An Introduction to MCNP

- Presented by:

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Outline

- MCNP: The Basics
  - What is it?
  - What does it do?
- History
- How does MCNP work?
  - Radiation Transport
  - Monte Carlo method
  - User input to the code

MCNP: What is it?

- MCNP : A General Monte Carlo Code for N-Particle Transport
- A general-purpose, continuous-energy, generalized-geometry, time-dependant, coupled Monte Carlo transport code
- MCNP contains approximately 50,000 lines of source coding.

MCNP: What does it do?

- MCNP solves particle transport problems
- Can be used in a number of different modes:
  - neutron transport only
  - photon transport only
  - electron transport only
  - neutron and photon transport
  - photon and electron transport
  - neutron, photon and electron transport

MCNP: What does it do?

- Uses a continuous energy scheme, rather than energy groups.
  - Neutron energy range: 10^{-11} MeV to 20 MeV
  - Photon and electron energy range from 1 keV to 1 GeV
- Has generalized 3-D geometry capabilities with elaborate plotter capabilities
- Has elaborate tally capabilities (answers can be expressed in flux, energy deposition, dose, etc.)

MCNP: What does it do?

- Can perform criticality calculations
- Has extensive cross section libraries
- Can be run interactively or in batch mode
- Used primarily for shielding calculations and interaction rate calculations.

History

- The use of the Monte Carlo method as a radiation transport research tool springs form work done at Los Alamos National Laboratory during WWII.
- Credit for the so-called invention of Monte Carlo as a mathematical discipline is generally given to Fermi, von Neumann, and Ulam.

History

- 1947: Fermi invents a mechanical device called FERMIAC to trace neutron movement through fissionable material by the Monte Carlo method.
- Early 1950's: Ulam leads a group of scientists in creating the Monte Carlo neutron transport code, called MCS.
- 1965: Features are added to MCS to produce the Monte Carlo neutron code MCN.

History

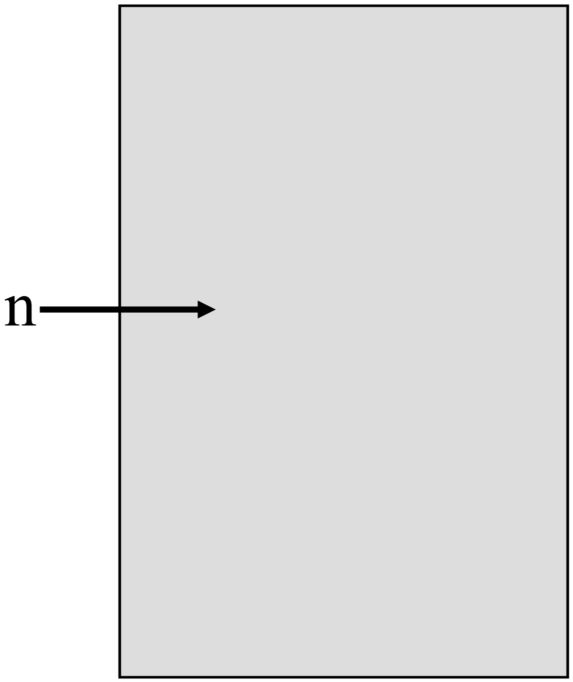
- The photon codes MCG and MCP are added to the LANL family of Monte Carlo codes
  - MCG dealt with photon transport at high energies.
  - MCP handled photon transport down to 1 keV.
- 1973: MCN and MCG are merged to form MCNG, the predecessor of MCNP
- June 1977: MCNP results from the culmination of all the above codes.

History

- Since the first version of MCNP, the Radiation Transport Group (Group X-6) at LANL has maintained it.
- Group X-6 improves MCNP and releases a new version about every 18 months.
- The most recent version is MCNP5.
- Latest our available version is MCNP4C.

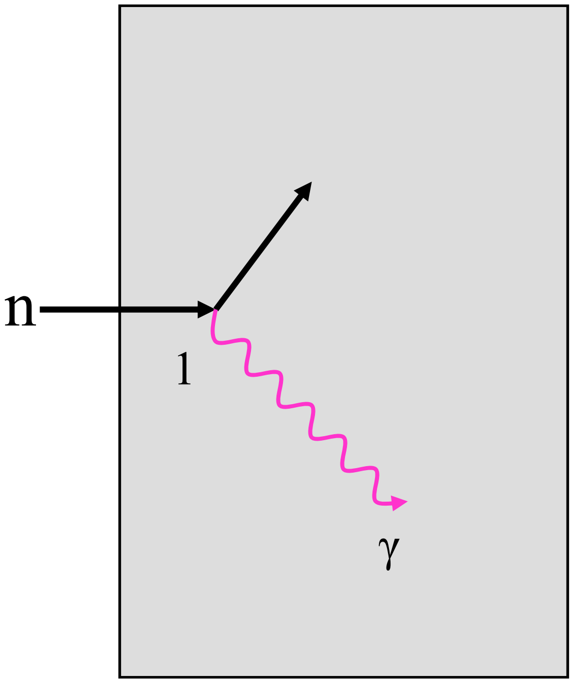
Monte Carlo Method

- Numbers between 0 and 1 are selected randomly to determine what (if any) and where interaction takes place, based on the rules (physics) and probabilities (transport data) governing the processes and materials involved.



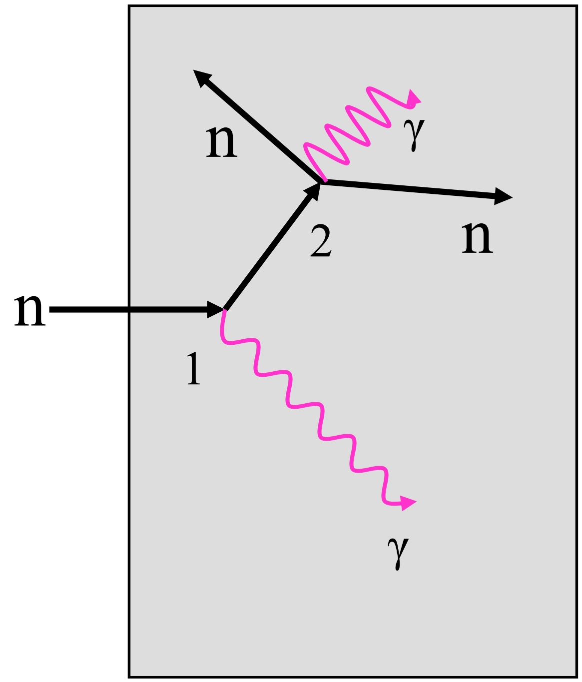
Monte Carlo Method

- In this particular case, a neutron collision occurs at event 1.
- The neutron is scattered in the direction shown, which is selected randomly from the physical scattering distribution.
- A photon is also produced and is temporarily stored, or banked, for later analysis.



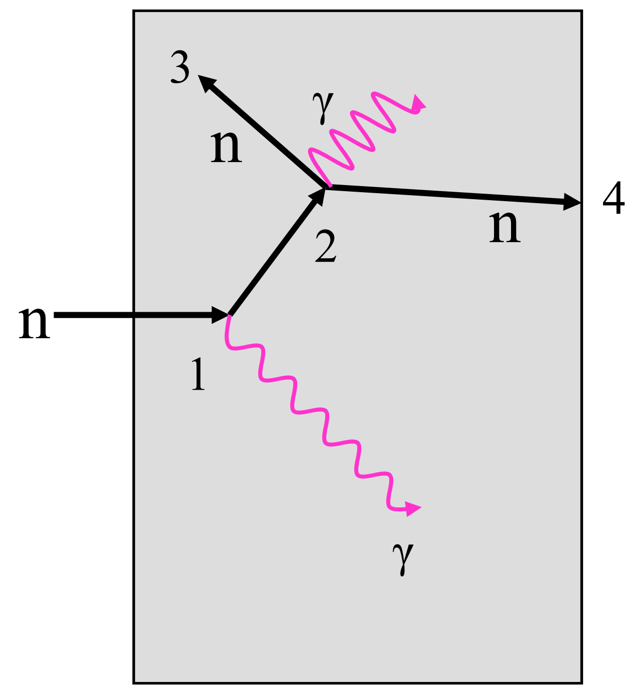
Monte Carlo Method

- At event 2, fission occurs, resulting in the termination of the incoming neutron and the birth of two outgoing neutrons and one photon.
- One neutron and the photon are banked for later analysis.



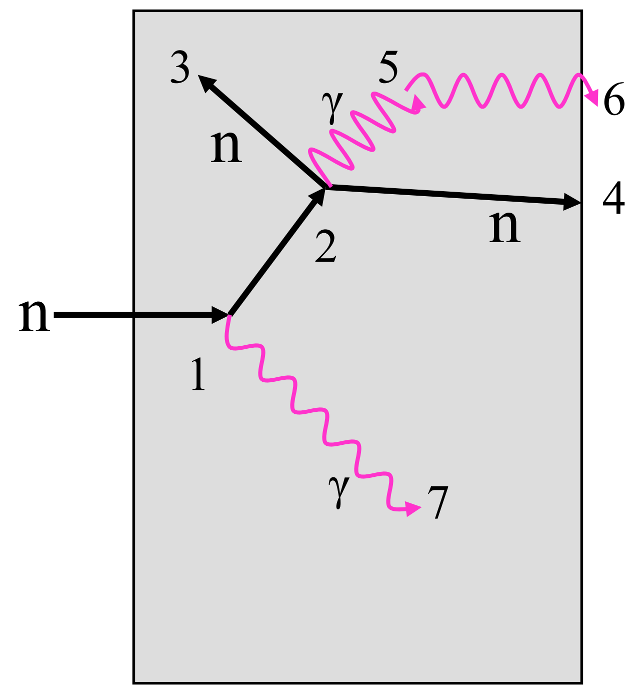
Monte Carlo Method

- The banked neutron is now retrieved and, by random sampling, leaks out of the slab at event 4.
- The first fission neutron is captured at event 3 and terminated.



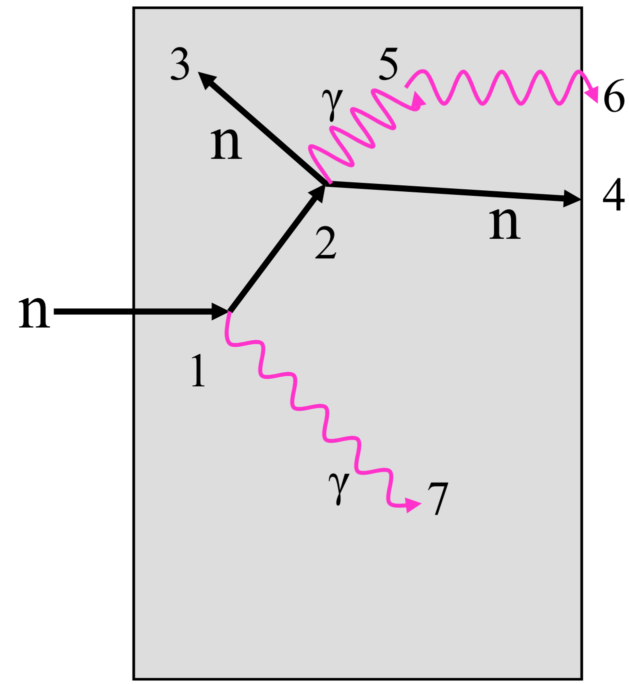
Monte Carlo Method

- The fission-produced photon has a collision at event 5 and leaks out at event 6.
- The remaining photon, created at event 1 is now followed with a capture at event 7.
- Note that MCNP retrieves banked particles such that the last particle stored in the bank is the first particle taken out.



Monte Carlo Method

- The neutron history is now complete.
- As more and more such histories are followed, the neutron and photon distributions become better known.
- The quantities of interest are tallied, along with estimates of the statistical precision of the results.



User Input to the Code

- The user creates an input file that is read by MCNP.
- This file contains information about the problem in areas such as:
  - geometry specification
  - material descriptions
  - location and characteristics of the source
  - type of answers or **“tallies”** desired

User Input to the Code

- The format of the input deck is <u>very</u> specific.
- Three major sections:
  - **Cell cards** - used to define the shape and material content of physical space.
  - **Surface cards** - defines the boundaries in space used to “create” cells (spheres, cylinders, planes)
  - **Data cards -** defines sources, materials, tallies and other information needed for problem solving.

User Input to the Code

- Specific unit expressions:
  - Length (cm)
  - Energy (MeV)
  - Time (shakes, 10^{-8} s)
  - Mass density (g cm^{-3})
  - Atom density (10^{-24} * cm^{-3} = #/(cm-barn))
  - Cross section (barns)

User Input to the Code

- Input decks are required to be both line and column specific.
  - Input is limited to columns 1 to 80
  - certain entries can appear only in a certain range of columns within a specified line
  - blank lines are required in certain places, and not allowed in other
  - spaces only may fall between entries, no tabbing

Input Structure in MCNP

Outlooks:

- Geometry Definition
- Format of Input File
- Running MCNP
- Geometry Plotting
- Material Specification

The meaning of Cell :

- Each finite medium that is filled by a determined material is called a “cell”
- A media with zero importance can be infinite
- Any cell is defined with surrounding surfaces

Cell Cards :

Form: **j m d geom params**

j = cell number and must begin in the first five columns (1< **j** <99999)

m = 0 if the cell is a void.(1 < **m** < 99999)

= material number if the cell is not a void. This indicates that the cell is to contain material *m*, which is specified on the M*m* card.

d = absent if the cell is a void.

= cell material density. A positive entry is interpreted as the atomic density in units of $\mathbf{10^{-24}}$ **atoms/** $\mathbf{cm^3}$

A negative entry is interpreted as the mass density in units of **g/** $\mathbf{cm^3}$.

Cell Cards :

Form: **j m d geom params**

geom = specification of the geometry of the cell. It consists of signed surface numbers and Boolean operators that specify how the regions bounded by the surfaces are to be combined.

params = optional specification of cell parameters by entries in the **keyword** = value form.

Geometry definition :

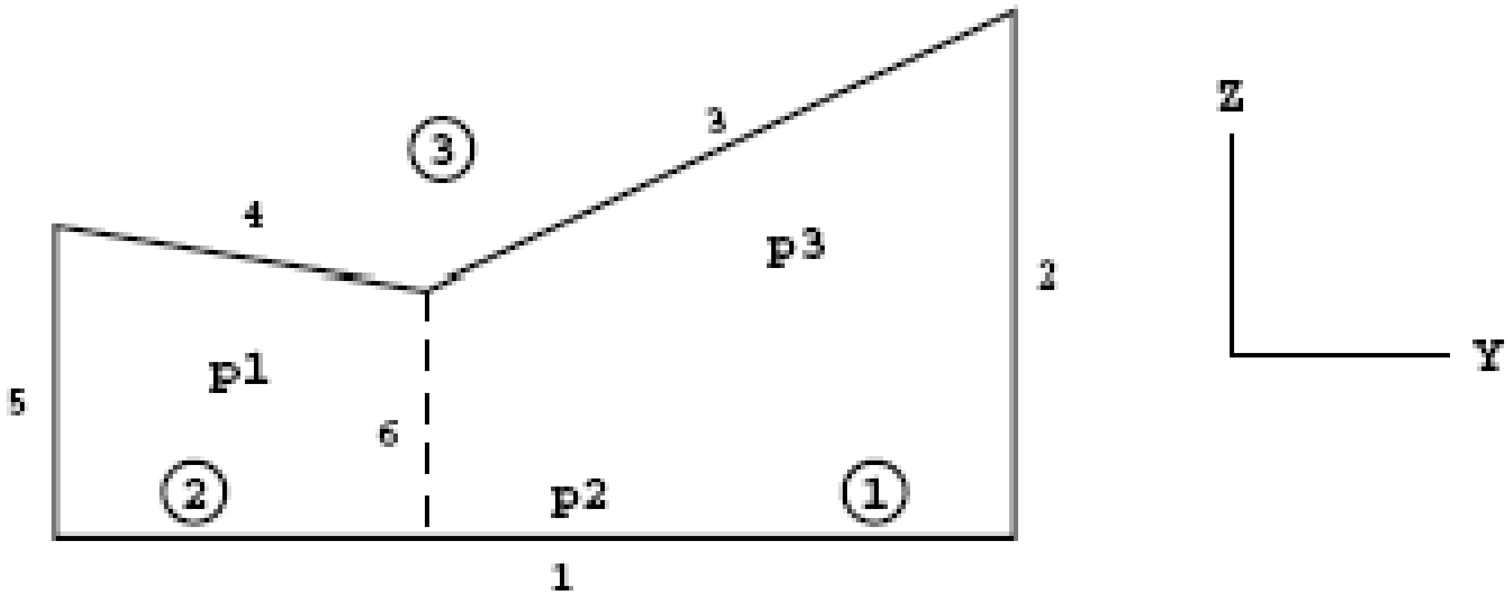
- The cells are defined by the intersections, unions, and complements of the regions bounded by the surfaces

1. Cells Defined by Intersections of Regions of Space

2. Cells Defined by Unions of Regions of Space

3. Cells Defined by Complement operator

Cells defined by intersections

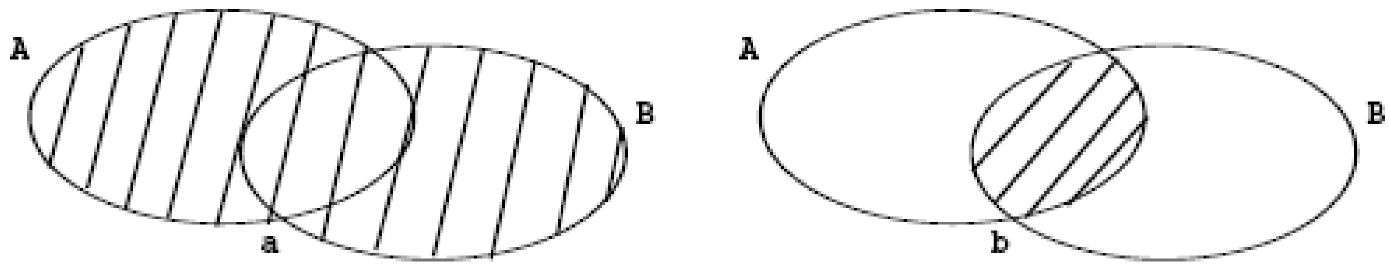


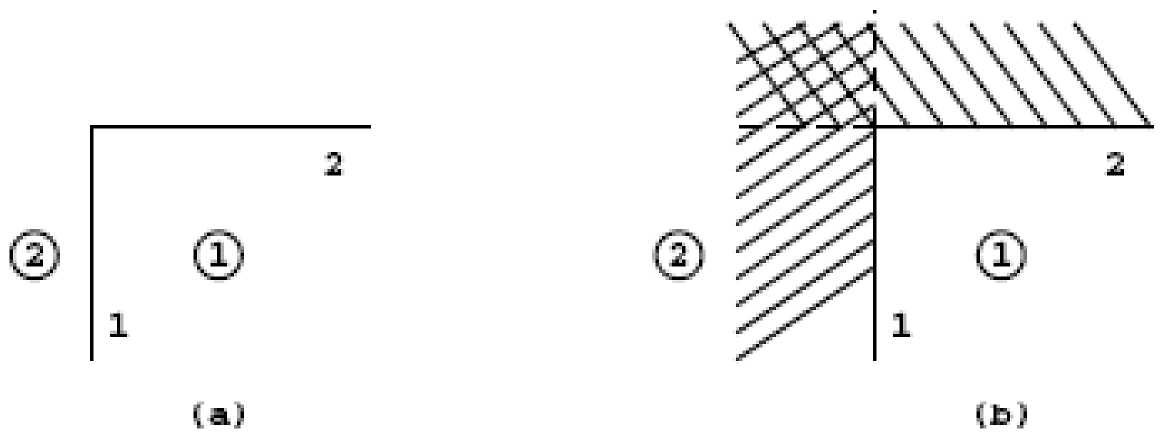
1 0 1 –2 –3 6

2 0 1 –6 –4 5

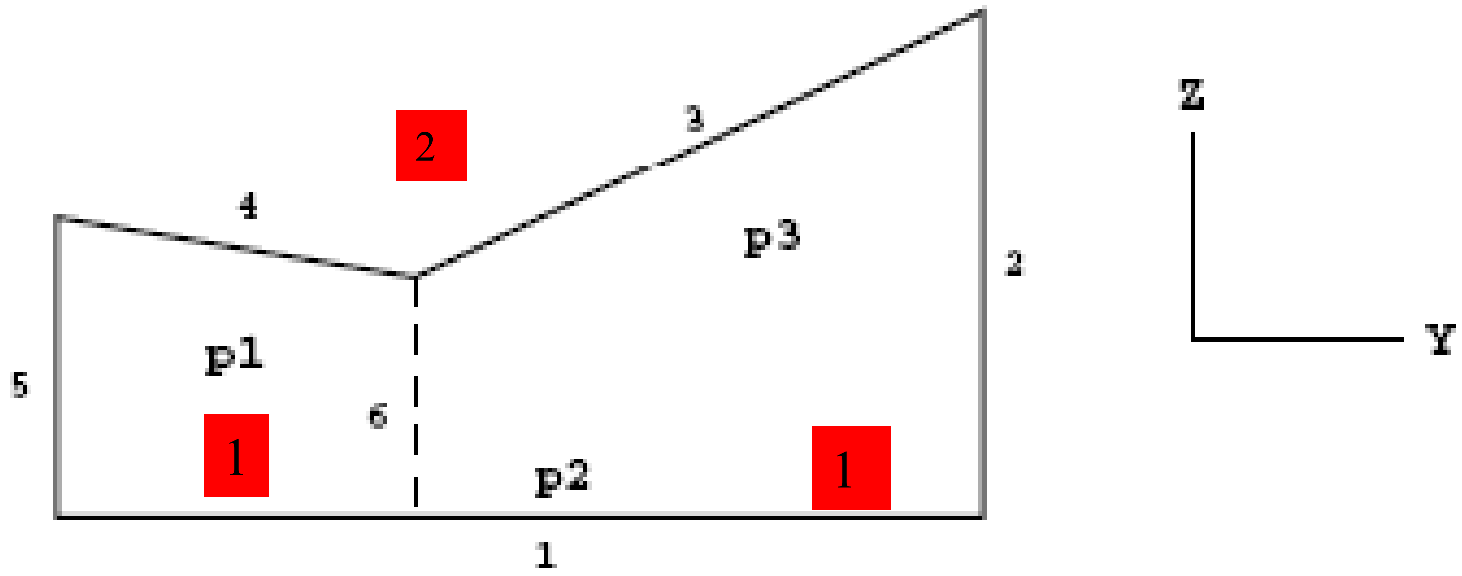
Cell 3 cannot be specified with the intersection operator.

Cells defined by unions





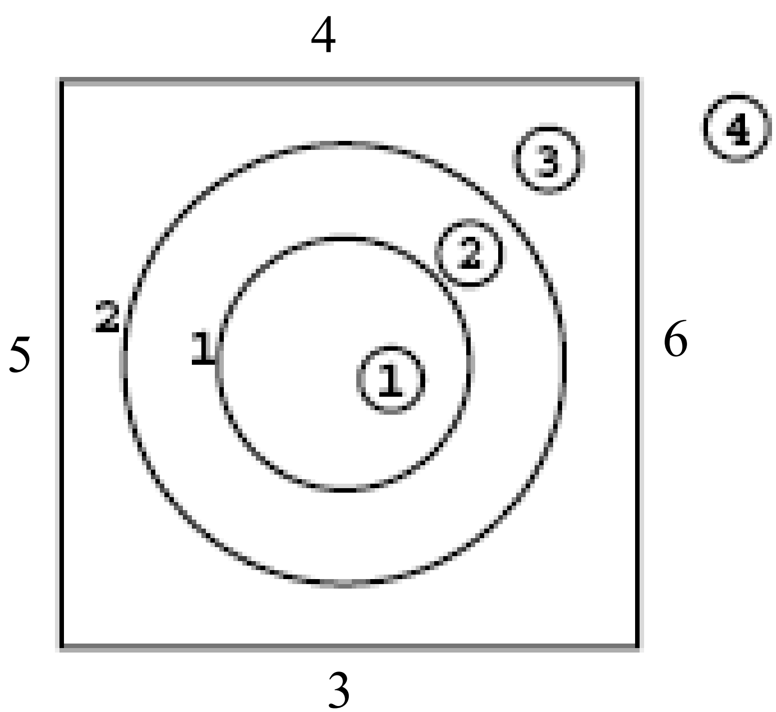
Cells defined by unions



1 0 1 –2 (–3 : –4) 5

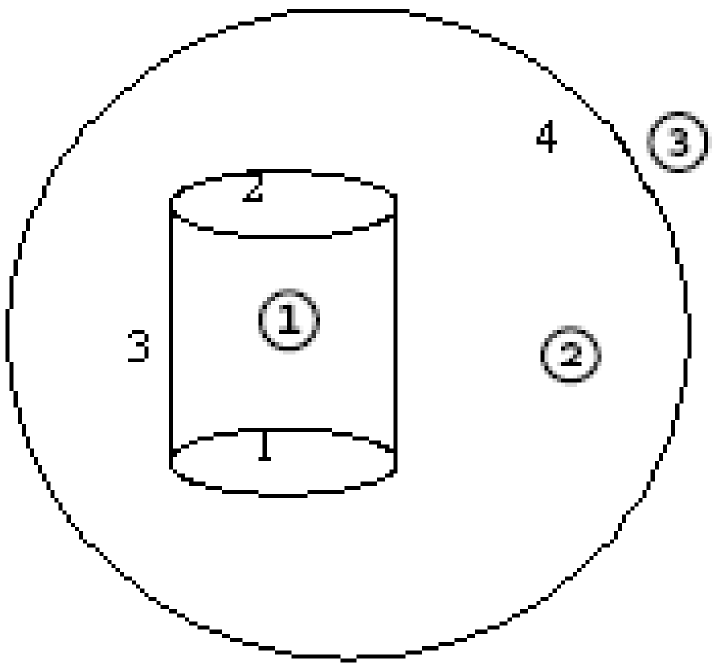
2 0 –5 : –1 : 2 : 3 4

Cells defined by complement operator



1 0 -1
2 0   1 -2
3 0   2 3 –4 5 –6 7 –8
4 0   -3:4:-5:6:-7:8   **or**
4 0   #(3 –4 5 –6 7 –8) or
4 0   #1 #2 #3

Cells defined by complement operator



cell 1: (Cylinder)
1 0 1 -2 -3
cell 2: (inside sphere and outside of cylinder)
2 0 -4 #1
2 0 -4 (-1:2:3)
2 0 -4 #(1 -2 -3)
cell 3:(outside sphere)
3 0 4
3 0 #1 #2
4 0 #3 #2 #1

Cell Definition Examples

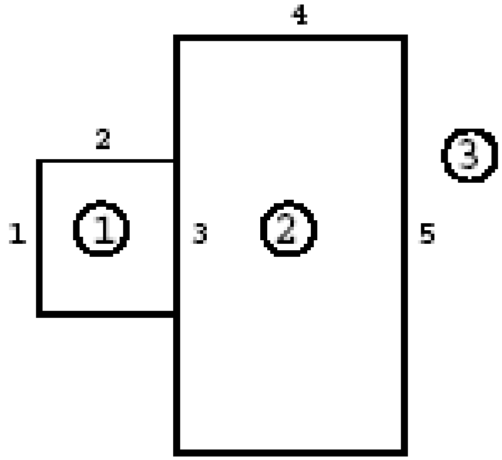


Figure 4-1a.

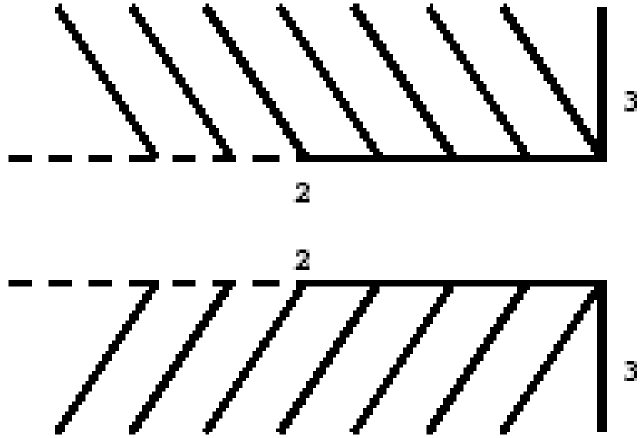


Figure 4.1b 2 -3

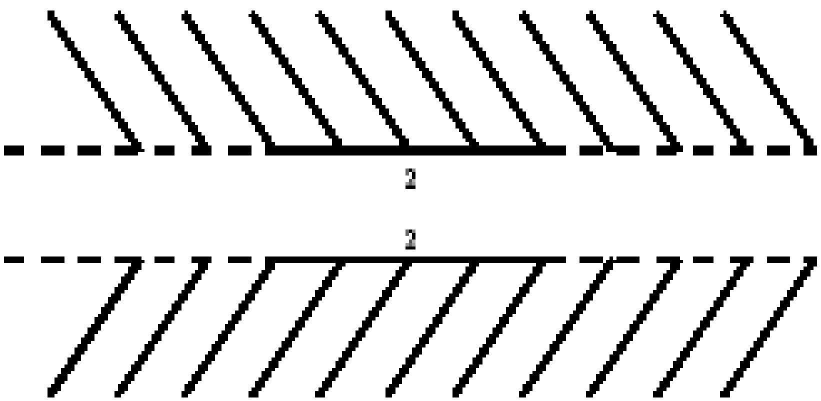


Figure 4.1c 2

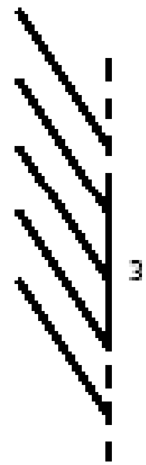


Figure 4.1d -3

Cell Definition Examples

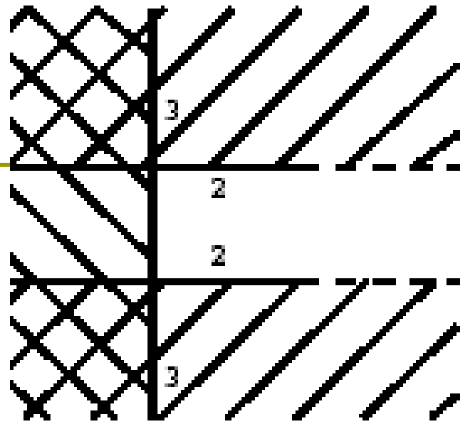


Figure 4.1e

-3:2

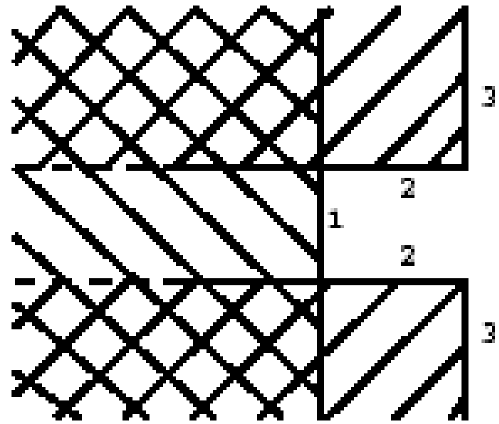


Figure 4.1f

-1:(2 -3)

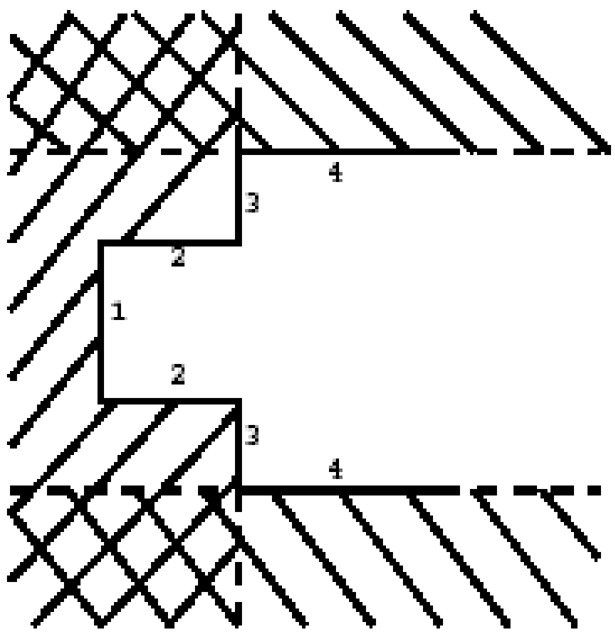


Figure 4-1g.

2 –3 :–1:4

Cell 3:

2 –3 :–1: 4:5

Cell Definition Examples

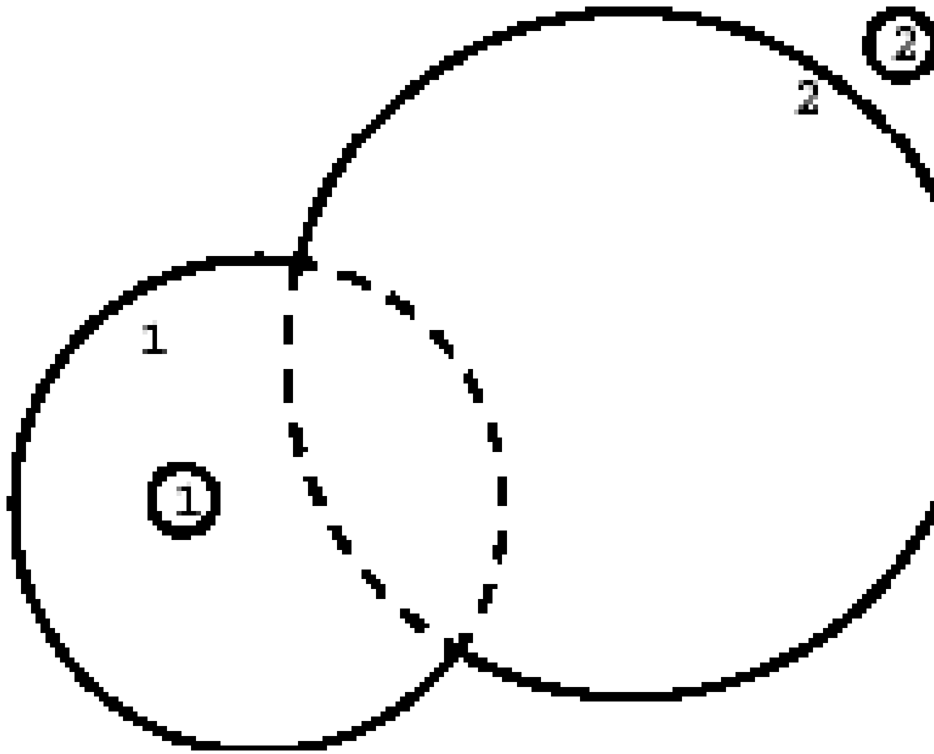


Figure 4-2.

Cell 1 is everything interior to the surfaces 1 and 2:

1	0	−1	:	−2
2	0	1		2

Cell Definition Examples

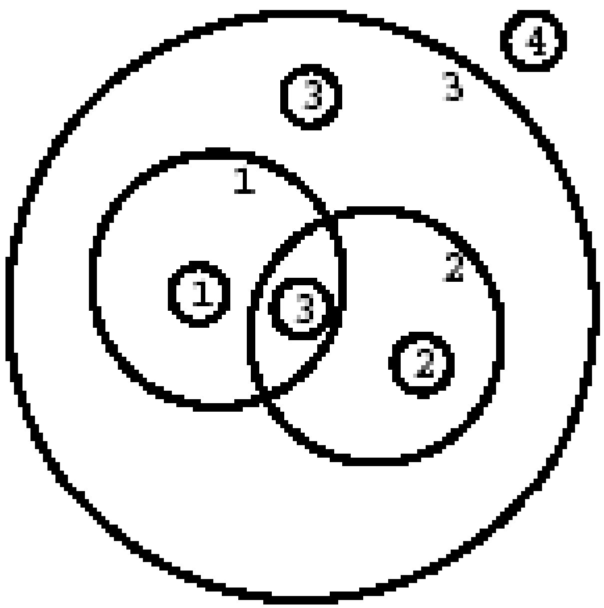


Figure 4-3.

1 0 -1 2

2 0 1 -2

3 0 (-3 1 2):(-1 –2)

4 0 3

Cell Definition Examples

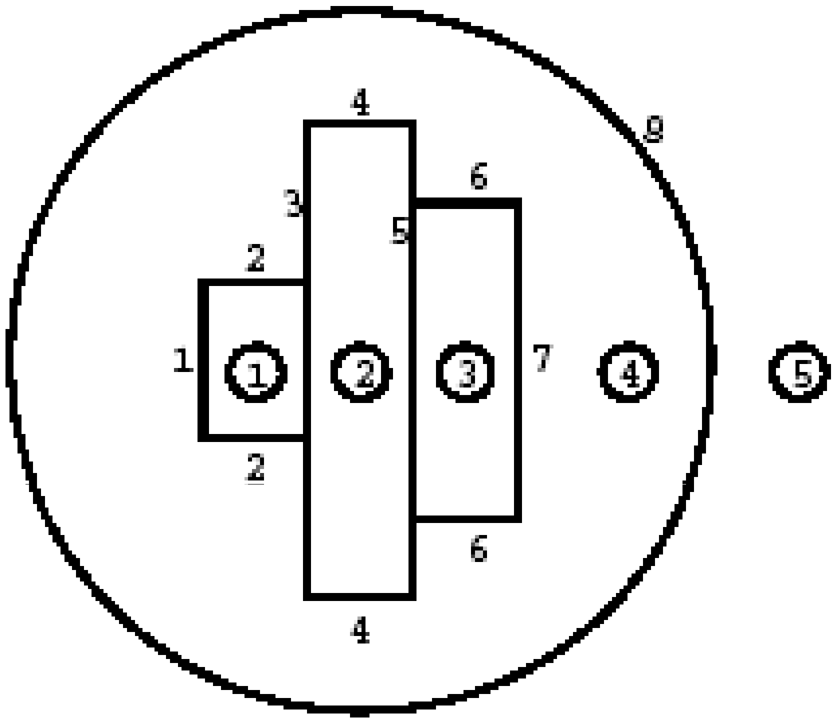


Figure 4-4.

1	0	1	-2	-3	
2	0	3	-4	-5	
3	0	5	-6	-7	
4	0	-8	#1	#2	#3
5	0	8			

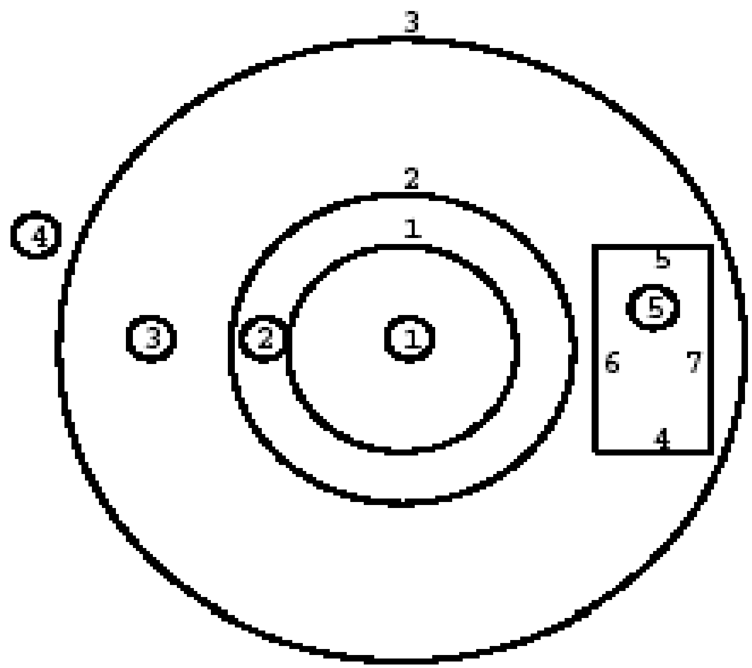


Figure 4-6.

This is three concentric spheres with a box cut out of cell 3. Surface 8 is the front of the box and 9 is the back of the box. The cell cards are

```
1    0   -1
2    0   -2   1
3    0   -3   2 (-4:5:-6:7:8:-9)      $ These parentheses are required.
4    0    3
5    0    4 -5 6 -7 -8 9
```

Cell 3 is everything inside surface 3 intersected with everything outside surface 2 but not in cell 5. Therefore, cell 3 could be written as

```
        3    0    -3  2  #(4 -5 6 -7 -8 9)
or      3    0    -3  2  #5
or      3    0    -3  2  (-4:5:-6:7:8:-9)
```

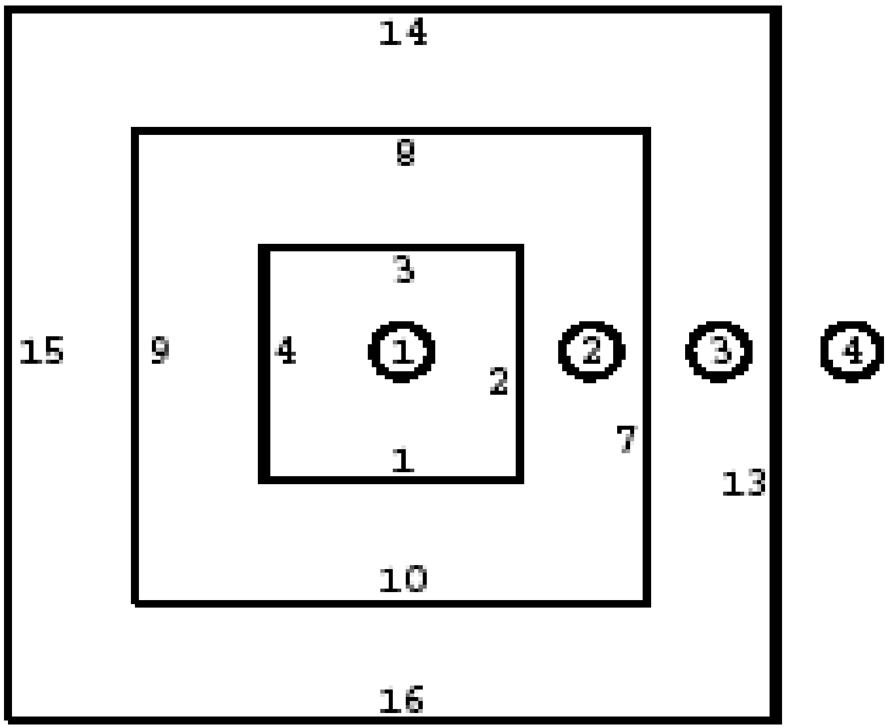


Figure 4-7.

1	0	–2	–3	4	1	5	–6
2	0	–7	–8	9	10	11	–12
		(2 :	3 :	–4 :	–1 :	–5 :	6)
3	0	–13	–14	15	16	17	–18
		(7 :	8 :	–9 :	–10 :	–11 :	12)
4	0	13 :	14 :	–15:	–16 :	–17 :	18

Surface Cards :

Form: j a list

j = surface number: 1<=j<=99999 ,

with asterisk (*) for a reflecting surface

or plus (+) for a white boundary.

a = equation mnemonic from Table 3.1

list = one to ten entries, as required.

TABLE 3.1: MCNP Surface Cards

Mnemonic	Type	Description	Equation	Card Entires
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

MCNP Surface Cards

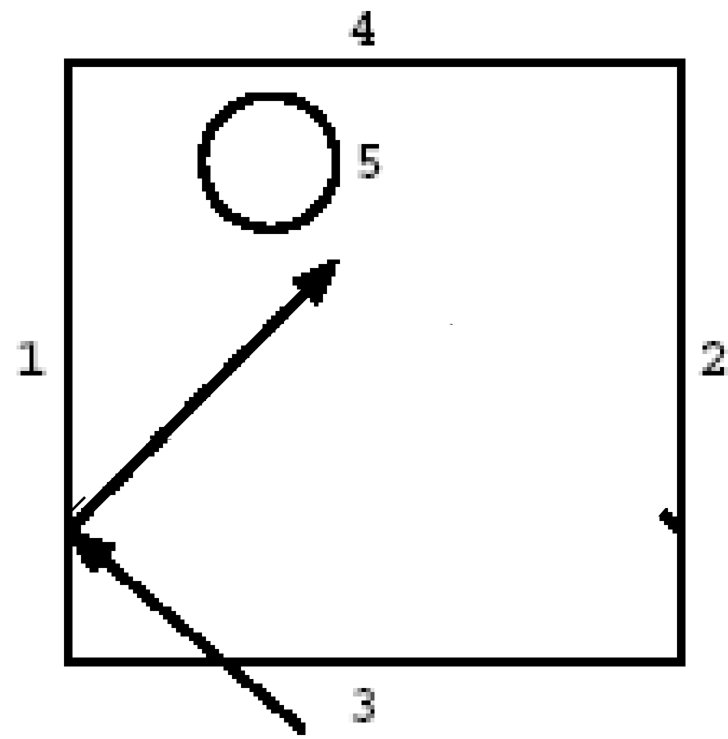
K/X K/Y K/Z KX KY KZ	Cone	Parallel to X–axis Parallel to Y–axis Parallel to Z–axis On X–axis On Y–axis On Z–axis	$\sqrt{(y-\bar{y})^2+(z-\bar{z})^2}-t(x-\bar{x}) = 0$ $\sqrt{(x-\bar{x})^2+(z-\bar{z})^2}-t(y-\bar{y}) = 0$ $\sqrt{(x-\bar{x})^2+(y-\bar{y})^2}-t(z-\bar{z}) = 0$ $\sqrt{y^2+z^2}-t(x-\bar{x})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

Special Surfaces :

1. Reflecting Surfaces
2. White Boundaries
3. Periodic Boundaries

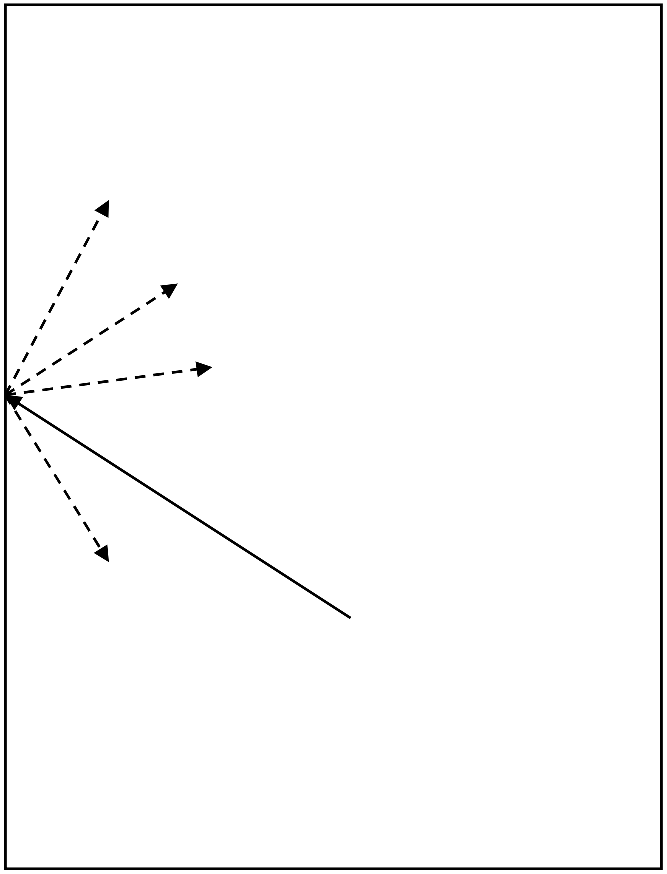
Reflecting Surfaces :

A surface can be designated a reflecting surface by preceding its number on the surface card with an asterisk. Any particle hitting a reflecting surface is specularly (mirror) reflected. Reflecting planes are valuable because they can simplify a geometry setup (and also tracking) in a problem.



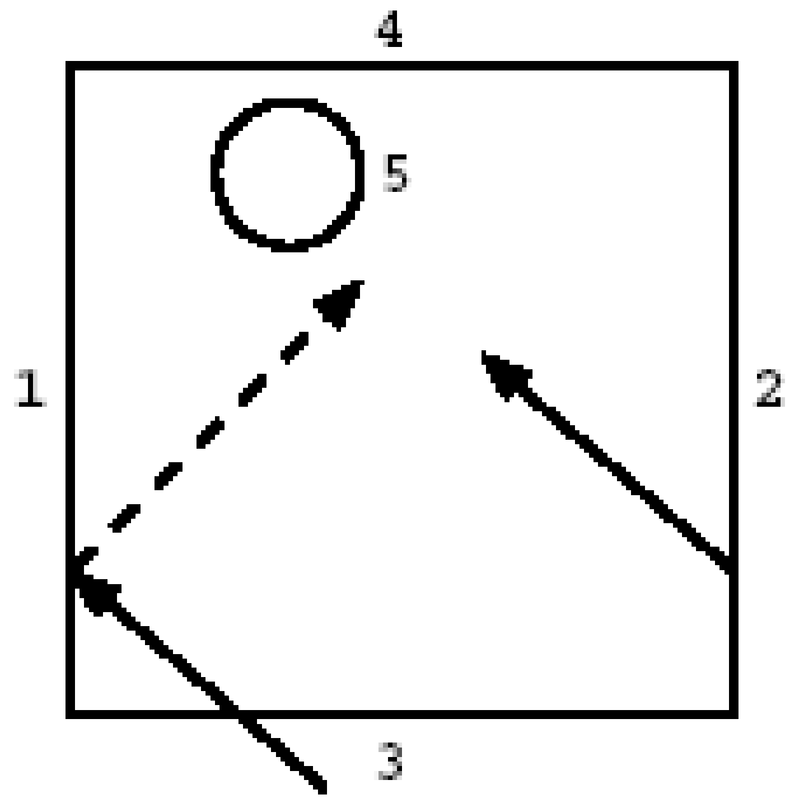
White Boundaries :

A surface can be designated a white boundary surface by preceding its number on the surface card with a plus. A particle hitting a white boundary is reflected with a cosine distribution, $p(\mu) = \mu$, relative to the surface normal; that is, $\mu^2 = \xi$, where ξ is a random number.



Periodic Boundaries :

Periodic boundary conditions can be applied to pairs of planes to simulate an infinite lattice. Although the same effect can be achieved with an infinite lattice, the periodic boundary is easier to use, simplifies comparison with other codes having periodic boundaries, and can save considerable computation time.



Format of Input File

Message Block
Blank Line Delimiter
Title Card
Cell Cards #1
…………………
Cell Cards #N
Blank Line Delimiter
Surface Cards #1
…………………
Surface Cards #N
Blank Line Delimiter
Data Cards #1
…………………
Blank Line Terminator
Anything Else

Main Data Cards

Problem mode	mode n mode p mode e mode n p mode p e mode n p e
Cell importance	imp:n imp:p imp:e
Source	sdef pos=x y z erg=E
Tally (particle current)	F1:n *S1 S2* …
Material Specification	Mn *ZAID1 f1* …
Problem cutoff	NPS n

Input File Example

Message Block (optinal)	Message: Sample Problem Input Deck
Blank Line Delimiter	
Title Card	Cell cards for sample problem
Cell Cards #1	1 1 -0.0014 -7
Cell Cards #2	2 2 -7.86 -8
Cell Cards #3	3 3 -1.60 1 -2 -3 4 -5 6 7 8
Cell Cards #4	4 0 -1:2:3:-4:5:-6
Blank Line Delimiter	
C description (optional)	C end of cell cards for sample problem
C description (optional)	C Beginning of surfaces for cube
Surface Cards #1	1 PZ -5
Surface Cards #2	2 PZ 5
Surface Cards #3	3 PY 5
Surface Cards #4	4 PY -5
Surface Cards #5	5 PX 5
Surface Cards #6	6 PX -5
C description (optional)	C End of cube surfaces
Surface Cards #7	7 S 0 -4 -2.5 .5 $ oxygen sphere
Surface Cards #8	8 S 0 4 4.5 $ iron sphere
Blank Line Delimiter	

Input File Example :

Data Card #1	IMP:N 1 1 1 0
Data Card #2	SDEF POS=0 -4 -2.5
Data Card #3	F2:N 8 $ flux across surface 8
Data Card #4	F4:N 2 $ track length in cell 2
Data Card #5	E0 1 12I 14
Data Card #6	M1 8016 1 $ oxygen 16
Data Card #7	M2 26000 1 $ natural iron
Data Card #8	M3 6000 1 $ carbon
Data Card #9	NPS 100000
Blank Line Delimiter	
Anything Else	End of Input deck

Running MCNP

Execution Line:

Mcnp inp=mcin outp=mcout runtpe=mcruntpe

mcnp i=mcin o=mcout r=mcruntpe

Default File Name	Description .
INP	Problem input specification
OUTP	ASCII output file
RUNTPE	Binary start-restart data
XSDIR	Cross-section directory

Execution Options

Mnemonic	Module	Operation
i	IMCN	Process problem input file
p	PLOT	Plot geometry
x	XACT	Process cross sections
r	MCRUN	Particle transport
z	MCPLOT	Plot tally results or cross section data

Execution Interrupts

(ctrl c)<cr> (default)	MCNP status
(ctrl c)s	MCNP status
(ctrl c)m	Make interactive plots of tallies
(ctrl c)q	Terminate MCNP normally after current history
(ctrl c)k	Kill MCNP immediately

Geometry Plotting

- To look at the geometry with the PLOT module using an interactive graphics terminal, type in :

  MCNP ip i = inpfile

- After the plot prompt *plot* > appears, geometry plotting commands can be used.

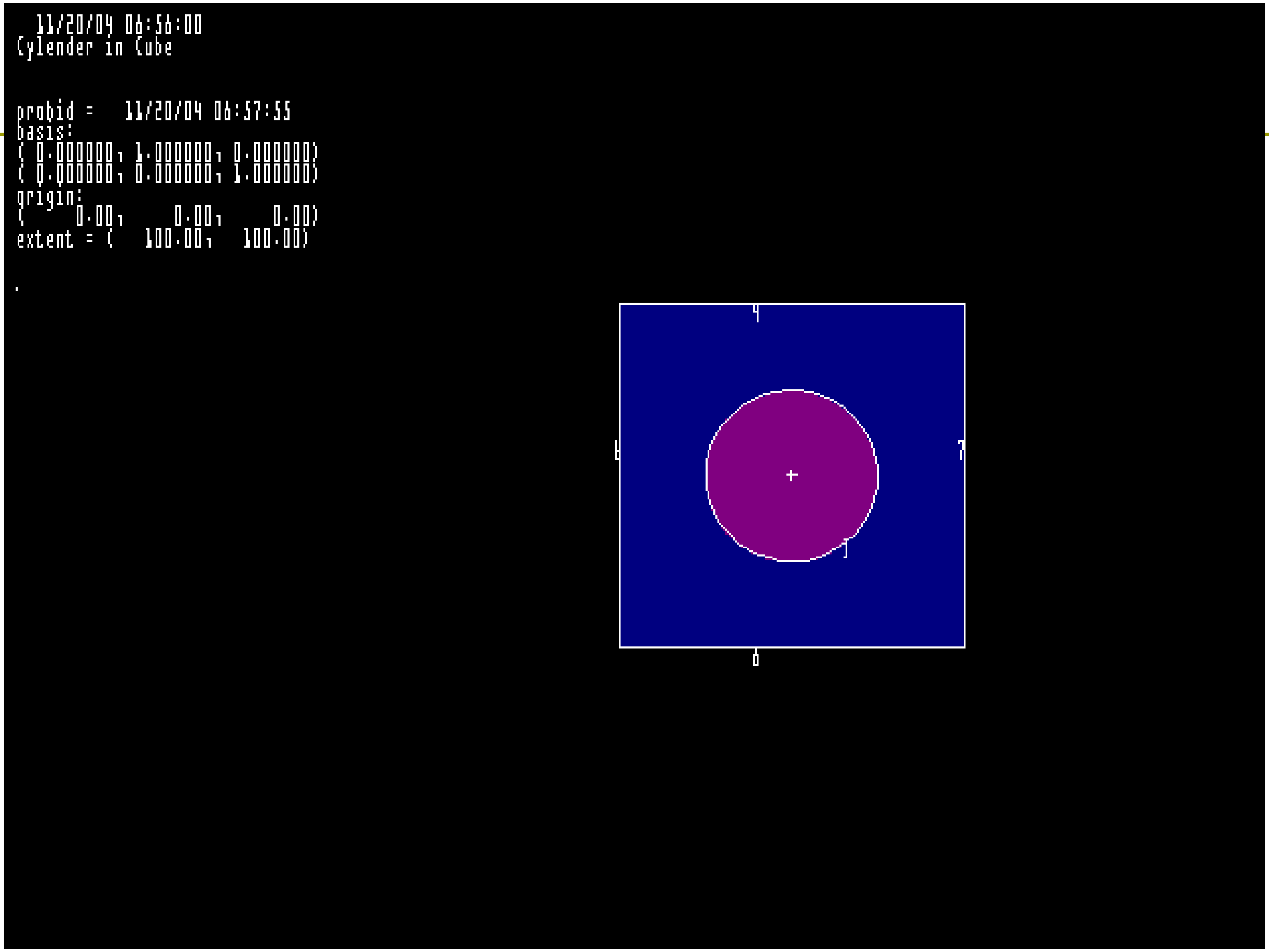
Geometry Plotting Commands

Mnemonic	Operation
px = a	intersection of the surfaces of
py = b	the problem by the plane X=a,
pz = c	Y=b and Z=c
ex = d	length of window around origin
la = S C	Put labels of size S on the surfaces and labels of size C in the cells.Default values S=1, C=0.
end	end of geometry plotting

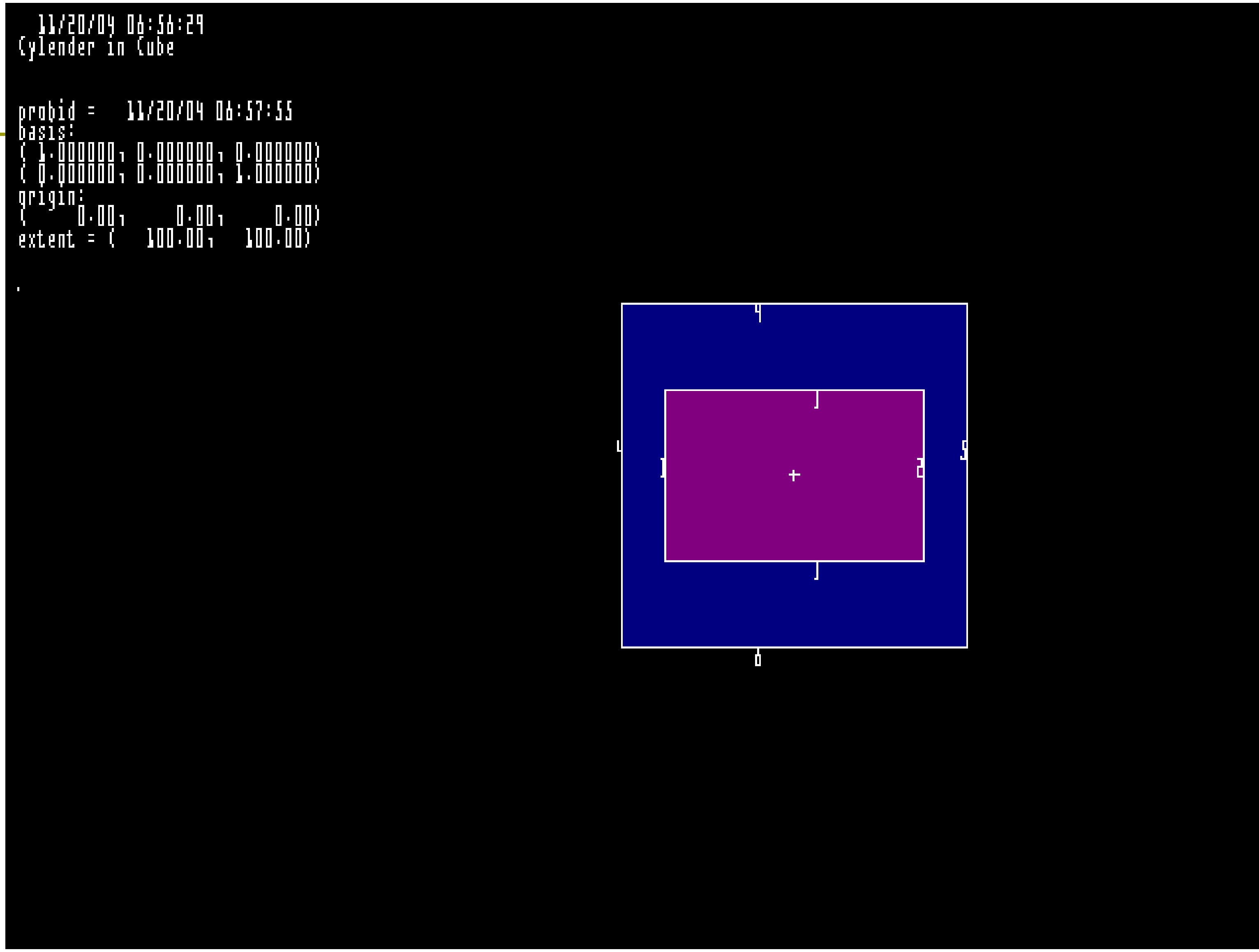
Geometry Plotting Commands

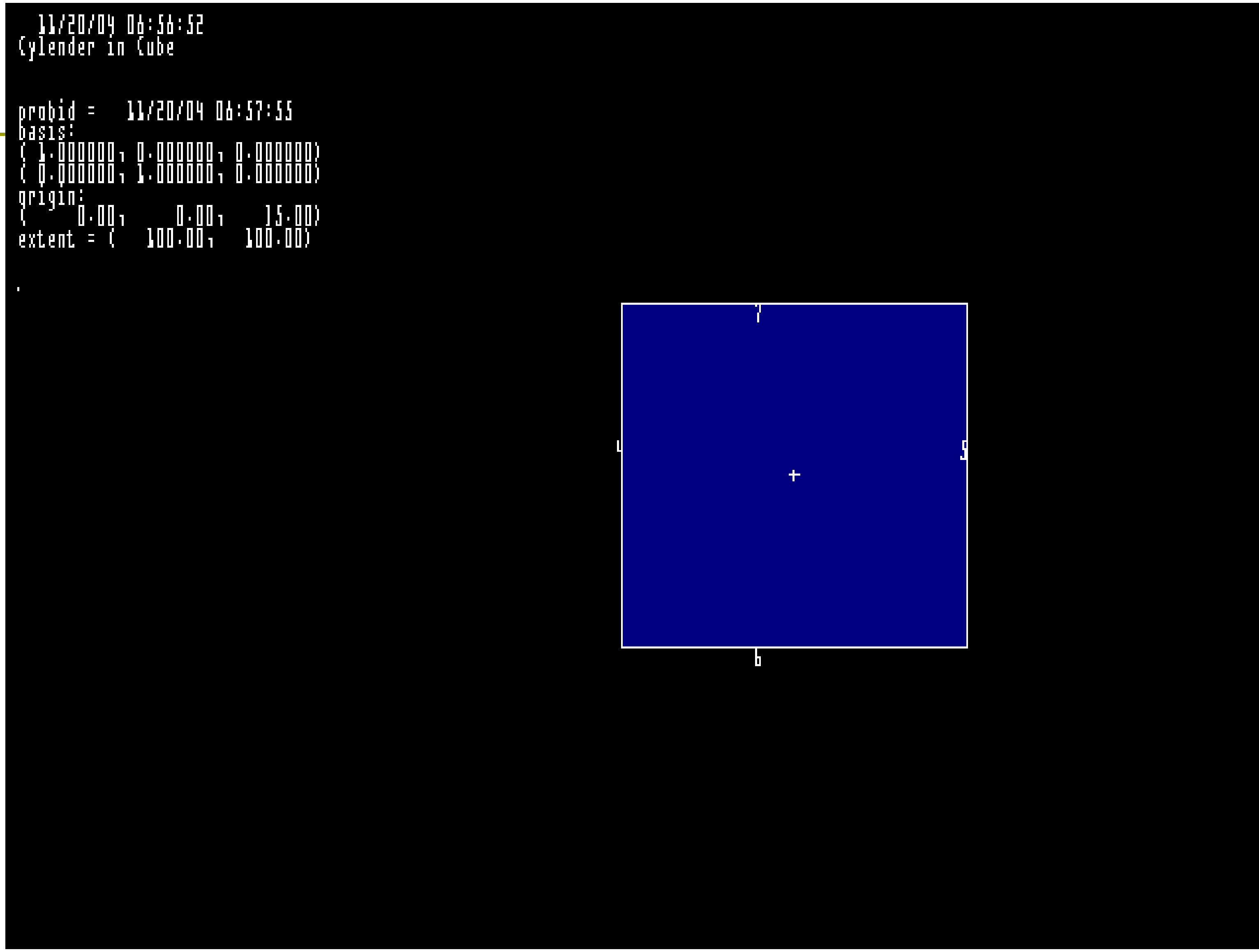
```
Cylender in Cube
1  1 -1.0  1 -2 -3
2  2 -1.0 #1 4 -5 6 -7 8 -9
3  0 #1 #2

1  px -30
2  px  30
3  cx  20
4  px -40
5  px  40
6  py -40
7  py  40
8  pz -40
9  pz  40

imp:n 1 1 0
m1 1001 1
m2 1002 1
```

Mcnp ip I=test.I
plot> **px = 0**
plot> py=0
plot> pz=35



11/20/04 06:56:00
Cylender in Cube
probid = 11/20/04 06:57:55
basis:
(0.000000, 1.000000, 0.000000)
(0.000000, 0.000000, 1.000000)
origin:
(0.00, 0.00, 0.00)
extent = (100.00, 100.00)


11/20/04 06:56:29
Cylender in Cube
probid = 11/20/04 06:57:55
basis:
(1.000000, 0.000000, 0.000000)
(0.000000, 0.000000, 1.000000)
origin:
(0.00, 0.00, 0.00)
extent = (100.00, 100.00)


11/20/04 06:56:52
Cylender in Cube
probid = 11/20/04 06:57:55
basis:
(1.000000, 0.000000, 0.000000)
(0.000000, 1.000000, 0.000000)
origin:
(0.00, 0.00, 15.00)
extent = (100.00, 100.00)
7
4
5
6

Material Specification :

- **Mm** ***ZAID1 fraction1 ZAID2 fraction2 …***

m **=** corresponds to the material number on the cell cards

ZAIDi = either a full **ZZZAAA.nnX** or partial **ZZZAAA** element or nuclide identifier for constituent *i*, where **ZZZ** is the atomic number, **AAA** is the atomic mass, ***nn*** is the library identifier, and **X** is the class of data

fractioni = atomic fraction (or weight fraction if entered as a negative number) of constituent ***i*** in the material.

Material Specification :

H-1	1001	Na-23	11023	U-235	92235
H-2	1002	Al-27	13027	U-238	92238
Li-6	3006	Si-28	14028	B-nat	5000
Li-7	3007	Fe-55	26055	Si-nat	14000
Be-9	4009	Pb-207	82207	Fe-nat	26000
O-16	8016	Fe-55	26055	Pb-nat	82000

Class of Data :

ZZZAAA.nnC	continuous-energy neutron
ZZZAAA.nnD	discrete-reaction neutron
ZZZAAA.nnY	dosimetry
XXXXXX.nnT	thermal S(α,β)
ZZZ000.nnP	continuous-energy photon
ZZZ000.nnM	neutron multigroup
ZZZ000.nnG	photon multigroup
ZZZ000.nnE	continuous-energy electron

Examples: 1001.35c 1001.50c 1001.60c

1001 2 8016 1 NLIB=60c

Thermal S(〈,®) Cross section Libraries:

- **MTm X1 X2 …**
- ***Xi*** = S(α,β) identifier corresponding to a particular component on the Mm card. (most significant below 2 eV)

Examples: m1 1001 2 8016 1
mt1 lwtr.01t
m2 1001 2 6000 1
mt2 poly.01t
m3 6012 1
mt3 grph.04t

See Appendix G for details