How to use the MCNP

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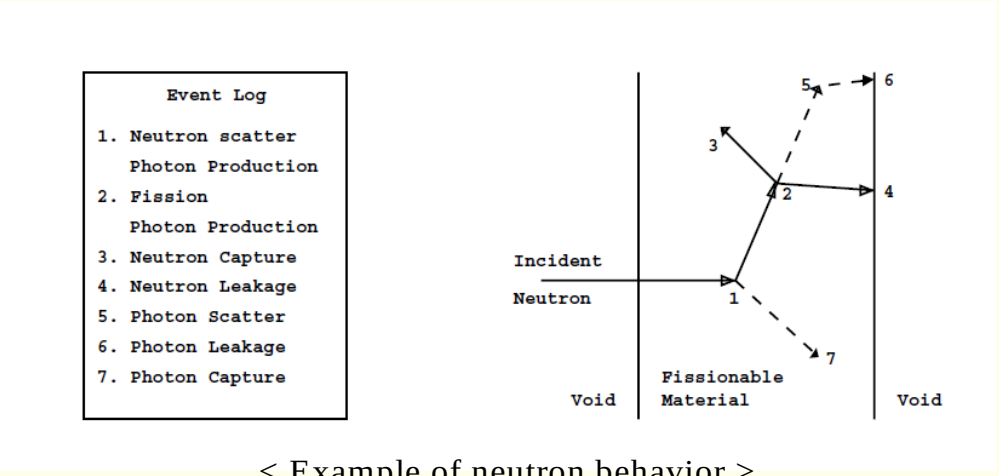
Contents

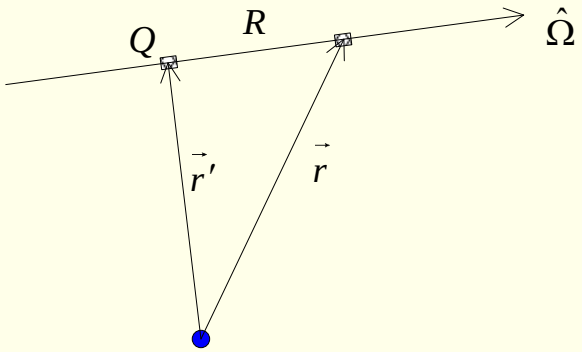
Code overview

- MCNP : a general purpose Monte Carlo N-Particle code by Los Alamos Nat'l Lab. (ver. 4C2)
- Monte Carlo simulates individual particles and recording results (tallies) of their average behavior.



< Example of neutron behavior >

- Averaging particle behavior represents solving integrated Transport eq.



$$R = \left|\vec{r} - \vec{r}'\right| [\text{cm}]$$

$$\varphi_g(\vec{r}, \hat{\Omega}) = \int_s \frac{Q_g(\vec{r}')}{4\pi R^2} e^{-\Sigma_t R} d\vec{r}'$$

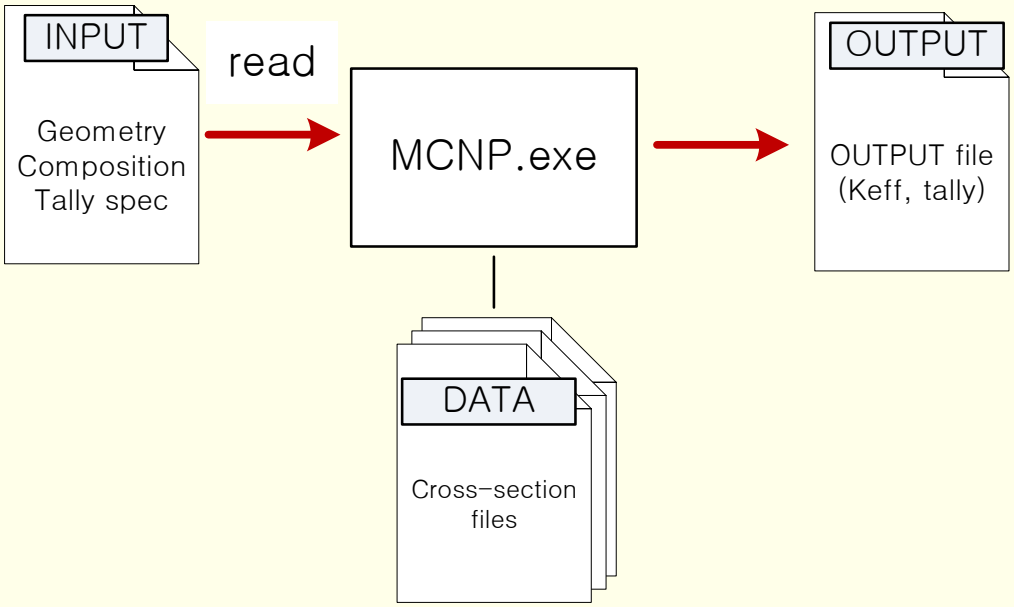
SNURPL

Code overview

- Consists of three part : Input, Nuclear-data, and Output

- Input file contains

1. geometry specification
2. description of materials
3. location and characteristics of (neutron) sources
4. types of answers or tallies desired



Input overview

- Compose of 4 cards
- The message block ends with a blank line delimiter
- The columns of line : 1~80
- Comment : c
- Termination of data entry on a line : $

Message Block (Optional)
Blank Line Delimiter (Optional)
One Line Problem ***Title Card***
Cell Cards
…
…
Blank Line Delimiter
Surface Cards
…
…
Blank Line Delimiter
Data Cards
…
…
Blank Line Terminator (Optional)

Input overview

- Facilitation of Input Card Preparation

- The FIDO (Flexible Input Deterministic Output) system

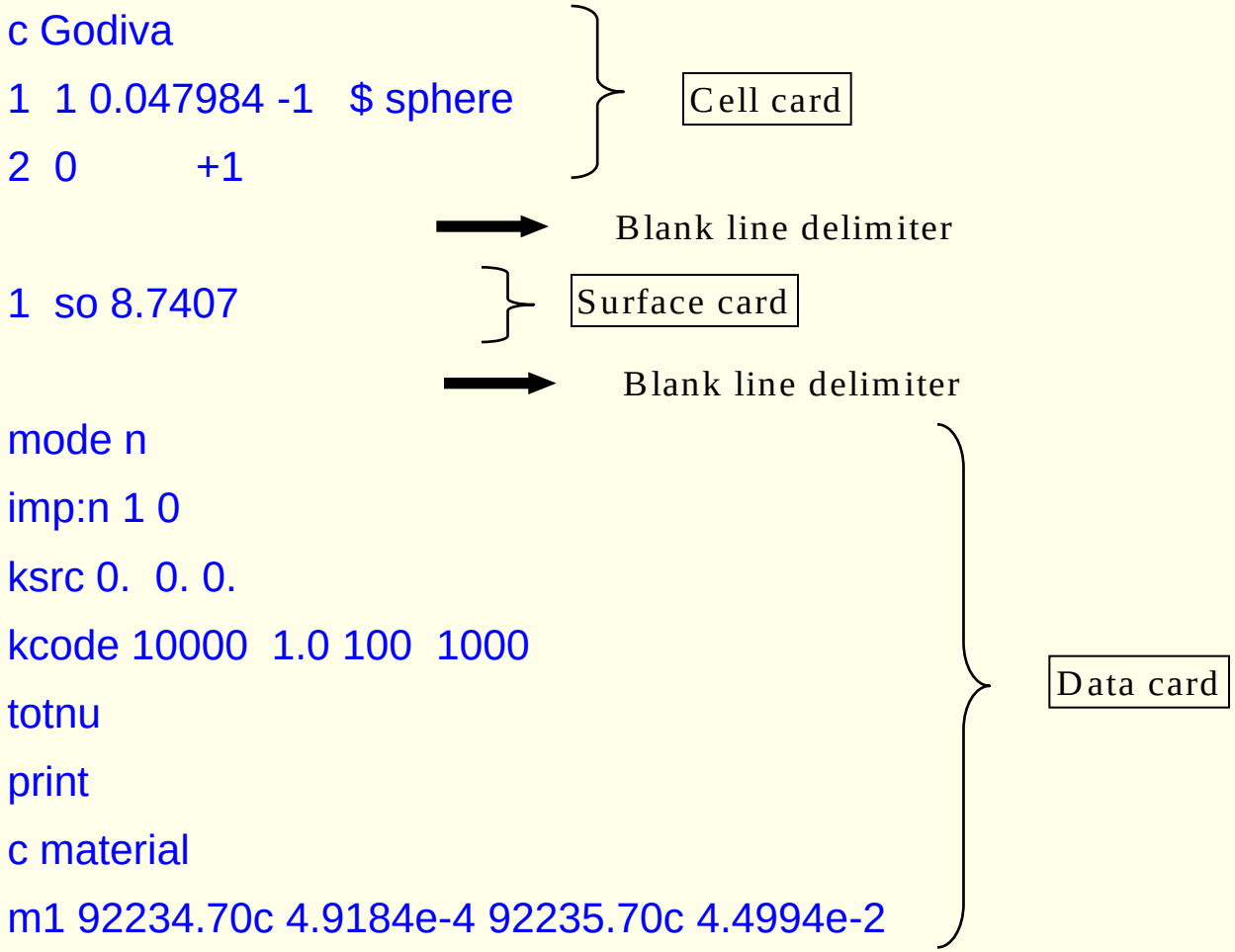
nR : Repeat the preceding entry n times One Line
2 4R : 2 2 2 2 2

nl : Insert n linear interpolates between the entries preceding and following
1.5 2I 3.0 : 1.5 2.0 2.5 3.0

xM : Multiply the previous entry by x
1 1 2M 2M 2M 4M 2M = 1 1 2 4 8 32 64

nJ : Jump over n items and use the default value
1 2 J 0.5

Input Example



Cell card (1)

- Cell cards
  - Cell cards describe the cells - geometry, material, density, cell parameter.
- Form

cell number	cell material number	cell material density	geometry	cell parameter (optional)

- cell number

cell number	cell material number	cell material density	geometry	cell parameter (optional)

The first entry and must begin in the first five columns

$1 \leq$ cell number ≤ 99999

- Example

1) 1 1 6.72E-2 −7 9 −10 u=1 imp:n=1

2) 4 0 1 :-2 :3 :-4 :5 :-6 :-9 :10

3) 99998 3 6.63E-02 -1 2 -3 4 -5 6 12 imp:n=1 vol=1.336

Cell card

- cell material number

cell number	cell material number	cell material density	geometry	cell parameter (optional)

  - 0 if the cell is a void.
  - material number if the cell is not a void.
  - Material number is corresponding to the number, which is specified on the M*m* card. (Data cards)

```
1   21 -10.0 -7 9 -10
2   22 0.0429 7 -8 9 -10
3   23 -0.9961 -1 2 -3 4 -5 6 8 9 -10
4    0        1 :-2 :3 :-4 :5 :-6 :-9 :10

*1       p 1.732050807569 -1 0  1.611854897735
*2       p 1.732050807569 -1 0 -1.611854897735
*3       p 1.732050807569  1 0  1.611854897735
*4       p 1.732050807569  1 0 -1.611854897735
*5      py 0.8059274488677
*6      py -0.8059274488677
 7      cz 0.44
 8      cz 0.5
*9      pz -0.75
*10     pz 0.75

mode  n
kcode 10000 1.0 50 1050 20000 0 13000
m21  92235.60c  -0.02468  92238.60c -0.8568  8016.60c  -0.1185 $Fuel Region
m22  40000.60c      1                                  $ Zr
m23  1001.60c -0.1119   8016.60c -0.8881                $ H2o
```

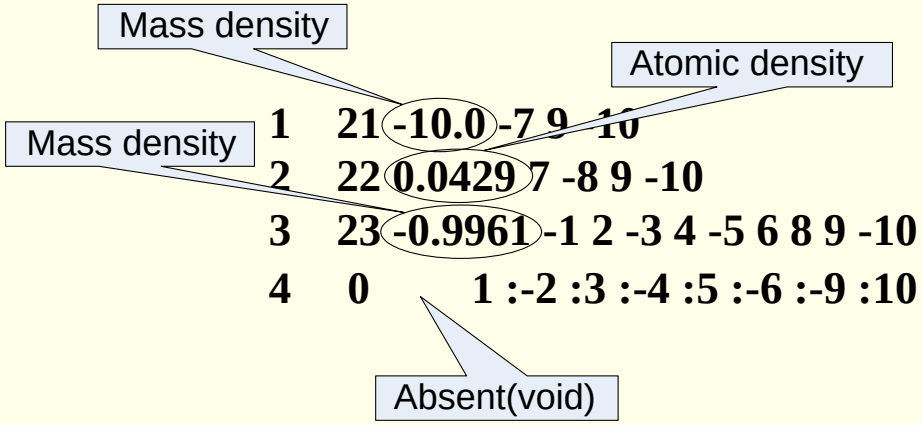
Cell card

- cell material density

cell number	cell material number	cell material density	geometry	cell parameter (optional)

  - Absent if the cell is a void
  - postive value : atomic density $\left(\text{unit} : 10^{24}\ \text{atoms/cm}^3\right) or \left(1\ \text{atoms/barn} \cdot \text{cm}\right)$

    negative value : mass density $\left(\text{unit:}\ \text{g/cm}^3\right)$



Cell card

- geometry

cell number	cell material number	cell material density	**geometry**	cell parameter (optional)

– Consist of signed surface number and boolean operator

– Surface number is corresponding to the number that is in the surface cards

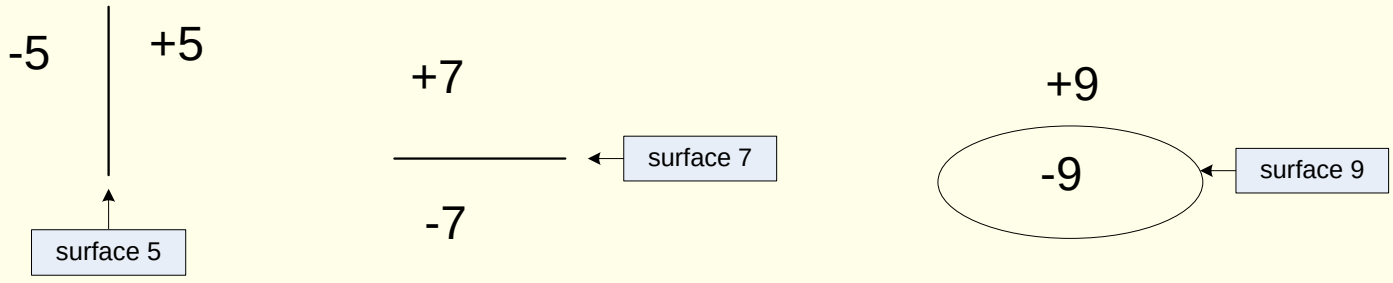
– A surface divide space into positive & negative sides

– The regions are combined by boolean operator

1) intersection → no symbol

2) union → : (colon)

3) complement → # (sharp)



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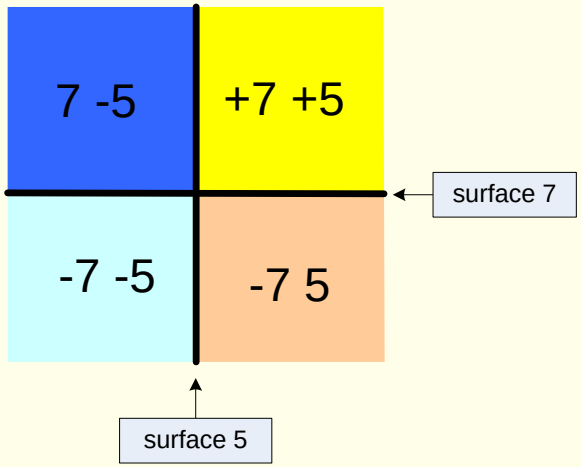
Cell card

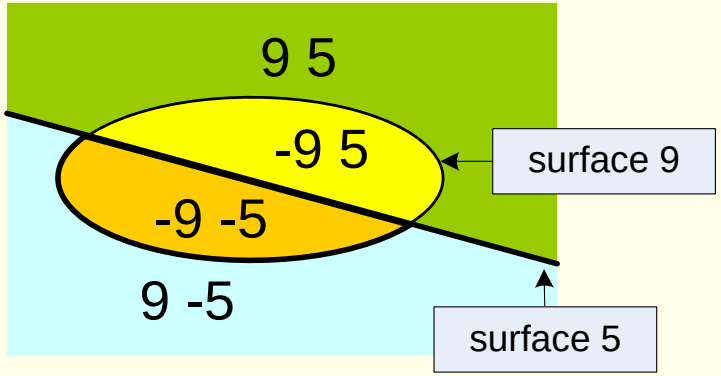
- geometry

cell number	cell material number	cell material density	geometry	cell parameter (optional)

1) intersection → no symbol




```
10  0    7   5
11  0    7  -5
12  0   -7   5
13  0   -7  -5
```




```
1  0   -9  5
2  0   -9  -5
5  0    9  5
7  0    9  -5
```

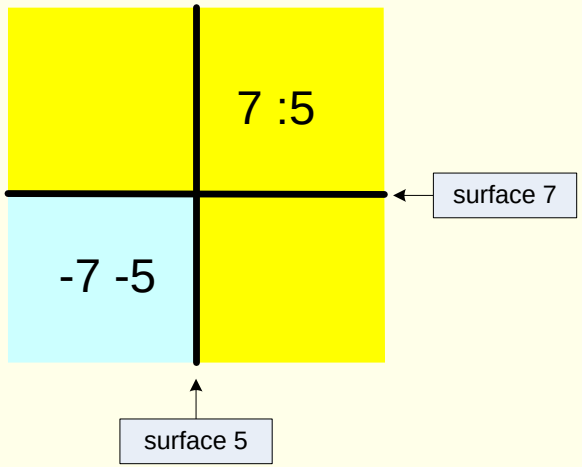

SNURPL

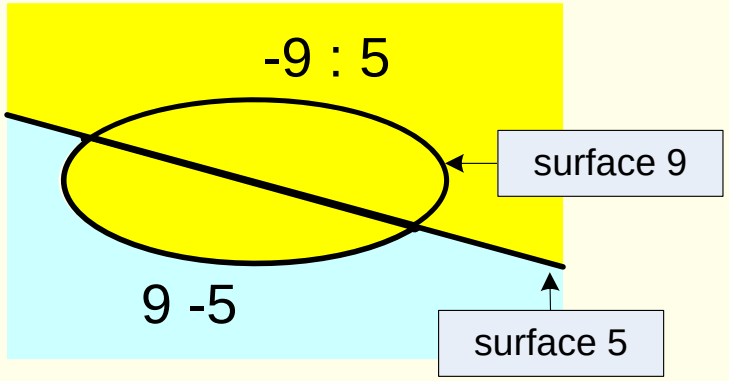
Cell card

- geometry

cell number	cell material number	cell material density	geometry	cell parameter (optional)

2) union → : (colon)






```
1  0    7 : 5
7  0   – 7  – 5
```

```
5  0   – 9 : 5
9  0    9   – 5
```


SNURPL

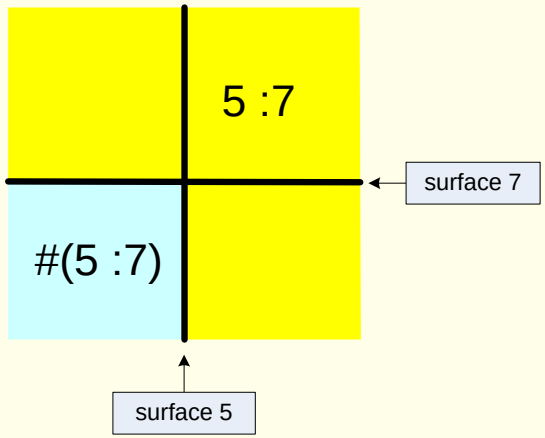
Cell card

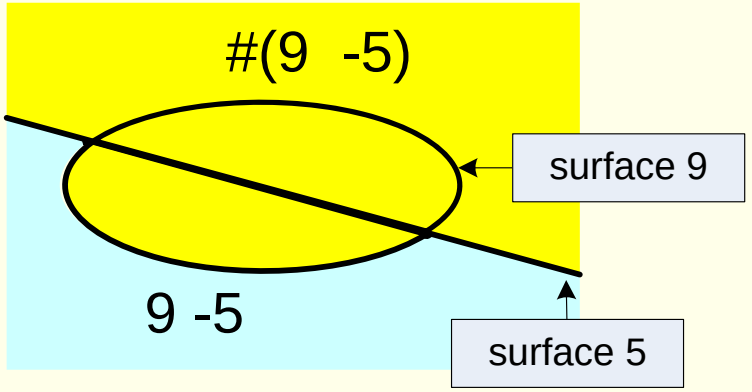
- geometry

cell number	cell material number	cell material density	geometry	cell parameter (optional)

3) complement → # (sharp)

The number immediately after a #, without parentheses, is interpreted as a cell number.






```
10  0   5 7          =   10  0   5 7
20  0  #(5 7)           20  0  #1
```

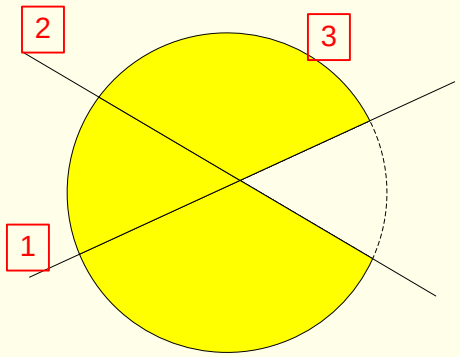
```
1  0   9 -5          =   1  0   9 -5
2  0  #(9 -5)           2  0  #1
```


SNURPL

Cell card

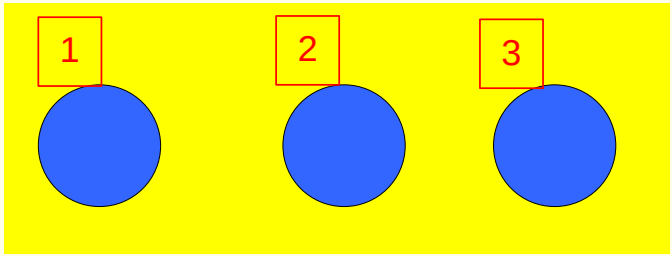
- geometry

cell number	cell material number	cell material density	**geometry**	cell parameter (optional)



Yellow part

$$1\ 0\ -3\ \ (1:-2)$$
$$\Leftrightarrow\ 1\ 0\ (-3\ 1):(-1\ -2\ -3)$$
$$\Leftrightarrow\ 1\ 0\ (-3\ 1)\ :(-3\ \ -2)$$



Blue part $\quad 1\ 0\ -1\ :\ -2\ :\ -3$

Yellow part $\quad 2\ 0\ \#1$

$$\Leftrightarrow\ 2\ 0\ 1\ 2\ 3$$

SNURPL

Cell card

- cell parameter

cell number	cell material number	cell material density	geometry	cell parameter (optional)

  - Form : keyword = value
  - cell parameter related with Data cards.
  - Allowed keyword : IMP, VOL, PWT, FCL, WWN, DXC, NONU, PD, TMP U, TRCL, LAT, FILL
  - example

```
8    3  6.63E-02   -1 2 -3 4 -5 6 12  imp:n=1 vol=1.336
```

Surface card

- Form

surface number	eqaution mnemonic	entry

– surface number

$1 \leq$ surface number ≤ 99999

the surface number with asterisk ('*'): for a reflecting surface

– equation mnemonic

– entry

The surface type, equations, mnemonics are given.

To specify a surface by this method,

find the surface and determine the coefficients for the equation.

– example

```
*1      p  1.7  -1  0  1.6
 7     cz  0.6225
21     so  0.0175
```

Surface card

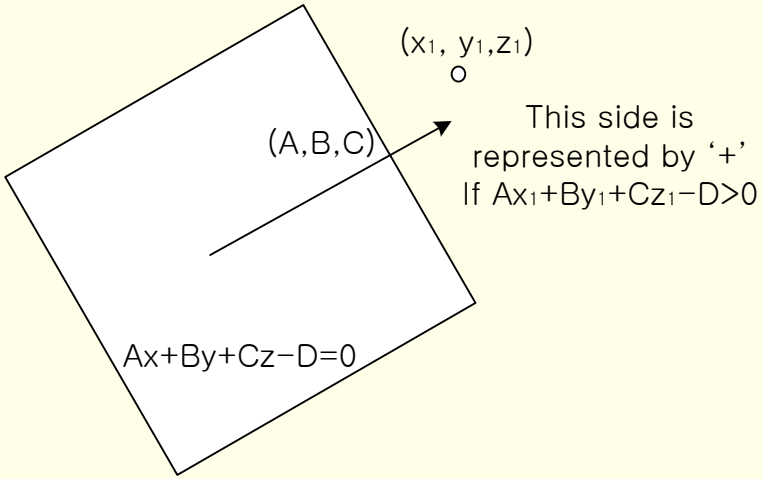
Mnemonic	Type	Description	Equation	Card Entries
P	Plane	General	$Ax + By + Cz - D = 0$	ABCD
PX		Normal to X-axis	$x - D = 0$	D
PY		Normal to Y-axis	$y - D = 0$	D
PZ		Normal to Z-axis	$z - D = 0$	D
SO	Sphere	Centered at Origin	$x^2 + y^2 + z^2 - R^2 = 0$	R
S		General	$(x-\bar{x})^2 + (y-\bar{y})^2 + (z-\bar{z})^2 - R^2 = 0$	$\bar{x}\ \bar{y}\ \bar{z}\ R$
SX		Centered on X-axis	$(x-\bar{x})^2 + y^2 + z^2 - R^2 = 0$	$\bar{x}\ R$
SY		Centered on Y-axis	$x^2 + (y-\bar{y})^2 + z^2 - R^2 = 0$	$\bar{y}\ R$
SZ		Centered on Z-axis	$y^2 + y^2 + (z-\bar{z})^2 - R^2 = 0$	$\bar{z}\ R$
C/X	Cylinder	Parallel to X-axis	$(y-\bar{y})^2 + (z-\bar{z})^2 - R^2 = 0$	$\bar{y}\ \bar{z}\ R$
C/Y		Parallel to Y-axis	$(x-\bar{x})^2 + (z-\bar{z})^2 - R^2 = 0$	$\bar{x}\ \bar{z}\ R$
C/Z		Parallel to Z-axis	$(x-\bar{x})^2 + (y-\bar{y})^2 - R^2 = 0$	$\bar{x}\ \bar{y}\ R$
CX		On X-axis	$y^2 + z^2 - R^2 = 0$	R
CY		On Y-axis	$x^2 + z^2 - R^2 = 0$	R
CZ		On Z-axis	$x^2 + y^2 - R^2 = 0$	R

Surface card

K/X	Cone	Parallel to X-axis	$\sqrt{(y-\bar{y})^2+(z-\bar{z})^2}-t(x-\bar{x}) = 0$	$\bar{x}\ \bar{y}\ \bar{z}\ t^2 \pm 1$
K/Y		Parallel to Y-axis	$\sqrt{(x-\bar{x})^2+(z-\bar{z})^2}-t(y-\bar{y}) = 0$	$\bar{x}\ \bar{y}\ \bar{z}\ t^2 \pm 1$
K/Z		Parallel to Z-axis	$\sqrt{(x-\bar{x})^2+(y-\bar{y})^2}-t(z-\bar{z}) = 0$	$\bar{x}\ \bar{y}\ \bar{z}\ t^2 \pm 1$
KX		On X-axis	$\sqrt{y^2+z^2}-t(x-\bar{x}) = 0$	$\bar{x}\ t^2 \pm 1$
KY		On Y-axis	$\sqrt{x^2+z^2}-t(y-\bar{y}) = 0$	$\bar{y}\ t^2 \pm 1$
KZ		On Z-axis	$\sqrt{x^2+y^2}-t(z-\bar{z}) = 0$	$\bar{z}\ t^2 \pm 1$ ± 1 used only for 1 sheet cone
SQ	Ellipsoid Hyperboloid Paraboloid	Axis parallel to X-, Y-, or Z-axis	$A(x-\bar{x})^2+B(y-\bar{y})^2+C(z-\bar{z})^2$ $+2D(x-\bar{x})+2E(y-\bar{y})$ $+2F(z-\bar{z})+G = 0$	A B C D E F G $\bar{x}$ $\bar{y}$ $\bar{z}$
GQ	Cylinder Cone Ellipsoid Hyperboloid Paraboloid	Axes not parallel to X-, Y-, or Z-axis	$Ax^2+By^2+Cz^2+Dxy+Eyz$ $+Fzx+Gx+Hy+Jz+K = 0$	A B C D E F G H J K
TX	Elliptical or circular torus. Axis is parallel to X-, Y-, or Z-axis	$(x-\bar{x})^2/B^2+(\sqrt{(y-\bar{y})^2+(z-\bar{z})^2}-A)^2/C^2-1 = 0$		$\bar{x}\ \bar{y}\ \bar{z}$ A B C
TY		$(y-\bar{y})^2/B^2+(\sqrt{(x-\bar{x})^2+(z-\bar{z})^2}-A)^2/C^2-1 = 0$		$\bar{x}\ \bar{y}\ \bar{z}$ A B C
TZ		$(z-\bar{z})^2/B^2+(\sqrt{(x-\bar{x})^2+(y-\bar{y})^2}-A)^2/C^2-1 = 0$		$\bar{x}\ \bar{y}\ \bar{z}$ A B C
XYZP		Surfaces defined by points		See pages 3–15 and 3–17

Surface card

- defining the sign of surface
  - substitute a point ($\vec{v}$) which is not on the surface into the equation $f(x) = 0$
  - the value $f(\vec{v})$ represents the sign of surface regard to the point



Data card

- Categories

1) Problem type Card
2) Geometry Cards
3) Variance Reduction
4) Source Specification
5) Tally Specification
6) Material Specification
7) Energy and Thermal Treatment Specification
8) Problem Cutoff Cards
9) User Data Arrays
10) Peripheral Cards

Problem type Card

– Form

MODE | particle designator

particle designator

N : neutron transport (default)

P : photon transport

E : electron transport

N P : neutron and neutron-induced photon transport

P E : photon and electron transport

N P E : neutron, neutron-induced photon and electron transport

– example

```
MODE N
```

Geometry Cards

a) VOL (optional) : Cell volume

b) AREA (optional) : Surface Area

c) U (optional) : Universe

d) TRCL (optional) : Cell Transformation

e) LAT (optional) : Lattice

f) FILL (optional) : Fill

g) TRn (optional) : Coordinate Transformation

Geometry Cards

- VOL : Cell volume

  Form : VOL $x_1\ x_2\ \cdots\ x_i$

  or VOL NO $x_1\ x_2\ \cdots\ x_i$

  – can write in cell cards (cell parameter)

  – MCNP attempts to calculate the volume of all cells unless "NO" appears on the VOL cards.

  – If no value is entered for a cell on the VOL card, the calculated volume is used.

  – MCNP calculate the volume basically.
  However MCNP cannot calculate the volume of complicated region.

  – Volumes of cells are required for tallies.
  So we write the volume manually, when MCNP cannot calculate it.

Geometry Cards

- Repeated Structures Cards

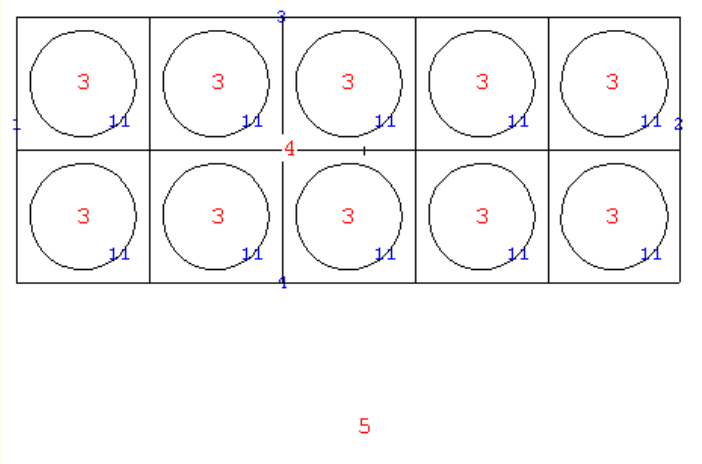
U : Universe

FILL : Fill

– example

```
1 0   1 -2 -3 4 -5 6  fill=1
2 0   -7 1 -3 8      u=1 fill=2 lat=1
3 0  -11   u=2
4 0   11   u=2
5 0  -1:2:3:-4:5:-6

1  px   0
2  px   50
3  py   10
4  py   -10
5  pz    5
6  pz   -5
7  px   10
8  py    0
11 s   5 5 0 4
```



SNURPL

Variance Reduction

a) IMP (required) : Cell importances

b) ESPLT (optional) : Energy splitting and roulette

c) PWT (optional) : Photon production weights

d) EXT (optional) : Exponential transform

e) VECT (optional) : Vector input

f) BBREM (optional) : Bremsstrahlung biasing

Variance Reduction

- Cell importances
  - Form IMP:n x_1 x_2 $\cdots$ x_I

    n : N for neutrons, P for photons, E for electrons

    x_i : importance for cell i

    I : number of cells in the problem
  - The importance of a cell is used to terminate the particle's history if the importance is zeros.
  - can write in the cell parameter in the cell cards
  - example

```
1 1 0.5 1 -2 -3 4 -5 6 fill=1
2 1 0.5 -7 1 -3 8      u=1 fill=2 lat=1
3 4 0.5 -11   u=2
4 1 0.5  11   u=2
5 0 -1:2:3:-4:5:-6

1 px 0
2 px 50
.
.
11 s 5 5 0 4

mode  n
kcode 10000 1.000000 100 500
imp:n 1 1 1 1 0
m1    8016.60c      0.035432  $MAT
     6000.60c      0.011811 92235.60c     0.0024754
m4    6000.60c      0.048081  $MAT
sdef pos=20 0 0 cel=1
```

```
1 1 0.5 1 -2 -3 4 -5 6 fill=1 imp:n=1
2 1 0.5 -7 1 -3 8      u=1 fill=2 lat=1 imp:n=1
3 4 0.5 -11   u=2 imp:n=1
4 1 0.5  11   u=2 imp:n=1
5 0 -1:2:3:-4:5:-6 imp:n=0

1 px 0
2 px 50
.
.
11 s 5 5 0 4

mode  n
kcode 10000 1.000000 100 500
m1    8016.60c      0.035432  $MAT
     6000.60c      0.011811 92235.60c     0.0024754
m4    6000.60c      0.048081  $MAT
sdef pos=20 0 0 cel=1
```

Source Specification

a) SDEF : General source

b) SIn : Source information

c) SPn : Source probability

d) SBn : Source bias

e) DSn : Dependent source

f) SCn : Source comment

g) KCODE : Criticality source

h) KSRC : Source points

Source Specification

- Criticality source

Form : KCODE NSRCK RKK IKZ KCT MSRK KNRM MRKP KC8 ALPHA

NSRCK = number of source histories per cycle

RKK = initial guess for keff

IKZ = number of cycles to be skipped before beginning tally accumulation

KCT = number of cycles to be done

MSRK = number of source points to allocate storage for

KNRM = normalize tallies by 0=weight / 1=histories

MRKP = maximum number of cycle values on MCTAL or RUNTPE

KC8 = summary and tally information averaged over:

0 = all cycles

1 = active cycles only.

Defaults: NSRCK=1000; RKK=1.0; IKZ=30; KCT=IKZ+100; MSRK=4500 or 2* NSRCK; KNRM=0; MRKP=6500;KC8=1;

The KCODE card specifies the MCNP criticality source that is used for determining

Source Specification

- Source points

Form : KSRC $x_1\ y_1\ z_1 \quad x_2\ y_2\ z_2 \quad \cdots$

$x_i\ y_i\ z_i$: location of initial source points

At least one point must be in a cell containing fissile material
and points must be away from cell boundaries.

The energy of each particle in the initial source is sampled
from a Watt fission spectrum hardwired into MCNP

If this card is absent, an SRCTP source file or SDEF card must be supplied
to provide initial source points for a criticality calculation

Source Specification

- General source

SDEF : General source

SIn : Source information

SPn : Source probability

POS : Reference point for position sampling

CEL : Starting cell number

ERG : Starting energy

WGT : Starting weight

TME : Time (Shakes)

RAD : Radial distance of the position

SUR : Surface

VEC : Reference vector for DIR

Tally Specification

a) Fna : Tally type

b) FCn : Tally comment

c) En : Tally energies

d) FQn : Tally print hierarchy

e) FMn : Tally multiplier

f) DEn/DFn : Dose energy/Dose function

g) EMn : Energy multiplier

h) SDn : Segmented volume/area

Tally Specification

- Tally type

Fna : Tally type

Mnemonic	Tally Description	Fn units	*Fn units
F1:N or F1:P or F1:E	Current integrated over a surface	particles	MeV
F2:N or F2:P or F2:E	Flux averaged over a surface	particles/cm^2	MeV/cm^2
F4:N or F4:P or F4:E	Flux averaged over a cell	particles/cm^2	MeV/cm^2
F5a:N or F5a:P	Flux at a point or ring detector	particles/cm^2	MeV/cm^2
F6:N or F6:N,P or F6:P	Energy deposition averaged over a cell	MeV/g	jerks/g
F7:N	Fission energy deposition averaged over a cell	MeV/g	jerks/g
F8:P or F8:E or F8:P,E	Energy distribution of pulses created in a detector	pulses	MeV
+F8:E	Charge deposition	charge	N/A

Tally Specification

- Surface and Cell tallies (tally types 1, 2, 4, 6 and 7)
  - Form

$$Fn\ :pl\ \ S_1\ S_2\ \cdots\ S_k$$

n = tally number

pl = N or P or N,P or E

S_i = problem number of surface or cell for tallying

  - example

```
f104:n   1
fc104   Flux averaged over a cell 1
fm104    1
```

```
f104:n   ( 1 2 3 )
fc104   Flux averaged over a cell 1 2 3
fm104    1
```

Tally Specification

- Tally energies
  - Form

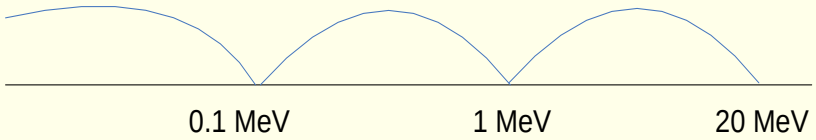
$$En \quad E_1 \; E_2 \; \cdots \; E_k$$

n = tally number.

E_i = upper bound (in MeV) of the i-th energy bin for tally n.

  - the order of increasing magnitude
  - An E0 card can be used to set up a default energy bin structure for all tallies.
  - A specific En card will override the default structure for tally n.
  - example

```
E11 .1 1 20
```



Material Specification

a) Mm : Material

b) DRXS : Discrete reaction

c) TOTNU : Total fission $\bar{\nu}$

d) NONU : Fission turnoff

e) AWTAB : Atomic weight

f) XSn : Cross-section files

g) VOID : Negates materials

h) MGOPT : Multigroup card

Material Specification

- Material
  - Form

$$M_n \; ZAID_1 \; fraction_1 \; ZAID_2 \; fraction_2$$

n : corresponds to the material number on the cell cards

$ZAID_i$ = either a full ZZZAAA.nnX

where ZZZ is the atomic number, AAA is the atomic mass,

nn is the library identifier and X is the class of data

$fraction_i$ = atomic fraction (or weight fraction if entered as a negativenumber).

  - example

```
m1    92235.60c 1.649991E-02
      92238.60c 3.168330E-01
      8016.60c  6.666671E-01    $Fuel Region
m2   40000.60c      1              $ Zr
m3   1001.60c 6.666667E-01   8016.60c 3.333333E-01     $ H2o
```

Energy and Thermal Treatment Specification

a) PHYS : Energy physics cutoff

b) TMP : Free-gas thermal temperature

c) THTME : Thermal times

d) MTm : S(α,β) material

Energy and Thermal Treatment Specification

- S(α,β) material
  - Form

    MTm X_1 X_2 $\cdots$

    m : material number on Mm card

    X_i : S(α,β) identifier corresponding to a particular component on the Mm card

  - When thermal treatments are needed, this card is used.

  - example

```
m3   1001.60c 6.666667E-01   8016.60c 3.333333E-01      $ H2o
mt3 lwtr.01t
```

Execution

- basic execution command

  mcnp i=[input file name] o=[output file name]

  – input file name and output file name don't have to over 8 character.

  – The treatment for ascii file of input file follow the treatment in the LINUX. (LF[Line Feed] only) so input file must not include the CR [Carriage Return].

  – when there is same output file name, MCNP generates output file with another name.

  – "xsdir" file must need to determine the xs data directory.

- plot geometry

  mcnp ip i=[input file name]

Contents

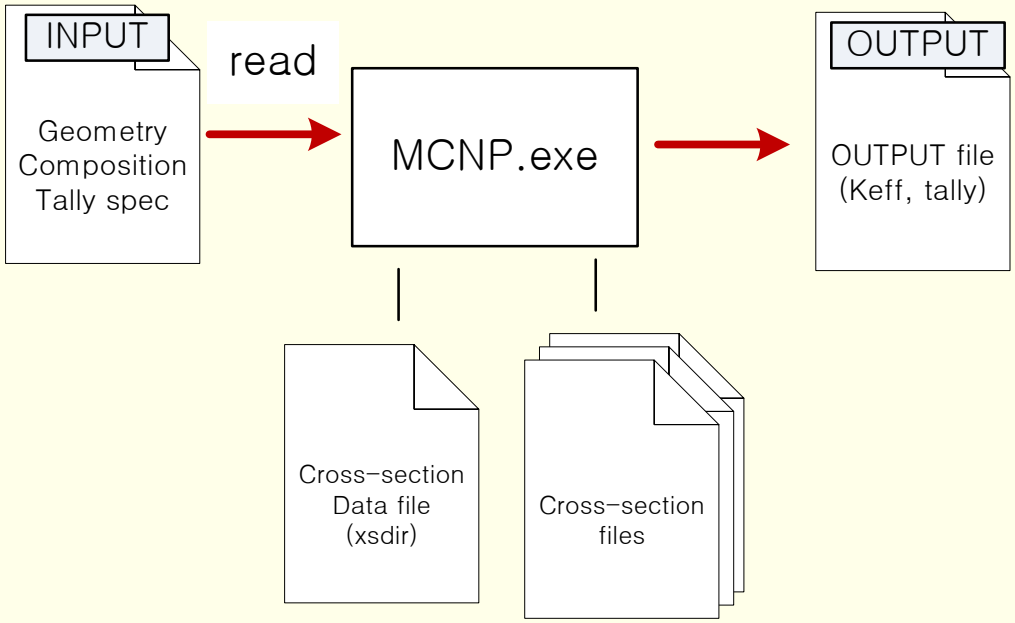
- MCNP execution
- C5G7 benchmark
- Source definition
- Tally definition
- Practice

Required files for MCNP

- Executable file : mcnp.exe
- Cross-section data file : xsdir

This data file specifies the location and name of xsec files of each material ID

- cross-section files : cross-section files located in ./xs/* for default
- input file



Execution

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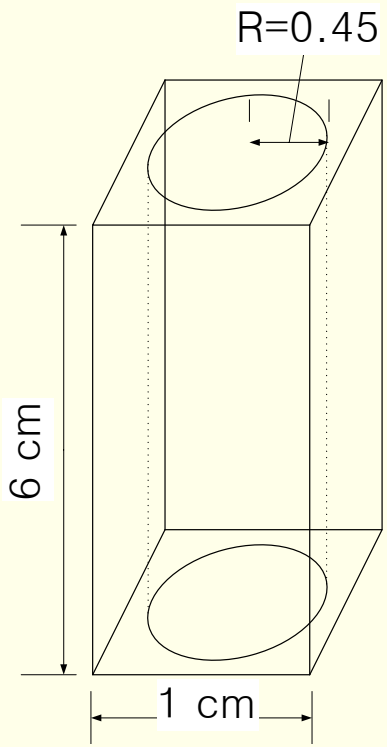
  – when there is same output file name, MCNP generates output file with another name.

  – "xsdir" file must need to determine the xs data directory.

Test Execution

- test file in your practice package contains a simple arbitrary pincell.
- run mcnp for test input. use the 'test' file in your mcnp folder as the input file.
- open the output file and confirm that estimated keff is near 0.63

  use searching option for finding "keff ='



- cylindrical pincell is consisted of low enriched UO2
- Pincell is barreled in water moderator.

  See that thermal treatment xsec is specified in data card.

C5G7 benchmark problem

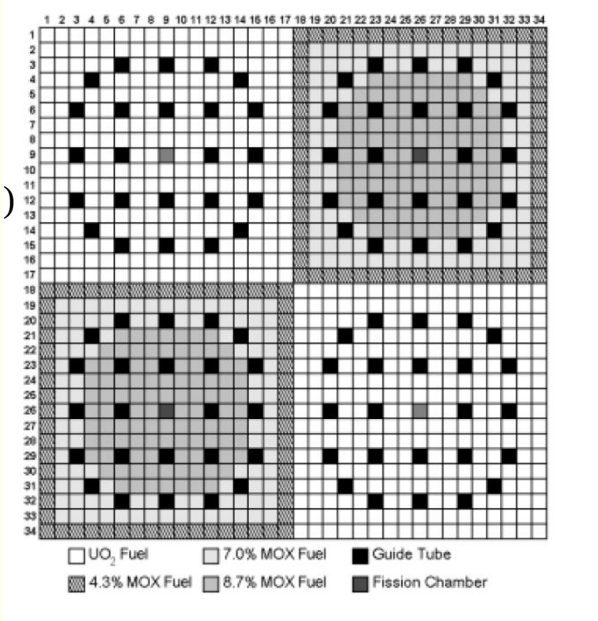
- try C5 Mox fuel assembly problem.

< Before calculation >

- Open the 'c5g7' input file and figure out the configuration of geometry.
- See that flux tally is specified in data card.
- As 7-group xsec is used for the materials,
  copy the cross section files into the folder mcnp file located.
  the xsec files for C5G7 is under C5G7 sub-folder. (mastab1~7)
- A documentation for C5G7 is prepared in pdf format.

< After calculation >

- Confirm that keff is near the calculation reference.
  (see page 4 in document)
- find the output for flux tally.



< Quarter core configuration of C5G7 >

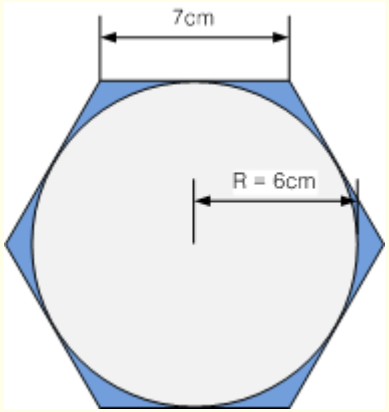
Practice

- Calculation keff

  2D pincell dimension : radius 6cm

  hexagonal moderator material : water ($1 g/cm^3$)

  critical calculation in all reflective boundary



Source Specification

- Four source type

  SDEF : General source

  KCODE : Criticality source

  SSR : Surface source

  SOURCE, SRCDX : User-supplied source

Source Specification

- SI card

  SIn option values

  options : Blank

  H - histogram bin boundaries (default)

  L - Discrete values

  A - points where probability density distribution is defined

  S : distribution numbers

- *Ex*)

  SDEF ERG=D1

  SI1 H 0.01 0.1 1 5 14

  SDEF POS=D5

  SI5 L 0 0 0 0 5 0

- SP card

  SPn option values

  options D - bin probabilities

  C - cumulative bin probability

  V - probability proportional to volume

- *Ex*)

  SDEF ERG=D1

  SI1 H 0.01 0.1 1 5 14

  SP1 D 0 2 1 1 6

  or SP1 C 0 0.2 0.3 0.4 1

  SDEF ERG=D5

  SI5 -2 1 $ maxwellian distribution

Source Specification

- Source Card Examples (SIn L)

```
SDEF ERG = D3 POS= 1 1 1
SI3 L 0.1 0.5 1 2 5
SP3 D 0.0 0.1 0.2 0.3 0.4
```

- Source Card Examples (SIn H)

```
SDEF ERG = D3 POS= 1 1 1
SI3 H 0.1 0.5 1 2 5
SP3 D 0.0 0.1 0.2 0.3 0.4
```

- Source Card Examples 2

```
SDEF CEL= D4
SI4 L 1 10 20
SP4 V
SC4 CellSource :proportional to cell volume
```

Source Specification

- Source Card Examples (SIn L)

```
SDEF ERG = D3 POS= 1 1 1
SI3 L 0.1 0.5 1 2 5
SP3 D 0.0 0.1 0.2 0.3 0.4
```

- Source Card Examples (SIn H)

```
SDEF ERG = D3 POS= 1 1 1
SI3 H 0.1 0.5 1 2 5
SP3 D 0.0 0.1 0.2 0.3 0.4
```

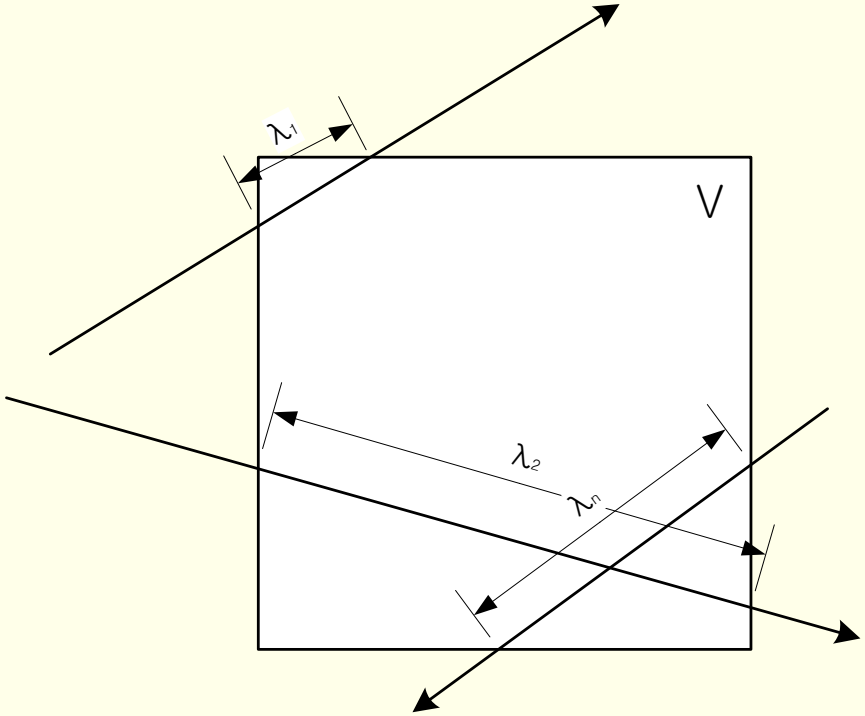
- Source Card Examples 2

```
SDEF CEL= D4
SI4 L 1 10 20
SP4 V
SC4 CellSource :proportional to cell volume
```

Tally Type and Estimator (F4)

- Track-length Estimator : F4 tally (Standard Tally)

Fluence : $\sum_i \frac{w_i}{V} \lambda_i \left[particle / cm^2 \right]$



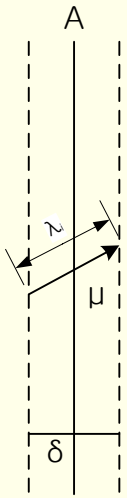
< concept of track length estimator >

Tally Type and Estimator (F2)

- Surface Estimator : F2 tally

Fluence : $\sum_i \frac{w_i}{V}\lambda_i = \sum_i \frac{w_i}{\delta A |\mu_i|}\frac{\delta}{}$

$$= \sum_i \frac{w_i}{A|\mu_i|}\left[particle/cm^2\right]$$



< concept of surface estimator >

Tally Card

- Fn card

  Fn:pl S_1 S_2 S_3 ...

  n: tally number

  pl: N, P, or E

  S : tally surface, cell number

- Ex) F4: N 2

  F14:N 1 (2 3) (4 5 6)

- FC Card

  tally comment card : the title heading of the tally in output file

  FCn comment

- Ex) F2:p 5

  FC2 surface detector #1

- FM Card

  tally multiplier card

  FMn C m R

$$= C \cdot \int \Phi(E) R_m(E) dE$$

Tally Card

- FS Card

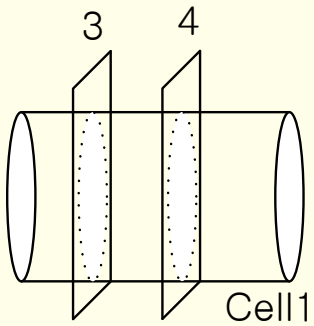
  FSn S_1 S_2 S_3 ...

  n: tally number

  S: signed problem number of a segmenting surface

- Ex) F4: N 1

  FS4 -3 -4



Practice

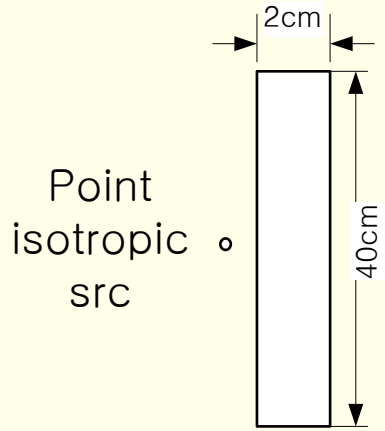
- Calculation of neutron flux

  slab dimension : W 2cm L 20cm H 40 cm

  material : water ($1g/cm^3$)

  isotropic point source of 1Mev

  1 cm away from the left surface



- Calculate the flux at the right end surface of the slab using f2 tally