COMPUTER CODE ANISN MULTIPLYING MEDIA AND SHIELDING CALCULATION
III. SAMPLE PROBLEMS

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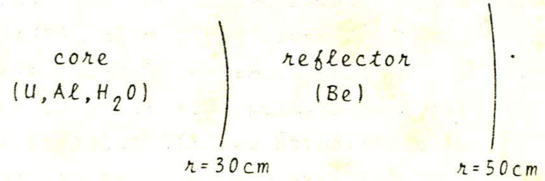
SAMPLE PROBLEM 1

TITLE : 4 GROUP P-1 REFLECTED CYLINDER EIGENVALUE TEST PROBLEM.

DESCRIPTION : The goal of this sample problem is an eigenvalue calculation (K_{eff}) for a two region cylindrical reactor. The inner region is the core, loaded with a mixture of uranium, aluminium, and water with a radius of 30cm (typical research reactor core), and the outer region with 20cm thickness represents a beryllium reflector. The calculation is to be done in 4 groups, P-1 scattering order, S-4 angular quadrature in cylindrical geometry.



INPUT : Two types of input data are available for this sample problem. One, in which the cross sections are entered through the input stream in the free format 14** array, and the second using the cross sections from the interface file ISOTXS.SP2. The difference between two types of input are in the variables MCR and MTP (array 15$), being MCR= 4 and MTP=0 in the first case, and MCR=0; MTP=4 in the second case. Also, in the second case instead of array 14, it is used array 13, and since in this case the input file is ANISNC4.LIB, one has to copy ISOTXS.SP2, before executing ANISN, i.e,

```
COPY ISOTXS.LIB    ANISNC4.LIB
```

Enclosed, the input files for these two cases are presented (ANISN.SP1 and ANISN.SP2). From these files, it is noted that the mixing tables (10$; 11$; 12*) are different, indicating that in case of cross section input file by ANISNC4.LIB the P_N components do not have to be multiplied by $2\ell+1$.

RESULTS: The output file for this sample problem was illustrated in part II (code description). The multiplication factor calculated was:

$$K_{eff}=1.16126,$$

after 9 outer iteration. The flux distribution for this sample problem is illustrated in Figure 1. The plot shows in figure 1 was obtained by the 1-D semi-log plot routine, ANIPLT /1/ which is distributed by RSIC together with ANISN-ORNL (CCC-254) and was designed to provide distribution plot of data from ANISN or DOT codes (This routine is not available in ANISN-PC).

EXERCICES : Using the interactive Executable File APE 32 it is suggested modifications in the original input file for sample problem 1 in order to verify the sensitivity of the results for the proposed changes, and also as training in the use of APE 32. The following exercices are suggested:

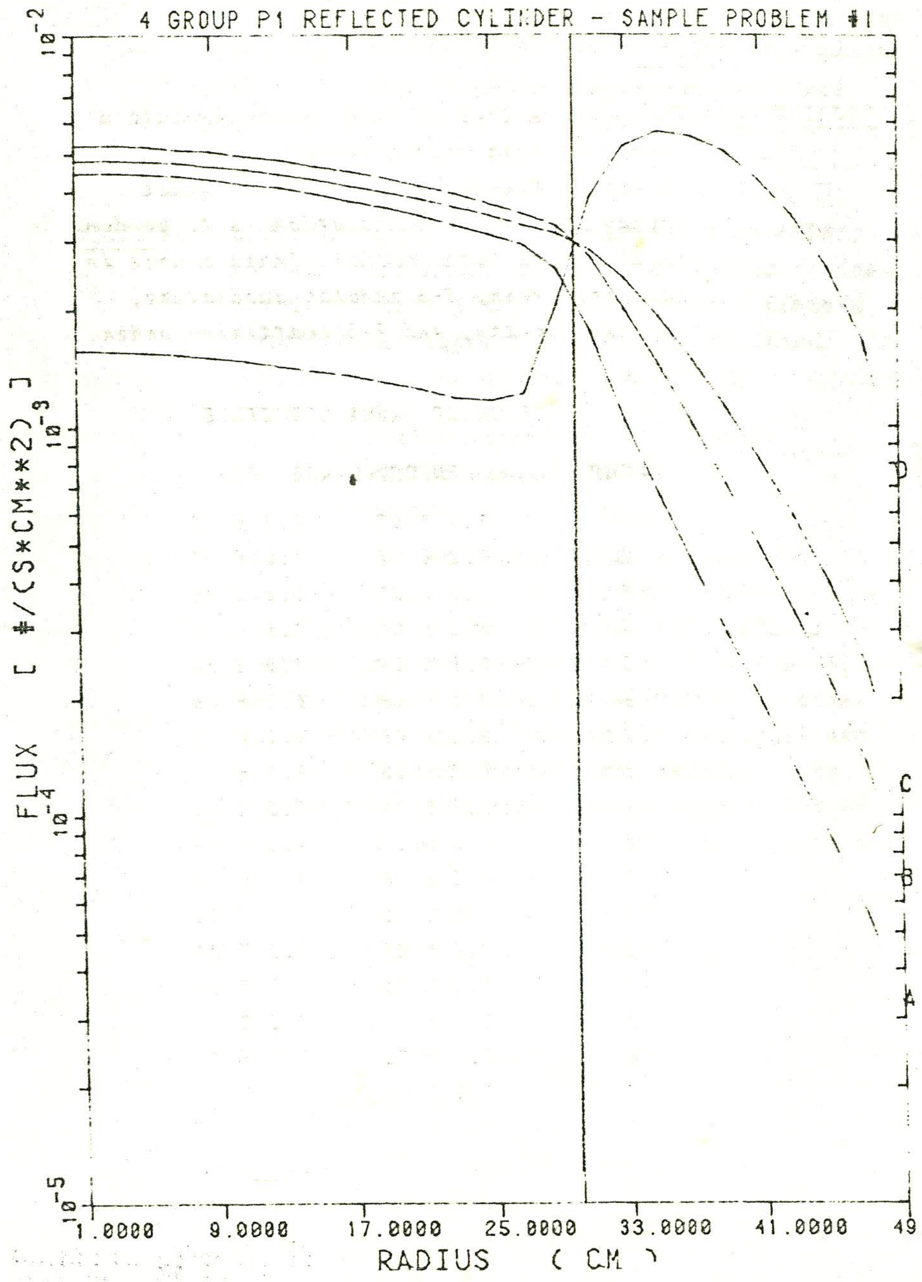
i) Sensitivity analysis of the eigenvalue (K_{eff}) with the variation in the quadrature order (S_2; S_4; S_6; S_8; S_{12}). The results that will be obtained by performing this exercice is shown below:

S-N	K_{eff}
S-2	1.16260
S-4	1.16128
S-6	1.16092
S-8	1.16082
S-12	1.16075

ii) Sensitivity analysis of the eigenvalue (K_{eff}) when axial leakages are considered through a buckling correction (DB^2). To perform this exercice consider the cylinder height as 60cm and BF=1.420892. (16** array needs to be modified). With this correction, K_{eff} will be reduced to the value 1.07216 (S-4).

```
ANISN 4 GROUP REFLECTED CYLINDER EIGENVALUE TEST PROBLEM 2 (ISOTXS)
15$$                1                0                1                4
                    2                1                0                2
                   25                1                4                3
                    5                8                4                0
                    4                8                0                0
                    0                0                0               20
                    0                1                0                0
                  100                0                0                0
                   -1                1                0                0
16**     +1.00000E+00     +0.00000E-01     +1.00000E-04  4R+0.00000E-01
         +1.00000E+00     +0.00000E-01     +5.00000E-01    +2.00000E-04
       3R+0.00000E-01
   T
13$$                1                2                3                4
   T
3**      F+1.00000E+00
   T
4**      +0.00000E-01     +2.00000E+00     +4.00000E+00    +6.00000E+00
         +8.00000E+00     +1.00000E+01     +1.20000E+01    +1.40000E+01
         +1.60000E+01     +1.80000E+01     +2.00000E+01    +2.20000E+01
         +2.40000E+01     +2.60000E+01     +2.80000E+01    +3.00000E+01
         +3.20000E+01     +3.40000E+01     +3.60000E+01    +3.80000E+01
         +4.00000E+01     +4.20000E+01     +4.40000E+01    +4.60000E+01
         +4.80000E+01     +5.00000E+01
6**      +0.00000E-01   2R+1.66667E-01     +0.00000E-01  4R+1.66667E-01
7**      -4.95005E-01     -3.50021E-01     +3.50021E-01    -9.36742E-01
         -8.68890E-01     -3.50021E-01     +3.50021E-01    +8.68890E-01
8$$                 1                1                1                1
                    1                1                1                1
                    1                1                1                1
                    1                1                1                2
                    2                2                2                2
                    2                2                2                2
                    2
9$$                 5                7
10$$                5                6                7                8
11$$                1                2                3                4
12**     F+1.00000E+00
19$$                1                1
   T    T
```

FIGURA 1



SAMPLE PROBLEM 2

TITLE : ANISN/PC LEAD SPHERE TEST PROBLEM

DESCRIPTION : The goal of this problem is to simulate a sphere of lead with a photon source in the center emitting particles in the first energy group. The calculation is to be done in 21 groups (see below), fixed source in the first mesh, S-6 angular quadrature, sphere geometry, and P-3 scattering order.

21 GROUP GAMMA STRUCTURE

GROUP	ENERGY RANGE (eV)	
1	1.4 E 07	1.2 E 07
2	1.2 E 07	1.0 E 07
3	1.0 E 07	8.0 E 06
4	8.0 E 06	7.5 E 06
5	7.5 E 06	7.0 E 06
6	7.0 E 06	6.5 E 06
7	6.5 E 06	6.0 E 06
8	6.0 E 06	5.5 E 06
9	5.5 E 06	5.0 E 06
10	5.0 E 06	4.5 E 06
11	4.5 E 06	4.0 E 06
12	4.0 E 06	3.5 E 06
13	3.5 E 06	3.0 E 06
14	3.0 E 06	2.5 E 06
15	2.5 E 06	2.0 E 06
16	2.0 E 06	1.5 E 06
17	1.5 E 06	1.0 E 06
18	1.0 E 06	4.0 E 05
19	4.0 E 05	2.0 E 05
20	2.0 E 05	1.0 E 05
21	1.0 E 05	1.0 E 04

INPUT : The input file for this sample problem is enclosed with this note (ANISN.SP3). The microscopic cross sections for lead were taken from FLUNGP.LIB master library using the utility program LMOD to create a ISOTX.SP3 library file, and so before run this sample problem one has to copy ISOTX.SP3 into ANISNC4.LIB.

EXERCICES : Using the interactive executable file APE32, and the executable file LMOD to generate cross sections input files from the master library FLUNGP it is suggested to change the lead shielding by the following materials:

- H_2O (ρ = 1g/cm^3)
- Depleted Uranium (ρ=19g /cm^3)
- Concrete (ρ= 2.3g/cm^3)

To use these materials it is necessary to compute the atomic density (atoms/barn-cm) of each element in the mixture. For depleted uranium it is assumed that it is pure u-238, and the atomic density is 4.8079x10^{-2} atoms/barn-cm. For water, the atomic densities are 6.686x10^{-2} atoms/barn-cm for hydrogen, and 3.3343x10^{-2} atoms/barn-cm for oxygen. For concrete it was used the composition given by:

Concrete Portland (From H.E. Hungerford, Reactor Materials, Vol. I, Materials, C.R. Tripton Jr (Ed.), Interscience Publishers, New York, 1960. (ρ= 2.3g/cm^3)

Element	Composition Wt%	Atomic Density (Atom/bar-cm)
Iron	1.4	3.4724 E-4
Hydrogen	1.0	1.3852 E-2
Oxigenio	52.9	4.5799 E-2
Magnesium	0.2	1.1395 E-3
Calcium	4.4	1.5207 E-3
Silicon	33.7	1.6621 E-2
Sodium	1.6	9.6403 E-4
Potassium	1.3	4.6052 E-4
Aluminium	3.4	1.7455 E-3
Carbon	0.1	1.1532 E-4

For each material a library of microscopic cross section was created and entered through array 13*. The macroscopic cross sections were calculated by the input given by the mixing table (array 10$; 11$ and 12$ of each case. Enclosed a comparison among the mixing table are illustrated.
Finally a comparison among of the results that will be obtained for the transmission (Right leakage, which is defined as the J$^+$ current multiplyied by the surface area) is shown for the different materials keeping constant the thickness of the shielding.

SHIELD	TRANSMISSION (RT LEAKAGE)
WATER	8.70344 E-1
CONCRETE	1.40834 E-1
LEAD	3.92294 E-9
DEPLETED URANIUM	6.45631 E-16

Note: RT LEAKAGE= (RT BDY J$^+$) x AREA

LEAD

```
*********************************************************************
             ANISN MIX TABLE (10$,11$,12*) MODULE
*********************************************************************

POSITION       MIXTURE(10$)       COMPONENT(11$)        DENSITY(12*)
    1               5                   1               3.35000E-02
    2               6                   2               3.35000E-02
    3               7                   3               3.35000E-02
    4               8                   4               3.35000E-02

Do you want to make changes? (Y/N/F/R; Default=No)
```

URANIUM

```
*********************************************************************
             ANISN MIX TABLE (10$,11$,12*) MODULE
*********************************************************************

POSITION       MIXTURE(10$)       COMPONENT(11$)        DENSITY(12*)
    1               5                   1               4.80790E-02
    2               6                   2               4.80790E-02
    3               7                   3               4.80790E-02
    4               8                   4               4.80790E-02

Do you want to make changes? (Y/N/F/R; Default=No)
```

WATER

```
*********************************************************************
             ANISN MIX TABLE (10$,11$,12*) MODULE
*********************************************************************

POSITION       MIXTURE(10$)       COMPONENT(11$)        DENSITY(12*)
    1               9                   1               6.68600E-02
    2              10                   2               6.68600E-02
    3              11                   3               6.68600E-02
    4              12                   4               6.68600E-02
    5               9                   5               3.34300E-02
    6              10                   6               3.34300E-02
    7              11                   7               3.34300E-02
    8              12                   8               3.34300E-02

Do you want to make changes? (Y/N/F/R; Default=No)
```

CONCRETE

```
*********************************************************************
             ANISN MIX TABLE (10$,11$,12*) MODULE
*********************************************************************

POSITION       MIXTURE(10$)       COMPONENT(11$)        DENSITY(12*)
    1              41                   1               3.47240E-04
    2              42                   2               3.47240E-04
    3              43                   3               3.47200E-04
    4              44                   4               3.47200E-04
    5              41                   5               1.38520E-02
    6              42                   6               1.38520E-02
    7              43                   7               1.38520E-02
    8              44                   8               1.38520E-02
    9              41                   9               4.57990E-02
   10              42                  10               4.57990E-02
   11              43                  11               4.57990E-02
   12              44                  12               4.57990E-02
   13              41                  13               1.13950E-03
   14              42                  14               1.13950E-03
   15              43                  15               1.13950E-03

Press 'ENTER' to continue .....
```

```
                                CONCRETE
*****************************************************************************
              ANISN MIX TABLE (10$,11$,12*) MODULE
*****************************************************************************

POSITION        MIXTURE(10$)          COMPONENT(11$)           DENSITY(12*)
   16                44                     16                 1.13950E-03
   17                41                     17                 1.52070E-02
   18                42                     18                 1.52070E-02
   19                43                     19                 1.52070E-02
   20                44                     20                 1.52070E-02
   21                41                     21                 1.66210E-02
   22                42                     22                 1.66210E-02
   23                43                     23                 1.66210E-02
   24                44                     24                 1.66210E-02
   25                41                     25                 9.64030E-04
   26                42                     26                 9.64030E-04
   27                43                     27                 9.64030E-04
   28                44                     28                 9.64030E-04
   29                41                     29                 4.60520E-04
   30                42                     30                 4.60520E-04

Press 'ENTER' to continue .....


                                CONCRETE
*****************************************************************************
              ANISN MIX TABLE (10$,11$,12*) MODULE
*****************************************************************************

POSITION        MIXTURE(10$)          COMPONENT(11$)           DENSITY(12*)
   31                43                     31                 4.60520E-04
   32                44                     31                 4.60520E-04
   33                41                     33                 1.74550E-03
   34                42                     34                 1.74550E-03
   35                43                     35                 1.74550E-03
   36                44                     36                 1.74550E-03
   37                41                     37                 1.15320E-04
   38                42                     38                 1.15320E-04
   39                43                     39                 1.15320E-04
   40                44                     40                 1.15320E-04

Do you want to make changes? (Y/N/F/R; Default=No)



*****************************************************************************
              ANISN MIX TABLE (10$,11$,12*) MODULE
```

SAMPLE PROBLEM 3

TITLE : RESEARCH REACTOR SHIELDING PROBLEM- P3 COUPLED NEUTRON-GAMMA TEST PROBLEM. (ANISN1. INP) /2/

DESCRIPTION : This problem is a Realistic Shielding calculation for a Research Reactor. The goal of the test problem is to calculate the gamma and neutron radial fluxes distribution in a typical research reactor shielding (see figure 2), as well as the Dose Rate (rem/hr) in the surface of the shielding. The primary source of gamma rays and neutrons in the core were calculated by:

i) Fission Neutrons

$$S^g_N = 3.1 \times 10^{10} \nu \chi_g \bar{p} \text{ neutrons/cm}^3\text{-s}$$

with ν the number of neutron emitted by thermal fission (ν=2.46), χ_g the fraction of neutrons emitted by fission, and calculated from the watt spectrum ($\chi_g = \int_g \chi(E)\,dE$)

$$\chi(E) = 0.484 \sin\sqrt{2E} \exp[-E] \;;$$

and the power density distribution ($\bar{p}$) taken from a typical 2MW research reactor /2/.

ii) Prompt Fission Gamma Rays.

$$S^g_{\gamma F} = 3.1\times10^{10} \, \Gamma^g_F \bar{p} \quad \text{gammas/cm}^3\text{-sec;}$$

with Γ^g_F the number of gammas emitted in the energy group, per fission and calculated using the spectrum /3',

$$\Gamma(E) = \begin{cases} 6.6; & 0.1<E<0.6 \text{ MeV} \\ 20.2 \exp[-1.78E]; & 0.6<E<1.5 \text{ MeV} \\ 7.2 \exp[-1.09E]; & 1.5<E<10{,}5 \text{ MeV} \end{cases}$$

iii) Fission Product Gamma Rays.

$$S^{g}_{\gamma FP} = 3.1 \times 10^{10}\ \Gamma^{g}_{FP}\ \overline{p} \quad \text{gammas/cm}^3\text{-sec},$$

with Γ^{g}_{FP} the number of Fission product gamma rays emitted in the group per fission, and calculated using the spectrum /3/.

$$\Gamma^{g}_{FP}\ (E) = 7.4 \exp[-1.1E] \text{ gamma/fission.MeV.}$$

In table 2, a summary of the fixed primary neutron-gamma source is presented. The calculation is to be made in cylindrical geometry, 4 neutrons and 4 gammas groups, angular quadrature S-8, and 85 spatial meshs.

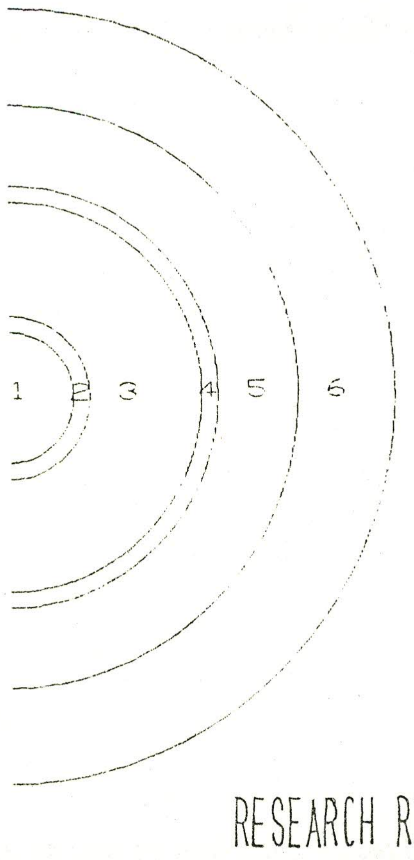
INPUT : The input file for this sample problem was done in the standard format, and thus it will not be possible to modify it using APE 32, interactive Executable File. To run this sample problem the input file is ANISN1.INP . The coupled neutron-gamma cross section entered through the input stream in the standard format 14* array, and were generated using the AMPX-II /4/ code system from nuclear data of VITAMIN-C /5/.

RESULTS: The fluxes distribution are illustrated in figure 3 (neutron groups) and in figure 4 (gamma groups).

SUGGESTED EXERCICES : Since this problem takes time to run in a PC, (~1 hour) it will not be suggeste any modification in the original file. The only suggested exercice is to calculate the dose rate in the shielding surface. To calculate the dose rate it will be used a executable program, DOSE. EXE /2/. This program calculates the gamma and neutron conversion fractors, (rem/hr)/(particle/cm²-sec) using ANSI.ANS 6.11 - Neutron and Gamma ray Flux-to-Dose-Rate Factor, and multiply it by the flux to obtain the Dose Rate /6'. The results that will be obtained by executing this exercice are illustrated in table 3.

Table 2 : GROUP STRUCTURE AND PRIMARY NEUTRON-GAMMA SOURCE FOR SAMPLE PROBLEM 3.

	GROUP	ENERGY RANGE (eV)	χ_g	Γ_{Fg}	Γ_{FPg}	SOURCE INTENSITY (particle/second)
	1	$15.0\times10^{6}-8.20\times10^{5}$	0.754	-	-	2.026×10^{15}
neutrons	2	$8.20\times10^{5}-5.53\times10^{3}$	0.246	-	-	6.623×10^{14}
	3	$5.53\times10^{3}-6.82\times10^{-1}$	1.874×10^{-4}	-	-	5.060×10^{11}
	4	$6.82\times10^{-1}-1.00\times10^{-5}$	2.573×10^{-11}	-	-	6.916×10^{5}
	1	$14.0\times10^{6}-4.00\times10^{6}$	-	8.434×10^{-2}	9.994×10^{-2}	2.006×10^{14}
Gammas	2	$4.0\times10^{6}-2.50\times10^{6}$	-	3.486×10^{-1}	4.200×10^{-1}	4.711×10^{14}
	3	$2.50\times10^{6}-1.00\times10^{6}$	-	1.983	2.189	4.541×10^{15}
	4	$1.00\times10^{6}-1.00\times10^{4}$	-	5.243	5.342	1.152×10^{16}



ZONE	NAME	RADIUS (CM)
1	CORE	24.83
2	GRAPHITE	36.35
3	WATER	150.00
4	SS	150.39
5	LIGHT CONCRETE	210.00
6	HEAVY CONCRETE	360.00

FIGURE 2

RESEARCH REACTOR SHIELDING-GEOMETRY & ZONES

FIGURE 3

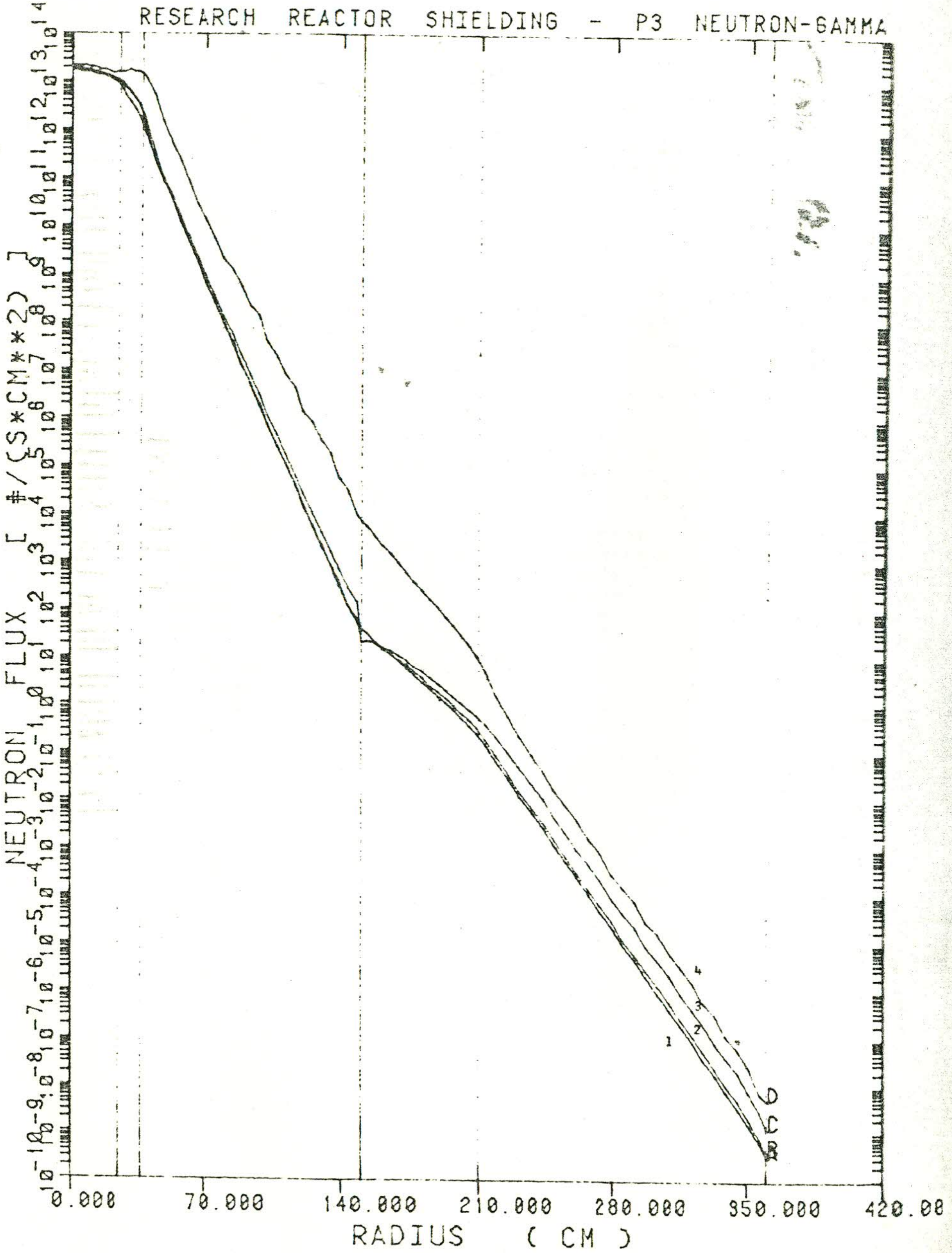


FIGURE 4

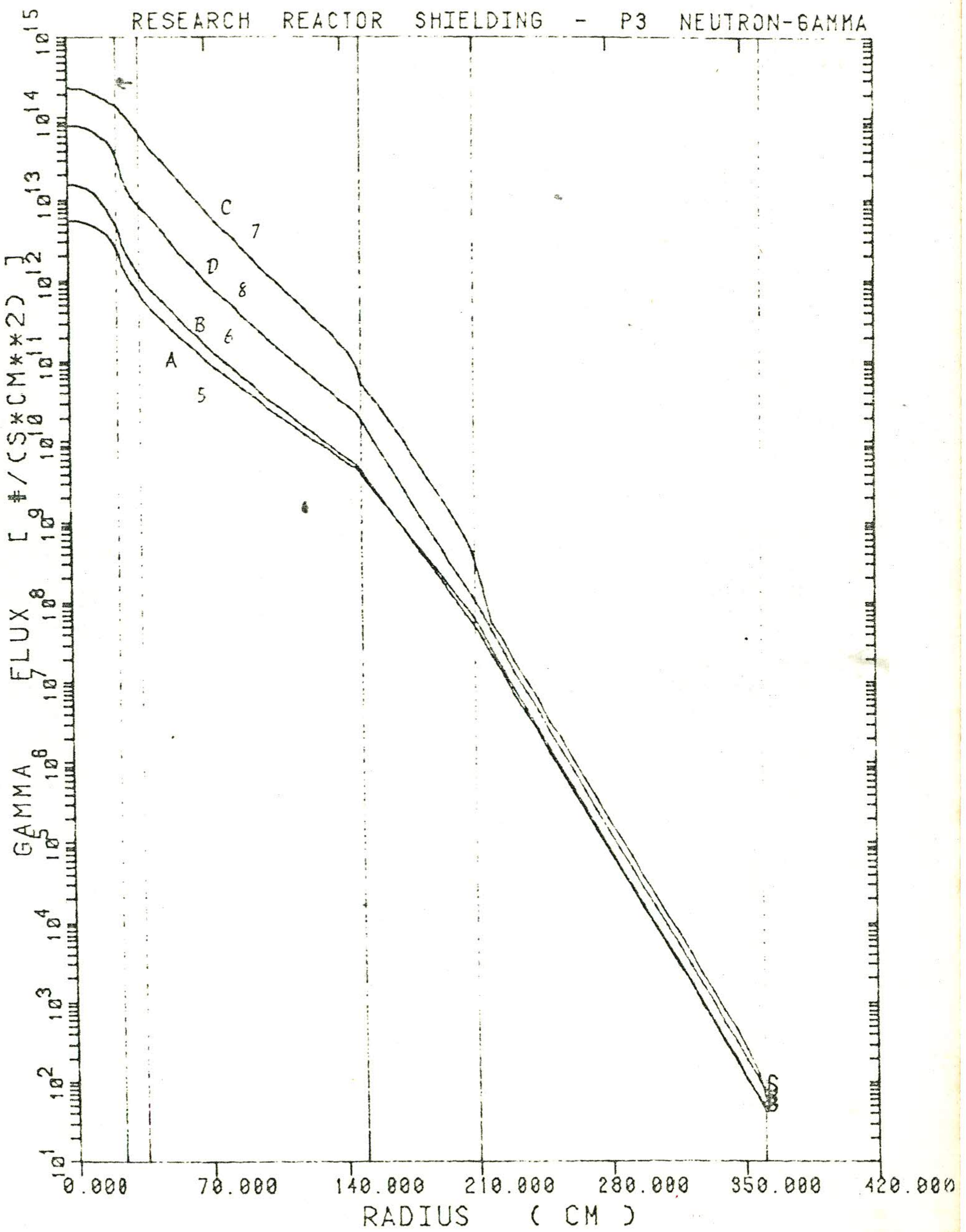


TABLE 3

a) *Neutron Dose Rate*

```
GROUP   AVERAGE ENERGY        FLUX                                            DOSE RATE
            (MEV)         (#/CM**2-S)           FACTOR                        (REM/HR)
                                        (REM/HR)/(#/CM**2-S)
  1       7.410E+00        3.170E-10          1.47E-04                        4.662E-1
  2       4.128E-01        3.670E-10          7.79E-05                        2.859E-1
  3       2.765E-03        1.290E-09          3.62E-06                        4.676E-1
  4       3.410E-07        5.060E-09          4.18E-06                        2.115E-1

 TOTAL NEUTRON DOSE RATE =  1.010E-13  REM/HR
TO CONTINUE PRESS ENTER
```

b) *Gamma Dose Rate*

```
GROUP   AVERAGE ENERGY        FLUX                                            DOSE RATE
            (MEV)         (#/CM**2-S)           FACTOR                        (REM/HR)
                                        (REM/HR)/(#/CM**2-S)
  1       9.000E+00        4.580E+01          8.771E-06                       4.017E-04
  2       3.250E+00        4.270E+01          4.413E-06                       1.884E-04
  3       1.750E+00        7.720E+01          2.930E-06                       2.262E-04
  4       5.050E-01        6.180E+01          1.191E-06                       7.361E-05

 GAMMA TOTAL DOSE RATE=  8.899E-04
Stop - Program terminated.

C:\APC >
```

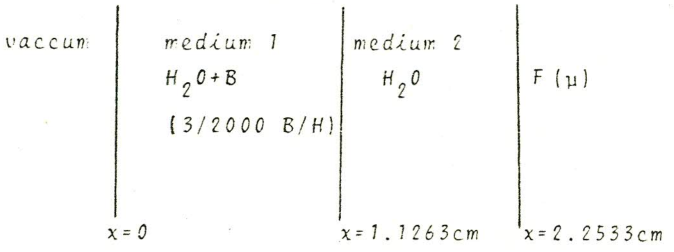
SAMPLE PROBLEM 4

TITLE : ANISN-PC-TWO-SLABS-SHELL SOURCE - TWO-GROUPS PROBLEM (BENCHMARK).

DESCRIPTION : *This problem is a S_N solution of two-media problem irradiated on the surface by a flux of particles. Thus, we consider a slab of thickness 1.1263cm of medium 1 (H_2O + B with 3/2000 barns/H) adjacent to another of thickness 1.1270 cm (medium 2 - H_2O) irradiated on the x=2.2533cm by a flux of neutrons $\underline{F}(\mu)$;*

$$\underline{F}(\mu) = \begin{vmatrix} 1 \\ 0 \end{vmatrix}; \quad \mu < 0 ,$$

i.e., isotropic incidence in group 1 (E>0.3eV). The geometry of the problem is illustrated below. The goal of this problem is to compare the S_N solution obtained by ANISN with that reported by Ishiguro & Garcia / 7 /, which solved the same problem using "Exact" methods, and therefore as a way to assess the accuracy of ANISN code.



INPUT: As mentioned in part I, ANISN does not have as option inhomogeneous boundary condition (specified incoming angular flux). Thus, to simulate this physical problem one has to use shell source type of calculation, i.e., IPM=1 (enter shell source by group and angle in array 18**, in the last mesh interval). When entering the shell source by angle, it should be observed that there are only incidence for $\mu<0$, and therefore only negative angle cosine should contain the shell source (Here it was used S-16, or 17 angle cosine, and thus there are only values in the first 9 angles). The cross sections were reported by Ishiguro & Garcia /7/, and reproduced below. Since, the cross sections are macroscopic it will be not necessary to have a mixing table (10$, 11$, 12*). The input file (ANISNSS1.INP) for this sample problem is enclosed with this notes.

The cross sections for this sample problem were generated by XSDRN code /8 /, and are reproduced in the table 4.

TABLE 4 : CROSS SECTION FOR SAMPLE PROBLEM 4

GROUP	TYPE	MEDIUM 1 (H_2O+B)	MEDIUM 2 (H_2O)
1	Σa_1	0.00231	0.00074
	$\nu\Sigma f_2$	-	-
	Σt_1	0.88731	0.88798
	$\Sigma_{2\to1}$	0.00106	0.000336
	$\Sigma_{1\to1}$	0.83912	0.83975
2	Σa_2	0.07774	0.018564
	$\nu\Sigma f_2$	-	-
	Σt_2	2.9664	2.9865
	-	-	-
	$\Sigma_{2\to2}$	2.8876	2.9676
	$\Sigma_{1\to2}$	0.04588	0.04749

RESULTS : The thermal flux (group 2) distribution obtained by ANISN-PC (S-16) was compared with that obtained by a "Exact" method / 7/ and are illustrated in Figure 5. From this results we note that ANISN can obtain 2 to 3 significant figures when comparing with "exact" solutions ("benchmark").

EXERCICES : To test the sensitivity of the results for different quadrature used, it is suggest to RUN the same problem using: i) Gauss-P_N, and ii) Gauss-DP_N, and compare with the result obtained by Level symmetric-LP_N which is the default option for ANISN-PC.

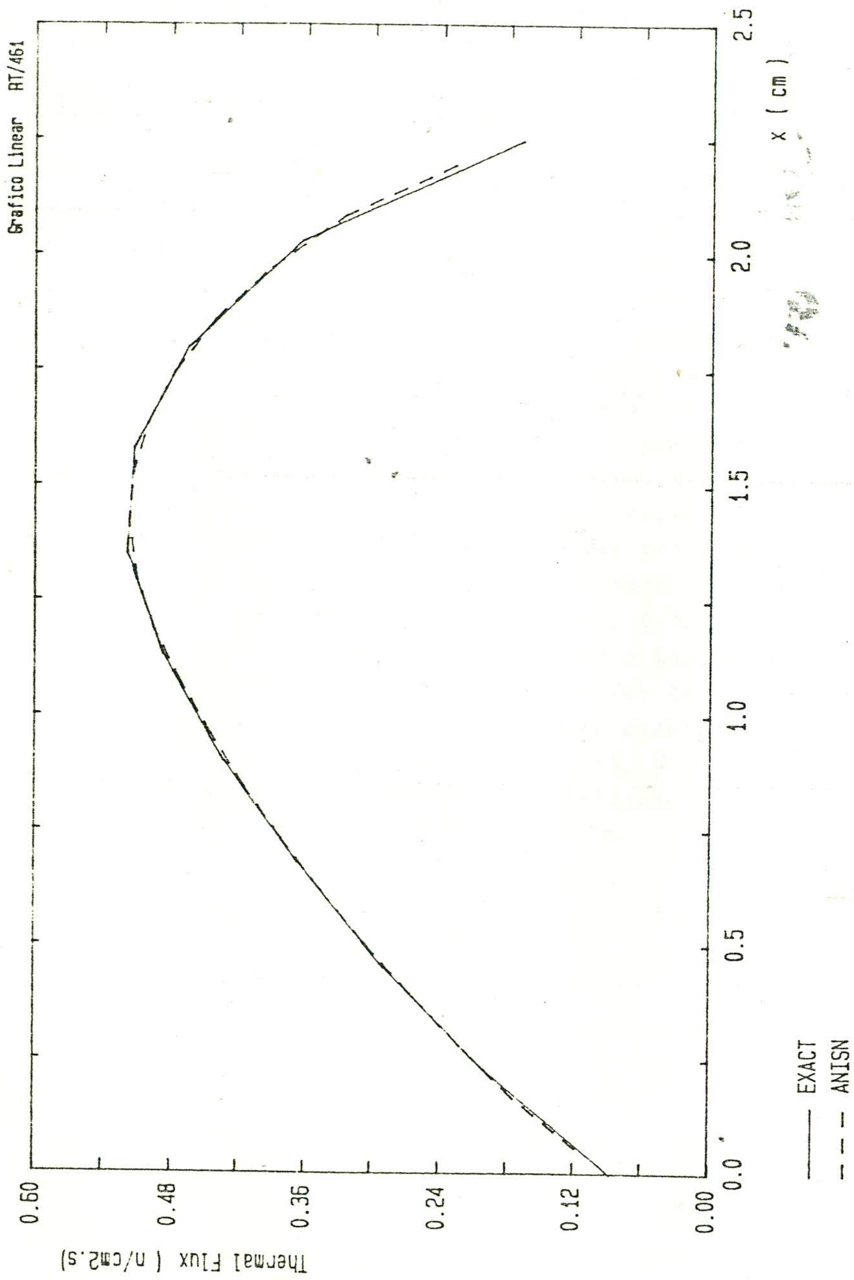


FIGURE 5

TWO SLABS - SHELL SOURCE - TWO GROUPS

```
ANISN PL-TWO-SLABS-SHELL  SOURCE-TWO-GROUP(BENCHMARK)
15$$                 1               0               0               16
                     1               0               0                2
                    20               0               2                3
                     5               6               0
                     0               2               0
                     0               1              20               50
                     0               0               0                0
                    50               0               0                0
                     0               1               0                0
16**    2R+0.00000E-01  +1.00000E-04  4R+0.00000E-01  +1.00000E+00
          +0.00000E-01  +5.00000E-01    +2.00000E-04  3R+0.00000E-01
   T
14**      +2.31000E-03  +0.00000E-01    +8.87310E-01    +1.06000E-03
          +8.39120E-01  +0.00000E-01    +7.77400E-02    +0.00000E-01
          +2.96640E+00  +0.00000E-01    +2.88760E+00    +4.58800E-02
          +7.40000E-04  +0.00000E-01    +8.87980E-01    +3.36000E-04
          +8.39750E-01  +0.00000E-01    +1.85640E-02    +0.00000E-01
          +2.98650E+00  +0.00000E-01    +2.96760E+00    +4.74900E-02
   T
18**    9R+1.11111E-01 25R+0.00000E-01
   T
3**      F+1.00000E+00
   T
1**      F+0.00000E-01
4**       +0.00000E-01  +1.12630E-01    +2.25260E-01    +3.37890E-01
          +4.50520E-01  +5.63150E-01    +6.75780E-01    +7.88410E-01
          +9.01040E-01  +1.01367E+00    +1.12630E+00    +1.23900E+00
          +1.35170E+00  +1.46440E+00    +1.57710E+00    +1.68980E+00
          +1.80250E+00  +1.91520E+00    +2.02790E+00    +2.14060E+00
          +2.25330E+00
5**      F+1.00000E+00
6**       +0.00000E-01  +2.44936E-02    +4.13296E-02    +3.92569E-02
          +4.00796E-02  +6.43754E-02    +4.42097E-02    +1.09085E-01
        2R+1.37170E-01  +1.09085E-01    -4.42097E-02    +6.43754E-02
          +4.00796E-02  +3.92569E-02    -4.13296E-02    -2.44936E-02
7**       -9.90298E-01  -9.80501E-01    -9.09285E-01    -8.31997E-01
          -7.46751E-01  -6.50426E-01    -5.37097E-01    -3.92289E-01
          -1.38957E-01  +1.38957E-01    +3.92289E-01    +5.37097E-01
          +6.50426E-01  +7.46751E-01    +8.31997E-01    +9.09285E-01
          +9.80501E-01
8$$                  1               1               1                1
                     1               1               1                1
                     1               1               2                2
                     2               2               2                2
                     2               2               2                2
9$$                  1               2
   T    T
```

SAMPLE PROBLEM 5

TITLE : ANISN-PC-ZONE THICKNESS SEARCH-CRITICAL HALF THICKNESS.

DESCRIPTION : The goal of this problem is to determine the value of the critical thickness for reflected slab reactors. The core of multiplying medium 1 extends from - x to x surrounded by a finite reflector of non multiplying medium 2 with thickness B, and thus our aim is to determine the value of x which makes the reactor critical (k=1). Also, since this problem has an "exact" solution / 7 / it is an opportunity to determine the acurracy of ANISN in criticality search. The problem is to be done in two group, S-12, and with B=5.63076cm (equivalent to 5 fast mean free path). The cross sections for the reflector is that reported in sample problem 2(H_2O), and for the core (H_2O+U-235, U/H=1/1000) are these reported by Ishiguro and Garcia / 7 /, and illustrated below,

$\chi_1=1.0$	$\chi_2=0.0$
$\Sigma_{a_1}=0.00194\,cm^{-1}$	$\Sigma_{a_2}=0.05363\,cm^{-1}$
$(\nu\Sigma_f)_1=0.00209\,cm^{-1}$	$(\nu\Sigma_f)_2=0.07391\,cm^{-1}$
$\Sigma_{t_1}=0.88721\,cm^{-1}$	$\Sigma_{t_2}=2.9727\,cm^{-1}$
$\Sigma_{2\to 1}=0.00767\,cm^{-1}$	$\Sigma_{2\to 2}=2.9183\,cm^{-1}$
$\Sigma_{1\to 1}=0.83892\,cm^{-1}$	$\Sigma_{1\to 2}=0.04635\,cm^{-1}$
(Fast Group, $E>0.3eV$)	(Thermal Group, $E<0.3eV$)

INPUT : This problem is a zone thickness search (IEVT=4 in 15$ array), and the main characteristics in the input file are in array 15$ (IPVT=1), and consequentely in the 16**array the value of K as 1.00. Also in this type of problem (criticality search) is necessary to enter the 20* array (Radius Modifier by zone). In this array one should give a value for the core, and zero for the reflector, since the search is for the critical thickness of the core. The input file for this sample problem (ANISNCRT.INP) is enclosed with this note.

RESULTS : The critical half-thickness obtained by ANISN-PC is illustrated in table 5 , for several reflector thickness and compared with "exact" results obtained by Ishiguro and Garcia / 7 /. From these results we observe that ANISN gives less than 0.1% of error comparing with a standard "exact" result.

Table 5 . Critical half-thickness for critical reflected slab reactor.

Reflector Thickness	5.63076cm	3.3784cm	2.25230cm
Core Half Thickness (Exact)	4.8634cm	5.3546cm	5.8921cm
Core Half Thickness (ANISN)	4.865cm	5.358cm	5.900cm

EXERCICES : The suggested exercices are to change the reflector thickness to 3.3784cm and 2.25230cm, and compare Critical half-thickness with exact solution (see table 5).

```
ANISN-PC-ZONE THICKNESS SEARCH-CRITICAL HALF THICKNESS
15$$             1                 0                 0               12
                 1                 1                 0                3
                20                 4                 2                1
                 5                 6                 0               50
                 0                 2                 0                0
                 0                 0                 0                0
                 0                 0                 0                0
                50                 0                 0
                 0                 1                 0
16**    +0.00000E-01      -1.00000E-01      +1.00000E-04    4R+0.00000E-01
      2R+1.00000E+00      +5.00000E-01      +2.00000E-04      +5.00000E-08
        +1.00000E-03      +7.50000E-01
   T
14**    +1.94000E-03      +2.09000E-03      +8.87210E-01      +7.67000E-04
        +8.38920E-01      +0.00000E-01      +5.36300E-02      +7.39100E-02
        +2.97270E+00      +0.00000E-01      +2.91830E+00      +4.63500E-02
        +7.40000E-04      +0.00000E-01      +8.87980E-01      +3.36000E-04
        +8.39750E-01      +0.00000E-01      +1.85640E-02      +0.00000E-02
        +2.98650E+00      +0.00000E-01      +2.96760E+00      +4.74900E-02
   T
3**     F-1.00000E+00
   T
1**     -1.00000E+00      +0.00000E-01
4**     +0.00000E-01      +4.00000E-01      +8.00000E-01      +1.20000E-00
        +1.60000E+00      +2.00000E+00      +2.40000E+00      +2.80000E-00
        +3.20000E+00      +3.60000E+00      +4.00000E+00      +4.56308E+00
        +5.12615E+00      +5.68923E+00      +6.25230E+00      +6.81538E+00
        +7.37846E+00      +7.94153E+00      +8.50461E+00      +9.06768E+00
        +9.63076E+00
5**     F-1.00000E+00
6**     +0.00000E-01      +3.53813E-02      +5.58811E-02      +6.24786E-02
        +6.31890E-02      +1.19089E-01    2R+1.63981E-01      +1.19089E-01
        +6.31890E-02      +6.24786E-02      +5.58811E-02      +3.53813E-02
7**     -9.85921E-01      -9.71638E-01      -8.72271E-01      -7.60021E-01
        -6.28019E-01      -4.59548E-01      -1.67213E-01      -1.67213E-01
        +4.59548E-01      +6.28019E-01      +7.60021E-01      +8.72271E-01
        +9.71638E-01
8$$              1                 1                 1                1
                 1                 1                 1                1
                 1                 1                 2                2
                 2                 2                 2                2
                 2                 2                 2                2
9$$              1                 2
20**    +1.00000E-03      +0.00000E-01
   T    T
```

SAMPLE PROBLEM 6

TITLE : ACTIVITIES (REACTION RATE) IN A PHANTON HEAD, IRRADIATED BY NEUTRONS. BNCT (BORON NEUTRON CAPTURE THERAPY) / 8/.

DESCRIPTION : *In order to illustrate the activities calculation by ANISN, a simulation of a phanton head with a selective concentration of Boron (B-10), irradiated by an isotropic beam of neutrons with a energy spectrum in the phanton surface, was performed. This calculation is basic in a technique called "Boron Neutron Capture Therapy" (BNCT) for treating brain tumors depends on selective loading of tumortissue with a B-10 compound and subsequent irradiation of the brain with neutrons. The alpha particles produced in the $^{10}B(n,\alpha)^{7}Li$ reaction destroy the tumor cells. A tissue equivalent head model was simulated as one dimensional slab of 23cm thickness, and using a buckling factor correction for finite transverse dimension (17cm). The tissue composition is illustrated in table 6 , and it was assumed a tumor size of 5cm tickness, localized at 5cm deep from the head surface. Also, it was assumed a concentration of 30 µg of B-10 per gram of tissue in the tumor zone ($1.65x10^{-6}$B-10 atoms/barn-cm), and 10µg of B-10 per gram of tissue, in the tissue zones ($0.55x10^{-6}$ B-10 atoms/barn-cm). Thus, the problem was simulated as three region (zones). All* the calculations were made in a coupled neutron-gamma group structure of 12 groups (6 neutrons, 6 gammas), as shown in table 7, and it was assumed a neutrons incidence at the botton surface (shell source) with a energy distribution given by :

Group 1 : 0.01
Group 2 : 0.27
Group 3 : 0.27
Group 4 : 0.24
Group 5 : 0.21
Group 6 : 0.00

The goal of the sample problem is to calculate the reaction rate (activities) in the normal tissue elements due to neutron capture, such as O(n,γ); H(n,γ), as well as the reaction rate due to (n,α) reaction in the B-10, for the three zones of the phanton head. These reaction rate give an indication of the dose rate due to different nuclear reaction, as well an indication of the BNCT efficacy:

Table 6 : Tissue Composition in a Phanton Head (ρ= 0.9869gr/cm^3).

Element	Composition Wt%	Atomic Density (atom/barn-cm)
O	74	2.749 E-2
C	15.4	7.620 E-3
H	10.4	6.133 E-2
Na	0.2	5.000 E-5

^{10}B(normal tissue)=5.5E-5at/barn-cm
^{10}B(tumor tissue)=1.65E-6at/barn-cm.

Table 7 : Group Structure

	Group	Energy Range (MeV)
	1	1.0 - 5.0
	2	0.5 - 1.0
neutrons	3	0.1 - 0.5
	4	0.01 - 0.1
	5	0.5x10^{-6} - 0.01
	6	0 - 0.5x10^{-6}
	1	10 - 14
	2	5 - 10
gammas	3	2 - 5
	4	1 - 2
	5	0.5 - 1
	6	0.01 - 0.5

<u>INPUT</u> : The sample problem is a shell source type of calculation (IEVT=0) in slab geometry (IGE=1), P5-S8 approximation, 45 spatial mesh (IM); 12 groups (IGE), 3 zones (IZM), and with the option to calculate activities by zone (ID3). Since it is an activity calculation in the table of cross section one has to enter the activity cross sections for the desired reactions. The number of activities to be calculated are 5 per zone (1=O(n,γ); 2=C(n,γ); 3=H(n,γ); 4=Na(n,γ); 5=^{10}B(n,α)). All the cross section, including the capture cross sections, and (n,α) reaction in B-10, were calculated from VITAMIN-C / 5 /, by AMPX-II / 4 /. The Input File (TUMOR.DAT) is in standard format, and since it is too big (cross section sets are read in array 14*), it will not be reproduced here.

RESULTS : The activities by zone for each reaction are illustrated in table 8 .

Table 8 : Activities by Zone (reactions per second)*

ZONE	REACTION				
	O(n,γ)	C(n,γ)	H(n,γ)	Na(n,γ)	B(n,α)
1	$1.27x10^{-5}$	$6.66x10^{-5}$	$4.97x10^{-2}$	$6.98x10^{-5}$	$5.46x10^{-3}$
2	$1.32x10^{-5}$	$6.95x10^{-5}$	$5.18x10^{-2}$	$7.24x10^{-5}$	$1.71x10^{-2}$
3	$4.95x10^{-6}$	$2.60x10^{-5}$	$1.94x10^{-2}$	$2.71x10^{-5}$	$2.13x10^{-3}$

*Source Normalization Factor =1.

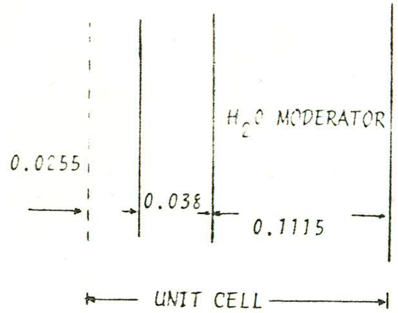
SAMPLE PROBLEM 7

TITLE : ANISN-PC-CELL CALCULATION FOR PLATE TYPE REACTOR

DESCRIPTION : This sample problem is a cell calculation for a typical plate type reactor (fuel region) to generate the macroscopic cross section in four groups (macroscopic cell weighted cross section), using microscopic cross sections library in 60 groups (36 thermal groups with up-scattering cutoff 1.855eV). The microscopic cross sections library (reactor independent) was generated using basic data from ENDF-B-IV (Evaluated Nuclear Data File) manipulated by AMPX-II /4/ with a weighting spectrum: Maxweelian (300^oK), 10^{-5} to 1.855eV; 1/E from 1.855eV to 0.85MeV; and a fission spectrum from 0.85MeV to 10MeV. The collapsed group structure for the macroscopic cross sections was taken as:

Group 1 : 10MeV - 0.8208MeV
Group 2 : 0.8208MeV - 5.53KeV
Group 3 : 5.53KeV - 0.625eV
Group 4 : 0.625eV - 10^{-5}eV

The unity cell consist of a U-Al_x meat, with 0.0255cm half-thickness, a Aluminium cladding with 0.038cm thickness, and a Moderator region (H_2O) with 0.115cm thickness. This unit cell represents a standard (23 plates/Element) Fuel Element with 93 w/o enriched as that reported in Tec.Doc233 /9/. Below, the unit cell is illustrated, as well as the atomic densities in each zone.



Atomic Densities (at/barns-cm)

Meat
Nu-235=1.6181E-3
Nu-238=1.2026E-4
Nal=5.7021E-2

Clad: Nal=6.0271E-2

Moderator: No=3.3431E-2
NH=6.6863E-2

INPUT : The microscopic cross sections (U-238;U-235;H;O;Al) in 60 groups, P-1 scattering order, are entered through the input stream in the free format 14** array. The boundary conditions are reflection in both boundaries (IBL=IBR=1), the number of mesh intervals are 11, being 5 in the meat, 1 in the clad, and 5 in the moderator, and the order of quadrature is S-4. The sample problem is an eigenvalue calculation (k_∞), with flux weight for cell calculation (IFG=1), and thus it will be necessary enter array 27$$ (Few Group Parameters), and array 28$$ (Few Group Number for each Multigroup). The input file for this sample problem is ANISNCEL.INP, and since it is too big (98008 location required) it will not be reproduced here. It is interesting to emphasize that this problem requires more location than that available (80002), and in this case ANISN will internally change IDAT1 to 1, i.e, the location required for cross sections will be storage in scratch files (Hard Disk).

RESULTS : The infinity multiplication factor (K_∞) obtained by ANISN-PC is illustrated in table 9, and compared with that obtained by XSDRNPM /4/, and HAMMER /10/ codes for the same problem. Also, enclosed with this note, the cross section weighting data, and the collapsed cross sections for each zones (4 groups), are illustrated.

Table 9 : Infinity Multiplication Factor For Sample Problem 7.

Code	K_∞
ANISN-PC (P-1; 60 groups)	1.77075
HAMMER (Spectrum code)	1.77329
XSDRNPM (P-1;85 groups)	1.77072
XSDRNPM (P-3; 85 groups)	1.77046

```
ANISN-PC-CELL CALCULATION FOR PLATE TYPE REACTOR

CROSS SECTION WEIGHTING DATA

ICON     0/1/2/ZONE/MIXED/MACRO/BOTH (MINUS IMPLIES CELL)           -2
IHTF     POSITION OF SIGMA TOTAL                                     3
IHSF     POSITION OF SIGMA GG (MINUS IMPLIES REMOVE UPSCATTER)      -4
IHMF     TABLE LENGTH                                                7
IPUN     PUNCH TRIGGER  -1/0/1=ISOTXS/NO/ANISN PUNCH                 0
ICOM     NO. OF FEW GROUPS                                           4

FEW GROUP   1 CONTAINS GROUP(S)   1   2   3   4   5   6   7   8   9  10  11
12

FEW GROUP   2 CONTAINS GROUP(S)  13  14  15  16  17  18  19  20  21  22  23
24  25  26  27  28

FEW GROUP   3 CONTAINS GROUP(S)  29  30  31  32  33  34  35  36  37  38  39

FEW GROUP   4 CONTAINS GROUP(S)  40  41  42  43  44  45  46  47  48  49  50
51  52  53  54  55  56  57  58  59  60

ANISN-PC-CELL CALCULATION FOR PLATE TYPE REACTOR

            MATERIAL#       ZONE        L OF F(L)
     1         11             1             0
     2         12             1             1
     3         15             2             0
     4         16             2             1
     5         13             3             0
     6         14             3             1

  LEGEND FOR COLLAPSED CROSS SECTION TABLES

    POS.      DESCRIPTION

      1    SIGMA ABSORPTION FOR GROUP g
      2    NU SIGMA FISSION FOR GROUP g
      3    SIGMA TOTAL        FOR GROUP g
      4    SIGMA        SCATTERING FROM g     TO g
      5    SIGMA DOWN-SCATTERING FROM g- 1 TO g
      6    SIGMA DOWN-SCATTERING FROM g- 2 TO g
      7    SIGMA DOWN-SCATTERING FROM g- 3 TO g
      8    SIGMA DOWN-SCATTERING FROM g- 4 TO g
      9    SIGMA DOWN-SCATTERING FROM g- 5 TO g
     10    SIGMA DOWN-SCATTERING FROM g- 6 TO g
     11    SIGMA DOWN-SCATTERING FROM g- 7 TO g
     12    SIGMA DOWN-SCATTERING FROM g- 8 TO g
     13    SIGMA DOWN-SCATTERING FROM g- 9 TO g
     14    SIGMA DOWN-SCATTERING FROM g-10 TO g
     15    SIGMA DOWN-SCATTERING FROM g-11 TO g

 FLUX WEIGHTED CROSS SECTIONS FOR MATERIAL 11 IN ZONE  1
 ZONE FLUX/CELL FLUX INCLUDED IN WEIGHTING

 POS.  GRP.   1    GRP.   2    GRP.   3    GRP.   4
  1   4.52370E-04 9.82054E-03 1.78123E-02 1.05185E-01
  2   8.28457E-04 1.38026E-02 3.59247E-02 2.14154E-01
  3   3.15931E-02 2.44244E-02 3.22495E-02 1.20109E-01
  4   3.11021E-02 1.44467E-02 1.21913E-02 1.49238E-02
  5   0.00000E-01 1.39012E-04 1.55187E-04 2.24571E-03
  6   0.00000E-01 0.00000E-01 0.00000E-01 0.00000E-01
  7   0.00000E-01 0.00000E-01 0.00000E-01 0.00000E-01

 FLUX WEIGHTED CROSS SECTIONS FOR MATERIAL 12 IN ZONE  1
 ZONE FLUX/CELL FLUX INCLUDED IN WEIGHTING

 POS.  GRP.   1    GRP.   2    GRP.   3    GRP.   4
  1   0.00000E-01 0.00000E-01 0.00000E-01 0.00000E-01
  2   0.00000E-01 0.00000E-01 0.00000E-01 0.00000E-01
  3   3.15931E-02 2.44244E-02 3.22495E-02 1.20109E-01
  4   1.95521E-02 9.42064E-04 1.77635E-03 8.90958E-04
  5   0.00000E-01-1.59440E-04-7.56015E-05-8.28164E-04
  6   0.00000E-01 0.00000E-01 0.00000E-01 0.00000E-01
  7   0.00000E-01 0.00000E-01 0.00000E-01 0.00000E-01

 FLUX WEIGHTED CROSS SECTIONS FOR MATERIAL 15 IN ZONE  2
 ZONE FLUX/CELL FLUX INCLUDED IN WEIGHTING

 POS.  GRP.   1    GRP.   2    GRP.   3    GRP.   4
  1   6.32254E-05 1.01098E-04 5.12517E-04 2.06037E-03
  2   0.00000E-01 0.00000E-01 0.00000E-01 0.00000E-01
  3   4.62493E-02 1.77108E-02 1.81403E-02 1.97703E-02
  4   4.54590E-02 1.75904E-02 1.45701E-02 1.77090E-02
  5   0.00000E-01 2.15420E-04 2.19372E-04 3.05770E-03
  6   0.00000E-01 0.00000E-01 0.00000E-01 0.00000E-01
  7   0.00000E-01 0.00000E-01 0.00000E-01 0.00000E-01

 FLUX WEIGHTED CROSS SECTIONS FOR MATERIAL 16 IN ZONE  2
```

```
 POS.  GRP.   1    GRP.   2    GRP.   3    GRP.   4
  1   0.0[illegible]E-01 0.0[illegible] [illegible] [illegible]
  2   0.00000E-01 0.00000E-01 0.00000E-01 0.00000E-01
  3   4.62493E-02 1.77108E-02 1.81403E-02 1.977[illegible]
  4   2.75[illegible] 1.42[illegible]E-02 [illegible] 1.54[illegible]E-02
  5   0.0[illegible]E-01-2.48727E-0[illegible]-1.13[illegible]E-04 1.21117E-0[illegible]
  6   0.00000E-01 0.0[illegible] [illegible] 0.000[illegible]
  7   0.00[illegible]E-01 [illegible] [illegible] [illegible]

 FLUX WEIGHTED CROSS SECTIONS FOR MATERIAL 13 IN ZONE  3
 ZONE FLUX/CELL FLUX INCLUDED IN WEIGHTING

 POS.  GRP.   1    GRP.   2    GRP.   3    GRP.   4
  1   1.22877E-04 4.28473E-04 2.35569E-03 5.79063E-03
  2   0.00000E-01 0.00000E-01 0.00000E-01 0.00000E-01
  3   3.74914E-01 9.48502E-01 9.98590E-01 1.81730E+00
  4   3.25146E-01 8.47337E-01 4.13581E-01 1.80752E+00
  5   0.00000E-01 4.96279E-02 5.87044E-02 5.82636E-01
  6   0.00000E-01 0.00000E-01 7.39295E-06 4.21307E-02
  7   0.00000E-01 0.00000E-01 0.00000E-01 5.22264E-06

 FLUX WEIGHTED CROSS SECTIONS FOR MATERIAL 14 IN ZONE  3
 ZONE FLUX/CELL FLUX INCLUDED IN WEIGHTING

 POS.  GRP.   1    GRP.   2    GRP.   3    GRP.   4
  1   0.00000E-01 0.00000E-01 0.00000E-01 0.00000E-01
  2   0.00000E-01 0.00000E-01 0.00000E-01 0.00000E-01
  3   3.74914E-01 9.48302E-01 9.98590E-01 1.81730E+00
  4   5.60276E-01 1.61670E+00 8.74205E-01 1.31661E+00
  5   0.00000E-01 6.53153E-02 9.32350E-02 6.53815E-01
  6   0.00000E-01 0.00000E-01 1.52432E-06 3.32134E-02
  7   0.00000E-01 0.00000E-01 0.00000E-01 1.96861E-06

 ANISN-PC-CELL CALCULATION FOR PLATE TYPE REACTOR

 INTEGRATED FLUX

 GRP.  ZONE   1    ZONE   2    ZONE   3    ZONE   4
  1   2.87252E+00 4.27912E+00 1.25483E+01 1.97000E+01
  2   1.28254E+00 1.91108E+00 5.61035E+00 8.80397E+00
  3   1.25015E-01 1.86289E-01 5.47173E-01 8.58477E-01
  4   1.07974E+00 1.61083E+00 4.78458E+00 7.47515E+00

 AVERAGE FLUX

 GRP.  ZONE   1    ZONE   2    ZONE   3    ZONE   4
  1   1.12641E+02 1.12614E+02 1.12535E+02 1.12567E+02
  2   5.02727E+01 5.02944E+01 5.03141E+01 5.03067E+01
  3   4.90226E+00 4.90259E+00 4.90710E+00 4.90542E+00
  4   4.23402E+01 4.23926E+01 4.29085E+01 4.27137E+01

 AVERAGE FLUX RATIO

 GRP.  ZONE   1    ZONE   2    ZONE   3    ZONE   4
  1   1.00065E+00 1.00042E+00 9.99702E-01 1.00000E+00
  2   9.99722E-01 9.99755E-01 1.00015E+00 1.00000E+00
  3   9.99357E-01 9.99424E-01 1.00034E+00 1.00000E+00
  4   9.91255E-01 9.92482E-01 1.00456E+00 1.00000E+00

 INTEGRATED FLUX RATIO

 GRP.  ZONE   1    ZONE   2    ZONE   3    ZONE   4
  1   1.45814E-01 2.17215E-01 6.36972E-01 1.00000E+00
  2   1.45676E-01 2.17073E-01 6.37255E-01 1.00000E+00
  3   1.45625E-01 2.16995E-01 6.37376E-01 1.00000E+00
  4   1.44444E-01 2.15492E-01 6.40064E-01 1.00000E+00

 ZONE    VOL. FRAC.      VOLUME
    1   1.45716E-01   2.55016E-02
    2   2.17124E-01   3.79980E-02
    3   6.37158E-01   1.11506E-01
    4   1.00000E+00   1.75006E-01
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