NENG 685 | Fall 2017
HW 4 | Due Nov. 29, 2017

Name: Kevin Choe

On homework:

- Well organized and documented work scores better. If I cannot figure out what is going on, then I am less likely to "intuit" what you intended, and the score will be reflective of this fact.
- If you work with anyone else, document what you worked on together.
- Show your work.
- Always clearly label plots (axis labels, a title, and a legend if applicable).
- Homework should be done "by hand" (i.e. not with a numerical program such as MATLAB, Python, or Wolfram Alpha) unless otherwise specified. You may use a numerical program to check your work.
- If you use a numerical program to solve a problem, submit the associated code, input, and output.
- ***I should not have to run your code to see your answers.*** The attached code is an additional form of feedback for me and a method to give partial credit. If you want full credit, then include the outputs (plots, tables, answers, etc.) in your write-up.

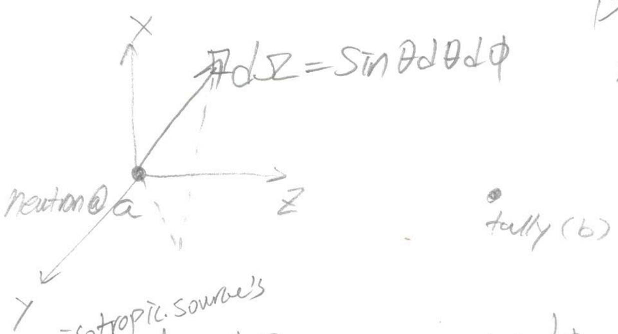
Do not write in the table to the right.

Problem	Points	Score
1	30	
2	30	
3	15	
4	25	
Total:	100	

1. (30 points) Using the direct inversion of CDF sampling method, derive sampling algorithms for

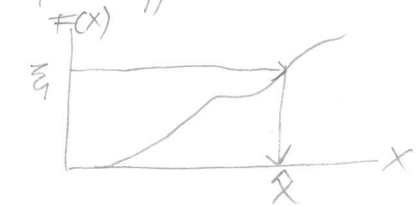
   (a) The neutron direction in 3D if the neutron source is isotropic. *Note: each unit direction should have a specified sampling.*

$d\Omega = \sin\theta \, d\theta \, d\phi$



Direct inversion of CDF

$\hat{x} \leftarrow F^{-1}(\xi)$



Isotropic source's surf Area

$$\frac{d\Omega}{\rightarrow 4\pi} = \frac{\sin\theta \, d\theta \, d\phi}{4\pi} = \frac{-d\cos\theta \, d\phi}{4\pi} = -\frac{d\mu \, d\phi}{4\pi}.$$

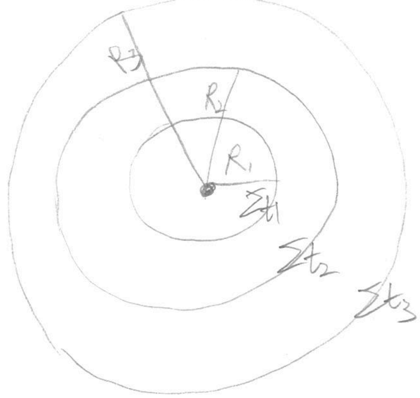
apply direct inversion of CDF method

$$f(\Omega) = f_1(\mu) f_2(\phi) = \frac{1}{2}\frac{1}{2\pi}$$

$$F_1(\mu) = \int_{-1}^{\mu} f_1(\mu')d\mu' = \frac{1}{2}(\mu + 1) = \xi_1 \quad \text{or} \quad \mu = 2\xi_1 - 1$$

$$F_1(\phi) = \int_0^{\phi} f_2(\phi')d\phi' = \frac{\phi}{2\pi} = \xi_2 \quad \text{or} \quad \phi \; 2\pi\xi_2$$

(b) The distance to the next collision in the direction of neutron motion if the neutron is in the center of the spherical volume that consists of three concentric layers with radii R_1, R_2, and R_3, each made of different materials with total cross sections Σ_{t1}, Σ_{t2}, and Σ_{t3}, respectively. *Note: Do not use the mfp algorithm to determine the distance; sample the distance explicitly.*



total cross section

$$\sum_{t=1}^{3} = \Sigma_{t1} + \Sigma_{t2} + \Sigma_{t3} = P_1 + P_2 + P_3$$

distance to collision, $S_c = \dfrac{n_c}{\Sigma_t}$

sample the number of Mean Free path.

$$g(n_c)\,dn_c = e^{-n_c}\,dn_c$$

$$G(n_c)\,dn_c = 1 - e^{n_c}$$

with direct inversion of CDF, we get $n_c = -\ln(1-\xi)$

plug n_c in to S_c we get,

$$S_c = -\frac{\ln(1-\xi)}{\Sigma_t} \propto -\frac{\ln(\xi_n)}{\Sigma_n}$$

1. $S_1 \le P_1$ for a particle moving from the center of the sphere to the first Boundary:

$$S_1 = -\frac{\ln(\xi_1)}{\Sigma_{t1}}$$

2. $P_1 \le S_2 \le P_1 + P_2$

$$S_2 = -\frac{\ln(\xi_2)}{\Sigma_{t2}}$$

3. $P_1 + P_2 \le S_3 < P_1 + P_2 + P_3$

$$S_3 = -\frac{\ln(\xi_3)}{\Sigma_{t3}}$$

4. $S_3 < P_1 + P_2 + P_3$: kill zone.

(c) The type of collision if it is assumed that the neutron can have both elastic and inelastic scattering, and can be absorbed in fission or (n,gamma) capture interactions. Assume monoenergetic neutron transport.

$$\sum_{i=1}^{4} P_i = \underbrace{P_1}_{\text{elastic}} + \underbrace{P_2}_{\text{inelastic}} + \underbrace{P_3}_{\text{fission}} + \underbrace{P_4}_{\text{n, gamma capture}}$$

$0 \le \xi < P_1$: elastic scattering

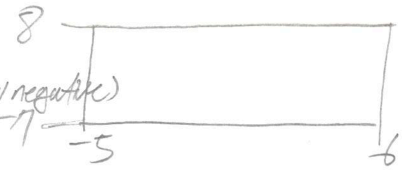
$P_1 \le \xi < P_1 + P_2$: inelastic scattering

$P_1 + P_2 \le \xi < P_1 + P_2 + P_3$: fission

$P_1 + P_2 + P_3 \le \xi < P_1 + P_2 + P_3 + P_4 = 1$: absorption (n, gamma) capture

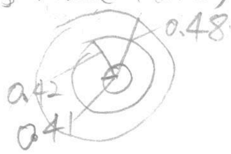
Vaccum M1

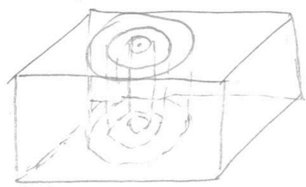
2. A sample of MCNP input is given below: cell cards



PWR Single Pin title line
1 1 -10.41 -1 -10 20 imp:n=1 filler density: 10.41 g/cm³ (w/ negative)
2 0 1 -2 -10 20 imp:n=1
3 3 -6.55 2 -3 -10 20 imp:n=1 M3
4 4 -0.7 3 -5 6 -7 8 -10 20 imp:n=1 material 4 (water) filling up the defined volume — density of 0.7 g/cm³
5 0 5:-6:7:-8:10:-20 imp:n=0 M4
outside X plane & outside Y plane & outside Z plane = 0 = kill zone
Vaccum

c Rods dimensions (surface card)





1 cz 0.41 reflective surface
2 cz 0.42
3 cz 0.48
c Basic lattice cell
*5 px 0.63 plane perpendicular to X axis @ X = 0.63cm — Create a Box
*6 px -0.63 // @ X = -0.63cm
*7 py 0.63 // Y axis @ Y = 0.63cm
*8 py -0.63 // @ Y = 0.63cm
c Axial limits
*10 pz 200.0 outside plane perpendicular to Z axis @ Z = 200cm — top & Botom
*20 pz -200.0 inside //

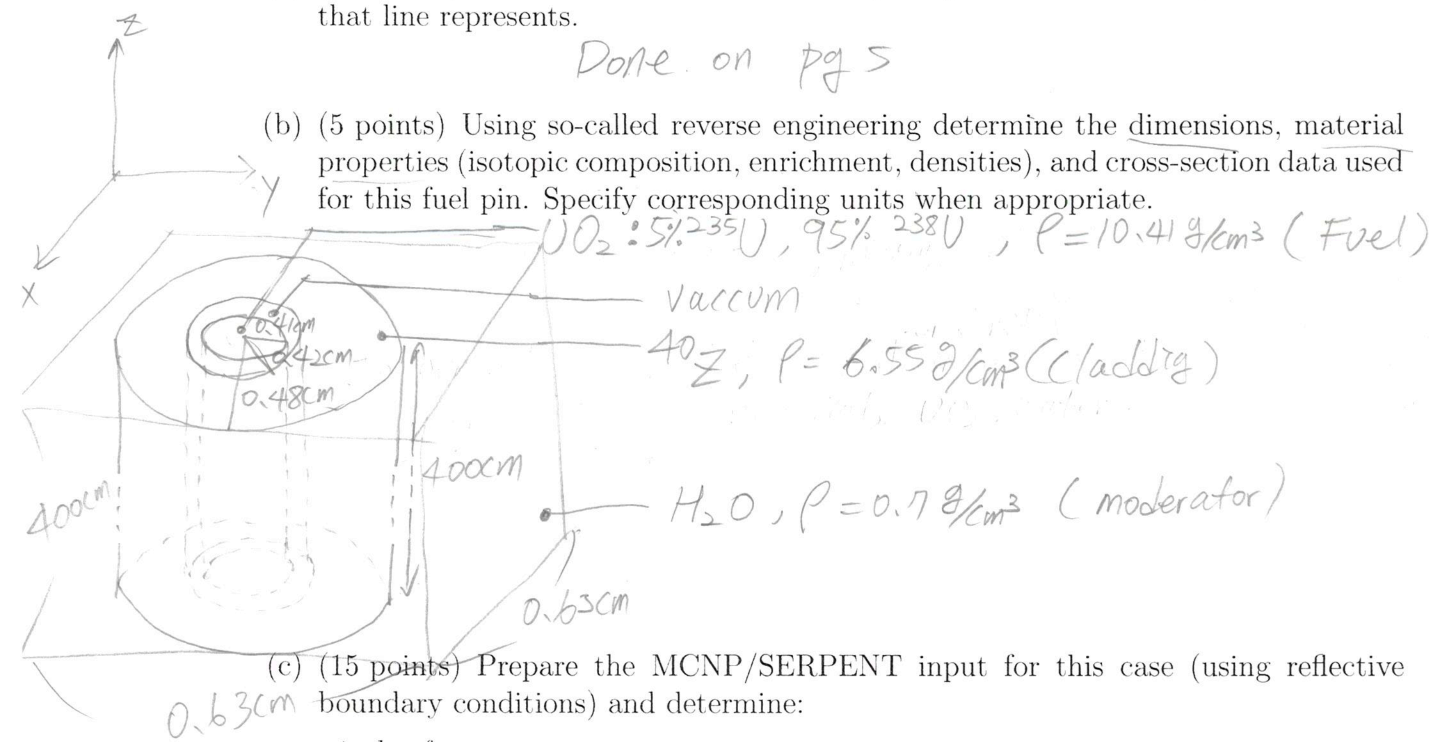
c Material 1 (UO_2)
m1 8016.13c 2.0 material 1 (Z = 8; A = 16) ENDF7LX (6.13c) ^{16}O / atomic fraction 2
92235.63c 0.05 endf66d ^{235}U / atom percent 0.05
92238.63c 0.95 endf66d ^{238}U / // 0.95
c Material 4 (light water)
m4 1001.78c 2 material 4 / endf 70a ^{1}H / atomic fraction 2
8016.78c 1 endf70a ^{16}O / atomic fraction 1
mt4 lwtr.04t material 4 / light water (endf5)
c Material 3 (Z for the wall)
m3 40000.12c 1. cladding — material 3 / 40 Z / atomic fraction 1
c ——— Tallies ———
fc4 Tally 4 Tally comment 4, provides Title for tally in output as "Tally 4"
f4:n 1 3 4 Flux averaged over cells: 1, 3, 4
e4:n 1e-6 1. 20. Energy bin for tally 4: 1eV – 20MeV
fc14 Tally 14 Tally coment 14, title "Tally 14"
f14:n 1 track length in cell 1
fm14 (1 1 (-2) (-6)) Tally multiplier card 14 / (Normalization factor = 1, material 1, absorption cross section, Total fission cross section)
c ———————
kcode 1000 1.00 50 250 criticality calculation / 1000 histories / initial Keffective = 1 / skip 50 cycles / # of cycles to be done 250 — output
ksrc 0. 0. 0. criticality source point (positioned at X = 0, Y = 0, Z = 0)
print output all tables
mode n Track neutrons

VOL 12J 345.510821 J
└ skip till cell 12, volume = 345.510821 cm³

(a) (10 points) Next to each line in this sample input add a short explanation what that line represents.

Done on pg 5

(b) (5 points) Using so-called reverse engineering determine the dimensions, material properties (isotopic composition, enrichment, densities), and cross-section data used for this fuel pin. Specify corresponding units when appropriate.



(c) (15 points) Prepare the MCNP/SERPENT input for this case (using reflective boundary conditions) and determine:

1. k_inf.

   1.38485 ± 0.00142

2. The average neutron flux in the fuel, clad and moderator (the actual onegroup scalar flux!).

   UO_2 Z H_2O

   in UO_2 : 6.93540E-2 ± 0.0010 n/cm² /src n
   in Z : 6.95094E-02 ± 0.0010 n/cm² src n
   in H_2O : 6.94880E-02 ± 0.0009 n/cm² /src n

3. The average one-group absorption and fission rates in the fuel zone.

   Fission rate : 2.66208E-03 ± 0.0022 fission/s
   Absorption rate : 1.90964E-3 ± 0.0019 fisson/s

4. The neutron flux spectra in the fuel and water zones, separately, with a minimum of 20 energy groups (structure should be fine enough to see resonance capture regions). Plot the neutron spectra. Also determine the two-group

neutron flux in the fuel and water zones.

fuel: • Thermal: 6.96748 E-3 ± 0.0023
• Fast: 5.93452E-2 ± 0.004
water: • Thermal: 1.28299E-3 ± 0.0018
• Fast: 5.88045E-2 ± 0.0014

5. The spatial distribution of a two-group flux (fast and thermal) along the central line of the fuel pin. You might want to add more cylindrical zones in the fuel region, and more zones in the water region. Explain why additional spatial subdivision is needed?

The fact that the fast fluxs are greater than the thermal fluxs for both fuel and moderator is concerned. We need to expand the volume of each zones to make sure that neutrons get thermalized enough to cool down the system.

6. Comment of the spectral and energy distribution of neutron flux in the fuel pin. the thermal flux is smaller than the fast flux because the relatively shorter mean free path & small quantity moderator. Neutrons don't get to thermalize as much as to be a favorable condition.

3. In Lesson #9, an outline was created for a simple 1D MC transport program. In class a notebook was started to implement the MC algorithm, but it was not fully completed. Complete the code by: pg 5-6 of Doc

(a) (5 points) Adding in the algorithm to transport the particle if $s_b < s_c$ in the transport function code attached in the back

(b) (5 points) Writing the main controller program to use the functions and classes defined in the notebook and transport N number of particles.

code attached in the back

(c) (5 points) Outputting the flux tally in each cell in units of n/cm^2/src particle. Include the relative error for each flux.

with $N = 2000$

	Cell 1:	Cell 2	Cell 3	Cell 4
flux (in n/cm^2/src p)	0.103521	0.0259104	0.074073	0.17444
relative error	± 0.004829	± 0.0019297	± 0.00675	± 0.00286630

4. In lesson 14, a miniature city was created and exposed to neutrons from a fission weapon (nw_effects.in). The small scale of the city made it easy to get reasonable statistics. Now let's investigate something closer to reality:

   (a) (5 points) Add in a ground made of asphalt. Make the internal volume of the buildings 30% wood, 20% aluminum, 20% steel, and 30% air by mass (this should be one homogenized material).

   Updated in the input file (attached)

   (b) (5 points) Scale the geometry by a factor of 10 and determine the flux for a car located between each block (use the previous definition of a "car"). Report the FOM and flux spectrum with errors (plot) for each.

   Scale changed (Please refer to the input file attached)

   I run MCNP 21+ hrs but I could not get an output file at the end; my computer turned off due to low battery issue during a class time... ugh.....

   (c) (10 points) Use ADVANTG to create weight windows for the problem considered in part b). Run the MCNP with the ADVANTG weight windows with the same number of particles and report the FOM and flux spectrum with errors (plot) for each car.

   Pg 39 for Adv ops

   (d) (5 points) Comment on your findings regarding the FOM and uncertainties between

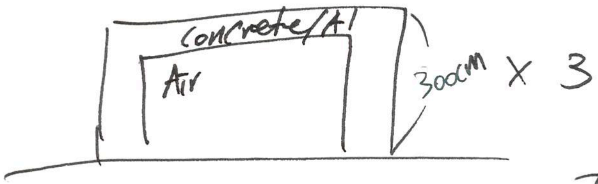
   Figure of Merit $= \frac{1}{R^2 t}$

the analog and ADVANTG variance reduction models.

I wasn't able to run file (my computer took hours to compute and didn't seem to end sooner...) Hypotetically, if the file was ran, I expect that ADVANTG would demonstrate to significantly increase the tally figure merit relative to an analong MCNP simulation. In class we were taught that ADVANTG provides a powerful efficient, and fully automated alternative to traditional methods for generating variance reduction parameters.

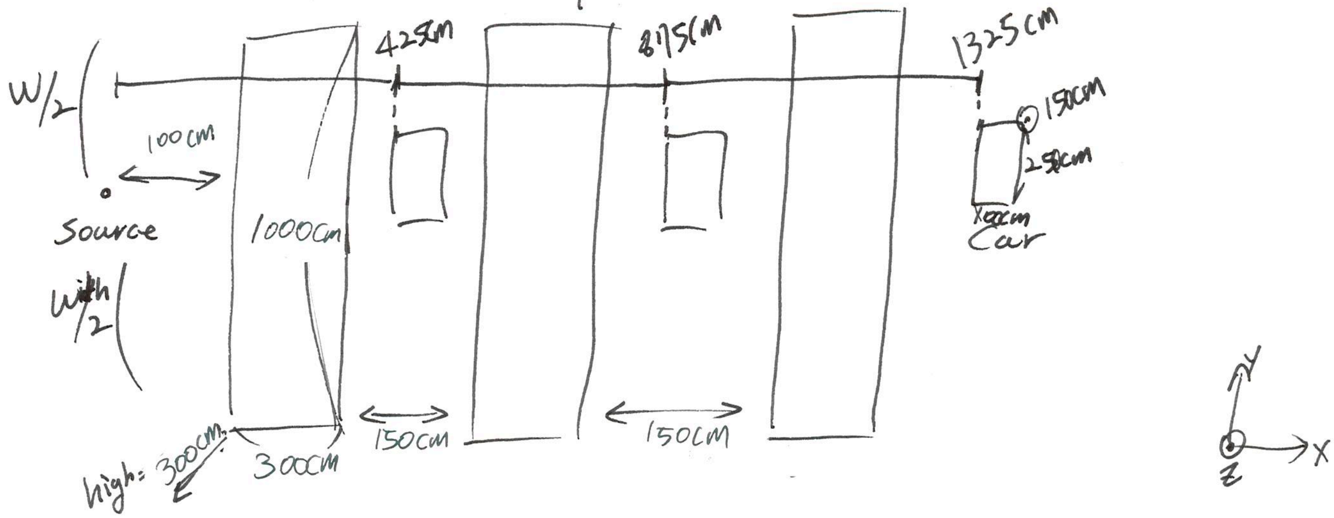
BONUS (5 points): submit your code by providing read/clone access to an online version control repository where your code is stored (e.g. github or bitbucket).

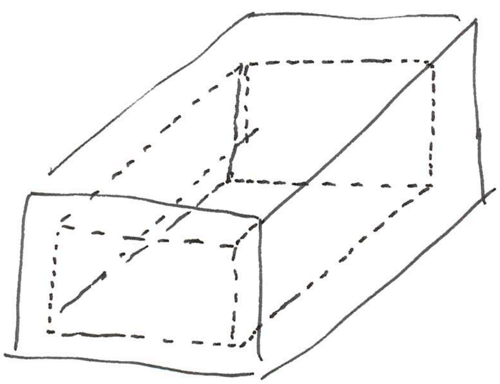
Prob 4 Hand Drawing



Scale by a factor of 10

Top View





PNNL Library (pg 22)

Change .txt → .i

Mcnp6 i =\name of the file

```
PWR Single Pin
c
*************************************************************************
******************
c  Cell Cards
c
*************************************************************************
******************
1 1 -10.41 -30 -10 20 imp:n=1     vol=1
2 1 -10.41 -31 30 -10 20 imp:n=1 vol=1
3 1 -10.41 -31 30 -10 20 imp:n=1 vol=1
4 1 -10.41 -31 30 -10 20 imp:n=1 vol=1
5 1 -10.41 -32 31 -10 20 imp:n=1 vol=1
6 1 -10.41 -33 32 -10 20 imp:n=1 vol=1
7 1 -10.41 -34 33 -10 20 imp:n=1 vol=1
8 1 -10.41 -35 34 -10 20 imp:n=1 vol=1
9 1 -10.41 -36 35 -10 20 imp:n=1 vol=1
10 1 -10.41 -37 36 -10 20 imp:n=1 vol=1
11 1 -10.41 -1 37 -10 20 imp:n=1 vol=1
12 0 1 -2 -10 20 imp:n=1 vol=1
13 3 -6.55 2 -3 -10 20 imp:n=1 vol=1
14 4 -0.7 3 -40 -7 8 -10 20 imp:n=1 vol=1
15 4 -0.7 40 -41 -7 8 -10 20 imp:n=1 vol=1
16 4 -0.7 41 -42 -7 8 -10 20 imp:n=1 vol=1
17 4 -0.7 42 -43 -7 8 -10 20 imp:n=1 vol=1
18 4 -0.7 43 -44 -7 8 -10 20 imp:n=1 vol=1
19 4 -0.7 44 -45 -7 8 -10 20 imp:n=1 vol=1
20 4 -0.7 45 -46 -7 8 -10 20 imp:n=1 vol=1
21 4 -0.7 46 -47 -7 8 -10 20 imp:n=1 vol=1
22 4 -0.7 47 -5 6 -7 8 -10 20 imp:n=1 vol=1
23 0 5:-6:7:-8:10:-20 imp:n=0 vol=1

c
*************************************************************************
******************
c Surfaces
c
*************************************************************************
******************
c Rods dimensions
1 cz 0.41
2 cz 0.42
3 cz 0.48
c Basic lattice cell
*5 px 0.63
*6 px -0.63
*7 py 0.63
*8 py -0.63
c Axial limits
*10 pz 200.0
*20 pz -200.0
c z x_sections for inner pin flux
30 cz 0.1
31 cz 0.2
```

```
32 cz 0.3
33 cz 0.4
34 cz 0.5
35 cz 0.6
36 cz 0.7
37 cz 0.8
c z x_sections for outter pin flux
40 cz 0.9
41 cz 1
42 cz 1.1
43 cz 1.2
44 cz 1.3
45 cz 1.4
46 cz 1.5
47 cz 1.6

c
*************************************************************************
******************
c Data Cards
c
*************************************************************************
******************
c Material 1
m1 8016.73c 2.0
     92235.73c 0.05
     92238.73c 0.95
c Material 4
m4 1001.71c 2
     8016.71c 1
mt4 lwtr.04t
c Material 3
m3 40000.58c 1.
c
*************************************************************************
******************
c Tallies
c
*************************************************************************
******************
fc4 Neutron flux, cladding, moderator
f4:n (1 2 3 4 5 6 7 8 9 10) 12 (14 15 16 17 18 19 20)
e4:n 1e-6 1. 20.
c
fc14 neutron flux spectra in the fuel and water, region 1
f14:n (1 2 3 4 5 6 7 8 9 10)
e14 0.625e-6 100ILOG 20
fc24 neutron flux spectra in the fuel and water, region 2
f24:n (1 2 3 4 5 6 7 8 9 10)(14 15 16 17 18 19 20)
e24 0.625e-6 100ILOG 20
fc34 neutron flux spectra in the fuel and water, region 3
f34:n (14 15 16 17 18 19 20)
e34 0.625e-6 100ILOG 20
c
```

```
fc44 Fuel absorption rate
f44:n (1 2 3 4 5 6 7 8 9 10)
fm44 (-1 1 (-2)) $(flux, atomic rho,absorption x_section)
c
fc54 Fuel fission rate
f54:n (1 2 3 4 5 6 7 8 9 10)
fm54 (-1 1 (-6)) $(flux, atomic rho, fission x_section)
c
c radial projection, neutron locations, 0.025ev for thermal neutrons
f64:n 1
e64 0 0.025 20
f74:n 2
e74 0 0.025 20
f84:n 3
e84 0 0.025 20
f94:n 4
e94 0 0.025 20
f104:n 5
e104 0 0.025 20
f114:n 10
e114 0 0.025 20
f124:n 11
e124 0 0.025 20
f134:n 12
e134 0 0.025 20
f144:n 13
e144 0 0.025 20
f154:n 14
e154 0 0.025 20
f164:n 15
e164 0 0.025 20
f174:n 16
e174 0 0.025 20
f184:n 17
e184 0 0.025 20
f194:n 18
e194 0 0.025 20
f204:n 19
e204 0 0.025 20
f214:n 20
e214 0 0.025 20
c
*************************************************************************
*****************
kcode 3000 1.00 50 250
ksrc 0. 0. 0.
print
mode n
```

Problem3 (a)

```
    elif s_c>s_b:
        particle.xLoc= particle.xLoc+s_b*particle.direction
        particle.tally[particle.cell]+=s_b
        particle.numMFP=particle.numMFP*(1-s_b*totXSec[particle.cell])
        if particle.direction>=1:
            particle.cell=particle.cell+1
        elif particle.direction<1:
            particle.cell=particle.cell-1
        else:
            print ('Particle is lost at the boundary')
        # Check if particle will leak from the slab
        if particle.xLoc==boundaries[0] and particle.direction<0:
            return
        elif particle.xLoc==boundaries[4] and particle.direction>0:
            return
        # Update s_b and s_c
        s_c=distToCol(particle.numMFP, totXSec[particle.cell])
        #print ('s_cBot',s_c)
        s_b=calcDistToBound(particle, bounds)
        #print ('S_b_bot',s_b)
    else:
        print('particle has not transported')
```

Problem3 (b)

```
N = 2000
particle.tally=np.array([0., 0., 0., 0.])
# This is here to reset the particle tally for a new run.
for i in range(N):
    # Init particle.

    particle.direction=1      # particle direction to +side(to the right)
    particle.energy=2450000   # given 2.45 MeV neutrons
    particle.cell=0           # start from the 0th boundary
    particle.xLoc=0          # Particle is at 0.
    # calculate s_b and s_c
    calcNumMFP(particle)    # Give an initial MFP
    s_c=distToCol(particle.numMFP,totXSec[0])
    s_b=calcDistToBound(particle,boundaries)
    #print ('sb1',s_b)
    #print ('sc1',s_c)
    transport(particle,s_b,s_c,xSecCDF, materials,boundaries,totXSec)
print ('The number of source particles is',N)
for i in range(0,len(materials)):
    print "flux for cell:", format(particle.tally[i]/N), "n/cm^2/src particle"
    print "relative errors in cell:", format(((particle.tally[i])**(1/2))
                                            /(particle.tally[i]))
```

Problem3 (c) Result

```
('The number of source particles is', 2000)
flux for cell: 0.103792454782 n/cm^2/src particle
relative errors in cell: 0.00481730585377
flux for cell: 0.268215515218 n/cm^2/src particle
relative errors in cell: 0.00186417254644
flux for cell: 0.0771969677252 n/cm^2/src particle
relative errors in cell: 0.00647693834012
flux for cell: 0.177397783004 n/cm^2/src particle
relative errors in cell: 0.0028185245133
```

```
This models a simple NW neutron burst and models a simple city-scape.
c
c Simple NW det scenario.  A NW is set off on the edge of the city.  To
start, let's create
c a scale model.  The key factors for the model:
c
c 1) Model three "city blocks" of long apartment buldings.  The buildings
are .3 m high,
c    .3m wide, and 1.0 m long separated by 0.15 m of air. The walls are
.03 m thick concrete.
c    (All dimensions scaled by a factor of 10 now)
c 2) The flux is tallied inside of a car modeled as a 0.1 wide, 0.15 m
tall, and 0.25 m long
c    aluminum box with 0.0025 m thick walls. The car is located on the
far side of the 3rd
c    block from the source.
c 3) Add a volume flux tally to the inside of the car.
c 4) The source is modeled as a point fission source (Watt spectrum) from
a HEU bomb.
c
c
*************************************************************************
******************
c  Cell Cards
c
*************************************************************************
******************
1 3 -2.2961415    -1 11     imp:n=1  $1st box
2 like 1 but trcl=(550 0 0)  $2nd box: duplicate the block1 and position
at x=550cm
3 like 1 but trcl=(1000 0 0) $3rd box: duplicate the block1 and position
at x=1000cm
4 2 -2.6989       -2 22     imp:n=1 $car1
5 like 4 but trcl=(875 0 0)  $2nd car: duplicate the car1 and position at
x=875cm
6 like 4 but trcl=(1325 0 0) $2nd car: duplicate the car1 and position at
x=875cm
7 4 -1.3           3 -111   imp:n=1 $Gound made of asphalt

c
*************************************************************************
******************
c Surfaces
c
*************************************************************************
******************
1   RPP 100 400      -500 500      0 300       $Block 1 outer
11  RPP 130 370      -470 470      30 270      $Block 1 inner
2   RPP 425 525      -125 125      0 150       $Car1 outer
22  RPP 427.5 522.5  -122.5 122.5  2.5 147.5   $Car1 inner
3   PZ  0                                      $Gound made of asphalt
111 RPP -150 15000 -550 550 -100 350           $Outer killzone
```

```
c
*************************************************************************
******************
c Data Cards
c
*************************************************************************
******************
c Concrete, Regular, rho =2.300
m1 1001  -0.013742 $Concrete H
     8016  -0.046056 $Concrete O
     11023 -0.001747 $Concrete Na
     13027 -0.001745 $Concrete Al
     14000 -0.016620 $Concrete Si
     20000 -0.001521 $Concrete Ca
     26000 -0.000347 $Concrete Fe
c Aluminum, rho =2.6989
m2 13027 -0.060238 $Al
c Material3: new material for building, total_rho=2.2961415
c 30% Wood(rho =0.192), 20%Al(rho =0.53978), 20%steel(rho =1.564),
30%air(rho =0.0003615)
m3 1001  -0.006842 $Wood H
     6000  -0.004785 $Wood C
     7014  -0.000041 $Wood N
     8016  -0.003089 $Wood O
     12000 -0.000009 $Wood Mg
     16000 -0.000018 $Wood S
     19000 -0.000006 $Wood K
     20000 -0.000006 $Wood Ca
     13027 -0.012047 $Al
     6000  -0.000392 $Steel C
     26000 -0.016781 $Steal Fe
c     6000  -0.000000 $Air C
     7014  -0.000012 $Air N
     8016  -0.000003 $Air O
c     18000 -0.000000 $Air Ar
c Asphalt, rho=1.3
m4 1001    -0.080564 $Asphalt H
     6000  -0.055277 $Asphalt C
     7014  -0.000338 $Asphalt N
     8016  -0.000198 $Asphalt O
     16000 -0.000920 $Asphalt S
     23000 -0.000006 $Asphalt V
c     28000 -0.000000 $Asphalt Ni
c
*************************************************************************
******************
c Physics
c
*************************************************************************
******************
SDEF PAR=n POS 0 500 150 ERG=d4
SP4 -3 0.988 2.249
```

```
c
**************************************************************************
******************
c Tallies
c
**************************************************************************
******************
fc4 Flux in cars
f4:n 4 5 6
e4 1e-8 25ILOG 20
PRINT
MODE n
NPS 1E12
```