR AN ARBITRARY SPATIAL EXTERNAL SOURCE AND ARBITRARY SCATTERING.
TWOTRAN solves two-dimensional particle transport problems in X-Y geometry. Both direct and adjoint, homogeneous (KEFF or parametric eigenvalue searches) or inhomogeneous time-independent problems are solved subject to vacuum, reflective, or input specification of boundary flux conditions. Both anisotropic inhomogeneous problems and general anisotropic scattering problems are treated.

GATT is a three-dimensional few-group neutron diffusion theory program for calculating the detailed spatial flux and power distribution for reactors with hexagonal core configuration. The program uses a uniform triangular mesh in the horizontal mesh planes and assumes a relatively simple region structure in the axial direction. It was designed to represent the special patch-type core structure of the HTGR reactor as closely as possible.
D. DEPLETION, FUEL MANAGEMENT, COST ANALYSIS, AND REACTOR ECONOMICS

55 AIMFIRE This program was designed to compare the costs of various fuel cycles. The program contains a library of fast and thermal microscopic cross sections, decay constants, and fission yields for 50 isotopes. The present version is used to investigate the economics of uranium fuel systems.

58 SIZZLE SIZZLE solves the one-dimensional, CR multigroup burnup problem in the diffusion theory approximation for fast intermediate reactors. After the initial calculation at T=0, average cross sections are computed for further calculations using one to six energy groups. Criticality may be maintained by use of a concentration search. The concentration of the various isotopes is permitted to vary only from region-to-region. Chains included are TH232, U238, and a fission product poison chain.

99 DUC DIMENSIONAL BURNOUT(DDB) The five-group, two-dimensional, neutron diffusion equations in cylindrical geometry are solved with burnout options and control rod search options.

117 FEVER FEVER performs one-dimensional few-group depletion calculations. Options are available to adjust control poisons in various regions of the reactor, self-shielding of lumped poisons, and to calculate hot maximum and cold shut-down multiplication.

134 NUCY The calculation of nuclide concentrations at a point in a reactor at successive time intervals, with exposure to a two-group neutron flux. Infinite system criticality is calculated.
NUCLEAR FUEL CYCLE COSTS IN DOLLARS PER YEAR ($/YR), AND IN MILLS PER KILOWATT-HOUR (MILLS/KWHR) ARE COMPUTED AND TABULATED FOR EACH REGION OF A MULTIREGION REACTOR CORE, ON THE BASIS OF AEC-LEASED NUCLEAR FUEL MATERIAL AND OF PRIVATELY-OWNED NUCLEAR FUEL MATERIAL. FUEL CYCLE COSTS ARE COMPUTED SEPARATELY FOR EACH REGION OR ZONE, FOR CORE DESIGNS OF ANY CONFIGURATION OR COMBINATION OF MATERIAL DEPLETION OR ENRICHED URANIUM FUEL OR OTHER SPECIAL NUCLEAR MATERIAL. PRINTED OUTPUT OF THE PROGRAM INCLUDES (A) DETAILED FUEL CYCLE COSTS FOR EACH ZONE IN TABULAR FORM, FOR AEC-LEASED AND PRIVATELY-OWNED NUCLEAR FUEL MATERIAL, RESPECTIVELY, (B) A SUMMARY TABULATION OF NUCLEAR FUEL COSTS FOR ALL REGIONS OF THE COMPLETE CORE, INCLUDING FIXED CHARGES ON WORKING CAPITAL REQUIRED FOR CORE FABRICATION AND FOR NUCLEAR FUEL MATERIAL, (C) A SUMMARY TABULATION OF CERTAIN COMPUTED PERFORMANCE AND ECONOMIC DATA, VIZ., AVERAGE RESIDENCE TIME, ANNUAL FUEL THROUGHPUT, UNIT ELECTRICAL ENERGY YIELD, ANNUAL POWER GENERATION, AND CORE FABRICATION COSTS, AND (D) A TABULATION OF ALL INPUT DATA FOR ALL REGIONS. THE PRINTING OF DATA DESCRIBED IN (A) AND (D) ABOVE IS OPTIONAL WITH PROGRAM USE.

ISOTOPES THIS PROGRAM CAN BE USED TO CALCULATE FOR ANY NEUTRON FLUX THE OPTIMUM TIME OF IRRADIATION FOR MAXIMUM YIELD, THE SPECIFIC ACTIVITY OF THE PRODUCT ISOTOPE IN CURIES PER GRAM OF TARGET MATERIAL, AND THE COMBINED SPECIFIC ACTIVITY OF THE TARGET AND PRODUCT ISOTOPES. THE PRODUCT ISOTOPE MAY BE PRODUCED BY ANY SIMPLE REACTION SUCH AS (N,GAMMA), (N,P), (N,2N), ETC., OR IT MAY BE PRODUCED BY DECAY OF A PARENT ISOTOPE.

ISOCRUNCH CAN BE USED TO COMPUTE THE AMOUNT OF EACH ISOTOPE IN A REACTION AND DECAY CHAIN FOR ANY SPECIFIED NEUTRON FLUX AND TIME, TO SUM THE CONTRIBUTIONS OF VARIOUS CHAINS TO THE SAME ISOTOPE, TO GRAPH ON AN ASSOCIATED ELECTROROTTER OR CALCONE THE YIELD OF AN ISOTOPE VS. TIME FOR A GIVEN FLUX, AND TO FIND THE OPTIMUM TIME FOR MAXIMUM YIELD OF AN ISOTOPE IN A CHAIN. THE PROGRAM DOES NOT TAKE INTO ACCOUNT THE SELF-SHIELING OF A TARGET IN A REACTOR OR THE DEPENDENCE OF REACTION CROSS SECTIONS ON NEUTRON ENERGY WHICH CAN BE HANDLED BY ADJUSTING THE INPUT DATA.

RELLOE-FEVER A FEW-GROUP, 1-D DEPLETION CALCULATION WHICH ALLOWS FUEL IN VARIOUS STAGES OF IRRADIATION TO BE HOMOGENIZED INTO THE SAME REGION FOR PURPOSES OF THE DIFFUSION CALCULATION BUT FOLLOWS THE DEPLETION OF EACH OF THE SUB-REGIONS SEPARATELY. THE CALCULATION MAY BE INTERRUPTED PERIODICALLY FOR REFUELING ONE OR MORE REGIONS. RECYCLING IS OPTIONAL AND THERE IS NO LIMIT TO THE NUMBER OF REFUELINGS WHICH MAY BE PERFORMED. A CONTROL POISON SEARCH IS AVAILABLE AND CONCENTRATION DEPENDENT SELF-SHIELING FACTORS MAY BE APPLIED TO ONE LUMPED POISON.
REVISED-GAD

This infinite-medium depletion program performs fuel cycle calculations on reactors employing partial refueling. The burn-up of up to 12 discrete fuel compositions (regions or stages of irradiation) may be followed simultaneously. The burn-up calculation may be interrupted periodically to remove the contents and to refuel one or more regions. Fuel may be partially or completely recycled.

WAMPUM

This program calculates fuel cycle costs on a detailed basis, using results of nuclear depletion calculations and certain specified economics assumptions. The purpose is to provide a measure of performance for comparing or optimizing fuel cycles and associated reactor core and fuel element characteristics.

TEMCO

TEMCO computes reactor temperature coefficients.

OPUS

The code generates a flow network equivalent to a gas-cooled nuclear power plant of specified electrical output (in the range of 100 to 1000 MW) according to input data and programmed rules, proceeds to evaluate the plant performance and price of the turbogenerator set (according to General Electric price data), and prints as a result a coded list of all plant components and a detailed performance map.

STMGEN

STMGEN will be used in a plant optimization program. This code determines the area of each section of a steam generator required to satisfy the design conditions of heat transfer, pressure drop and maximum tube temperature constraints. The cost of the generator is computed as a function of the total heat transfer area, the non-productive tube length required to connect the headers, plus the cost of the headers.

RAC2

This program calculates the fission product activity distributions in a high temperature gas-cooled reactor system.

ASSAULT

Assault is a multigroup, two-dimensional reactor depletion program. Given nuclide concentrations and microscopic cross sections, the steady-state multi-region, multigroup diffusion equations are solved in one or two dimensions over a finite-difference mesh point system. Calculated neutron fluxes are then used to determine nuclide concentrations after a specified period of exposure. These calculations are repeated for a specified number of time-steps.
Gargoyle is an infinite medium depletion code for fuel cycle calculations on reactors employing partial refueling. The burnup of up to 12 discrete fuel compositions (regions or stages of irradiation) may be followed simultaneously. The burnup calculation may be interrupted periodically to remove the contents of and to refuel one or more regions. Fuel may be completely or partially recycled. A control poison search may be performed at each time-step. Feed fuel searches are permitted at the end of each burnup cycle before refueling. Concentration dependent self-shielding factors may be applied to any nuclide except moderators. The nuclide scheme employed, although not completely general, is flexible.

DTF-Burn is a one-dimensional, multigroup burnup code based on transport theory. A different type of eigenvalue calculation (multiplication, period, nuclide concentration, zone thickness, or system dimension) can be performed at the beginning of each of a specified set of time-steps. This feature permits the simulation, in a single problem run, of a system in which various control methods are used as a function of time. The concentration of the various nuclides is permitted to vary from region-to-region or point-by-point. Nuclides considered are U235, U236, U238, NP239, Pu239, Pu240, Pu241, Pu242, and FP, a pseudo fission product pair.

PCQ7 solves few-group neutron diffusion-depletion problems in one, two, and three dimensions. Adjoint solutions are also available and two overlapping thermal groups may be used in one and two-dimensional problems. Either pointwise or regionwise depletion may be performed using the harmony depletion system. The geometry may be rectangular, cylindrical, or spherical in one dimension, rectangular, cylindrical, or hexagonal in two dimensions, and rectangular or hexagonal in three dimensions. All geometries provide for variable mesh spacing in all dimensions. Zero flux, zero current, and rotational symmetry boundary conditions are available, and boundary value problems may be solved by specifying the flux values on one or more boundaries.
301 FREVAP6

The FREVAP type of code for estimating the release of longer-lived metallic fission products from HTGR fuel elements has been developed to take into account the combined effects of the retention of metallic fission products by fuel particles and the rather strong absorption of these fission products by the graphite of the fuel elements. Release calculations are made on the basis that the loss of fission product nuclides such as strontium, cesium, and barium is determined by their evaporation from the graphite surfaces and their transpiration induced by the flowing helium coolant. The code is devised so that changes of fission rate (fuel element power), fuel temperature, and graphite temperature may be incorporated into the calculation. Temperature is quite important in determining release because, in general, both release from fuel particles and loss by evaporation (transpiration) vary exponentially with the reciprocal of the absolute temperature.

302 GAFFE

A zero-dimensional calculation of feed fuel requirements is performed to produce a specified end of cycle multiplication factor for the equilibrium fuel cycle, given feed composition, length of cycle and reactor power. It is alternately possible to compute cycle length or feed enrichment. The code is a survey tool which assumes periodic refueling and permits complete or partial recycling of materials. The segregated fuel concept can be handled within the framework of the calculation.

313 CINDER (M01C2)

CINDER is a nine-group, one-point depletion and fission product program based on the evaluation of a general analytical solution of nuclides coupled in any linear sequence of radioactive decays and neutron absorptions in a specified neutron flux spectrum. The desired depletion and fission product chains are all physical data and are specified by the problem originator. The program computes individual nuclide number densities, activities, disintegration rates, and macroscopic and barns/fission poisons at each time-step as well as selected summaries of these data.

314 NAP

NAP calculates the spectrum and spatial distribution in one dimension of activation gamma rays following neutron irradiation.

318 FEVER7

FEVER7 performs a multigroup, 1-dimensional depletion calculation which allows fuel in various stages of irradiation to be homogenized into the same region for purposes of the diffusion calculation but follows the depletion of each of the subregions separately. The calculation may be interrupted periodically for refueling one or more regions. Recycling is optional and there is no limit to the number of refuelings which may be performed. A control poison search is available and concentration dependent self-shielding factors may be applied to a number of lumped poisons.
ISOSEARCH

This program was developed to calculate the unknown reaction cross section, flux value, or product activity in an isotope-production scheme consisting of two or three nuclides.

2DB

2DB is a flexible, two-dimensional (X-Y, R-Z, R-Theta, hex geometry) diffusion code for use in fast reactor analyses. The code can be used to:

(A) Compute fuel burnup using flexible material shuffling scheme,
(B) Perform criticality searches on time absorption (alpha), material concentrations, and region dimensions using a regular or adjoint model. Criticality searches can be performed during burnup to compensate for fuel depletion,
(C) Compute flux distributions for an arbitrary extraneous source.

PDQS

The few-group two-dimensional neutron diffusion equations are solved. Up to five groups may be used with scattering allowed between adjacent groups. In addition, depletion problems may be solved with PDQS.

GAUGE

The two-dimensional few-group neutron diffusion theory equations for a uniform triangular mesh are solved to obtain the multiplication factor and the spatial flux and power distribution of reactors with hexagonal core configuration. Complete reactor life histories with partial refueling at a number of reload time points can be calculated. At each discrete time point a control rod search may be performed to maintain criticality at all times. The depletion scheme of all burnable nuclides is specified by the user at execution time. Three modes of operation are possible - (1) straight burnup calculation, (2) control rod criticality search, allowing the adjustment of a number of control rod banks according to a prescribed rod sequencing scheme, and (3) a series of static calculations with insertion of rods into fixed prescribed positions.

POWERCO

POWERCO calculates the cost of electricity produced by nuclear power stations, assuming all cash expenses such as investment and fuel costs, operating expenses, and taxes are known. The power cost is held constant throughout the project life.
CINCAS is a nuclear fuel cycle cost code which may be used for either engineering economy predictions of fuel cycle costs or for accounting forecasting of such costs. Features of CINCAS include:

1. Monthly calculation of dollar costs and mass inventory on a batch and case basis for each month of a period which is usually defined as (but not restricted to) beginning with the delivery of fuel to the reactor site and ending with the withdrawal of fuel from the reactor.

2. A general formula for the unit price of enriched uranium which allows for variable feed and tails enrichments, costs of feed, chemical conversion, separative work, and losses in conversion and fabrication.

ISOGEN calculates radioisotope generation and decay, using two-group neutron cross sections.

RAPFU calculates equilibrium fuel cycle isotopics in fast breeder reactors. The recycled plutonium is permitted to have different isotopic compositions in two different core zones, and several recycle schemes are available as options. Output data includes the initial, average, and discharged fuel isotopic concentrations for each region of the core zones and the blankets, breeding ratio, doubling time, and (optionally) fuel costs calculated using simplified relationships.
E. SPACE-INDEPENDENT KINETICS

121 AIREK3 AIREK3 FINDS THE NUMERICAL SOLUTION TO THE SPACE INDEPENDENT REACTOR KINETICS EQUATIONS BASED ON THE METHOD DEVELOPED BY E. R. COHEN. INPUT AND OUTPUT ARE SIMPLIFIED AND THE POWER, INVERSE PERIOD, FEEDBACKS, AND PRECURSORS ARE DISPLAYED GRAPHICALLY.

122 SNAPKIN5/SNAPKIN5A SNAPKIN5 PROVIDES A ONE-REGION TIME-DEPENDENT CALCULATION OF POWER, ENERGY, TEMPERATURE, REACTIVITY, INVERSE PERIOD, AND HYDROGEN LOSS IN A SNAP REACTOR AFTER A PERTURBATION FROM GIVEN INITIAL CONDITIONS. SNAPKIN-5A, IN ADDITION, WEIGHTS POWER, HEAT CAPACITY, AND REACTIVITY IMPORTANCE FOR TWENTY-FIVE OR FEWER REGIONS.

135 TRAFICORPORATION EXPERIMENTAL FREQUENCY RESPONSE DATA OBTAINED FROM A LINEAR DYNAMIC SYSTEM IS PROCESSED TO OBTAIN THE TRANSFER FUNCTION AS A RATIO OF TWO FREQUENCY-DEPENDENT POLYNOMIALS. THE TRANSFER FUNCTION MAY HAVE NON-MINIMUM PHASE.

163 AIROS AIROS SOLVES THE SPACE-INDEPENDENT REACTOR KINETICS EQUATIONS AND PROVIDES FOR THE DETERMINATION OF REACTIVITY BY SOLVING IN ADDITION THE DISCRETIZED EQUATIONS WHICH REPRESENT THE SPATIAL HEAT AND MASS TRANSFER MODEL FOR SEVERAL FUEL CHANNELS. IN ADDITION, VARIATION OF THE FILM COEFFICIENT WITH FLOW IS ACCOUNTED FOR AS WELL AS THE PROVISION FOR FLOW DECAY AND AFTERGLOW HEATING. SCRAMS CAN BE INITIATED BY DELAYED SIGNALS FROM INSTRUMENTS WHICH SENSE ANY QUANTITY CALCULATED, E.G., POWER, INVERSE PERIOD OR TEMPERATURE. GENERALIZED FEEDBACK EQUATIONS ARE USED TO PROVIDE FLEXIBILITY IN THE MODELS THAT REPRESENT MULTICHANNEL HEAT TRANSFER INCLUDING CONDUCTION AND CONVECTION, ENERGY, PRESSURE AND OTHER PHENOMENON. THE REACTIVITY EQUATION IS ALSO GENERALIZED. THE REACTIVITY FEEDBACK COEFFICIENTS CAN BE CONSTANT OR VARY AS THE SQUARE ROOT OR RECIPROCAL OF TEMPERATURE. FURTHERMORE ANY FEEDBACK VARIABLE CAN BE USED TO INITIATE A REACTIVITY SCRAM, EACH WITH A UNIQUE DELAY TIME.

168 INVERSE KINETICS (R102) GIVEN THE SPACE-INDEPENDENT, ONE ENERGY GROUP REACTOR KINETICS EQUATIONS AND THE INITIAL CONDITIONS, THIS PROGRAM DETERMINES THE TIME VARIATION OF REACTIVITY REQUIRED TO PRODUCE THE GIVEN INPUT OF FLUX-TIME DATA.

188 CMPXMAT A SYSTEM OF N LINEAR EQUATIONS DERIVED FROM THE LAPLACE TRANSFORM OF A SET OF LINEARIZED DIFFERENTIAL EQUATIONS IS SOLVED FOR AMPLITUDE AND PHASE ANGLE AS A FUNCTION OF FREQUENCY.
R101 SOLVES THE SPACE-INDEPENDENT, ONE-ENERGY GROUP REACTOR KINETICS EQUATIONS TO DETERMINE THE TIME VARIATION OF NEUTRON DENSITY GIVEN SPECIFIED INITIAL CONDITIONS. ANY OF FOUR PROGRAMMED REPRESENTATIONS OF EXCESS REACTIVITY CAN BE SELECTED.

GASA DETERMINES THE STABILITY OF ANY PHYSICAL SYSTEM WHOSE MOTION IS DESCRIBABLE BY A SET OF FIRST ORDER LINEAR DIFFERENTIAL EQUATIONS.

BLOOST6 COMBINES A REACTOR SPACE-INDEPENDENT KINETICS CODE WITH A TWO-DIMENSIONAL HEAT TRANSFER CODE, AND A TIME-DEPENDENT SPHERICAL GEOMETRY HEAT TRANSFER ROUTINE FOR FUEL PARTICLES. THE CODE IS APPLICABLE TO PROBLEMS FOR WHICH THE SPACE-INDEPENDENT FORM OF THE REACTOR KINETICS EQUATION IS APPLICABLE.

GAPOTKIN IS A POINT KINETICS CODE THAT SOLVES THE SPACE-INDEPENDENT KINETICS EQUATIONS FOR A VERY GENERAL FORM OF THE REACTIVITY FUNCTION.

AIROS2A SOLVES THE SPACE-INDEPENDENT REACTOR KINETICS EQUATIONS AND PROVIDES FOR THE DETERMINATION OF REACTIVITY BY SOLVING IN ADDITION THE DISCRETIZED EQUATIONS THAT REPRESENT THE SPATIAL HEAT AND MASS TRANSFER MODEL FOR SEVERAL FUEL CHANNELS. IN ADDITION, VARIATION OF THE FILM COEFFICIENT WITH FLOW IS ACCOUNTED FOR ALONG WITH THE PROVISION FOR FLOW DECAY AND AFTERGLOW HEATING. SCRAMS CAN BE INITIATED BY DELAYED SIGNALS FROM INSTRUMENTS THAT SENSE ANY QUANTITY CALCULATED, E.G., POWER, INVERSE PERIOD OR TEMPERATURE. GENERALIZED FEEDBACK EQUATIONS ARE USED TO PROVIDE FLEXIBILITY IN THE MODELS THAT REPRESENT MULTICHANNEL HEAT TRANSFER INCLUDING CONDUCTION AND CONVECTION, ENERGY, PRESSURE AND OTHER PHENOMENON SUCH AS FUEL MELTING, COOLANT BOILING AND VOIDING BURN-OUT. THE REACTIVITY EQUATION IS ALSO GENERALIZED. THE REACTIVITY FEEDBACK COEFFICIENTS CAN BE CONSTANT OR VARY AS THE SQUARE ROOT OR RECIPROCAL OF TEMPERATURE. FURTHERMORE, ANY FEEDBACK VARIABLE CAN BE USED TO INITIATE A REACTIVITY SCRAM, EACH WITH A UNIQUE DELAY TIME. AN INPUT GENERATOR COMPUTES THE CONDUCTION AND CONVECTION COEFFICIENTS FOR AN N X M NODAL, MULTICHANNEL SYSTEM USING BUILT-IN TABLES OF SPECIFIC HEAT, DENSITY, CONDUCTIVITY AND VISCOSITY FOR THE COMMON FUEL, STRUCTURE AND COOLANT MATERIALS, AND PERFORMS AN INITIAL TEMPERATURE CALCULATION. THE FILM COEFFICIENTS MAY BE SPECIFIED OR CALCULATED USING LYNCS EQUATION OR THE DITTUS-BOELTER EQUATION.
BLAST has been developed to study accident conditions in critical and subcritical thermal multiplying systems. The programme computes the time behaviour of the thermal neutron density and the system temperature following a step change in reactivity. The integrated thermal neutron density is also computed, from which the total number of fissions during an excursion may be obtained.
F. SPACE-TIME KINETICS, COUPLED NEUTRONICS-HYDRODYNAMICS- THERMODYNAMICS, AND EXCURSION SIMULATIONS

102 AX1

AX1 PERFORMS A COUPLED NEUTRONICS-HYDRODYNAMICS CALCULATION. GIVEN A SPHERICALLY SYMMETRIC, SUPERPROMPT CRITICAL SYSTEM, THE PROGRAM COMPUTES THE VARIATION IN TIME AND SPACE OF THE SPECIFIC ENERGY, TEMPERATURE, PRESSURE, DENSITY AND VELOCITY.

129 CONEC

CONEC IS A COUPLED NEUTRONIC-ELASTICITY CODE DESIGNED FOR APPLICATION TO PULSED FAST REACTOR SYSTEMS. IT IS A ONE-DIMENSIONAL CALCULATION, CAPABLE OF DEALING WITH SOLID SPHERES OR SPHERICAL SHELLS. SPECIFICALLY, CONEC CALCULATES THE ALPHA OF A SYSTEM, THE TEMPERATURE DISTRIBUTION, THE RADIAL AND TANGENTIAL STRESS DISTRIBUTIONS, AND FROM THESE THE ACCELERATIONS, VELOCITIES, AND DISPLACEMENTS THROUGHOUT THE SYSTEM.

145 WEAK EXPLOSION PROGRAM

THIS PROGRAM PERFORMS A COUPLED NEUTRONICS-HYDRODYNAMICS CALCULATION FOR A SPHERICALLY SYMMETRIC REACTOR CORE WITH A GIVEN COMPOSITION TO OBTAIN THE TIME-DEPENDENT ENERGY RELEASE THAT RESULTS FROM THE INSERTION OF REACTIVITY AT A GIVEN RATE. THE BASIC USE FOR THIS PROGRAM IS FOUND IN ANALYSIS OF FAST REACTOR CORE COLLAPSE ACCIDENTS.

153 HATCHET

HATCHET IS A MAJOR MODIFICATION OF THE AX1 CODE DESIGNED TO STUDY BURST CHARACTERISTICS OF A SUPERPROMPT CRITICAL, CONCENTRIC SHELL, PULSED REACTOR. IT COMPUTES SPECIFIC ENERGY, TEMPERATURE, PRESSURE, DENSITY AND VELOCITY VARIATIONS AS A FUNCTION OF TIME AND SPACE. THE CODE ALSO COMPUTES REACTIVITY AS A FUNCTION OF INVERSE REACTOR PERIOD, POWER, THE TOTAL AND KINETIC ENERGIES, AND THE POSITION OF THE SHELLS WHICH COMPRISSE THE SYSTEM.
FCRE/FORE2 FORE calculates reactor power and temperatures of fuel, coolant, clad, and structure as functions of time in response to a programmed reactivity insertion specified as a series of ramps. Temperature profiles are computed at specified axial positions for an average channel and for the peak power (central) channel. The heat of fusion accompanying fuel melting is taken into account. Feedback reactivity mechanisms that respond to changes of temperatures include the fuel Doppler effect and thermal expansion of the core (and axial blanket or reflector) materials. FORE2 is a coupled thermal hydraulics-point kinetics digital computer code designed to calculate significant reactor parameters under steady-state conditions, or as functions of time during transients. The transients may result from a programmed reactivity insertion or a power change. Variable inlet coolant flow rate and temperature are considered. The code calculates the reactor power, the individual reactivity feedbacks, and the temperature of coolant, cladding, fuel, structure, and additional material for up to seven axial positions in three channel types which represent radial zones of the reactor. The heat of fusion accompanying fuel melting, the liquid metal voiding reactivity, and the spatial and the time variation of the fuel cladding gap coefficient due to changes in gap size are considered.

PUMP IMPELLER DESIGN SYSTEM This system has been developed for the design and development of high speed turbo-machinery for pumping high temperature liquid metals. After initializing the analysis and design effort by establishing the pump conditions of service the codes are used to define the pump impeller design. In all programs, real fluid effects are not considered. The machining code produces a paper tape for use in the actual fabrication of the impeller.

AX-TNT AX-TNT solves (A) the coupled hydrodynamic, thermodynamic and neutronic equations which describe a spherical, super-prompt critical reactor system during an excursion, (B) the coupled equations of motion, and ideal gas equation of state for the detonation of a spherical charge in a gas.

WIGL2 WIGL2 is a one-dimensional two-group space-time diffusion theory program with zero, one, or six delayed neutron groups. The program will treat slab, cylindrical, and spherical geometries and includes non-boiling heat transfer. It accounts for xenon feedback and feedback effects due to fuel and coolant temperature. Control rod motion and control system feedback based on total core power or outlet coolant temperature can be simulated. Transients may be excited by prescribed changes in inlet coolant temperature, coolant flow rate, or rod position.
293 MARS

MARS performs a coupled neutronics-hydrodynamics calculation for a finite cylinder core with concentric regions of different compositions and characteristics to obtain the time-dependent energy release that results from the insertion of reactivity according to a prescribed program. The basic use for this program is found in analysis of fast reactor core collapse accidents. The reactivity changes due to material displacement is determined by the use of perturbation theory and the pressure generation for the hydrodynamics calculation is determined by the use of an exponential form for the saturated vapor pressure curve. The Doppler effect is determined by the use of a flexible T^*(-N) form where N is a parameter that is specified at the time of execution. The external reactivity insertion can be in the form of a step, a linear ramp or a parabolic ramp, with a limit on the amount of reactivity which can be inserted being a definable quantity. Tabulated values of the power and material worth distributions are accepted as input. The characteristics of each region (density, specific heat, Doppler coefficient, equation of state, geometric limits) are specified independently for each region to provide flexibility in the treatment of cores of unusual configuration. The neutron kinetics calculations are done with a point reactor model and delayed neutrons are used.

309 TSN

The time-dependent neutron transport equation is solved. Energy deposition is allowed to cause variation in the neutron cross sections for the core region, in the core density, and in core height. The result is a kinetics calculation including spatial dependence both in feedback effects and in neutron density. The results are summarized in a manner similar to point-kinetics codes, and spatial distributions are also given. Graphical summaries of the significant variables and spatial distributions are given. In the IBM360 version, either the neutron yield from fission or the thickness of a specified zone can be changed as an independent function of time as specified by the user. This provides two ways of allowing explicit reactivity variation with time.

310 GAKIN

The multigroup, 1-dimensional, time-dependent diffusion theory equations are solved in slab, cylindrical or spherical geometry with delayed neutrons taken into account. An arbitrary scattering matrix is allowed, together with a piece-wise, linear, time-dependent, inhomogeneous source term. Feedback is available from time-dependent cross section changes and build-up in xenon. The time integration is divided into time zones with unique feedback and source data for each zone.
TWIGL solves the two-dimensional, two-group, space-time neutron diffusion equations in rectangular or cylindrical geometry in the presence of temperature feedback. The neutron diffusion and delayed precursor equations are differenced in both space and time. The thermal-hydraulic description is based on a no-boiling, one-pass model formulated in terms of regionwise-averaged coolant and fuel metal temperatures.

RAUMZEIT solves systems of one-dimensional, time-dependent, multigroup diffusion-type equations using either of two treatments of the time dependence, finite differencing or the time-integrated approach (see Reference 2).

GAKIT solves the multigroup, one-dimensional, time-dependent diffusion theory kinetics equations including delayed neutron effects and temperature feedback based on two-dimensional heat transfer calculations. For the one-dimensional multigroup kinetics equations an arbitrary scattering matrix and arbitrary fission transfer are allowed, and plane, cylindrical, or spherical geometry might be used. A piecewise linear time-dependent inhomogeneous source can be specified. Feedback is available from xenon buildup and temperature dependence of cross sections. The heat transfer calculation is performed for two-dimensional R-Z fuel element models assuming predetermined axial power shape functions and time-dependent power amplitudes obtained from the one-dimensional kinetics calculations. For the fuel elements average fuel and moderator temperatures are calculated which determine, based on tables, the temperature-dependent cross sections. Transients may be introduced by step changes of cross sections, by piecewise linear time-dependent cross sections (ROC withdrawal accidents), by step changes of the flow rates or by step changes of the coolant inlet temperatures.

NOWIG is a program for solving the one-dimensional two-group neutron diffusion and delayed precursor equations using a shape-specified point kinetics approximation. Feedback due to changes in the fuel metal temperature and coolant density is accounted for by using a model which is identical with that used in the WIGL2 (ACC ABSTRACT 274) program.
G. RADIOLOGICAL SAFETY, HAZARD AND ACCIDENT ANALYSIS

47 CLOUD

The cloud program calculates the external gamma-ray dose rate and total integrated dose resulting from the continuous release of radioactive materials to the atmosphere.

172 AISITE2/R153

Aisite2 is largely based on methods proposed by the AEC in TID-14844 but differs in certain of the assumptions and models. The code automatically varies any one of 46 parameters such as reactor power, building leak rate, iodine clean-up rate, and halogen filter efficiency, computing the exclusion area, and low population boundary zones as functions of that parameter. The edit includes dose vs. distance data, fractional contribution by isotope group to the inhalation dose, and critical distances providing both printed and graphical data. Three MCELSs are available for fission product release with up to 4 levels of containment.

196 CURIE/DOSE/THUNDERHEAD

Curie calculates the fission product inventory produced in a reactor during constant power operation and releases different percentages of the noble gases, halogens, and particulates to the atmosphere at reactor shutdown. Dose calculates the total dose to 13 internal body organs resulting from inhalation of the passing radioactive cloud. Thunderhead calculates the external cloud gamma exposure dose from the released fission products.

200 SATURATED BLOWDOWN

Calculation of local pressures, mass flow rates, fluid qualities, specific volumes, temperatures and local heat transfer coefficients at wetted walls, during blowdown of the reactor primary coolant loop.

228 PRECON

The purpose of the program is to determine, for a gas-cooled reactor, the containment pressure as a function of time, after ruptures have been assumed to occur in the primary system or in conjunction with steam generator ruptures.

229 ELBOW

Deflections, restraint forces and stresses are determined in a fuel element subjected to asymmetric power generation, circumferentially non-uniform coolant temperature distribution, and asymmetric fast neutron flux damage.

RSAC GENERATES A FISSION PRODUCT INVENTORY FROM A GIVEN SET OF REACTOR OPERATING CONDITIONS AND THEN COMPUTES THE EXTERNAL GAMMA DOSE, THE DEPOSITION GAMMA DOSE, AND/OR THE INHALATION-INGESTION DOSE TO CRITICAL BODY ORGANS AS A RESULT OF EXPOSURE TO THESE FISSION PRODUCTS. PROGRAM OUTPUT INCLUDES REACTOR OPERATING HISTORY, FISSION PRODUCT INVENTORY, DOSAGES, AND INGESTION PARAMETERS.

TRANS-FUGUE1 IS A TRANSIENT SINGLE CHANNEL, TWO-PHASE FLOW, AND HEAT TRANSFER CODE FOR ANALYSIS OF POSTULATED REACTOR INCIDENTS INVOLVING BOILING. THE CODE IS BASED ON A HOMOGENEOUS HYDRODYNAMIC MODEL WHICH ASSUMES EQUAL PHASE TEMPERATURES AND VELOCITIES. IT ASSUMES VAPOR GENERATION TO BE HEAT TRANSFER LIMITED ONLY, AND CALCULATES AXIAL DISTRIBUTION OF VELOCITY, VOID FRACTION, PRESSURE, COOLANT TEMPERATURE, AND FUEL TEMPERATURE. TRANSIENTS SIMULATING POWER EXCURSIONS, LOSS OF FLOW, LOSS OF PRESSURE AND CHANNEL PLUGGING CAN BE STUDIED.

WHAM IS USED TO CALCULATE PRESSURE, VELOCITY, AND FORCE TRANSIENTS IN THE LIQUID FILLED PIPING NETWORKS. IT CAN BE APPLIED TO MULTI-LOOP, COMPLEX PIPING NETWORKS CONSISTING OF ANY NUMBER OF DEAD ENDS, ELBOWS, ORIFICES, MULTIPLE BRANCH TEES, CHANGES OF FLOW PASSAGE CROSS SECTIONS, CHECK VALVES, PUMPS, PRESSURIZERS OR TANKS, AND EXIT VALVES. HYDRAULIC LOSSES ARE CONSIDERED. TRANSIENTS CAN BE INITIATED EITHER, BY CLOSURE OR BY OPENING OF ONE OR MORE EXIT VALVES (OR BY THE SYSTEM RUPTURE AT ONE OR MORE PLACES), OR BY A PRESCRIBED CHANGE IN THE TANK (OR PRESSURIZER) PRESSURE, OR BY THE PUMP POWER FAILURE. WATER-HAMMER ANALYZES HYDRAULIC TRANSIENTS IMPOSED BY THE SUDDEN DISCHARGE OF LIQUID FROM THE PRIMARY COOLANT LOOP, BEFORE ANY GENERATION OF VAPOR HAS TAKEN PLACE. EFFECTS OF HYDRAULIC LOSSES ARE NEGLECTED, THE LOOP PRESSURIZER IS ASSUMED TO REMAIN CONSTANT AT INITIAL PRESSURE AND THE PRESENCE OF A PUMP IS IGNORED. IT WAS DESIGNED FOR THE EARLY BLOWDOWN ANALYSIS FOR LOFT.
R284 MC555 (ACT1) THIS PROGRAM WILL PREDICT THE PRESSURE TRANSIENT IN THE CONTAINMENT VESSEL AS A FUNCTION OF TIME AS A RESULT OF A MAJOR RUPTURE IN THE PRIMARY SYSTEM OF A PRESSURIZED WATER REACTOR. MEANS ARE AVAILABLE FOR INTRODUCING WATER AND HEAT INTO THE CONTAINMENT VESSEL. THERE ARE ALSO PROVISIONS FOR SPECIFYING VARIOUS TYPES OF HEAT SINKS FOR ENERGY ABSORPTION. THE TEMPERATURE DISTRIBUTION IN THESE HEAT SINKS IS CALCULATED AS A FUNCTION OF TIME. DISCHARGE RATES OF THE CONTAINMENT MAY ALSO BE DETERMINED AS A FUNCTION OF BOTH TIME AND THE INTERNAL PRESSURE.

R295 FLASH2 FLASH2 DETERMINES THE TRANSIENT RESPONSE OF A WATER REACTOR TO A LOSS-OF-COOLANT ACCIDENT OR SEVERE VARIABLE PRESSURE OPERATION.


328 NURLOC-1.0 NURLOC-1.0 PERFORMS CORE/PRESSURE VESSEL THERMAL ANALYSIS FOR A NUCLEAR REACTOR LOSS-OF-COOLANT ACCIDENT.

330 ECCSA1 ECCSA1 PREDICTS THE THERMAL AND HYDRAULIC PERFORMANCE OF REACTOR CORE CHANNELS DURING A LOSS-OF-COOLANT ACCIDENT AND SUBSEQUENT EMERGENCY CORE COOLING INJECTION.
CHEMLCC2 is used to compute the extent of core heating and metal-steam reaction following a hypothetical loss-of-coolant accident, including emergency core-cooling failure, in a water-cooled reactor. The program includes the effects of heat production by decay heating in the fuel as well as by steam-metal chemical reaction, heat transfer from the core to the bottom and top grid-support plates and to the wall surrounding the core, and methods of calculating the effects of core movement and material transfer on reaching slumping temperature.

RELAP2 calculates flow, mass inventories, temperatures, pressures, reactivities, and transient power for the primary system of a water reactor during a reactivity or a loss-of-coolant accident. Although retaining the simplified geometry (three volumes plus a core region) of the previous RELAP program, many improvements and extensions have been made. The geometry can be made to approximate either a pressurized or a boiling water reactor system. The core is treated as a two-point model for power generation, heat transfer, and reactivity feedbacks and as a one-point model for the reactor kinetics, pressure balances, and flow balances. Also, RELAP2 can be used for reactor system safety studies including large reactivity excursions as well as the loss-of-coolant and pump-failure accidents.
STEADY-STATE AND TRANSIENT HEAT TRANSFER

152 ARGUS (RE248)  
THIS PROGRAM CALCULATES TRANSIENT TEMPERATURES IN A CONCENTRIC, CYLINDRICAL CONFIGURATION. UP TO 25 CONCENTRIC REGIONS ARE ALLOWED, EACH CONTAINING EITHER A STATIONARY (SOLID OR NON-FLOWING LIQUID) OR TURBULENTLY FLOWING (LIQUID OR GAS) MATERIAL. ANY STATIONARY REGION CAN HAVE SPATIAL- AND TIME-DEPENDENT HEAT GENERATION. TEMPERATURES ARE CALCULATED AT NODE POINTS EQUALLY-SPACED WITHIN A REGION. FILM COEFFICIENTS ON FLOWING REGION BOUNDARIES ARE CALCULATED BY THE PROGRAM. TIME-DEPENDENT COOLANT VELOCITIES ARE PERMITTED. THE HEAT SOURCE IS ASSUMED TO BE ANGULAR INDEPENDENT. AXIAL HEAT CONDUCTION IS NEGLECTED, BUT AXIAL HEAT TRANSPORT DUE TO MATERIAL MOTION IS CONSIDERED IN THE FLOWING REGIONS.

155 PTH1  
THIS PROGRAM CALCULATES CONTAINMENT SHELL PRESSURE-TEMPERATURE HISTORY RESULTING FROM AQUEOUS COOLANT SYSTEM BLOWDOWN.

182 AXFLU  
AXFLU CALCULATES HEAT TRANSFER FROM A LATTICE OF FLUID CYLINDRICAL FUEL PINS TO COOLANT IN EITHER SLUG OR FULLY-DEVELOPED LAMINAR FLOW. SOLUTION IS ANALYTIC AND CLOSED, EXCEPT FOR SATISFYING BOUNDARY CONDITIONS AT A FINITE, (BUT ARBITRARY), NUMBER OF BOUNDARY POINTS.

183 AXTHRM  
AXTHRM CALCULATES HEAT TRANSFER FROM A TRIANGULAR LATTICE OF CLAD SOLID CYLINDRICAL FUEL PIN TO COOLANT IN EITHER SLUG OR FULLY-DEVELOPED LAMINAR FLOW. SOLUTION IS ANALYTIC AND CLOSED, EXCEPT FOR SATISFYING BOUNDARY CONDITIONS AT A FINITE, (BUT ARBITRARY), NUMBER OF BOUNDARY POINTS.

198 HEATING2  
HEATING2 IS A GENERALIZED HEAT TRANSFER CODE CAPABLE OF SOLVING TRANSIENT AND/OR STEADY-STATE COORDINATE SYSTEMS. THE SIMPLIFIED INPUT MAKES IT A VERY USEFUL CODE FOR THOSE PROBLEMS HAVING GEOMETRICAL CONFIGURATIONS WHICH CAN BE DESCRIBED BY PARALLEL AND PERPENDICULAR LINES OR PLANES, OR CONCENTRIC CIRCLES. THESE FIGURES CAN BE BROKEN UP INTO A MAXIMUM OF 100 ONE-MATERIAL REGIONS WITH POSITION AND TIME-DEPENDENT VOLUMETRIC HEAT GENERATION RATES. UP TO 40 MATERIALS WITH CONSTANT PROPERTIES CAN BE DESCRIBED, AND THE INITIAL TEMPERATURES OF THESE REGIONS ARE POSITION-DEPENDENT. THE BOUNDARIES OF THESE REGIONS CAN BE CONTACT, INSULATED, TIME-DEPENDENT TEMPERATURE CONTROLLED, OR FORCED CONVECTION WITH A TIME-DEPENDENT SINK TEMPERATURE. ALSO A RADIATION BOUNDARY WITH A TIME-DEPENDENT SINK TEMPERATURE IS INCLUDED.

205 BLOOST5  
BLOOST5 COMBINES A REACTOR KINETICS CODE WITH A TWO-DIMENSIONAL HEAT TRANSFER CODE, AND IS APPLICABLE TO PROBLEMS FOR WHICH THE SPACE-INDEPENDENT FORM OF THE REACTOR KINETICS EQUATION IS APPLICABLE.
242 RATH MESHER/RATH WANTON  THESE CODES SOLVE THE TWO- AND THREE-DIMENSIONAL HEAT CONDUCTION PROBLEM IN GENERAL GEOMETRIES. TIME, TEMPERATURE, AND POSITION-DEPENDENT PHYSICAL PROPERTIES CAN BE USED.

246 FLOW-MODEL  A MULTI-CHANNEL, TWO-DIMENSIONAL, TWO-PHASE FLOW MODEL, DESIGNED TO COMPUTE THE AXIAL AND RADIAL COOLANT DENSITY AND QUALITY PROFILES, THE AXIAL PRESSURE PROFILE AND THE WEIGHT FLOW DISTRIBUTION FOR AN OPEN MATRIX FLOW, BOILING WATER REACTOR.

256 MANTA  MANTA IS A PROGRAM WHICH PROVIDES A THERMAL-HYDRAULIC NODAL ANALYSIS IN THE STEADY STATE. IT WAS DESIGNED TO ANALYZE FUEL ELEMENT CONFIGURATION IN THE SUPER-HEAT DEVELOPMENT PROGRAM. MANTA ANALYZES MIXING BETWEEN COOLANT CHANNELS, ALLOWS FOR TEMPERATURE VARIANCE CONDUCTIVITY IN ADMITTANCE CALCULATION, AND MULTIPLE STACKED SEGMENTS THROUGH THE FUEL REGION FOR A 7 ELEMENT CLUSTER ANALYSIS OVER A LENGTH OF UP TO 8 FEET. MANTA IS DESIGNED FOR SINGLE-PASS STEAM FLOW. THE FLOW DIRECTION IN THE COOLANT CHANNELS MAY BE EITHER UP OR DOWN, THEREBY PERMITTING THE ANALYSIS OF TWO-PASS AS WELL AS SINGLE-PASS FUEL ELEMENTS. MANTA ACCOUNTS FOR THE HEAT TRANSFER AND PRESSURE DROP THAT MAY OCCUR BETWEEN COOLANT CHANNELS DUE TO MIXING AS WELL AS TO THE CONVENTIONAL HEAT TRANSFER AND PRESSURE DROP RELATIONSHIPS DUE TO FRICTION, DISCONTINUITIES, ACCELERATION, CONVECTION, CONDUCTION, AND RADIATION. MANTA ALLOWS FOR THE CALCULATION AT EACH NOCE OF THE MATERIAL PROPERTIES VISCOSITY, SPECIFIC HEAT, CONDUCTIVITY, AND SPECIFIC VOLUME TO CORRESPOND TO THE ACTUAL NODE TEMPERATURE BEING SOLVED FOR.

R267 WATER  WATER IS A SUBROUTINE USED TO EXTRACT THERMODYNAMIC AND TRANSPORT PROPERTIES OF LIQUID, VAPOR, AND SUPERCRITICAL WATER BY TABULAR INTERPOLATION OVER THE RANGE OF STATES - 14.5 TO 14,500 PSIA AND 32 TO 1472 DEGREES F. THESE PROPERTIES ARE SPECIFIC VOLUME, SPECIFIC ENTHALPY, DYNAMIC VISCOSITY, AND THERMAL CONDUCTIVITY, TABULATED FOR PRESSURE AND TEMPERATURE CONDITIONS.

R272 FIGRO  FIGRO CALCULATES THE ONE-DIMENSIONAL STEADY-STATE TEMPERATURE DISTRIBUTION AND TOTAL FUEL SWELLING FOR METAL-CLAD, AXISYMMETRIC, BULK-OXIDE CYLINDRICAL FUEL ELEMENTS. THE FUEL PELLET MAY BE SOLID, ANNULAR, OR CONTAIN TWO RADIAL ZONES. OXIDE FUEL THERMAL CONDUCTIVITY IS A FUNCTION OF TEMPERATURE, DEPLETION, AND POROSITY. FUEL SWELLING IS A FUNCTION OF TEMPERATURE, DEPLETION, INTERNAL HYDROSTATIC PRESSURE, AND FISSIONING RATE. EITHER THE CLAD SURFACE FLUX OR THE TEMPERATURE AT THE INSIDE RADIUS OF THE FUEL MAY BE SPECIFIED AS A BOUNDARY CONDITION FOR THE HEAT CONDUCTION EQUATION.
R286  HCT2
HOT2 is a digital computer program that solves two-dimensional plane and axially symmetric steady-state and transient heat conduction problems with diagonal boundaries and interfaces. Mesh spacing (at most 5000 points) is completely variable. As many as 99 regions are permitted in order to describe spatial variations in material properties, heat generation rates, and boundary conditions. The heat generation rate and boundary conditions may vary with time.

R294  MC899/HOH
By making calls on a subroutine called HOH, M0899 edits thermodynamic and transport properties of water over the range 14.5 to 2538 psia and up to 608 degrees Fahrenheit below saturation and 932 degrees Fahrenheit above saturation.

R299  LICN
LION is a digital computer program which will solve three-dimensional transient and steady-state temperature distribution problems. The input consists of geometry, physical properties, boundary conditions, internal heat generation rates, and coolant flow rates as a function of time. In addition to solving problems of heat conduction in a structure, LICN can handle forced convection, free convection, and radiation or a combination of these at the surface of the structure. The output consists of complete nodal temperature distribution along with surface fluxes and heat transfer coefficients. An option is included in the program for determining the mean temperature in any specified section of the structure.

346  THTE
THTE (TRANSIENT HEAT TRANSFER VERSION E) provides a solution capability for large complex, three-dimensional transient and steady-state heat transfer problems which can include conduction, convection, and radiation with the option to compute fluid flow rates on a one-dimensional basis.

R348  TOPS
The TOPS program is a digital simulation of pressurizer dynamics based on a rigorous application of the first law of thermodynamics and phenomenological heat and mass transfer laws with empirically determined coefficients. The program is useful in studying the thermodynamic paths of pressurizer transients and is convenient to use as a design tool.

349  TCODEE
The TCODEE program calculates temperatures at the center points of a two-dimensional array in X-Y, R-Z, or R-THETA geometry. The mesh in this array may be variably-spaced. Averaged material constants are used which may be spatial and temperature dependent. Provision is made in the program for material phase changes. Channels for forced flow coolant may be included at exterior boundaries. In addition to the temperature array, program output includes surface temperatures and heat fluxes.
RAPP computes the relationship among mass flow, pressure, and piping resistance (K-factor) for high velocity flow of a two-phase mixture of steam and water. The source fluid may be subcooled or saturated water, saturated steam, or a mixture of steam and water. The downstream pressure must be below the saturation pressure of the source fluid. Specific applications include pressurizer surge line pressure drop and pressure distribution downstream of a relief valve.
I. DEFORMATION AND STRESS DISTRIBUTION COMPUTATIONS, STRUCTURAL ANALYSIS AND ENGINEERING DESIGN STUDIES

48 FUGUE
THE FUGUE PROGRAM COMPUTES STEADY-STATE WALL AND BULK FLUID TEMPERATURE, VOID FRACTION, AND LOCAL PRESSURE IN LIQUID-COOLED CLOSED CHANNELS IN WHICH THE HEATING RATE IS SPECIFIED. THE REQUIRED RELATIONSHIPS ARE EXPRESSED IN GENERAL, NON-DIMENSIONAL FORM AND COMBINED IN AN INTERNALLY CONSISTENT MANNER TO ALLOW PREDICTIONS FOR A VARIETY OF COOLANTS AND SPECIFIED OPERATING CONDITIONS.

R 80 SCR2
SOR2 SOLVES FOR THE FORCES, STRESSES, DEFLECTIONS, AND STRAINS IN THIN SHELLS OF REVOLUTION. THE SHELLS MAY BE GENERAL SURFACES OF REVOLUTION WITH VARIABLE THICKNESSES AND ELASTIC MODULII. THIS INCLUDES THE MORE FAMILIAR FORMS - THE CIRCULAR FLAT PLATE, CONE, CYLINDER, SPHERE, ELLIPSE, AND TORII WITH CIRCULAR OR ELLIPTICAL CROSS SECTIONS, FOR WHICH A SIMPLIFIED INPUT IS USED. THE AXISYMMETRIC LOADINGS CONSIDERED INCLUDE ARBITRARY DISTRIBUTIONS OF NORMAL, TANGENTIAL AND MOMENT SURFACE LOADINGS, AS WELL AS EDGE FORCES AND DEFLECTIONS. THE EFFECTS OF RADIAL AND AXIAL TEMPERATURE DISTRIBUTIONS, CENTRIFUGAL LOADING DUE TO ROTATION ABOUT THE AXIS AND VIBRATION ARE INCLUDED. THE ADDITIONAL EFFECTS OF MISALIGNMENT, LINE LOADS, AND ELASTIC SUPPORTS AT THE SHELL INTERSECTIONS ARE CONSIDERED.

1C9 4RESTRAINT PIPE STRESS CODE
THIS PROGRAM EVALUATES A FOUR RESTRAINT PIPING SYSTEM DESIGNED FOR HIGH TEMPERATURE OPERATION.

112 CRCCK
CROCK SELECTS THE MINIMUM WEIGHT DESIGN FOR A SPACE POWER PLANT IN WHICH THE WASTE CYCLE HEAT IS RADIATED DIRECTLY TO SPACE FROM THE CONDENSER. IT ACCOUNTS FOR HEAT TRANSFER, FLUID FLOW, METEOROID PROTECTION, AND THE GEOMETRIC PROPERTIES OF A RADIATOR-CONDENSER.

114 SHCCK
SHOCK SELECTS THE MINIMUM WEIGHT DESIGN FOR A HEAT-REJECTION SYSTEM FOR A SPACE POWER PLANT IN WHICH THE SENSIBLE HEAT LOST FROM A SINGLE-PHASE FLUID IS RADIATED TO SPACE. IT ACCOUNTS FOR HEAT TRANSFER, FLUID FLOW, METEOROID PROTECTION AND RADIATOR GEOMETRY.

190 WCPEXPRT
THE PURPOSE OF THE CODE IS TO CHOOSE THE LIGHTEST REACTOR RESULTING FROM ALL POSSIBLE COMBINATIONS OVER THE RANGE OF REACTOR INDEPENDENT VARIABLES SPECIFIED. THE ENGINEERING CALCULATIONS FOR EACH REACTOR DESIGN ARE BASED ON A LIQUID-COOLED CYLINDRICAL REACTOR HAVING PIN-TYPE FUEL ELEMENTS. REACTOR CONTROL IS ASSUMED TO BE EFFECTED BY MOVABLE SIDE REFLECTORS SURROUNDING THE CORE. NECESSARY PHYSICS DATA IS INPUT IN THE FORM OF SECOND-DEGREE EQUATIONS AS A FUNCTION OF THE INDEPENDENT VARIABLES.
GAZELLE5 DETERMINATION OF GAS-COOLED FAST REACTOR CORE PROPORTIONS REQUIRED TO SATISFY SPECIFIED DESIGN CONSTRAINTS AND COMPUTATION OF RESULTING PERFORMANCE CHARACTERISTICS.

CORE THE REACTOR CORE CONFIGURATION DESIGNED BY THE CORE PROGRAM CONSISTS OF CYLINDRICAL FUEL ELEMENTS ARRANGED ON AN EQUILATERAL PITCH SPACING WITH INTERNAL AND EXTERNAL COOLING. THE FUEL ELEMENT IS COMPOSED OF THREE CONCENTRIC GRAPHITE RINGS, AN INTERNAL COOLANT CHANNEL, AND A SPINE. THE COOLANT CHANNEL IS FORMED BY THE SPINE AND INNER GRAPHITE RING. THE FUEL IS CONTAINED IN THE MIDDLE GRAPHITE RING. CORE WILL DETERMINE THE NUMBER OF FUEL ELEMENTS OF A SPECIFIED DIAMETER AND LENGTH REQUIRED TO SATISFY A GIVEN CENTRAL FUEL TEMPERATURE. IT ALSO ADJUSTS THE PITCH SPACING REQUIRED TO PRODUCE A DESIRED PRESSURE DROP AS WELL AS PROVIDING A SPECIFIED INTERNAL COOLING RATE.

CYCLOPS1 CYCLOPS1 IS A PROGRAM FOR THE ANALYSIS OF THERMODYNAMIC SYSTEMS. IT IS A GENERAL PURPOSE PROGRAM THAT PERFORMS A HEAT AND MASS BALANCE FOR THERMODYNAMIC SYSTEMS COMPOSED OF PUMPS, TURBINES, SEPARATORS, HEAT EXCHANGERS, COMPRESSORS AND FLOW THROTTLING DEVICES. THESE COMPONENTS MAY BE CONNECTED IN ANY DESIRED WAY. THE THERMODYNAMIC FLUIDS THAT MAY BE USED ARE WATER, AIR, NITROGEN, AND PARA-HYDROGEN.

SAFE-PCRS SAFE-PCRS NUMERICALLY DETERMINES THE STRESS AND STRAIN DISTRIBUTION WITHIN EITHER HOMOGENEOUS OR HETEROGENEOUS THICK-WALLED BODIES OF REVOLUTION. IT IS DESIGNED FOR THE ANALYSIS OF MULTI-MATERIAL AXISYMMETRIC COMPOSITE STRUCTURES SUCH AS REINFORCED AND/OR PRESTRESSED CONCRETE VESSELS. DEFORMATIONS MUST BE WITHIN THE ELASTIC LIMIT OF THE MATERIALS CONSIDERED AND ONLY BODIES OF REVOLUTION SUBJECTED TO AXISYMMETRIC LOADING CAN BE TREATED.

SAFE-AXISYM SAFE-AXISYM IS A PROGRAM FOR THE ANALYSIS OF MULTI-MATERIAL AXISYMMETRIC COMPOSITE STRUCTURES. IT IS DESIGNED FOR THE ANALYSIS OF HETEROGENEOUS STRUCTURES SUCH AS REINFORCED AND/OR PRESTRESSED CONCRETE VESSELS. THE STRUCTURE IS ASSUMED TO BE LINEARLY ELASTIC, AND ONLY BODIES OF REVOLUTION SUBJECTED TO AXISYMMETRIC LOADING CAN BE TREATED.

SAFE-PLANE SAFE-PLANE IS APPLIED TO TWO-DIMENSIONAL STRUCTURES OF ARBITRARY GEOMETRY UNDER IN-PLANE LOADS. EITHER PLANE STRESS OR PLANE STRAIN CONDITIONS MAY BE IMPOSED. MECHANICAL AND/OR THERMAL LOADS ARE PERMITTED.
SAFE-SHELL is used to design and analyze axisymmetric thin shell structures of arbitrary generatrices under axisymmetric mechanical and/or thermal loading conditions. The intersection of two or more shells can be treated.

CYGR02 is used to determine stresses and strains during steady-state and transient power operation of an oxide-fueled, metal clad rod-type fuel element in a pressurized environment. Major loading conditions include fuel swelling, fission gas and coolant pressure, clad growth and differential thermal expansion. The application for which the program has been developed is zircaloy tubes containing bulk oxide fuel. Axial and azimuthal symmetry of temperature and stresses is assumed.

The SEALSHELL2 program determines stresses, strains, deflections, and reactions in a thick shell of revolution with axisymmetric loading. The loading consists of a temperature distribution, inside and outside pressure distributions, and circumferential forces and moments applied to the middle surface. The shell is linear-elastic with tensile, bending, and shear strains.

MO552 solves the transient response problem of linear elastic, lumped-mass systems subjected to a unidirectional foundation transient that can be either a velocity or acceleration transient. Normal mode theory is used and the input to the program consists of the mode shapes, frequencies, and foundation transient. Element effects are also evaluated as a function of time. Modal damping coefficients may be specified.

SAFE-CREEP performs a viscoelastic analysis of plane or axisymmetric composite concrete structures with age and temperature dependent creep data.

WIREX produces wiring lists containing all the information normally found on detailed wiring drawings. These lists are presented to the electricians as a job book containing information in the order in which it is required for logical procedure through a wiring job. WIREX lists were used extensively during installation for the UHTREX project.

PIPE performs an elastic stress analysis of a 3-dimensional piping structure with thermal stresses, redundant loops, and concentrated loads.
R331  FLOT1 will predict the steady-state flow and the flow transient due to the subsequent loss of power to all pumps and terminate the transient at a specified time or it will predict the flow transient in which only some of the pumps are lost. This latter transient may be terminated by a maximum transient time or by check valve closures in all loops in which pumping power is lost. In the latter event, the program will predict the subsequent steady-state flow distribution.

R332  SAFE-3D

SAFE-3D is a finite element program for the three-dimensional elastic analysis of heterogeneous composite structures. The program uses the following types of finite elements - (1) tetrahedral elements to represent the continuum, (2) triangular plane stress membrane elements to represent inner liner or outer case, and (3) uniaxial tension-compression elements to represent internal reinforcement. The structure can be of arbitrary geometry and have any distribution of material properties, temperatures, surface loadings, and boundary conditions.

R337  STEM

STEM calculates and punches out partially-coupled mass, stiffness and internal load function matrices for a structural system of beams having prismatic segments. Shear deformation and rotational inertia are included in the calculations.

R344  GEM

GEM is intended primarily to perform vibration studies with the capability of generating input for the VEP (vibration eigenvalue problem) routine and performing additional operations on the output from the SHO (shock) segment. Given a system of masses and springs, the VEP routine computes the natural frequencies of the vibrating system as well as the mode shapes for each frequency. Given the mode shapes, frequencies, and masses of a vibrating system, the SHO routine will compute the deflections and forces at the mass points.
SWELL2 was developed to provide the fuel-element lifetime data required for power plant optimization studies. The program commences by calculating radial and axial temperatures in a fuel element. The fuel which is hotter than the effective-fission-gas-release-temperature is considered to release its fission gas immediately to a plenum, and the pressure of all plenum gas is calculated. The pressure of the gas retained for a while in the colder fuel is also calculated. At each axial position the plenum pressure is compared with the pressure exerted by retained fission gas, and the larger of the two pressures is considered to be the effective pressure as far as cladding damage is concerned. The cladding is assumed to fail when its cumulative damage equals unity at any axial position. Damage components are primary creep, secondary creep, ratcheting growth due to thermal cycling, fatigue due to thermal cycling, and strain due to inexorable fuel swelling. At each step of the calculations the fuel element is subjected to several conditions of abnormal operation (overpower, flow coastdown, etc.) to ensure they could not produce enough additional damage to cause failure at that time.

SUPORAN solves for steady-state deformation and strength characteristics of a nuclear reactor core support structure. This structure is assumed to be made of two circular plates which are located above each other and are interconnected perpendicularly with concentric rows of tubular members. The geometry of the structure and its loading (transverse pressure and internal temperature gradients) are all of axisymmetrical nature.

BCW2 is used to calculate deflections of closely-spaced parallel beams, each with limited-pivot support at one end, possible beam interactions at the other end and at one intermediate position, assuming an arbitrary temperature distribution.

TUBE solves for the stresses due to pressure and temperature in a U-tube type heat exchanger. Specifically, it handles a configuration consisting of a spherical head, primary transition cylinder, and secondary cylinder. The transition cylinders may be conical and tapered in thickness, but the remaining shells are of uniform thickness. The spherical head and the transition cylinders may be omitted from a problem.

SAFE-2D performs the elastic stress analysis of general axisymmetric, plane, and combined axisymmetric and plane composite structures.
R383 MC266 MQ266 COMPUTES THE DYNAMIC
SHOCK FORCES AND MODAL FREQUENCIES ACTING ON A LUMPED MASS, LINEAR
ELASTIC MODEL OF A STRUCTURE SUBJECTED TO SHOCK SPECTRUM INPUTS.
THE MODEL EMPLOYED IS A COLLECTION OF LUMPED MASSES CONNECTED BY
WEIGHTLESS FLEXIBLE ELEMENTS. IF THE ORIGINAL STRUCTURE IS NOT
STATICALLY DETERMINATE, REDUNDANT FORCES MUST BE INTRODUCED TO
ENSURE A PRIMARY STRUCTURE THAT IS.
J. GAMMA HEATING AND SHIELD DESIGN PROBLEMS

45 GRACE1
GRACE1 is a multigroup, multi-region, gamma-ray attenuation program designed primarily for computing gamma-ray heating and gamma-ray dose rates in multi-region finite or semi-infinite slab shields. A different buildup factor may be specified for each source region considered.

46 GRACE2
GRACE2 is a multigroup, multi-region, gamma-ray attenuation program to compute the total dose rate or heat generation rate from either a spherical or a cylindrical source. The source, which may be located in either the central region of the system or in a concentric shell region surrounding it, may be uniform, exponential, or have a polynomial variation in the radial direction. In the case of cylindrical geometry, it may also have a polynomial variation in the axial direction.

91 FARSE1A
The program computes the neutron leakage from a shield annulus. The removal cross sections incorporate multiscattering effects. Dose deposit at the target mesh is then determined from the angular distribution of the leakage neutrons, integrated over the shield surface.

110 SCARF2
SCARF2 evaluates the first-order approximation of the fast neutron current at the payload surface due to neutrons which scatter from the radiator fins. It uses SNAP geometry and is a complementary program to FARSE.

111 SCAR1
SCAR1 performs single scatter ray tracing and evaluates scattering from a ring. It is complementary to FARSE.
LIPRECANI is a two-dimensional Monte Carlo program to compute the penetration and energy deposition of neutrons in pure hydrogenous media. The code offers three possible geometries, i.e., conical, cylindrical and one-dimensional infinite slab geometry (this latter option to the extent that an infinite slab may be approximated by a cylinder of large radius). The code may be used with a mono-directional point or beam source and is modified easily to handle angular distributions. Isotropic scattering in the center of mass system is assumed. The following results are tabulated with each summary -

(A) Energy deposition distribution,
(B) Particle deposition distribution,
(C) Particle leakage fraction,
(D) Particle absorption fraction,
(E) The fraction of particles reaching the cutoff energy which have neither leaked nor have been absorbed,
(F) The average energy per particle leaking from the system,
(G) The average energy per particle remaining in the system as heat generation,
(H) The average number of collisions per history,
(I) The total number of histories currently being summarized,
(J) The average energy deposit per history through absorption,
(K) The average energy of particles absorbed.

RATRAP computes the dose rate at specified spatial points about a system of SNAP geometry.

MORTIMER computes the dose rate at specified spatial points about a system of SNAP geometry.

MAC performs slab geometry, multigroup neutron and gamma ray penetration analysis for a multi-region reactor shield. The code calculates the following information as a function of distance through a reactor shield assembly -
neutron fluxes for up to 35 energy groups,
neutron dose rates,
the approximate neutron spectrum,
gamma ray fluxes for 7 energy groups,
total gamma dose rate, with a breakdown of the contribution from sources in each reactor and shield region,
the approximate gamma ray spectrum.

The Shoe code makes use of the method of steepest descent to find the dimensions of a minimum-weight, three-layer shield in spherical geometry. The weight minimization is carried out subject to the constraining condition of a constant dose rate at some selected point outside the shield.
MUSCAT computes the incident scattered neutron currents as a function of position within (1) the cavity formed by two truncated concentric spheres, (2) the cavity between two concentric circular cylinders, or (3) a cylindrical cavity.

LETO will solve the gamma ray transport and energy deposition problem in one-dimensional laminar slab geometry. The energy group scheme is employed to account for photon energy degradation. An arbitrary external spatial isotropic source may be specified with an arbitrary energy spectrum. The boundary conditions may be (a) free boundaries with arbitrary incident, (b) symmetry on the left arbitrary incident on the right, and (c) symmetry on both ends.

Included, also, considerable data checking and evaluation are performed. Combined with the flexible narrative input, these features make the code useful in the preparation of error-free input for complex problems.
K. TOTAL SYSTEMS ANALYSIS

THREDES IS A SCIENTIFIC APPLICATION PROGRAM THAT GENERATES THE BASIC FINITE ELEMENT REFERENCE GRID WORK FOR THE SAFE CODES. THIS GENERATED GRID CONSISTS OF TRIANGULAR ELEMENTS AND NODES, UNIAXIAL ELEMENTS, SUCH AS REINFORCEMENT BARS, TENDONS, AND ANCHORS, AND BIAXIAL MEMBRANES, SUCH AS ANY THIN SHELL OR LINER. THE PUNCHED OUTPUT SERVE AS DIRECT INPUT DATA TO THE SAFE CODES.

GAND PREPARES THE CROSS SECTIONS NEEDED FOR DETAILED COMPUTATIONS OF NEUTRON ENERGY SPECTRA IN FAST REACTORS FROM A FILE OF BASIC NUCLEAR DATA IN THE ENDF/B FORMAT.

BL47 IS A PLOTTING ROUTINE DESIGNED FOR PLANE STRUCTURES THAT ARE TO UNDERGO STRESS ANALYSIS. POINTS AND LINES ARE INPUT IN VARIOUS PARAMETRIC FORMS, AND CURVED SEGMENTS ARE DRAWN BETWEEN GIVEN POINTS ALONG THE GIVEN LINES. THE PROGRAM MAY BE USED AS A DRAFTING TOOL TO CONSTRUCT ENGINEERING DRAWINGS. BL47 USES THREE POINTS ON A STRAIGHT LINE SEGMENT TO OBTAIN DIMENSIONS FOR SEAL SHELL2 (ACC ABSTRACT 282) INPUT DATA.
**DATA MANAGEMENT**

159 MCMUS

MOMUS IS THE PROGRAM USED TO CONSTRUCT AND MAINTAIN THE NEUTRON CROSS SECTION MASTER TAPE DEVELOPED UNDER THE ATOMICS INTERNATIONAL AUTOMATED CROSS SECTION PROGRAM. THE LIBRARY CONTAINS 21 ELEMENTS WITH DATA FOR 11 PARAMETERS IN THE RANGE 0.001 EV TO 10 MEV. MOMUS WILL PERFORM THE FOLLOWING TASKS -

(A) MAKE THE BINARY MASTER TAPE FROM CARDS,
(B) LIST SELECTED ELEMENTS,
(C) UPDATE - ADD, CORRECT AND REPLACE ANY DATA,
(D) MAKE A SHORT TAPE CONTAINING SELECTED ELEMENTS,
(E) PROVIDE GRAPHICAL DISPLAY OF SELECTED DATA, AND
(F) PUNCH MICROSCOPIC DATA.

181 XLIBIT

USING THE ANL STANDARD CROSS SECTION DATA CARDS AS INPUT, THE PROGRAM CAN PERFORM THE FOLLOWING OPERATIONS -

(A) PREPARE A LIBRARY TAPE CONSISTING OF THE CROSS SECTION DATA FOR SETS OF MATERIALS, INCLUDING A DIRECTORY GIVING THE SETS ON THE TAPE AND THE MATERIALS IN EACH SET,
(C) ANY OF THE DATA ON A LIBRARY TAPE MAY BE PRINTED, AND THE PROGRAM WILL PUNCH THE CROSS SECTION DATA CARDS THAT WERE USED TO PLACE ANY OF THE MATERIALS OR SETS ON THE TAPE WITH ALTERED SET AND MATERIAL NAMES IF DESIRED.

206 UNPACK

UNPACK RETRIEVES NEUTRON CROSS SECTION DATA AND RELATED INFORMATION FROM A SCISRS (BROOKHAVEN NATIONAL LABORATORY) DATA TAPE. THE FORMAT AND STRUCTURE OF THE SCISRS TAPE IS DESCRIBED IN BNL-883 BY FRIEDMAN AND PLATT, JULY 1964.

236 DFSR

DATA FILE SERVICE ROUTINES (DFSR) ARE USED TO STORE, DISTRIBUTE, AND CHECK THE DATA ON THE EVALUATED NUCLEAR DATA FILE (ENDF). THERE ARE 6 ROUTINES -

DFSR1 - STORES DATA ON MASTER LIBRARY TAPES.
DFSR2 - PROCESSES REQUESTS FOR DATA BY FINDING THE APPROPRIATE DATA ON THE MASTER TAPE AND COPYING THEM ONTO THE REQUESTORS TAPE.
DFSR3 - CORRECTS DATA ON THE MASTER TAPE.
DFSR4 - COPIES/PRINTS/PUNCHES SELECTED DATA FROM THE MASTER TAPE.
DFSR5 - PRINTS LIST OF DATA RECENTLY ADDED TO MASTER TAPE AND PREPARES A CROSS-REFERENCE LIST OF ALL DATA ON THE MASTER TAPE.
DFSR6 - MAKES COPIES AND CHECKS THE MASTER TAPE.
CLIP IS THE CROSS SECTION LIBRARY PREPARATION AND MAINTENANCE PROGRAM FOR FORM AND THREDEx.

ETOE (ENDF/B TO MC**2 DATA CONVERSION) ACCEPTS CROSS SECTION DATA FROM A MODE 2 ENDF/B TAPE (SEE REFERENCE 3) AND PREPARES THE BINARY CROSS SECTION AND LEGENDRE POLYNOMIAL TAPE FOR THE MC**2 CODE WRITTEN BY ARGONNE NATIONAL LABORATORY.


SCORE IS AN INTERACTIVE NEUTRON CROSS SECTION EVALUATION SYSTEM.
This package of five programs is designed for processing ENDF/B (Evaluated Nuclear Data File Version B) tapes.

Checker checks that the ENDF/B tapes are in proper format and all fields are within specified limits, rather than the physics of the data library. Angular distributions reconstructed from Legendre coefficients are everywhere positive.

Crect provides a means of correcting assembled data on a tape by insertion and deletion of data.

Dammet selectively merges data from one or two ENDF/B library tapes onto a final tape. The mode (BCD or binary) and arrangement (standard or alternate) may be changed during this process.

Plotfb processes ENDF/B library tapes which contain data embedded within a necessary library structure in order to produce comprehensive listings and/or plots. The listings and/or plots contain an extensive amount of information related to the data, such as temperature dependence, physical units of the data, interpolation laws for the data, cryptic titles defining the reaction type, etc.

Slave3 provides modular subroutines which can be assembled to retrieve and process ENDF/B data for a specific problem.
N. SUBSIDIARY CALCULATIONS

133 WE C THIS PROGRAM EDITS THE MAGNETIC TAPE PRODUCED BY W-DSN CALCULATING REACTION RATES BY ENERGY AND BY VOLUME WITH TOTALS. IT CAN ALSO PRODUCE REACTION RATES FOR FED-IN CROSS SECTIONS.

207 CROSSPLOT AUTOMATIC PLOTS ARE GENERATED FROM NEUTRON CROSS SECTION DATA.

210 DTX THE DTX CODE CALCULATES EFFECTIVE MACROSCOPIC, HOMOGENEOUS, GROUP CROSS SECTIONS WHICH ARE SPACE-AVERAGED OVER THE FLUXES AND CURRENTS PRE-CALCULATED IN A ONE-DIMENSIONAL NEUTRON TRANSPORT CODE SUCH AS DTK OR DSN. ISOTROPIC AND ANISOTROPIC CROSS SECTIONS MAY BE INCLUDED.

239 CPS CPS PROVIDES A GRAPHICAL MEANS OF COMPARING EXPERIMENTAL CROSS SECTION VALUES OBTAINED FROM THE SCISRS LIBRARY TAPE WITH AN OPTION TO INCLUDE READING IN NEW EXPERIMENTAL OR CALCULATED VALUES.
EXPERIMENTAL DATA PROCESSING

154 CRCC90  THE CRCC90 CODE WAS DEVELOPED FOR USE IN THE DATA REDUCTION OF OUT-OF-PILE FLUID FLOW EXPERIMENTS ON THE ML-1 FUEL ELEMENTS. THE CODE IS SPECIFICALLY DESIGNED TO EVALUATE FRICTION FACTORS, ENTRANCE AND EXIT COEFFICIENTS, AND ORIFICE CALIBRATIONS FROM HYDRODYNAMIC DATA OBTAINED FROM SINGLE PHASE EXPERIMENTAL FLUID FLOW TESTS IN AXIAL FLOW DUCTS.

164 BURP1  THE PROGRAM CALCULATES ABSOLUTE TOTAL EFFICIENCY FOR MCNOENERGETIC GAMMA RAY INTERACTIONS IN CYLINDRICAL SCINTILLATION DETECTORS. THE ABSOLUTE TOTAL EFFICIENCY IS DEFINED AS THE FRACTION OF SOURCE GAMMAS WHICH INTERACT AT LEAST ONCE WITH THE CRYSTAL DETECTOR. CALCULATIONS ARE MADE FOR THE POINT ISOTROPIC SOURCE LOCATED ALONG THE AXIS OF SYMMETRY FOR SOLID CYLINDRICAL CRYSTALS, WITH OR WITHOUT A COAXIAL CYLINDRICAL WELL.

165 BURP2  THE PROGRAM CALCULATES THE ABSOLUTE TOTAL EFFICIENCY FOR MCNOENERGETIC GAMMA RAYS INTERACTING IN CYLINDRICAL SCINTILLATION DETECTORS. THE ABSOLUTE TOTAL EFFICIENCY IS DEFINED AS THE FRACTION OF SOURCE GAMMAS WHICH INTERACT AT LEAST ONCE WITH THE CRYSTAL DETECTOR. CALCULATIONS ARE MADE FOR HOMOGENEOUS, ISOTROPIC CIRCULAR DISK OR CYLINDRICAL VOLUME SOURCES. SOURCES MUST BE SYMMETRICAL WITH THE AXIS OF SYMMETRY FOR SOLID CYLINDRICAL CRYSTALS, WITH OR WITHOUT A COAXIAL CYLINDRICAL WELL. SOURCE ABSORPTION AND SCATTERING MAY BE INCLUDED FOR VOLUME SOURCES.

166 BURP3  THE PROGRAM CALCULATES ABSOLUTE TOTAL EFFICIENCY FOR MCNOENERGETIC GAMMA RAY INTERACTIONS IN CYLINDRICAL SCINTILLATION DETECTORS. THE ABSOLUTE TOTAL EFFICIENCY IS DEFINED AS THE FRACTION OF SOURCE GAMMAS WHICH INTERACT AT LEAST ONCE WITH THE CRYSTAL DETECTOR. CALCULATIONS ARE MADE FOR POINT ISOTROPIC SOURCES LOCATED BOTH ON AND OFF THE AXIS OF SYMMETRY FOR SOLID CYLINDRICAL CRYSTALS, WITH OR WITHOUT A COAXIAL CYLINDRICAL WELL. THE EFFICIENCIES FOR POINTS LOCATED AT SPECIFIED RACIAL OFF-AXIS POSITIONS, FOR A SINGLE AXIAL SOURCE-CRYSTAL DISTANCE, ARE NORMALIZED TO THE ON-AXIS EFFICIENCY FOR THE SAME AXIAL DISTANCE. THE ABSOLUTE EFFICIENCY FOR THE ON-AXIS POINT IS GIVEN.
BURP4 calculates the photo-fraction for monoenergetic gamma rays interacting in solid cylindrical scintillation detectors. Photo-fraction is defined as the fraction of interacting source gammas that are totally absorbed (including secondaries) in the crystal. Isotropic source geometries allowed are isotropic points (on or off-axis), disks, cylindrical volumes. Allowed monodirectional sources, normal to crystal face, are narrow beam collimated to crystal axis, broad beam illuminating entire crystal face, and collimated beam of any specified diameter.

BURP5 calculates the photo-fraction for monoenergetic gamma rays interacting in well-type cylindrical scintillation detectors. Photo-fraction is defined as the fraction of interacting source gammas that are totally absorbed (including secondaries) in the crystal. Isotropic source geometries allowed are isotropic points (on or off-axis), disks, cylindrical volumes. Also allows narrow monodirectional beam collimated to crystal axis, incident normal to well bottom.

CCINC coincidence counting data are treated to obtain specific disintegration rates, channel efficiencies and count rates, weighted means, and all associated standard errors. Corrections are made for unequal deadtime loss in each channel, coincidence resolving time losses, decay during counting, decay from a reference time, and background in each of the three channels. Input variables include sample identification, start time for counting, sample reference time, counting interval, total number of counts in two single channels and one coincidence channel, and normalizing sample volume. Input parameters consist of deadtimes of each single channel, coincidence resolving time for all three channels, decay constant, background count rate for all three channels, and standard errors for each of the above parameters. Optional input allows date, group classification, and a 3-digit user code. Output contains corrected single channel, coincidence, and disintegration rates referred to start of count, specific disintegration rate (counts/second/unit volume or weight) referred to reference time, weighted mean and error of any number of problems in a group, efficiencies of the two independent detector channels, date, summary of background values used, identification number, count start time, and upper and lower limit (one standard deviation) of each of the computed quantities.
EXPN ANALYZES PULSED NEUTRON DATA USING THE GARELIS-RUSSEL TECHNIQUE (REFERENCE 1). BASICALLY THE CODE COMPUTES THE PROMPT DECAY CONSTANT, ALPHA, AND THE PARAMETER (K*Beta/L) FROM EXPERIMENTAL DATA, WHICH IS DIRECTLY EXTRACTED FROM A TIME ANALYZER STORAGE MEMORY AND READ ONTO A PUNCHED PAPER TAPE. THE ALPHA-DETERMINATION PART OF THE CODE WAS ORIGINALLY OBTAINED UNDER THE NAME EXPLICIT FROM KNOLLS ATOMIC POWER LABORATORY BUT HAS SINCE BEEN MODIFIED. THE CODE PROVIDES OPTIONS FOR A PRE-BURST OR A POST-BURST BACKGROUND ANALYSIS. THAT IS, THE PARAMETERS ALPHA AND (K*Beta/L) ARE OBTAINED USING A BACKGROUND MEASURED PRIOR TO THE BURST OR MEASURED AFTER THE BURST.


TOAD IS USED TO PROCESS AND ANALYZE GAMMA RAY SPECTRA.
P. GENERAL MATHEMATICAL AND COMPUTING SYSTEM ROUTINES

R 43 CURFIT CURFIT IS A COMPOSITE PROGRAM FOR FITTING EXPERIMENTAL DATA POINTS WITH DIFFERENT TYPES OF COMMON ANALYTIC CURVES. THERE ARE AT PRESENT FIVE FITS AVAILABLE -

1. POLYNOMIAL Y = SUMMATION OVER I OF A(I)*X**I
2. EXPONENTIAL Y = A*EXP(B*X)
3. CCSINE Y = A*COS(B*(X+C))
4. SERIES OF CUBICS Y = A(J)+B(J)*X+C(J)*X**2+D(J)*X**3
5. FCURIER SERIES

62 LOS ALAMOS LEAST SQUARES THIS PROGRAM PERFORMS LEAST SQUARES FITTING OF LINEAR OR NONLINEAR FUNCTIONS IN SEVERAL INDEPENDENT VARIABLES. THE PROGRAM WILL DETERMINE AN ESTIMATE OF A IN THE FUNCTION Y=F(X,A) BY MINIMIZING THE SUM OF SQUARES. IN THE FUNCTION, X IS A VECTOR OF OBSERVED VARIABLES AND A IS A VECTOR OF PARAMETERS TO BE DETERMINED. IN THIS CONTEXT, A LINEAR FUNCTION IS ONE WHOSE PARTIAL DERIVATIVES WITH RESPECT TO THE ELEMENTS OF A ARE ALL INDEPENDENT OF A. A NONLINEAR FUNCTION HAS AT LEAST ONE OF THE ELEMENTS OF A APPEARING IN AT LEAST ONE OF THESE PARTIAL DERIVATIVES.

186 LAG1/LAG2 LAG IS A SINGLE PASS LOAD AND GO ASSEMBLER DESIGNED TO ACCEPT IBM7090 FLOCO II INSTRUCTIONS.

321 EXPALS THIS PROGRAM FITS BY LEAST SQUARES A FUNCTION WHICH IS A LINEAR COMBINATION OF REAL EXPONENTIAL DECAY FUNCTIONS. THE FUNCTION IS

Y(K) = SUMMATION OVER J OF A(J) * EXP(-LAMBD(J) * K)

VALUES OF THE INDEPENDENT VARIABLE (K) AND THE DEPENDENT VARIABLE Y(K) ARE SPECIFIED AS INPUT DATA. WEIGHTS MAY BE SPECIFIED AS INPUT INFORMATION OR SET BY THE PROGRAM (W(K) = 1/Y(K)).

324 FRANTIC FRANTIC IS DESIGNED TO PROCESS RAW COUNTING DATA AND TO FIT IN THE LEAST SQUARES SENSE THESE DATA TO THE MULTIPLE EXPONENTIAL GROWTH AND DECAY EQUATIONS. THE PROGRAM CAN BE USED FOR SUMS OF EXPONENTIALS WITH POSITIVE, NEGATIVE, OR ZERO EXPONENTS AND POSITIVE OR NEGATIVE COEFFICIENTS.

R327 DAFT1 DAFT1 IS A PROGRAM FOR WEIGHTED LEAST SQUARES FITTING OF 0.0253 EV NEUTRON DATA FOR FISILE NUCLIDES. THE PROGRAM ALSO CARRIES OUT COMPUTATIONS RELEVANT TO DISCRIMINATING OVERALL GOODNESS OF FIT, PARTICULARLY DEVIAN DATA, AND DATA WHOSE IMPROVEMENT WOULD LEAD TO LARGE REDUCTIONS IN ERROR OF EACH FITTED PARAMETER.
SNEQ CONSISTS OF THE TWO CODES, SNAP AND EQPLT, WHICH HAVE BEEN MERGED DUE TO THEIR COMMON USE OF THE SLIP COMPILER. SNAP INTERPRETS AND SOLVES PSEUDO-FORTRAN INPUT EQUATIONS REPRESENTING NONLINEAR ALGEBRAIC SYSTEMS. EQPLT INTERPRETS PSEUDO-FORTRAN INPUT EQUATIONS AND CALCULATES AND PLOTS MULTIPLE CURVES ON A SINGLE GRAPH. EQPLT IS USEFUL FOR PARAMETER STUDIES.
A benchmark study of ZPR-III assembly 48 using ENDF/B cross sections was undertaken to identify possible cross section discrepancies in the microscopic ENDF/B data. This work was done for the CSEWG testing subcommittee as part of their phase I data testing. This package contains the cross section data generated for this study in the form of the ultra-fine group cross sections of the materials of ZPR-III assembly 48 in the format of the GGA GAF/GAR program data tapes.
VII. KWIC Index
STANT CALC C TC IC 1G MEV Bnw 11C7 F4
FLSICN SLAB CYL SPHERE Bnw 11C7 F4
AFGAR ENCF/B DATA TAPES GGA 11C8 BIN
ESS ANALYSIS, 2-C BODIES GGA 11C8 F4
ALYSIS THIN SHELLS GGA 11C8 F4
HIEELING CCFE CAVITY GECM 11C8 F4
CIDENT ANALYSIS CCFE CALC GGA 11C8 F4
ERINC LAW CALCULATION GGA 11C8 F4
X-SECTION CALC CECCERATOR GGA 11C8 F4
SIS REACTOR KINETICS ECONS GGA 11C8 F4
X-SECTIONS HEX LATTICE GGA 11C8 F4
ESCALANCE PARAMETER CALC GGA 11C8 F4
ICNS FAST THERMAL SPECTRA GGA 11C8 F4
TIC ANALYSIS CONCRETE GGA 11C8 F4
C FISSION PRODUC T RELEASE GGA 11C8 F4
EL CYCLE CALCULATION GGA 11C8 F4
ETICS 2-C HEAT TRANSFER GGA 11C8 F4
CA XY RZ RTHETA GEOMETRY Bnw 11C8 F4
E FAST REACTOR X-SEC CALC Bnw 11C8 F4
TIME-DEPENDENT DIFFUSION GGA 11C8 F4
GRCUP-AVERAGE X-SEC CALC GGA 11C8 F4
ENCENT REACTOR KINETICS GGA 11C8 F4
P DIFFUSION AND DEPLETION GGA 11C8 F4
ER DISTRIBUTION SEARCH GGA 11C8 F4
CEFFICIENT CALCULATION GGA 11C8 F4
IFFUSION AND DEPLETION Bnw 11C8 F4
E STRUCTURE STRESS STUDY GGA 11C8 F4
NALYZER GAMMA-RAY SPECTRA GGA 11C8 F4
GECM DIFFUSION DEPLETION GGA 11C8 F4
CHANCE X-SECTION CALC GGA 11C8 F4
N LIBRARY PREPARATION GGA 11C8 F4
TRANSPOR TC CCE X-Y GEOM LASL 11C8 F4
METICS WITH TEMP FEEDBACK GGA 11C8 F4
ST X-SECTION GENERATION BnwL 11C8 F4
YMMETRIC STRESS ANALYSIS GGA 11C8 F4
ULSICN CALC HEX-Z MESH GGA 11C8 F4
GENERATION AND DECA Y Bnw 11C8 F5
DIFFUSION XY RZ GEOMETRY GGA 11C8 F+B AL
AC+ INELASTIC SCATTERING CDC 16C4 F63
SICNA SLAB CYLINDER SPHERE CDC 16C4 F63
FFUSION SLAB CYL SPHERE CDC 16C4 F63
N MCP AIM AND FCG CODES CDC 16C4 F63
P DIFFUSION XYZ GEOMETRY CDC 16C4 F63
ECTRM X-SECTION CALC CDC 16C4 F63
P DIFFUSION SLAB CYLINDER CDC 16C4 F63
ENUATION SLAB GEOMETRY CDC 16C4 F63
ENUATION CYL SPHERE GEOM CDC 16C4 F63
RATE FRCM A CLOUD CDC 16C4 F63
TRCA SPECTRUM X-SECTIONS CDC 16C4 F63
ECTRM X-SECTION CALC CDC 16C4 F63
RXXIMATION SLAB GEOMETRY CDC 16C4 F63
1-GF S4 APPROXIMATION CDC 16C4 F63
P DIFFUSION DEPLETION CDC 16C4 F63
DIFFUSION CYLINDER SLAB CDC 16C4 F63
GRAL X-SECTION CALC CDC 16C4 F63
FFUSION SLAB CYL SPHERE CDC 16C4 F63
LIDE CHAIN ECLATIONS CRNL 16C4 F63
ISCRETE ORDCATE CALC UNC 16C4 F63
1-GF S4 APPROXIMATION RZ PW 16C4 F63
CRETE CREATION LCCCE UNC-LASL 16C4 F63
LIDE CHAIN ANALYSIS CRNL 16C4 F63

RSP LT= GAMTEC2, MULTI-GP CON 185
RS P T= HFIN, 1-D MULTI-GP DIF 241
R LT= ZPR-III ASSEMBLY 48 G 356
RS P = SAFE-PLAN, PLANE STR 252
RS P = SAFE-SHELL, STRESS AN 253
RS P T= MUSCAT, VIEW FACTOR S 259
RS T= GADCAE, DSCET, HTGR AC 261
RS P = GASKET, THERMAL SCATT 263
RS P = GAKER, INELASTIC SCAT 289
RS P = GASA, STABILITY ANALY 290
RS P = HEXSCAT, ELASTIC SCAT 291
RS P = PSELDO, STATISTICAL R 292
RS P = FREEVAP, HTGR METALLI 3:1
RS T= GAFFE, EQUILIBRIUM FU 3:2
RS P T= BLCCST4, CCMBINED KIN 3:3
RS P = PERT4, 2-C Perturbati 3:4
RS P T= FCC4, FUNDAMENTAL MOD 3:6
RS P T= GAKIN, 1-D MULTIGROUP 3:11
RS P T= GAF, GTR, SPECTRA AND 3:16
RS P = GAPCTIN, SPACE-INDEP 3:17
RS P T= FEV7, 1-D MULTIGROUP 3:18
RS P T= GASP7, 1-D BURNUP POW 3:19
RS P T= TEMCC7, TEMPERATURE C 3:20
RS P T= DB2D, 2-D MULTIGROUP D 3:25
RS P T= TACTRAN, 2-D MULTI-GP 3:58
RS P T= GAKIT, 1-D MULTIGROUP 3:70
RS P T= IDX, 1-D DIFFUSION FA 3:74
RS P T= SAFE-2D, PLANE + AXIS 3:79
RS P T= GATT, 3-D FEW-GP DIFF 3:80
RS BP LT= ISOGCN, RADIONUCLED 3:67
RS P T= GAMBLE5, 2-D MULTI-GP 2:22
RS P = HAFEVER, HAUSER-FESHB 14
RS P = FCG, 1-D FEW-GP DIFFU 28
RS P T= AIM6, 1-D MULTI-GP DI 29
RS P T= PERT, 1-D Perturbatio 3:6
RS P T= WHIRLAW, 3-D 2-GROU 3:28
RS P T= GAM1, FAST NEUTRCA SP 3:35
RS P T= 2GRAND, 2-D FEW-GRCU 3:40
RS P T= GRACE, GAMMA-RAY ATT 4:5
RS P = GRACE2, GAMMA-RAY ATT 4:6
RS P T= CLOUD, GAMMA-RAY COSE 4:7
RS P T= TEMPEST2, THERMAL NEU 5:0
RS P T= FCRM, FAST NEUTRCA SP 5:1
RS = SAIL, 1-D 1-GP SN APP 5:2
RS = S4 CYL CELL CCCE, 1-D 5:3
RS LT= SIZZLE, 1-D MULTIGROUP 5:8
RS P T= EUPHISE3A, 2-D 2-GP 8:7
RS P T= ARES2, RESONANCE INE 8:9
RS = 16, SOLUTION OF NUC 13:4
RS P T= DF2, 1-D MULTIGROUP D 14:4
RS P T= FCRTAMTCC, 2-D MULT 16:1
RS P T= 2DF, 2-D MULTI-GP D1S 17:3
RS P = ISCTOPES, MAXIMUM YIE 17:9
RS P = ISOCRUNCH, REACTION D18:0
NL 260C F63 RS P T = MACH1, 1-C MULTI-GP DIFFUSION SLAB CYL SPHERE A 262
URC 650C F4 RS PLT = MACH1, 1-C MULTI-GP DIFFUSION SLAB CYL SPHERE P 262
703C F4 RS P T = DTF-BURN, 1-C MULTI-GP DTFT WITH DEPLETION LASL 269
F4 RSBPLT = CAESAR4, LILST, 1-C MULTI-GP DIFFUSION + LIB AI 350, 270
360C F36 RS P T = SNARG-1D, 1-C MULTI-GP DISCRETE ORBIT WITH CALC ANL 288
G 11C8 F4 RS P T = GAKIN, 1-C MULTIGROUP TIME-DEPENDENT DIFFUSION G 318
A 11C8 F4 RS P T = FEVER7, 1-C MULTIGROUP DIFFUSION AND DEPLETION G 318
CA F63 RS F T = GAKIT, 1-C MULITIGP KINETICS WITH TEMP FEEDBACK G 376
AI 7C5C F2 RS = PERT, 1-C PERTURBATION FCA AIM AND FCG CCDES 3C
C 1604 F62 RS = PERT, 1-C PERTURBATION FOR AIM AND FCG CCDES 3C
PKC 360 F4 RS P T = GAKIT, 1-C PERTURBATION FCA AIM AND FCG CCDES 3L
AFL 660C F4 RS T = MC64A, 1-C SLAB TRANSPORT WITH FLUXING DOWN B 342
600 F4 RS P T = ML75ELETO, 1-C SLAB GAMMA-RAY TRANSPORT BAPL 6 343
660C F4 RS P T = RALMRZSAF, 1-C TIME-DEPENDENT DIFFUSION CALC KAPL 352
RS = S4 CYL CELL CCDE, 1-C 1-GP S4 APPROXIMATION AI 7C9C F2 53
RS = S4 CYL CELL CCDE, 1-C 1-GP APPROXIMATION AI 16C4 F63 53
F 703C F2 RS P T = SAIL, 1-C 1-GP SN APPROXIMATION SLAB GEOMETRY AI 52
F 1604 F63 RS P T = SAIL, 1-C 1-GP SN APPROXIMATION SLAB GEOMETRY 52
9C F2 RS = SNAPKIN5/5A, 1-REGION KINETICS SNAP GEOMETRY AI 7U 122
CA BNW 11C8 F4 RS P T = 2DE, 2-D MULTIGROUP DIFFUSION AND DEPLET 325
CA LAL 660C F4 RS P T = 2DE, 2-D MULTIGROUP DIFFUSION AND DEPLET 325
UNC-LASL 16C4 F63 RS = 2DF, 2-C MULTI-GP DISCRETE ORBIT WITH CALC 173
TFP, 2-D PERTURBATION TCC CR 2DXY FLUX INPUT PW 16C4 F63 RS = 199
M CYL PW 16C4 F63 RS T = 2DXYL, 3-D MULTI-GP FLUX SYNTHESIS PRCRA 192
CP AGC 7C5C FLCC RSBP = 2DXY, 2-D MULTI-GP SN APPROXIMATION XY GE 18
LASL 7054 FAP RS P T = RATH, 2- CR 3-D HEAT CONDUCTION LUMPED MASS 242
LASL 7030 F4 RS P T = RATH, 2- CR 3-D HEAT CONDUCTION LUMPED MASS 242
G 90 F2 RS P T = EQUIPCISE3, 2-C 2-GROUP DIFFUSION CYL CRL 7 39
G 90 F2 RS P T = EQUIPCISE3A, 2-C 2-GROUP DIFFUSION CYLINDER SLAB CRL 70 87
G 9C F36 RS P T = EQUIPCISE3, 2-C 2-GROUP DIFFUSION CYLINDER SLAB CRL 16 87
AFL 660C F4 RS T = TWIGLE, 2-C 2-GP SPACE-TIME DIFFUSION FEEDBACK B 333
360 F4 RS P T = TWIGLE, 2-C 2-GP SPACE-TIME DIFFUSION ANL 338
= FLCH-MODEL, MULTI-CHANNEL 2-C 2-PHASE FLOW AI 7C94 F2 RS P 246
-PLANE, PLANE STRESS ANALYSIS 2-C BCDIES GGA 11C8 F4 RS P = SAFE 252
T = CMC, DATA PREPARATION FOR 2-C DESIGN PROGRAMS LASL 7C94 F2 RS P 234
AFL 660C F4 RS T = MC80C7, 2-C DIFFUSION ABSCORTION REMOVAL X-SECS B 280
AFCA 7C54 F4 RS P T = MARS, 2-C EXCURSION CALCULATION R-Z GEOMETRY 293
L 7030 F2 RS P T = 2GRAND, 2-C FEW-GROUP DIFFUSION CYLINDER CRN 40
L 1604 F63 RS P T = 2GRAND, 2-C FEW-GROUP DIFFUSION CYLINDER SLAB C 87
CC 7050 F+FAP RSBP T = CDB, 2-C FEW-GROUP DIFFUSION BURNUP RX GEOMETRY 99
IRM 360 F+EAL RS P T = PCQ5, 2-C FEW-GROUP DIFFUSION AND DEPLET 336
G 11C8 F4 RS P T = GAUGE, 2-C FEW-GP HEX GEOMETRY DIFFUSION DEPLET 399
G 11C8 F4 RS P T = GAUGE, 2-C FEW-GP HEX GEOMETRY DIFFUSION DEPLET 399
G 11C8 F4 RS P T = GAUGE, 2-C FEW-GP HEX GEOMETRY DIFFUSION DEPLET 399
BLCSCS6, PCINT-KINETICS WITH 2-C HEAT TRANSFER GGA 7C44 F4 RS P 281
T = ELCSCS6, COMBINE KINETICS WITH 2-C HEAT TRANSFER GGA 11C8 F4 RS P 33
326 F4 RS P T = HSCASAFE, 2-C MCNTE CARLO CARLO CALCULATION AI 37
AGC 7050 FLCC RSBP = 2CXY, 2-C MULTI-GP SN APPROXIMATION XY GEM 18
G F+FAP RSBP = CRAM, 1-C ANS 2-C LLMLI-GP DIFFUSION PROGRAM UK-R 7C9 133
G F+EAL RS P T = CRAM, 1-C ANS 2-C LLMLI-GP DIFFUSION PROGRAM AEC 133
G F+FAP RS P T = EXTERMINA, 2-C LLMLI-GP DIFFUSION PROGRAM CRN 156
F4 RS P T = EXTERMINA2, 2-C LLMLI-GP DIFFUSION PROGRAM ORNL 156
F4 RS T = EXTERMINA2, 2-C LLMLI-GP DIFFUSION PROGRAM ORNL 156
C 4 F63 RS F T = FORTRAN TCC, 2-C LLMLI-GP SN APPROXIMATION RZ PW 161
-LASL 1604 F62 RS = 2DF, 2-C MULTI-GP DISCRETE ORBIT WITH CALC 173
RS P T = VARI-CUR, TIME-DEP, 2-C MULTI-GP DIFFUSION WANN 660C F4 212
RS P T = VARI-CUR, TIME-DEP, 2-C MULTI-GP DIFFUSION UMCC 360 F4 212
7C44 F+FAP RSBP T = GAMBLE4, 2-C MULTI-GP DIFFUSION XY RZ GEOMETRY GGA 222
7C44 F+FAP RSBP T = GAMBLE5, 2-C MULTI-GP DIFFUSION XY RZ GEOMETRY GGA 222
L 705C F+FAP RS P T = ASSALIT, 2-C MULTI-GP DIFFUSION DEPLET CCDE CRN 24C
C 54 F4 RS P T = VARI-QLIR3, 2-C MULTI-GP DIFFUSION XY RZ RTH WANN 7 264
EC 635 F4 RS PLT = BISYN, 2-C MULTI-GP DIFFUSION SYNTHESIS CALC 328
LER 7030 F+FAP RS P T = TCSN, 2-C MULTIGROUP DISCRETE ORBIT WITH CALC 312
ENW 11C8 F4 RS P T = 2DB, 2-C MULTIGROUP DIFFUSION AND DEPLET 325
LASL 660C F4 RS P T = 2DB, 2-C MULTIGRP DIFFUSION AND DEPLET 325
LIFETIME ANALYSIS ANL 36C0 F36 RSBP = SWELL2, FUEL ELEMENT 353
P X-SECTION CALCULATION ANL 36C0 F36 RS P T = MC**2, ENDF MULTIGROUP 355
SUPERCR STRESS ANALYSIS ANL 36C0 F36 RS P T = SUPCRAN, REACTOR CORE 357
ULTRASONIC PARALLEL BEAMS ANL 36C0 F36 RSBP = B0W1, DEFLECTION CALC 365
G CLADDING-STEAM REACTION ANL 36C0 F36 RSBP = CHEMLOC2, CORE HEATING 366
IPING SYSTEM ANALYSIS KE 36C0 F4 RSBP = WHAM, LIQUID-FILLED P 378
ECTRUM X-SECTION CALC ANL 36C0 F36 RSBP T = GAM2, FAST NEUTRON SP 389
CS-FYCR CONDUCTION SPHERE CDC 36C0 F36 RSBP = AX1, COUPLED NEUTRON 392
NSFER FUNCTION SYNTHESIS ANL 36C0 F36 RSBP T = TRAFICCRPCRATION, TRA 403
LSTICRCN PARALLEL BEAMS ANL 36C0 F36 RSBP = B0W2, EFFECTIVE CALC 415
G CLACCING-STEAM REACTION ANL 36C0 F36 RSBP = CHEML0C2, CORE HEATING 426
IPING SYSTEM ANALYSIS KE 36C0 F4 RSBP T = ARGUS, TRANSIENT TEMP 438
REQUIREMENT STUDY ANL 36C0 F36 RS P T = R153, PARAMETRIC SITE 450
RARY UTILITY ROUTINE ANL 36C0 F36 RSBP T = FLARE, 3-C REACTIVITY 461
SPECTRUM X-SECTION CALC ANL 36C0 F36 RS P T = FLARE, 3-C REACTIVITY 472
INTEGRAL CALC 2-REG CELL ANL 36C0 F36 RSBP T = 2PLUS, NON-SPHERICAL 483
INTEGRAL CALC FMCGENECLS ANL 36C0 F36 RSBP T = 2PLUS, NON-SPHERICAL 494
VATING DATA REDUCTION ANL 36C0 F36 RSBP T = 2PLUS, NON-SPHERICAL 505
CPTICAL MCCDL X-SECTIONS ANL 36C0 F36 RSBP T = 2PLUS, NON-SPHERICAL 516
NT KINETICS KEX CPTIONS ANL 36C0 F36 RSBP T = 2PLUS, NON-SPHERICAL 527
IFUSION SLAB CYL SPHERE ANL 36C0 F36 RSBP T = 2PLUS, NON-SPHERICAL 538
1C90 F2 RS P T = WHIRLAWAY, 1-C 2-GP DIFFUSION DEPLETION 549
1604 F36 RS P T = WHIRLAWAY, 1-C 2-GP DIFFUSION DEPLETION 560
11C8 F4 RS P T = SAFE-3C, F4 FASC RS T = PEQO7, 1,2 OR 3-C FEW-GP DIFFUSION DEPLETION 571
6C F4 RS P T = PEQ07, 1,2 OR 3-C FEW-GP DIFFUSION DEPLETION 582
CC 11C8 F4 RS P T = GATT, 3-C FEW-GP DIFFUSION CALC HEX-Z MESH 593
6C F4 RS P T = GATT, 3-C FEW-GP DIFFUSION CALC HEX-Z MESH 604
GGA 11C8 F4 RS P T = GATT, 3-C FEW-GP DIFFUSION CALC HEX-Z MESH 615
C30 F4 RS P T = RAY, 2-GP DIFFUSION CALC HEX-Z MESH 626
W 16C4 F63 RS T = AX1, REACTIVITY AND POWER DISTRIBUTION 637
EC 635 F4 RS P T = FLARE, REACTIVITY AND POWER DISTRIBUTION 648
CC 36C6 F63 RSBP T = FLARE, REACTIVITY AND POWER DISTRIBUTION 659
KAPL 66C6 F4 RS P T = LION, 3-C TEMPERATURE DISTRIBUTION PROGRAM 670
GEC 635 F4+GMP RSBP T = THETE, 3-C TRANSIENT HEAT TRANSFER PROGRAM 681
1-C 2-GP SPACE-TIME DIFFUSION 692
LC AI 7C95 F+FAP RS T = 3-CEDW, BAPL 66C0 F4 RS P T = WIGL2, 3-C TRANSIENT HEAT TRANSFER PROGRAM 703
ICN SLAB CYLINDER SPHERE BC 625 F4 RS T = FCG, 1-C FEW-GP DIFFUSE 714
TI-GP DIFFUSION PROGRAM BC 625 F4 RS T = EXTERMINATOR2, 2-D MUL 725
TANT CALC C TC 1C MEV BC 625 F4 RS T = GATEC2, MULTI-GP CCSN 736
CRETE CIRCULAR PROGRAM BC 625 F4 RS T = GATEC2, MULTI-GP CCSN 747
SECTION CALCULATION GEC 625 F4 RS P T = AVOID, ANNULAR VELOCITY 758
AND FLOW DISTRIBUTION NED 625 F4 RS P T = AVOID, ANNULAR VELOCITY 769
CLRSICN CALCULATIONS NED 625 F4 RS P T = AVOID, ANNULAR VELOCITY 780
ERMAL-HYDRALIC ANALYSIS NED 625 F4 RS P T = AVOID, ANNULAR VELOCITY 791
CE EFFIT-ERMAL X-SECTIONS NED 625 F4 RS P T = AVOID, ANNULAR VELOCITY 802
EC NEUTRON SCLRE CATA NED 625 F4 RS P T = AVOID, ANNULAR VELOCITY 813
FFUSION SYNTHESIS CALC NED 625 F4 RS P T = AVOID, ANNULAR VELOCITY 824
ISCTCPIC BURNUP ANALYSIS GEV 625 F4 RS P T = AVOID, ANNULAR VELOCITY 835
METERS FAST BREECERS APO 625 F4 RS P T = AVOID, ANNULAR VELOCITY 846
AT TRANSFER PRGCM GEC 635 F4+GMP RSBP T = THETE, 3-D TRANSIENT HE 857
NT + EMERGENCY COOLING BCL 644 F4 RS P T = ECCC5A1, LCCS-OF-CCLLA 868
IFFUSION SLAB CYL SPHERE PURC 65C0 F4 RS PLT = MACHI, 1-C MULTI-GP 879
SHELFS CE REVOLUTION KAPL 66C0 F4 RS P T = SCR2, STRESS ANALYSIS 890
CRETE CIRCULAR PROGRAM BC 66C0 F4 RS P T = SCR2, STRESS ANALYSIS 901
SCRETE CIRCULAR PROGRAM BC 66C0 F4 RS P T = SCR2, STRESS ANALYSIS 912
IS CYL FUEL ELEMENT BAPL 66C0 F4 RS T = CYGRO2, STRESS ANALYSIS 923
4-5-14.5GCPspa 32-472CECF BAPL 66C0 F4 RS P T = NUCLEUS, 1-C 2-GP SPACE 934
LING TEMPERATURE STUDY BAPL 66C0 F4 RS P T = FIGRO, 1-C 2-GP SPACE 945
-E TIME DIFFUSION 2-GECM BAPL 66C0 F4 RS P T = FIGRO, 1-C 2-GP SPACE 956
ABSORPTION REMCVAL X-SECS BAPL 66C0 F4 RS P T = MC87, 2-D DIFFUSION 967
ESS ANALYSIS AXISYM LOAD BAPL 66C0 F4 RS P T = SEALFUSH2, SHELL STR 978
<table>
<thead>
<tr>
<th>Process/Program</th>
<th>Code/Codebook</th>
</tr>
</thead>
<tbody>
<tr>
<td>Accident Analysis</td>
<td>BAPL 66CC F4</td>
</tr>
<tr>
<td>Accident Analysis</td>
<td>BAPL 66CC F4</td>
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<td>Accident Analysis</td>
<td>BAPL 66CC F4</td>
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</tbody>
</table>

**Note:** The codes and codebooks listed are likely references to specific programs or systems used in accident analysis and related studies.
SICPLCT, RESCLVEC MULTILEVEL B-W X-SEC CALC BNL 660C F4 RS P = 377
RES BAPL 660C F4 RS P T = BL47, DRAFTING TOOL TC PLOT PLANE STRUCTU R373
STUCY AEB 360C F4 RS P = BLAST, REACTOR KINETICS TEMPERATURE DlST 363
NSFER GGA 7C44 F4 RS T = BLCOST5, POINT-KINETICS WITH 2-C HEAT TRA 205
FER GGA 11C8 F4 RS P T = BLCOST6, COMBINED KINETICS 2-C HEAT TRANS 303
KE 7094 F4 RSEP T = PTH1, BLCWDWN PRESSURE TEMPERATURE HISTORY 155
7G94 F+MAP RSEP T = SATURATEC BLCWDWN ANALYSIS LCFT KE 7094 F+MAP 20C
7G94 F+MAP RSEP T = SATURATEC BLCWDWN, BLOWDOWN ANALYSIS LCFT KE 20C
RS P = WATER-HAMMER, LIQUID BLCWDWN ANALYSIS LCFT UGA 36C F4 278
F4 RS P T = RELAP2, REACTOR BLCWDWN - EXCURSION ANALYSIS INC 7C44 369
VEC REGION AVERAGE X-SEC CALC BNL 6600 F4 RS P = AVERAGE, UNRESOL 376
C MULTIPLE B-W X-SEC CALC BNL 6600 F4 RS P = SIGPLCT, RESOLVE 377
SERVICE ROUTINES ENDF TAPES BNL 7090 F+FAP RS P T = DFSR, DATA FILE 236
VED REGION AVERAGE X-SEC CALC BNL 7094 F4 RS P = AVERAGE, UNRESOL 376
C MULTILEVEL B-W X-SEC CALC BNL 7094 F4 RS P = SIGPLCT, RESOLVE 377
AL EXPERIMENT ANALYSIS SYSTEM BNL-DP 7090 F2 RS P T = HAMMER, CRITIC 277
AL EXPERIMENT ANALYSIS SYSTEM BNL-DP 360 F4 RS P T = HAMMER, CRITIC 277
P CONSTANT CALC 0 TO 16 MEV BNW 1107 F4 RS P T = GAMEC2, MULTI-G 185
P DIFFUSION SLAB GEOMETRY BNW 1107 F4 RS P T = HFN, 1-D MULTI-G 241
RATATION XY RZ RT-EETA GEOMETRY BNW 1108 F4 RS P T = FFCC, FUNDAMENTA 306
CUP DIFFUSION ANC DEPLETION BNW 1108 F4 RS P T = 2CB, 2-D MULTIG 325
LIDE GENERATION ANC DECA BNW 1108 F5 RSBP T = ISOGEN, RADICNU 367
MULTI-GP COLLE D N AFPRCX BNW 7090 FLCC F4 RS P T = GE-HAP0-S13, 1-D 75
GA MULTIGROUP SLAB GEOMETRY BNW 7090 F2 RS P T = MAC, SHIELD DESI 143
P CONSTANT CALC C TO 1C MEV BNW 1107 F4 RS P T = SAFE-PLA 252
AT MCCE FAST X-SECTION GENERATION BNW 1107 F4 RS P T = 1CX, 1-D DIFFUSI 374
NE, PLANE STRESS ANALYSIS, 2-C BCCIES GGA 1108 F4 RS P T = SAFE-PLA 252
NT AGC 7090 F2 RS P = BOLNCE, FLUX DIST IN MULTI-PIN FUEL ELEME 237
RS ANL 360C F36 SBP P = BAW2, DEFLECTION CALCULATICN PARALLEL BEA 365
FL, FUEL CYCLE PARAMETERS FAST BREEDERS APO 635 F4 RS P = RAP 372
S P = WELDING, MATERIAL BUCKLING CYL FUEL ELEMENTS AEB 36C F4 362
HEAT TRANSFER MOLTEN FUEL TUBE BUNCLASSES LASL 7094 F2 RS P = AXFLU, 182
HEAT TRANSFER SOLID FUEL TUBE BLNACES LASL 7694 F2 RS P = AXTHRM, 183
T = CCB, 2-C FEW-GP DIFFUSION BURNUP RZ GEOMETRY GGA 769C F+MAP RS P T = 99
BURNP, HEAVY ELEMENT ISOTOPIC BURNUP ANALYSIS GEV 635 F4 RS P T = 311
1100 F4 RS P T = CAMP7, 1-C BURNUP POWER DISTRIBUTION SEARCH GGA 319
LYSIS GEV 625 F4 RS P T = BURNUP, HEAVY ELEMENT ISOTOPIC BURNUP ANA 311
UM 769C MAC RSB = BURNUP, DEFOCTOR EFFICIENCY POINT SOURCE 164
UM 769C MAC RSB = BURNUP, DEFOCTOR EFFICIENCY DISCO RIS 165
UM 769C MAC RSB = BURNUP, DEFOCTOR EFFICIENCY POINT SOURCE 166
TAL UM 769C MAC RSB = BURNUP, GAMMA-RAY PHOTOCFRACTION SOLID CRYST 169
AL UM 769C MAC RSB = BURNUP, GAMMA-RAY PHOTOCFRACTION WELL CRYST 170
RESCANCE INTEGRAL CALCULATION BN 2000 F4 RS P = STRIP, RESCVLED 305
LIB AI 36C F4 RSBPL = CAESAR4. LIBVST, 1-C MULTI-GP DIFFUSION + 270
AT, VIEW FACTOR SHIELDING CODE CAVITY GECN GGA 11C8 F4 RS P T = MUSC 259
FESF-EACH INELASTIC SCATTERING CCC 16C4 F63 RS P = HAFEVER, HAUSER- 14
CILFUSION SLAB CYLINDER SPHERE CCC 16C4 F63 RS P T = FFG, 1-D FEW-GP 28
GP DIFFUSION SLAB CYL SPHERE CCC 16C4 F63 RS P T = AIM6, 1-D MULTI- 29
BATICF FCR AIM ANC FCG CODES CCC 16C4 F63 RS P T = PERT, 1-D PERTUR 32
-CRLCP DIFFUSION XYZ GEOMETRY CCC 16C4 F63 RS P T = WHIRLAWAY, 3-D 2 32
CN SPECTRUM X-SECTION CALC CCC 16C4 F63 RS P T = GAM1, FAST NEUTR 33
-CN SPECTRUM X-SECTION CALC CCC 16C4 F63 RS P T = GAM1, FAST NEUTR 33
Y ATTENTION SLAB GEOMETRY CCC 16C4 F63 RS P T = SAIL, 1-D 1-GP 46
Y ATTENTION CYL SPHERE GEOM CCC 16C4 F63 RS P T = SAIL, 1-D 1-GP 46
CCSE RATE FRCM A CLOUC CCC 16C4 F63 RS P T = SAIL, 1-D 1-GP 46
L KELTRHCA SPECTRUM X-SECTIONS CCC 16C4 F63 RS P T = TEMPEST2, THERMA 50
CN SPECTRUM X-SECTION CALC CCC 16C4 F63 RS P T = TEMPEST2, THERMA 50
N APPROXIMATION SLAB GEOMETRY CCC 16C4 F63 RS P T = SAIL, 1-D 1-GP 52
1-1 1-GP S4 APPROXIMATION CCC 16C4 F63 RS P T = S4 CYL CELL CODE 53
CRCLP DIFFUSION DEPLETION CCC 16C4 F63 RS LT = SIZZLE, 1-C MULT 58
2-GP DIFFUSION CYLINDER SLAB CCC 16C4 F63 RS P T = ECUIPCISE3A, 2-D 87
7, 1-C MULTI-GP DIFFUSION SLAB CYL SPHERE BNW 11C7 F4 RS P T= HF 241
1, 1-C MULTI-GP DIFFUSION SLAB CYL SPHERE ANL 3600 F63 RS P T= MAC 262
1, 1-C MULTI-GP DIFFUSION SLAB CYL SPHERE PURD 650C F4 RS PLT= MAC 262
FIRE5, 1-C AGE-DIFFUSION SLAB CYL SPHERE AEB 36u F4 RS PLT= 9
FCG, 1-C FEW-GP DIFFUSION SLAB CYLINDER SPHERE AI 7090 F2 RS P T= 28
FCG, 1-C FEW-GP DIFFUSION SLAB CYLINDER SPHERE CDC 164F F63 RS P T= 28
FCG, 1-C FEW-GP DIFFUSION SLAB CYLINDER SPHERE BC 625 F4 RS T= 28
, 2-C FEW-GERLP DIFFUSION SLAB CYLINDER CRNL 709G F2 RS P T= 2OGRAND 4C
, 2-C FEW-GERLP DIFFUSION SLAB CYLINDER CDC 164F F63 RS P T= 2OGRAND 4C
CLITPI3E, 2-C 2-GP DIFFUSION SLAB CYLINDER SLAB ORNL 750C F2 RS P T= 28
CLITPI3E, 2-C 2-GP DIFFUSION SLAB CYLINDER CDC 164F F63 RS P T= 28
, 2-C FEW-GP SN APPROXIMATATION CYLINDER PPCC 7090 F2 RS P T= 28
, 2-C FEW-GP SN APPROXIMATATION CYLINDER PPCC 7090 F2 RS P T= 28
LS, TRANSIENT TEMPERATURE CALC CYLINDER ANL 3600 F63 RS P T= ARG 152
URTQN PENETRATION CALCULATION DAC 7CG F2 RS P T= LIPRECML MC NE 123
DATA BAPL 660C F4 RS P T= DAFT1, LEAST SQUARES FIT FISSILE NUCLIDE R327
C F4 RS P T= CHECKER.CRECI, DAMM.EET.PLCTFBE.SLAKE3 ENDF/SC NSC 66G 384
NCT AEG 7CG F2 RS = DCANOFF JR, MODERATOR SPACE CHORD DIST FU 150
, FUEL CYCLE CCS TS PERFORMANCE DATA KE 7CGC F2 RS P T= ANPRFCSCP 146
X-SECTIONS FROM B-w RESONANCE DATA GGA 704F F4 RS P = FASCO 216
LYSIS CF PULSEC NEUTRON SOURCE DATA NEC 635 F4 RS P T= EXP, ANAL 258
ST SQUARES FIT FISSILE NUCLUC DATA BAPL 660C F4 RS P T= DAFT1, LEA R327
CRE CCEC CALC FCR ANGULAR DIST DATA AI 7094 F+MAP RS P = CHAD, LEGEN 215
RS P T= ETIOE, ENCF/B TC MC=2 DATA CONVERSION APDA 360V ASAFC 350
BNL 7CG0 F+FAP RS P T= DFJ, DATA FILE SERVICE Routines ENDF TAPES 236
CCONTS CALC FROM TO REA L DATA LASL 660C F4 RS P T= GLEN, GROUP 361
= ECSIL, EXPERIMENTAL NEUTRON DATA LIBRARY LRL 709G F2 RS PLT= CFSL 351
IT, CURVE FITTING EXPERIMENTAL DATA POINTS KAPL 660C F+ASC RS P T= CURF 430
LASL 709G F2 RS P T= CPC, DATA PREPARATION FCR 2-C DESIGN PRGRAMS 234
NCE2, ENCFB THERMAL SCATTERING DATA PRCC DP 364 F4 RS P T= FLA 368
= CCINC, COINCIDENCE CCLNT DATA REDUCTION ANL 360C F63 RS P = CCF 248
, -III ASSEMBLY 48 GAFGAR ENCF/B DATA TAPES GGA 1108 BIN R LT= ZPR 356
AST SQUARES ANALYSIS RESONANCE DATA ULL 360C F4 RS P = CDFIL, LE 347
6CG F4 RS P T= RESQ1, RESQ2, DBF1, RESONANCE INTEGRAL HEX CELL BAPL 6 R285
TRY GGA 7CGC F+FAP RS P T= CCE, 2-C FEW-GP DIFFUSION BURNUP RZ GEGME 99
K, RACIUNUCLE GENERATION AND DECAY BNW 11C8 S5 RS PBLT= ISGEC 367
RS P = ISCRNLCH, REACTION DECAY CHAIN ANALYSIS ORNL 7090 F2 18C
RS P = ISCRNLCH, REACTION DECAY CHAIN ANALYSIS ORNL 164F 16C 18G
ALS, LEAST SQUARES EXPERTICAL DECAY CURVE LRL 7CG4 F2 RS P = EXP 321
MAXIMUM YIELD FROM REACTION OR DECAY ORNL 164F 26z RS P = ISCTCPES, 179
ANL 3600 F63 S6P B BOW2, DEFLECTION CALCULATION PARALLEL BEAMS 365
CHEE, MCNTE CARLEC SLC INC-DGW DENSITY CALC AI 7094 F+FAP RS PLT= T 149
ZZLE, 1-C MULTIGRPDP DIFFUSION DEPLETION AI 709C F2 RS LT= S1A 58
ZZLE, 1-C MULTIGRPDP DIFFUSION DEPLETION CDC 164F F63 RS LT= S1A 58
ZZLE, 1-C MULTIGRPDP DIFFUSION DEPLETION AL 364 F4 RS P T= S1A 58
FIVE, 1-C FEW-GP DIFFUSION DEPLETION PROGRAM GGA 764 F2 RS T= 17
AC-FEVER, 1-C FEW-GP DIFFUSION DEPLETION GGA 764 F4 RS P T= REL0 221
GSAI, 2-C MULTI-GP DIFFUSION DEPLETION CCDE ORNL 709C F+FAP RS P T= A 241
F-PURN, 1-C MULTI-GP DTFA WITH DEPLETION BAPL 660C F+ASC RS P T= A 269
7, 1,2 CR 3-C FEW-GP DIFFUSION DEPLETION BAPL 660C F+ASC RS P T= A 275
7, 1,2 CR 3-C FEW-GP DIFFUSION DEPLETION IBP 364 F4 RS P T= A 275
RS P = CINCEH.PMTC2, POINT DEPLETION FISSION PRODUCTS BAPL 660C F4 313
1, 1-C MULTIGRPDP DIFFUSION ANC DEPLETION GGA 11C8 F4 RS P T= FEV7R 38
, 2-C MULTIGRPDP DIFFUSION ANC DEPLETION BNW 11C8 F4 RS P T= 2DB 325
, 2-C MULTIGRPDP DIFFUSION ANC DEPLETION LAL 66G F4 RS P T= 2DB 325
5, 2-C FEW-GRPDP DIFFUSION ANC DEPLETION IBP 364 F+ASC RS P T= PDQ 336
2-C FEW-GP HEX CCEM DIFFUSION DEPLETION GGA 11C8 F4 RS P T= GAUGE, 339
= CCRCK, SPACE POWER PLANT DESIGN OPTIMIZATION AI 7CGC F2 RS 112
= SHOC, SPACE POWER PLANT DESIGN OPTIMIZATION AI 76C F2 RS 114
LSC F2 RSPLT= MAC, SHIVEL DESIGN MULTIGRPDP SLAB GEOMETRY BNW 7 143
6CG F63 RSPLT= MAC, SHIVEL DESIGN MULTIGRPDP SLAB GEOMETRY ANL 3 143
PIP, CENTRIFICAL TMP IMPPELLER DESIGN STLCY PW 164F 63 RS P T= 187
P T= STMGAL, STEAM GENERATOR DESIGN CRITERIA CCSIS GGA 764 F4 RS 227
X-SECTION CALC
AI 7654 F+MAP RS PLT= TRIX1, RESONANCE INTEGRAL 268
CR ANGULAR DIST DATA AI 7654 F+MAP RS P = CHAD, LEGENCRE CCEF CALC F 215
KL LATTICE EFFECTS GGA 7044 F+MAP RS P T= GARCL, RESONANCE OVERLAP A 219
LSICN XY RZ GEOMETRY GGA 7644 F+MAP RSBP T= GAMBLE4, 2-C MULTI-GP DIFF 222
ACE AND PRICE STLCY GGA 7044 F+MAP RS P T= OPUS, POWER PLANT PERFORMA 226
BRRNL P CYL LATTICE WAPC 7654 F+MAP RS PLT= LASER, SPECTRUM CALC WITH 249
ANALYSIS PROGRAM PPCO 7044 F+MAP RS P T= RSAC, RACIOLOGICAL SAFETY 265
ACCIDENT ANALYSIS PPCO 7044 F+MAP RS P T= CONTEMPT, LCSS-CF-COOLANT 297
ETE CRDINATE PROGRAM LER 7090 F+MAP RS P T= TDSN, 2-D MLLTIGROUP DISCR 312
HEAT CONDUCTION PPCO 7044 F+MAP RS P T= TOODEE, 2-D TIME-DEPENDENT 349
PARFU, FUEL CYCLE PARAMETERS FAST BREEDERS APC 635 F4 RS P = 372
GGA 7654 F2 RS PLT= GAM1, FAST NUTRON SPECTRUM X-SECTION CALC 33
CCE 1644 F63 RS PLT= GAM1, FAST NUTRON SPECTRUM X-SECTION CALC 33
ANL 3660 F63 RS LT= GAM1, FAST NUTRON SPECTRUM X-SECTION CALC 33
AI 7652 F63 RS PLT= FCRM, FAST NUTRON SPECTRUM X-SECTION CALC 51
CCE 1564 F63 RS PLT= FORM, FAST NUTRON SPECTRUM X-SECTION CALC 51
ECC 2060 F4 RS P T= FCREE, FAST REACTCR EXCURSION CALCULATIONS 174
EE 635 F4 RS P T= FCRE2, FAST REACTCR EXCURSION CALCULATIONS N 174
NL 3664 F4 RS P T= FCRE2, FAST REACTOR EXCURSION CALCULATIONS A 174
S P T= FCC4, FUNDAMENTAL MODE FAST REACTOR X-SECTION CALC BNW 118 F4 R 36
7652 F2 RSBPLT= AGN-GAM, FAST SPECTRUM MULTI-GP CONSTANT CALC ACG 204
LT= GGA4, MULTI-GP X-SECTIONS FAST THERMAL SPECTRA GGA 118 F4 RS P 298
RS P T= 1CX, 1-C DIFFUSION FAST X-SECTION GENERATION BNL 118 F4 374
CALC BNW 118 F4 RS P T= FCC4, FUNDAMENTAL MODE FAST REACTOR X-SEC 36
2-C 2-GP SPACE-TIME DIFFUSION FEEDBACK BAPL 666 F4 RS P T= TWWGL, R338
1-C MULTI-GP KINETICS WITH TEMP FEEDBACK GGA 118 F4 RS P T= GAKIT, 70
1-C 2-GP KINETICS TEMPERATURE FEEDBACK BAPL 666 F4 RS P T= NCWIG, R371
ETICH GGA 118 F4 RS P T= FEVERT, 1-C MULTI-GP DIFFUSION AND DEPL 318
GRAH GGA 7650 F2 RS T= FEVER, 1-C FEW-GP DIFFUSION DEPLETION PRO 117
7652 F2 RS P T= FCG, 1-C FEW-GP DIFFUSION SLAB CYLINDER SPHERE AI 28
1644 F63 RS P T= FCG, 1-C FEW-GP DIFFUSION SLAB CYLINDER SPHERE CDC 28
6254 F4 RS P T= FCG, 1-C FEW-GP DIFFUSION SLAB CYLINDER SPHERE BC 28
7652 F2 RS P T= MIST, 1-C FEW-GP SN COUPLE SN APPROX SLAB GECM PPCO 59
7652 F+MAP RSBP T= CCB, 2-C FEW-GP DIFFUSION BURNUP RZ GEOMETRY GGA 97
7650 F2 RS P T= FSR, 1-C FEW-GP DIFFUSION DEPLETION GGA 117
7644 F4 RS P T= TOPIC, 1-C FEW-GP DIFFUSION DEPLETION PROGRAM GGA 148
7650 F4 RS P T= TOPIC, 1-C FEW-GP DIFFUSION DEPLETION PPCO 148
7644 F4 RS P T= GASPS, 1-C FEW-GP DIFFUSION PCWER DIST SEARCH GGA 220
RS P T= RELCAC-FEVER, 1-C FEW-GP DIFFUSION DEPLETION D GGA 221
7644 F4 RS P T= TEMCC, 1-C FEW-GP DIFFUSION TEMP CCEF CALC GGA 225
66 F4 RSBP T= THEREDS, 1-C FEW-GP DIFFUSION DESIGN SYSTEM AI 3 273
+ASC RS T= PCC7, 1,2 CR 3-C FEW-GP DIFFUSION DEPLETION BAPO 666 F R275
4 RS P T= PCC7, 1,2 CR 3-C FEW-GP DIFFUSION DEPLETION IBM 36. F R275
118 F4 RS P T= GCLCE, 2-C FEW-GP HEX GEOM DIFFUSION DEPLETION GGA 339
1184 F4 RS P T= GATT, 3-C FEW-GP DIFFUSION CALC HEX-Z MESH GGA 386
6L F4 RS P T= 2CRAND, 2-C FEW-GROUP DIFFUSION SLAB CYLINDER CRNL 70 40
64 F63 RS P T= 2CRAND, 2-C FEW-GROUP DIFFUSION SLAB CYLINDER CDC 16 4
36C F+MAP RS P T= PCC5, 2-C FEW-GROUP DIFFUSION AND DEPLETION IBM R336
LY BAPL 666 F4 RS P T= FCRO, LSBR FUEL SWELLING TEMPERATURE STUD 272
7654 F+MAP RS P T= DFSR, DATA FILE SERVICE ROUTINES ENDF TAPES BNL 236
SCARF2, SCATTER FROM RADIATOR FINS SNAP GECM AI 7654 F2 RS = 140
HERE AEB 26C F4 RS PLT= FIRE5, 1-D AGE-DIFFUSION SLAB CYLINDER SP 9
AL 7654 F4 RSBP = MLFFLE, FISSILE NUCCLIDE X-SECTION EVALUATION OR 323
P T= CAFT1, LEAST SQUARES FINT FISSILE NUCCLIDE DATA BAPL 666 F4 RS R327
7644 F4 RS P T= KACZ, HTGR FISSION PRODUCT DEACTIVITY DIST STUDY GGA 231
S P = FREVAF6, HTGR METALLIC FISSION PRODUCT RELEASE GGA 118 F4 R 312
CINCER, MG102, POINT DEPLETION FISSION PRODUCT BAPL 666 F4 RS P = 313
N, CRNL 1664 F63 RS P = JLJ, TCR1, COUPLED-CHANNEL X-SEC EVALUATION 3.8

LATTICE DYNAMICAL PROGRAM KAPL 666G F4 RS P T = LION, 3-D TEMPER R299

ERATION FOR A SYSTEM OF BEAMS KAPL 666G F4 RS P = STEM, MATRIX GEN R337

pcblem ECR VIBRATING SYSTEMS KAPL 666G F4 RS P T = GEM, EIGENVALUE R344

ME-CEPENDING DIFFUSION CALC KAPL 666G F4 RS P T = RAUMZEIT, 1-D TI R352

TANGENTIAL CURVE PLOT KAPL 666G F4 RS P T = SNEQ, NONLINEAR R364

1624, SPACE-INDEPENDENT INVERSE KINETICS CALC KAPL 666G F4 RS P T = TUBE, U-TUBE HEA R378

LED FIPING SYSTEM ANALYSIS KE 36L0 F4 RSBP = WHAM, LICEFIL-2 278

CLE CCSTS PERFORMANCE DATA KE 7,94 F2 RSBP T = PTHL, BLCWMCN P 155

RESLUE TEMPERATURE HISTORY KE 7,94 F4 +MAP RSBP T = SATURATED BLCKDO 204

SLMMT, CRYSTALLINE SCATTERING KERNEL CALC GGA 7,90 F2 RS T = 56

SI, SPACE-INDEPENDENT KINETICS KEX OPTIONS ANL 36C6 F63 RSBP = R1 255

31, SPACE-INDEPENDENT KINETICS KEX OPTIONS WANNL 7094 F4 RS P = R1 255

PI T = AIREEK3, SPACE-INDEPENDENT KINETICS W/FEEDBACK A1 7019 F2 RS 121

PI T = AIREEK3, SPACE-INDEPENDENT KINETICS W/FEEDBACK AEB 36C6 F4 RS P 121

RS P = SNAPKINS/EA, 1-REGION KINETICS SNAP GEOMETRY A1 7019 F2 122

IC2, SPACE-INDEPENDENT INVERSE KINETICS CALC ANL 36C6 F63 RSBP = R1 168

IC2, SPACE-INDEPENDENT INVERSE KINETICS CALC WANNL 7094 F4 RS = R 168

BP = RIC1, SPACE-INDEPENDENT KINETICS KEX OPTIONS ANL 36C6 F63 RS 255

P = RIC1, SPACE-INDEPENDENT KINETICS KEX OPTIONS WANNL 7094 F4 RS 255

SA, STABILITY ANALYSIS REACTOR KINETICS ECNS GGA 1148 F4 RS P = GA 290

RS P T = ELCCST16, COMBINED KINETICS 2-D HEAT TRANSFER GGA 1148 F4 3.3

N, SPATIALLY-CEPENDING REACTOR KINETICS AI 7019 F4 RS P T = TS 309

N, SPATIALLY-CEPENDING REACTOR KINETICS AI 36C6 F4 RS P T = TS 309

KIN, SPACE-INDEPENDENT REACTOR KINETICS GGA 1148 F4 RS = GAPOT 317

F4 RS P = BLAST, REACTOR KINETICS TEMPERATURE DIST STUDY AEB 360 363

4F4 RS P T = CAYK1, 1-C MULTIGP KINETICS WITH TEMP FEEDBACK GGA 1148 F4 370

4F4 RS P T = NCHIG, 1-C 2-GP KINETICS TEMPERATURE FEEDBACK BAPL 66,0 R371

ALC FRMC CSN CTNPUT PW 16L4 LAC1 RS P = DT, EFFECTIVE X-SECTION C 218

RCX SLAB SYL SPHERE PW 16C4 LAC1 RS P = MGDSN, 1-D MULTI-GP SN APP 218

PW 16C4 F4+CPP RS P T = LAC, ASSEMBLER FOR FLCON2, CURVE FITTING 186

ICE WAPC 7054 F+MAP RS PLT = LASER, SPECTRUM CALC WITH BURNUP CYL LATI 249

ti-CF TRANSPORT CCCE X-Y GEOM LASL 710C F4 RS P T = XOTNTRAN, 2-D MUL 358

GP CIRCULATE CIRCULATE PROGRAM LASL 666G F4 RS P T = DTF3, 1-D MULTI- 219

CUP DIFFUSION AND THERMAL ITERATING CRYSTALLO MATERIALS LASL 666G F4 RS P T = TCT, THERMAL SCA 369

TANT CALC FROM TCR CLTPLT DATA LASL 666G F4 RS P T = GLEN, GROUP CCNS 361

ES, GENERAL CURVE FITTING LASL 7054 F4 RS = LASL LEAST SQUARE 62

PSIOE CF MULTI-GP X-SECTIONS LASL 705C FLCCG RS P = ZCT, GROUP-CNCLA 113

SFER MOLTEN FUEL TUBE BLNCLES LASL 7094 F2 RS P = AXFLU, HEAT TRAN 182

AESR SLOCL FUEL TUBE BLNCLES LASL 7094 F2 RS P = AXTHUR, HEAT TRA 183

HELITCEN PENETRATION SILC LASL 709C FLCCG RS P = MCS, MCNTE CARL 182

BP CIRCUATE CIRCUATE PROGRAM LASL 7030 F4 RS P T = DTF4, 1-D MULTI- 229

ATCH FOR 2-C DESIGN PROGRAMS LASL 7090 F2 RS P T = DPP, DATA PREP 234

HEAT CONDUCTION LUMPED MASS LASL 7094 FAP RS P T = RATH, 2-C 3-D 242

HEAT CONDUCTION LUMPED MASS LASL 703C F4 RS T = RATH, 2-C 3-D 242

LTI-CP CTDF WITH DEPLETION LASL 7030 F4 RS P T = DTF-BURN, 1-C MU 269

PPCCELCE WIRING LISTS UFEX LASL 7090 F2 RS P = WIREX, COMPUTER- 315

LASL 7054 F4 RS = LASL LEAST SQUARES, GENERAL CURVE FITTING 62

7CSC F2 RSBP = FERESY, LATTICE PARAMETERS HETEROGENEOUS CALC FMA 36

GAPCL, RESONANCE OVERLAP AND LATTICE EFFECTS GGA 7044 F+MAP RS P T = GARCL, RESONANCE OVERLAP AND LATTICE EFFECTS LER 7094 F4 RS P T = GARCL, RESONANCE OVERLAP AND LATTICE EFFECTS LER 7094 F4 RS P T = LER 219

SPECTRUM CALC WITH BURNUP CYL LATTICE WAPC 7094 F+MAP RS PLT = LASER, T, ELASTIC SCAT X-SECTIONS HEX LATTICE GGA 1148 F4 RS = HEXSCA 291

T, ELASTIC SCAT X-SECTIONS HEX LATTICE GGA 1148 F4 RS P 263

RS = FLANGEI, SCATTERING LATTICE X-SECTION CALCULATION GGA 7,44 F4 247
KAPL 66CC F4 RS P = STEM, MATRIX GENERATION, FCR A SYSTEM OF BEAMS R337
T= 4RESTRAINT PIPE STRESS, MAXIMUM MC MT CALC AI 7C9F F+RAP RS 1-5
1260 F63 RS P = TSCDCPES, MAXIMUM YIELD FROM REACTION OR DECAY CRNL 179
7C9G F2 RSEP = LIPRECA, MC NEUTRON PENETRATION CALCULATION DAC 123
C4 F+CCP RS P T = FMC-N+FMC-G, MC NEUTRON, GAMMA-RAY HISTORIES PW 195
ASAFA4 RS P T = EICE, ENF/B TC MC++2 DATA CONVERSION APDA 36J 35J
ICA ANL 36CC F36 RS P T = MC++2, ENDF MULTIGRP X-SECTION CALCULAT
Y LASL 7C5C FLCCG RS P = MCS, MCNE CARLC NEUTRAN PENETRATION STUD 222
FAP RS LT = QUICKIE, INFINITE MEDIUM SPECTRUM X-SECTIONS AI 7C9F F+ 119
8 F4 RS P = FREEPAF, HTC METALLIC FISSIATION PRODUCT RELEASE GGA 11L 3-1
HERE PW 16C4 LAG1 RS P = MCCSN, 1-D MULTI-GRP SN APPROX SLAB CYP SP 2-4
NECLS ANL 36CC F63 RSEP = MISH-MASH, RESONANCE INTEGRAL CALC HCPDGE 2-4
CEC PCC 7C5C F2 RS P = MIST, 1-D FEW-GRP SN DUALC SN APPRX SLAB 59
CLARES FIT SLW OF EXPONENTIALS MIT 7C9G F2 RS P = FRANTIC, LEAST S 324
C 7C9G F2 RS P = CRCC9C, ML-1 FLUID FLOW EXPERIMENT ANALYSIS AG 154
RS P T = FCC4, FLNCAMENTAL MCEC FAST REACTOR X-SEC CALC BNW 11J F4 3-6
2PLUS, NC-SPERICAL OPTICAL MCEC X-SECTIONS AI 7C9F 42 RS P T = 254
2PLUS, NC-SPERICAL OPTICAL MCCXL X-SECTIONS ANL 36CA F6J RSEP T = 254
C5C F2 RS = CANOFF JR, MCCERATOR SPACE CHORD DICT FUNCT 15-1
INELASTIC SCAT X-SECTION CALC MCCERATOR GGA 11L4 F4 RS P = GAKER, 289
RS F = AFXLL, HEAT TRANSFER MLTEN FUEL TUBE BUNDLES LASL 7C9F 2 182
RESTRAINT PIPE STRESS, MAXIML MCCENTM CALC AI 7C9F F+RAP RS T = 4-9
AI 7C9F F+FAP RS LT LIT = MCMUS, X-SECTION LIBRARY UTILITY PRGMRAM 15-9
7C94 F+FAP RS PLT = TYCHE3, MGNE CARLC SLWING-DCWN DENSITY CALC AI 149
LASL 7C9F FLCCG RS P = MCS, MCEX CARLC NEUTRON PENETRATION STUDY 202
F4 RS P T = MCCR-SAFE, 2-C MCEX CARLC CELL CALCULATION AI 36J 3-7
ENTRY AI 7C9F F2 RS = MCTIMER, DCSE RATE CALCULATION SNAP GEOM 142
AFL 66CC F4 RS P = CINDER, M102, POINT DEPLETICN FISSICA PRODCT B 3-13
BAPL 66CC F4 RS P T = FLCT1L, ML-219, PWR FLOW TRANSIENT ANALYSIS R331
BAPL 66CC F4 RS P T = ML266, LINEAR ELASTIC STRUCTURAL DYNAMICS R383
EM BAPL 66CC F4 RS P T = ML457, PIPE, ELASTIC STRESS OF PIPING SYST R329
STEMS BAPL 66CC F4 RS P T = M552, DYNAMIC ANALYSIS LINEAR ELASTIC SY R283
YSIS BAPL 66CC F4 RS P T = ML555,ACT1, LOSS OF-C-CLLENT ACCIDENT ANAL 284
WA BAPL 66CC F4 RS T = ML648, 1-C SLAB TRANSPORT WITH SLOWING DO R342
BAPL 66CC F4 RS P T = ML756, LETC, 1-D SLAB GAMMA-RAY TRANSPRT 343
BAPL 66CC F4 RS P T = ML899, HCH, STEAM TABLES 14-5-2538 PSIA R294
ANL 16L4 F63 RS P T = ML899, HCH, STEAM TABLES 14-5-2538 PSIA R294
-SERO BAPL 66CC F4 RS T = MC67, 2-C DIFFUSION ABSORTION REMVAL X R287
ICN CRNL 7B5C F4 RSEP = MUFFLE, FISSILE NUCLIDE X-SECTION EVALUAT 323
1108 F4 RS P T = GAKIT, 1-C MULIIGP KINETICS WITH TEMP FEEDBACK GGA 37-
1260 F2 RS LT = SIZZLE, 1-C MULTIGRP DIFFUSION DEPLETIN AI 7 58
604 F63 RS LT = SIZZLE, 1-C MULTIGRP DIFFUSION DEPLETIN CDC 1 58
360 F4 RSEP = SIZZLE, 1-C MULTIGRP DIFFUSION DEPLETIN AI 58
RSBPLT= MAC, SHELC DESIGNS MULTIGRP SLAB GEOMETRY BNW 7C9F F2 143
RSBPLT= MAC, SHELC DESIGNS MULTIGRP SLAB GEOMETRY ANL 36J F2 143
16C4 F63 RS PLT = CTF, 1-C MULTIGRP DISCRETE CRDINE CALC UNC 144
1108 F4 RS P T = GAKIN, 1-C MULTIGRP TIME-DEPENDENT DIFFUSION GGA 31C
7C5C F+MAP RS P T = TCSN, 2-C MULTIGRP DISCRETE CRDINE PROGRAM LER 312
1C8 F4 RS F T = FEVER7, 1-C MULTIGRP DIFFUSION AND DEPLETIN GGA 1 318
1108 F4 RS P T = 2CB, 2-C MULTIGRP DIFFUSION AND DEPLETIN BNN 325
16C4 F4 RS F T = 2CB, 2-C MULTIGRP DIFFUSION AND DEPLETIN LAS 325
16C4 F63 RS F T = MC++2, ENDF MULTIGRP X-SECTION CALCULAT ANL 3 355
16F4 F2 RS F = FLOW-MCCCL, MULI-CHAIN 2-D 2-PHASE FLCC AI 7 246
7CSC FLCCG RSEP = 2DXY, 2-C MULI-GP SN APPROXIMATON XY GECM AGC 18
7CSC F+FAP RS PLT = AIM6, 1-C MULI-GP DIFFUSION SLAB CYL SPHERE AI 29
16C4 F63 RS PLT = AIM6, 1-C MULI-GP DIFFUSION SLAB CYL SPHERE CDC 29
LCCC RSEP T = CEE-FAPC-1331, 1-C MULI-GP CCUBE SN APPRX BNW 7C9F F+ 73
FAP RSBPLT= CRAW, 1-D AND 2-C MULI-GP DIFFUSION PROGRAM UK-R 7C9F F+ 1-3
BAPL RS P T = CRAW, 1-D AND 2-C MULI-GP DIFFUSION PROGRAM AEC 36J F+ 1-3
P = ZCT; GRCUC-CCLAPSING CF MULI-GP X-SECTIONS LASL 7C9L FLCCG RS 113
7C9C F+FAP RS LT = LLGER, 1-C MULI-GP DIFFUSION SLAB CYL SPHERE AI 118
7C9C F+FAP RS LT = FAIM, 1-C MULI-GP DIFFUSION SLAB CYL SPHERE AI 120
16C4 F63 RS PLT = FAIM, 1-C MULI-GP DIFFUSION SLAB CYL SPHERE CDC 120
CA POTTENTIAL SHAPE CALCULATION ORNL 7090 F2 RS = LYNNE, WOODS-SAXON 381
7C90 F2 RS = wEC, w-DSN OUTPUT TAPE EDIT REACTION RATES UK-W 133
ECTIVE X-SECTION CALC FROM DSN OUTPUT PW 16C4 LAG1 RS P = DTX, EFF 210
, GROUP CONSTANT CALC FROM TOR OUTPUT DATA LASL 66GC F4 RS P T = GLEN 361
+MAP RS P T = GAROL, RESONANCE OVERLAP AND LATTICE EFFECTS GGA 7044 F2 219
4 RS P T = GAROL, RESONANCE OVERLAP AND LATTICE EFFECTS LER 7094 F2 219
BCW2, DEFLECTION CALCULATION PARALLEL BEAMS ARL 36GC F36 SBP = 365
GARCYCLE, FUEL CYCLE ANALYSIS PARTIAL REFUEL GGA 7444 F4 RS P T = 260
RNL 1644 F63 RS P = RAMES, PARTICLE WAVE FUNCTION RADIAL INTEGRALS G 335
CA IBM 26C F4+AL RS P = PDC5, 2-D FEW-GROUP DIFFUSION AND DEPLETION R336
CA BAPL 66GC F4+ASC RS T = PCC7, 1,2 CR 3-D FEW-GP DIFFUSION DEPLETION R275
LYSIS CRNL 16C4 F63 RS P = PEGGY, ELASTIC SCATTERING PHASE-SHIFT ANA 334
RSBP = LIPREC4, MC NEUTRONS PENETRATION CALCULATION DAC 7490 F2 123
P = MCS, MCNE CARLO NEUTRON PENETRATION STUDY LASL 7090 F2 202
= NRPFCP, FUEL CYCLE COSTS PERFORMANCE DATA KE 7490 F2 RSBP 146
P = WAPUMP, FUEL CYCLE COSTS PERFORMANCE STUDY GGA 7044 F4 RS 224
ETRY BNW 11C8 F4 RS P = PERT4, 2-D PERTURBATION XY RZ RTHEETA GEOM 304
705C F2 RS P = PERT, 1-C PERTURBATION FOR AIM AND FCG CCDES AI 36
1604 F63 RS P = PERT, 1-C PERTURBATION FOR AIM AND FCG CCDES CDC 30
36C F4 RS P = PERT, 1-C PERTURBATION FOR AIM AND FCG CODES BSC 30
16C4 F63 RS P = TCP, 2-C PERTURBATION TOC OR 2DXY FLUX INPUT PW 199
11C8 F4 RS P = PERT4, 2-C PERTURBATION XY RZ RTHEETA GEOMETRY BNW 304
CEAS AI 705C F2 RS = PERT, 1-D PERTURBATION FOR AIM AND FCG CO 30
CEES CDC 16C4 F63 RS P = PERT, 1-D PERTURBATION FOR AIM AND FCG CO 30
CEES BSC 36C F4 RS P = PERT, 1-D PERTURBATION FOR AIM AND FCG CO 30
= PEQGY, ELASTIC SCATTERING PHASE-SHIFT ANALYSIS CRNL 1604 F63 RS P 334
AC RSB = BURP4, GAMMA-RAY PHOTOFRAC TION SCL1D CRYSTAL UM 7090 MC 169
AC RSB = BURP5, GAMMA-RAY PHOTOFRAC TION WELL CRYSTAL UM 7090 M 170
7C90 F4+FAP RS T = 4RESTRAINT PIPE STRESS, MAXIMUM MOMENT CALC AI 109
BAPL 66GC F4 RS P T = MC457, PIPE, ELASTIC SCATTER CF PIPING SYSTEM R329
RSBP = WAPM, LIQUID-FILLED PIPING SYSTEM ANALYSIS KE 360C F4 278
MC457.PIPE, ELASTIC STRESS OF PIPING SYSTEM BAPL 6600 F4 RS P T = R329
CY PW 16C4 F63 RS P T = PIP, CENTRIFUGAL PUMP IMPELLER DESIGN STU 187
11C8 F4 RS P = SAFE-2D, PLANE + AXISYMMETRIC STRESS ANALYSIS GGA 379
16C4 F4 RS P = SAFE-PLAN, PLANE STRESS ANALYSIS, 2-D BODIES GGA 1 252
= BL47, CRAFING TOOL TO PLANE STRUCTURES BAPL 6600 F4 RS P T R373
RS = CRCCK, SPACE POWER PLANT DESIGN OPTIMIZATION AI 7090 F2 112
RS = SC-CCK, SPACE POWER PLANT DESIGN OPTIMIZATION AI 7090 F2 114
C44 F4+FAP RS P T = CPUS, POWER PLANT PERFORMANCE AND PRICE STUDY GGA 7 226
INEAR ALGEBRAIC EQUATION CURVE KAPL 6600 F4 RS P T = SNEC, MCNL R364
S P T = BL47, CRAFTING TOOL IC PLANET STRUCTURES BAPL 6600 F4 R 737
RS T = CHECKER.CRECTOR,MMET.PLCT.PLTFR.SLAPEE NDIF/8 PRCC NCSC 66GC F4 384
RSBP T = CRCSSPLCT, SC40GC PLCTS FROM X-SECTION TAPES GGA 7044 F4 2 7
CS4 F2 RS T = CPS, SC40GC PLCTS FROM SCSRS X-SECTION TAPES WNL 7 239
C4 F4 RS P = CINCE4,MOICLZ, PCINT DEPLETION FISSION PRODUCT BAPL 6600 313
= ELRP1, DETECTOR EFFICIENCY PCINT SOURCE UM 7090 MAD RSB 164
= ELRP3, DETECTOR EFFICIENCY PCINT SOURCE UM 7090 MAD RSB 166
7044 F4 RS T = BLOCT5, PCINT-KINETICS WITH 2-D HEAT TRANSFER GGA 205
LRE FITTING EXPERIMENTAL DATA PCNTS KAPL 6600 F4+FASC RS P T = CURFIT, C R 43
CC, 1-D CONTAINS-DENSITY PCST RUPPLAI GGA 7044 F4 RS T = PRE 228
RS = LYNNE, WOODS-SAXON POTENTIAL SHAPE CALCULATION ORNL 7690 F2 381
T = FLARE, 2-C REACTIVITY AND POWER DISTRIBUTION NED 635 F4 RS P 167
T = FLARE, 3-C REACTIVITY AND POWER DISTRIBUTION CDC 3600 F63 RSBP 167
= GASPS2, 1-C FEW-GP DIFFUSION POWER DISIT SEARCH GGA 7044 F4 RS P T 220
RS P T = GASPS7, 1-C BURNUP POWER DISIT SEARCH GGA 1108 F4 319
SC F2 RS = CRCEC4, SPACE POWER PLANT DESIGN CPTIMIZATION AI 74 1.2
SL F2 RS = SSHCCK, SPACE POWER PLANT DESIGN CPTIMIZATION AI 7L 1.4
GGA 7644 F4+FAP RS P T = CPUS, POWER PLANT PERFORMANCE AND PRICE STUDY 226
S CRNL 16C4 F63 RS P = POWERCO, NUCLEAR STATION ELECTRICITY C0ST 34
SN DCUBLE SN APFCX SLAB GEOM PPCC 7050 F2 RS P = MIST, 1-D FEW-GP 59
S P SA APPROXIMATION CYLINDER PPCC 7444 F4 RS P T = TCPIC, 1-C FEW-G 146
P SN APPROXIMATION CYLINDER
E-W MULTI-LEVEL CCNVOLUTION
AL SAFETY ANALYSIS PROGRAM
F-CYCLIC ANNEAL ANALYSIS
-CEPHOOD HEAT CONDUCTION
PTURE GGA 7C44 F4 RS T = F2 RSBP= T= FHT, BLOWCWN
T= PRECCN, HTGR CONTAINMENT
S, TRANSIENT THERMODYNAMICS OF
S, POWER PLANT PERFORMANCE ANC
ENCFE THERMAL SCATTERING DATA
CT..CMAT..PLICF, SLAVE 2 ENCF/B
GGA 118C F4 RS P = RCAC, HTGR FISSION
FREVAP6, HTGR METALLIC FISSION
MUL2, PCs DELETION FISSION
RS P = ISCSARCH, I5STOPE
7CS4 F4 RS P = GAMMA-P,
RS P = ISCSARCH, I5STOPE
ALC GGA 118C F4 RS P =
RY KE 7CS4 F2 RSBP T =
63 RS P T = EXPN, ANALYSIS OF
CS BAPL 66CC F4 RS P =
-CP DIFFUSION SLAB CYL SPHERE
MULTI-CP SN APPROXIMATION RZ
CR FLCC2 INSTRUCTION SET
PUMF IMPELLER DESIGN STUDY
R FUNCTION EVALUATION
CF SN CCNTS FCR CNS TCC
R WEIGHT OPTIZATION STUDY
NEUTRONICS-HYDRODYNAMICS SPH
-CP FLUX SYNTHESIS PROGRAM CYL.
ICN LIBRARY TAPE PREPARATION
CN LIBRARY TAPE PREPARATION
NEUTRON, GAMMA-RAY HISTORIES
ATCN TCC CR 2CXY FLUX INPUT
-SECTION CALC FROM CNS OUTPUT
-CP SN APPROX SLAB CYL SPHERE
UC F4 RS P T = FLCT1,M=219,
CNS AI 7C50 F+FAP RS LT =
AAL 36CC F63 RS P T =
CNS ANL 36CC F63 RSBP =
CNS WANL 7C54 F4 RSBP =
CALC 36CC F63 RSBP =
CALC WVL 7C54 F4 RSBP =
ELL AAL 36CC F36 RSBP T =
STUDY GGA 7C44 F4 RS P =
RAMES, PARTICLE WAVE FUNCTION
RS = SCARF2, SCATTER FROM
RADIATOR FINS SNAP GECM AI 7C9F F2
W, NELTRCN-INCLOCED GAMMA-RAY
PPCC 7040 F+MAP RS P T = RSAC, RADILOGICAL SAFETY ANALYSIS PROGRM
w 118C F5 RSBP= I5SCGN, RACIOUCILIGE GENERATION ANC DECAY BN
GRALS CRNL 16C4 F63 RS P =
S APC 635 F4 RS P =
R MIX KAPL 66CC F4 RS P =
C NEC 2C00 F2 RS P =
-CPN CUTPUT TAPE EDIT REACTION RATES
SS LASL 7C54 FAP RS P T =
SS LASL 7C30 F4 RS T =
RY AI 7C50 F2 RS P =
LC KAPL 66CC F4 RS P =
= WEC, w-CPN CUTPUT TAPE EDIT REACTION RATES
UK-W 7C90 F2 RS 133
RF1, DETECTOR EFFICIENCY PCINT SCLRC F UM 7C9G MAC RSB = BU 164
RF2, DETECTOR EFFICIENCY DSK SCLRC F UM 7L9G MAD RSB = B 165
RF3, DETECTOR EFFICIENCY PCINT SCLRC F UM 7C9U MAD RSB = BU 166
PH, ANALYSIS CF PULSED NEUTRON SCLRC F DATA NED 635 F4 RS P T= EX 258
RS = CANCERF JK, MCCERATCR SPACE CHORD DIST FLACT AEG 709U F2 156
I 7C9G F2 RS = CROCK, SPACE POWER PLANT DESIGN OPTIMIZATION A 112
I 7C9G F2 RS = SHOCK, SPACE POWER PLANT DESIGN OPTIMIZATION A 114
7C90 F2 RS = AIREK3, SPACE-DEPENDENT KINETICS W/FEEDBACK A 121
P 360 F4 RS P = AIREK3, SPACE-DEPENDENT KINETICS W/FEEDBACK AE 121
I 7L94 F4 MAP RS P T= AIROS, SPACE-DEPENDENT INVERSE KINETICS CALC 168
WANL 7G54 F4 RS = RICOZ, SPACE-DEPENDENT INVERSE KINETICS CALC 168
ANL 36CC F62 RS = RICOZ, SPACE-DEPENDENT KINETICS KEX CPTIONS 265
ANL 7L94 F4 RS = RICOZ, SPACE-DEPENDENT KINETICS KEX CPTICS 265
11C8 F4 RS = GAPCTKIN, SPACE-DEPENDENT REACTOR KINETICS GGA 317
F4 RS P T= WIGL2, 1-C 2-GP SPACE-TIME DIFFUSION 3-GEOM BAPL 66J 153
F4 RS P T= WIGL2, 2-C 2-GP SPACE-TIME DIFFUSION FEEDBACK BAPL 66J 153
S NEC 2C00 F2 RS = SPARTA, SPATIALLY-AVERAGED DOPPLER EFFECT 178
D 2C00 F2 RS P = SPARTA, SPATIALLY-AVERAGED DOPPLER EFFECTS NE 178
AI 7C94 F4 RS P T= TSN, SPATIALLY-DEPENDENT REACTOR KINETICS 309
AI 36C F4 RS P T= TSN, SPATIALLY-DEPENDENT REACTOR KINETICS 309
LTI-CP X-SECTIONS FAST THERMAL SPECTRA GGA 11C8 F4 RS PLT= GGC4, MU 298
11C8 F4 RS P T= THERMCSCANL), THERMAL SPECTRUM X-SECTION CALC ANL 36CC F63 184
CCESSING CF ANALYZER GAMMA-RAY SPECTRA GGA 11U8 F4 RS P T= TCAG, PR 333
RS PLT= GAM1, FAST NEUTRON SPECTRUM X-SECTION CALC GGA 7090 F2 33
RS LT= GAM1, FAST NEUTRON SPECTRUM X-SECTION CALC CCC 1664 F63 33
LT= TEMPEST2, THERMAL NEUTRON SPECTRUM X-SECTIONS AI 7U9C F+FAP RS P 50
LT= TEMPEST2, THERMAL NEUTRON SPECTRUM X-SECTIONS CDC 16C4 F63 RS P 50
LT= TEMPEST2, THERMAL NEUTRON SPECTRUM X-SECTIONS BHSC 36C F4 RS P 50
RS PLT= FCRM, FAST NEUTRON SPECTRUM X-SECTION CALC AI 7090 F2 51
RS PLT= FCRM, FAST NEUTRON SPECTRUM X-SECTION CALC CCC 16C4 F63 51
LT= CUICKIE, INFINITE MEDIUM SPECTRA GGA 11C8 F4 RS PLT= GGC4, MU 298
RS F T= THERMCSCANL), THERMAL SPECTRUM X-SECTION CALC ANL 36CC F63 184
CS4 F+FAP RS P T= EPITHERMOS, SPECTRUM AND X-SECTION CALCULATION GEV 7 201
C F2 RSBPLT= AGN-GAM, FAST SPECTRUM MULTIGP CONSTANT CALC AGC 709 242
AF 7C94 F4 MAP RS PLT= LASER, SPECTRUM CALC WITH BURNUP CYL LATTICE W 249
UPPPEE NEUTRONICS-HYDRODYNAMICS SPH PW 1664 F63 RS P = AX-TNT, CO 191
-C AGE-CIffUSION SLAB CYLINDER SPHERE AEB 36C F4 RS PLT= FIERES, 9
-FEW-GP CIffUSION SLAB CYLINDER SPHERE AIE 709C F2 RS P T= FGC, 1-D 28
-FEW-GP CIFFUSION SLAB CYLINDER SPHERE CDC 16C4 F63 RS P T= FGC, 1-D 28
- C MULTI-GP CIFFUSION SLAB CYL SPHERE AIE 709U F+FAP RS PLT= AIM1, 9
- C MULTI-GP CIFFUSION SLAB CYL SPHERE CDC 1604 F63 RS PLT= AIM1, 9
CE2, GAMMA-RAY ATTENUATION CYL SPHERE GECM AI 709C F2 RS P = GRA 46
CE2, GAMMA-RAY ATTENUATION CYL SPHERE GECM CDC 1604 F63 RS P = GRA 46
UPPLEE NEUTRONICS-HYDRODYNAMICS SPHERE CCC 360G F63 RS BP = AX1, CO 112
-C MULTI-GP CIFFUSION SLAB CYL SPHERE AI 709C F+FAP RS LT= ULCER, 1 118
-C MULTI-GP CIFFUSION SLAB CYL SPHERE AI 709C F+FAP RS LT= FAIM1, 1 120
-C MULTI-GP CIFFUSION SLAB CYL SPHERE CCC 16C4 F63 RS PLT= FAIM1, 120
-C MULTI-GP CIFFUSION SLAB CYL SPHERE BHSC 36C F4 RS P T= FAIM1, 120
UPPLEE NEUTRONICS-HYDRODYNAMICS SPHERE LRL 709C F2 RS P = CCNCE, CO 129
-C MULTI-GP SA APPROX SLAB CYL SPHERE UK-M 7C90 F2 RS P T= W-CSE, 1 132
-C MULTI-GP SA APPROX CYL SPHERE PW 1664 LAQG RS P = MGDA, 1 211
-C MULTI-GP CIFFUSION SLAB CYL SPHERE BNW 11C7 F4 RS P T= HFA, 1 241
-C MULTI-GP CIFFUSION SLAB CYL SPHERE ANL 360U F63 RS PLT= MACH1, 262
-GCA 11C8 F4 RS = GASAI, STABILITY ANALYSIS REACTOR KINETICS ECNS 296
63 RS P = PCNERCC, NLCLEAR STATION ELECTRICITY CCSZ CRNL 16C4 F 346
A 11C8 F4 RS P = PSEUDO, STATISTICAL RESONANCE PARAMETER CALC GG 292
I 7C90 F2 RS = FLUCUE, STEADY-STATE TEMPERATURE VICID FRACTION A 48
RS P T= HEATING2, TRANSIENT STEADY-STATE HEAT TRANSFER AI 7C94 F2 198
FLU, HEAT TRANSFER MOLTEN FUEL TUBE BUNDLES LASL 7094 F2 RS P = AX 182
THROW, HEAT TRANSFER SOLID FUEL TUBE BUNDLES LASL 7094 F2 RS P = AX 183
IS KAPL 66CC F4 RS P = CALC GGA 7C4F F+FAP RSB T = TUBE, U-TUBE HEAT EXCHANGER STRESS ANALYSIS R378
IS COL 36C F4 RS P = TUBE, U-TUBE HEAT EXCHANGER STRESS ANALYSIS R378
BACK BAPL 66CC F4 RS P T = THWIGL, 2-D 2-GP SPACE-TIME DIFFUSION FEED R338
GECM LASL 11CE F4 RS P T = TRETTRAN, 2-D MULTI-GP TRANSPORT CODE X-Y 358
GECM ANL 36C F4 RS P T = TRETTRAN, 2-D MULTI-GP TRANSPORT CODE X-Y 358
CALC AI 7C4 F+FAP RS PLT = TYCHE3, MCNE CARL SCWOING-DOWN DENSITY 149
CLIC BLCWCDW ANALYSIS LOFT UGR 36U F4 RS P = WATER-HAMMER, LI 278
CCOMPLETER-PRODUCED WIRING LISTS UHTREX LASL 7UC F2 RS P = WIREX, 315
QUARES ANALYSIS RESONANCE DATA UILL 36F F4 RS P = CCDILI, LEAST S 347
KAPL 66CC F4 RS P = TUBE, U-TUBE HEAT EXCHANGER STRESS ANALYSIS R378
C MULTI-GP DIFFUSION PROGRAM UK-R 7C90 F+FAP RS PLT = CRAM, 1-D AND 2-D MULTI-GP 103
-GP SN APPROX SLAB CYL SPHERE UK-W 7C9L F2 RS P T = W-DSN, 1-D MULTI-GP 132
T TAF IRE EDIT REACTION RATES UK-W 7C90 F2 RS PS = WED, W-DSN CUTUP 133
HERE AI 7C9C F+FAP RS LT = UCER, 1-C MULTI-GP DIFFUSION SLAB CYL SPHERE R341
EFFICIENCY POINT SOURCE UM 7090 MAD RSB = BURP1, DETECTOR 164
EFFICIENCY DISK SOURCE UM 7090 MAD RSB = BURP2, DETECTOR 165
EFFICIENCY POINT SOURCE UM 7090 MAD RSB = BURP3, DETECTOR 166
PHOTOCHEMICAL SLICID CRYSTAL UMCC 36U F4 RS P T = BURP4, GAMMA-RAY 169
PHOTOCHEMICAL WELL CRYSTAL UMCC 36U F4 RS P T = BURP5, GAMMA-RAY 170
CEP 2-C MULTI-GP DIFFUSION UMCC 36U F4 RS P T = VARI-GUIR, TIME-DEP 212
CUP DISCRETE CIRCUIT CALC UNC 16C4 F63 RS PLT = DTF, 1-D MULTI-GP 144
TUF-GP DISCRETE CIRCUIT CODE UNCC 36L F4 RS P T = VARI-GUIR, TIME-DEP 212
APE GGA 7C44 F+FAP RSB = UNPACK, RETRIEVAL FROM SCISRS X-SECTION T 2C6
GGA 7C44 F+FAP RSB = UNRESOLVED REGION RESONANCE INTEGRAL CALC 42
EAPL 66C0 F4 RS P = GANDY, UNRESOLVED RESONANCE X-SECTION G 341
BAPL 66C0 F4 RS P T = PUNI, UNRESOLVED RESONANCE INTEGRALS X-SECS R359
7054 F4 RS P = AVERAGE, UNRESOLVED REGION AVERAGE X-SEC CALC BNL 376
66CC F4 RS P = AVERAGE, UNRESOLVED REGION AVERAGE X-SEC CALC BNL 376
705C F2 RS P = AIMFIRE, URANIUM FUEL CYCLE COST ANALYSIS AI 55
LI = MCMUS, X-SECTION LIBRARY UTILITY PROGRAM AI 7C94 F+FAP RS 159
T = XLIBIT, X-SECTION LIBRARY UTILITY ROLTIME ANL 36C0 F63 RSBP 181
CLIF, FCRM CR RIPPED LIBRARY UTILITY ROLTIME AI 36C0 F4 RSB LT = 271
N WAIN 66CC F4 RS P T = VARI-GUIR, TIME-DEP 2-D MULTI-GP DIFFUSION R222
L WAIN 7C94 F4 RS P T = VARI-GUIR, TIME-DEP 2-D MULTI-GP DIFFUSION 222
RTH WAIN 7C94 F4 RS P T = VARI-GUIR3, 2-D MULTI-GP DIFFUSION 2-D 264
= GEM, EIGENVALUE PROBLEM FOR VIBRATING SYSTEMS KAPL 66CC F4 RS P T R344
A 11C8 F4 RS P T = MUSCAT, VIEW FACTOR SHIELDING CODE CAVITY GECM GG 259
107F4 RS P = SAFE-CREEP, VISCOELASTIC ANALYSIS CONCRETE GGA 1 360
UGUE, STEADY-STATE TEMPERATURE VCID FRACTION AI 7C9C F2 RS = 48
FCW4 RS P = AVCIC, ANNULAR VCID X-SECTION CALCULATION GEC 625 276
Y GGA 7C44 F4 RS P = WAPPUM, FUEL CYCLE CCSTS PERFORMANCE STUD 224
CEP 2-C MULTI-GP DIFFUSION WAIN 66CC F4 RS P T = VARI-GUIR, TIME-DEP 2-D MULTI-GP 222
PENDENT INVERSE KINETICS CALC WAIN 7C94 F4 RS P T = R1U2, SPACE-IND 168
FROM B-W RESONANCE PARAMETERS WAIN 7C94 F2 RS P = EXT, X-SECTIONS 238
S FROM SCISRS X-SECTION TAPES WAIN 7C94 F2 RS P T = CPS, SC42C PLOT 239
PENDENT KINETICS KEX OPTIONS WAIN 7C94 F2 RS P T = R1U1, SPACE-IND 255
MULTI-GP DIFFUSION XY RZ RTH WAIN 7C94 F4 RS P T = VARI-GUIR3, 2-D 264
CALC WITH BURNUP CYL LATTICE WAPD 7C94 F+MAP RS PLT = LASER, SPECTRUM 249
FCW4 RS P = WAPPUM, FUEL CYCLE CCSTS PERFORMANCE STUD 224
2ECGF BAPL 66C0 F4 RS P = WATER-HAMMER, LIQUID BLCWCDW ANALYSIS LO 278
F63 RS P = RAMES, PARTICLE WATER, STEAM TABLES 14,5-14,50PSIA 32-47 R267
MICS APCC 7C94 F2 RS P T = WEAK EXPLOSION, COUPLED NELTRCN-HYDRCYNA 145
S UK-W 7C94 F2 RS P T = WEC, W-DSN OUTPUT TAPE EDIT REACTION RATE 133
3 RS = WCEXPRT, REACTOR WEIGHT OPTIMIZATION STUDY PH 16G4 F69 190
S4 F2 RS P = SHC5E, SHIELD weight OPTIMIZATION CODE CEC AI 17 197
BURP5, GAMMA-RAY P-CUTOFFRACTION WELL CRYSTAL UM 7U90 MAD RSB = 170
RS = WAM, LIQUID-FILLED PIPING SYSTEM ANALYSIS 278
NTS AEB 36C F4 RS P = WHIRLAWAY, 3-D 2-GROUP DIFFUSION XYZ GEOM 32
S KE 36C F4 RSBP = WHIRLAWAY, 3-D 2-GROUP DIFFUSION XYZ GEOM 32
ETRY CRNL 7C94 F2 RS P T = WHIRLAWAY, 3-D 2-GROUP DIFFUSION XYZ GEOM 32
ETRY CCC 16C4 F63 RS P T = WHIRLAWAY, 3-D 2-GROUP DIFFUSION XYZ GEOM 32
RESCLVEC RESONANCE EPITHERMAL X-SECTIONS NED 635 F4 RS P T = REAX, 257
RS = HEXSCAT, ELASTIC SCAT X-SECTIONS HEX LATTICE GGA 1108 F4 291
F4 RS PLT = GGC4, MLLTI-GP X-SECTIONS FAST THERMAL SPECTRA GGA 1108 298
A, 2-C MULTI-GP TRANSPORT CODE X-Y GEOM LASL 11C8 F4 RS P T = TWCTRA 358
A, 2-C MULTI-GP TRANSPORT CODE X-Y GEOM ANL 36C F4 RS P T = TWCTRA 358
ANL 36CC F63 RSBP T = XLIBIT, X-SECTION LIBRARY UTILITY ROUTINE 181
2-C MULTI-GP SN APPROXIMATION XY GECM AGC 7090 FLCCD RSBP = 2CXY, 18
AMBELE4, 2-C MLLTI-GP DIFFUSION XY RZ GEOMETRY GGA 7C44 F+MAP RSBP T = G 222
AMBELE5, 2-C MLLTI-GP DIFFUSION XY RZ GEOMETRY GGA 1108 F+BAL RS P T = G 222
-CUIR3, 2-C MLLTI-GP DIFFUSION XY RZ RTH WNL 7094 F4 RS P T = VARI 264
P = PERT4, 2-C PERTURBATION XY RZ RTHETA GEOMETRY BNW 11C8 F4 RS 304
IRLAWAY, 3-D 2-GROUP DIFFUSION XYZ GEOMETRY ORNL 7090 F2 RS P T = WH 32
IRLAWAY, 3-D 2-GROUP DIFFUSION XYZ GEOMETRY CDC 16C4 F6 RS P T = WH 32
3 RS P = ISOTOPES, MAXIMUM YIELD FROM REACTION OR DECAY ORNL 16C4 F6 179
CNS LASL 7C5G FLCCC RS P = ZCT, GROUP-COLLAPSING OF MULTI-GP X-SECT 113
PES GGA 11C8 BIA R LT = ZPR-III ASSEMBLY 48 GAFGAR ENDF/B DATA TA 356
ALC GGA 7C5G F+FAP RSB T = ZLT, RESOLVED REGION RESONANCE INTEGRAL C 41
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