# Tallies

Introduction and Overview
Current and Flux Tallies
Reaction Rate Tallies
Mesh Tallies
Tally Output
Statistics & Checks
Tallies & Sources in Repeated Structures

# Print Card

- **PRINT (no entries) prints all possible print tables**

```
PRINT                    $ print everything
PRINT  110  40  130      $ print minimum + tables 110, 40, 130
PRINT  -170  -70  -110   $ Do NOT print tables 170, 70, 110
```

- **Tables in red are always printed - no way to turn off:**

-- Echo of input with error messages
10 Values of the source variables
20 Weight window information
30 Tally information
32 Mesh tally information
40 Material composition
50 Cell volumes and masses, surface areas
60 Cell information (e.g. density, importance, etc)
62 Forced collision and exponential transform
70 Surface information
72 Cell temperature information
80 ESPLT energy and TSPLT time importance ratios
85 Electron range and straggling tables
90 KCODE source data 98 Physical constants
100 Cross-section data summary
102 Assignment of S(a,b) data to nuclides
110 Source variables for first 50 histories (e.g., position, energy, direction cosines, etc.)
120 Analysis of quality of importance function
175 Keff by cycle
-- Neutron/photon/electron problem summary
126 Neutron/photon/electron activity by cell
128 Universe map
130 Neutron/photon/electron weight balance by cell (external, variance reduction, and physical events)
140 Neutron/photon activity by nuclide
--- Keff summary table
170 Source distribution frequency tables
178 Estimated keff results by batch size
190 Weight window generator summary
198 Weight window from multigroup fluxes
200 Weight window generated windows
--- Tally data
160 Tally statistical analysis summary page
161 Printed plot of probability density function
162 Cumulative plot of the tally density function
Cumulative unnormed tally density plot
--- Tally fluctuation chart

# What is a tally?

- **Tallies in MCNP are often called <u>edits</u> in many other codes**
  - Fluxes
  - Currents
  - Reaction rates

- **MCNP tally types:**
  - F1: Current on a surface
  - F2: Flux on a surface
  - F4: Flux in a cell (track-length estimate)
  - F5: Flux at a point or ring detector
  - F6: Energy deposition (track-length estimate)
  - F7: Fission energy deposition (track-length estimate)
  - F8: Pulse height tally
  - FMESH: Mesh tallies (MCNP5)
  - TMESH: Mesh tallies (MCNPX)

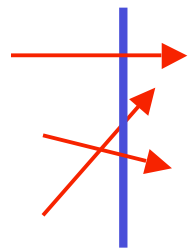
- **An energy weight can be applied to any tally by preceding it with an asterisk (e.g., *F4)**

- **A preceding plus-sign can be applied to F6 for collision heating (+F6) and F8 for charge deposition (+F8)**

# Basic Tallies

## F1 - Current across surface

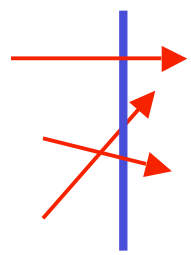
$$J = \frac{1}{W} \sum_{\substack{\text{all flights} \\ \text{crossing surface}}} wgt$$



W = total source weight

## F2 - Flux on surface

$$\phi = \frac{1}{A \cdot W} \sum_{\substack{\text{all flights} \\ \text{crossing surface}}} \frac{\text{wgt}}{|\mu|}$$



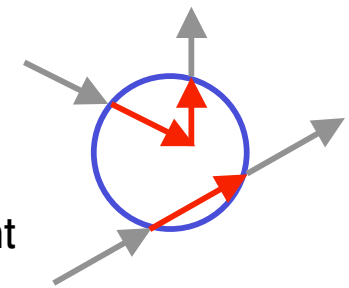
A = surface area

W = total source weight

$\mu = \Omega \bullet$ [surface normal]

## F4 - Flux in a cell

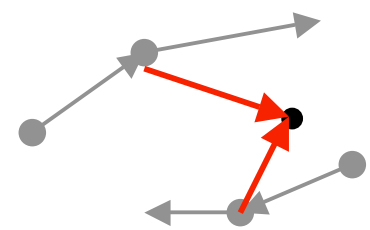
$$\phi = \frac{1}{V \cdot W} \sum_{\substack{\text{all flights} \\ \text{in cell}}} \text{wgt} \bullet \text{dist}$$



V = cell volume

W = total source weight

## F5 - Flux at a point



$$\phi = \frac{1}{W} \sum_{\substack{\text{all} \\ \text{collisions}}} \text{wgt} \cdot \frac{\text{p}(\mu)\text{e}^{-\Sigma_T R}}{2\pi R^2}$$

# Particle Types (Abbreviated)

mcnp5 & mcnpx, mcnpx

| IPT | Name | Symbol | IPT | Name | Symbol |
|---|---|---|---|---|---|
| 1 | Neutron | N | 20 | Pion$^+$ | / |
| 1 | Anti-neutron | N | 20 | Pion$^-$ | / |
| 2 | Photon | P | 21 | Neutral Pion | Z |
| 3 | Electron | E | 22 | Kaon$^+$ | K |
| 3 | Positron | E | 22 | Kaon$^-$ | K |
| 4 | Muon | I | 23 | $K_0$ short | % |
| 4 | Anti-muon | I | 24 | $K_0$ long | ^ |
| 6 | Electron neutrino | U | 31 | Deuteron | D |
| 6 | Anti-electron neut. | U | 32 | Triton | T |
| 9 | Proton | H | 33 | Helium-3 | S |
| 9 | Anti-proton | H | 34 | Helium-4 | A |

# Tally Quantities Scored

| Type | Tally on surface, cell, or point |
|---|---|
| F1: Surface Current . . . | surface |
| All particles | |
| F2: Surface Flux . . . | surface |
| All particles | |
| F4: Track length estimate of cell flux . | cell |
| All particles | |
| F5: Flux at a point or ring detector . | point or ring |
| N or P | |
| F6: Trk length est. of energy deposition . | cell |
| All particles | |
| F7: Trk-len est. of fission energy dep. . | cell |
| N | |
| F8: Pulse height tally . . . | cell |
| All particles | |

# Tally Quantities Scored (cont'd)

| Type | Estimator |
|---|---|
| F1: Surface Current . . . <br> All particles | wgt |
| F2: Surface Flux . . . <br> All particles | wgt / (\|μ\|*A) |
| F4: Track length estimate of cell flux . <br> All particles | wgt * dist / V |
| F5: Flux at a point or ring detector . <br> N or P | wgt * p(μ)*exp($-\Sigma_T R$)/($2\pi R^2$) |
| F6: Trk length est. of energy deposition . <br> All particles | wgt*dist*$\Sigma_T$(E)*H(E)*/mass |
| F7: Trk-len est. of fission energy dep. . <br> N | wgt*dist*$\Sigma_F$(E)*Q*/mass |
| F8: Pulse height tally . . . <br> All particles | (src wgt in bin $E_D$)*wgt/(src wgt) |

# Tally Quantities Scored (cont'd)

| Type | Fn: Units | *Fn: Units | +Fn |
|---|---|---|---|
| F1: Surface Current . . <br> All particles | # | MeV | |
| F2: Surface Flux . . <br> All particles | #/cm$^2$ | MeV / cm$^2$ | |
| F4: Track length estimate of cell flux <br> All particles | #/cm$^2$ | MeV / cm$^2$ | |
| F5: Flux at a point or ring detector <br> N or P | #/cm$^2$ | MeV / cm$^2$ | |
| F6: Trk length est. of energy deposition <br> N, P, or E | MeV / g | jerks / g | MeV/g |
| F7: Trk-len est. of fission energy dep. <br> N | MeV / g | jerks / g | |
| F8: Pulse height tally . . <br> All particles | pulses | MeV | charge |

1 shake = 10 nanosec, 1 jerk = $10^{16}$ erg = $10^{9}$ Joules = 1 GJoule

# Contents of the Basic Tally Card

- **Simple form:** Fn:<pl> $S_1$ $S_2$ . . . $S_m$
- **General form:** Fn:<pl> $S_1$ ($S_2$ . . . $S_i$) ($S_j$ . . . $S_k$) $S_m$
  - n = tally number = i + 10j  $1 \le i \le 8$ ($\neq 3$), $0 \le j \le 99$
    - Last digit is tally type
    - F4:n, F14:n, F124:n are all "F4" tallies
  - <pl> = particle type = particle <u>symbol</u>, n, p, e, etc.
  - $S_k$ = cell number, surface number, or T (T sums over previous tallies)
  - ($S_j$ . . . $S_k$) produces a <u>single</u> tally over items $S_i$ through $S_k$
  - Note: tally type 5 uses a different format

# Tally Examples

| | |
|---|---|
| F14:n 10 30 50 | F4 neutron cell flux, 3 bins |
| F994:n (10 30) 50 | F4 neutron cell flux, 2 bins |
| F44:p 10 11 12 T | F4 photon cell flux, 4 bins |
| F12:e (1 3) (2 4) 5 T | F2 electron surface flux, 4 bins |
| F2:n 1.1 | F2 neutron (macrobody) surface flux |

# Example - tal1

- **Copy file g1 to tal1**

- **Edit file tal1, add neutron tallies for**
  - Current escaping sphere
  - Flux at spherical surface
  - Flux in interior of sphere

- **Run the problem**

- **Examine the output file**
  - Search for "1tally" note: "1" is holdover from line-printer days!
  - Look at tally 1: info, then result & relative error
  - Look at tally 2: info, then result & relative error
  - Look at tally 4: info, then result & relative error

# Example - tal1 - Output

```
1tally   1        nps =       49756
     tally type 1    number of particles crossing a surface.
     tally for  neutrons
 number of histories used for normalizing tallies =        30000.00

 surface  1
                5.72049E-01 0.0029
- - - - - - - - - - - - - - - - - - - - - - - - - - - - - - - - - - - - - -
1tally   2        nps =       49756
     tally type 2    particle flux averaged over a surface.    units  1/cm**2
     tally for  neutrons
 number of histories used for normalizing tallies =        30000.00

          areas   surface:       1        9.60135E+02

 surface  1
                9.13105E-04 0.0049
- - - - - - - - - - - - - - - - - - - - - - - - - - - - - - - - - - - - - -
1tally   4        nps =       49756
     tally type 4    track length estimate of particle flux.   units  1/cm**2
     tally for  neutrons
 number of histories used for normalizing tallies =        30000.00

          volumes     cell:      10       2.79751E+03

 cell  10
                2.41661E-03 0.0029
```

# Review: Basic Statistics

- **MCNP tally results have the form**

  **RESULT RELERR**

  Where

  RESULT = average score for the tally, after N histories

  RELERR = relative error in the average score, after N histories

  **All tally results are normalized to be <u>per starting particle</u>**

- **Exception: For KCODE calculations, K-effective results are reported as**

  **RESULT STD**

  Where

  RESULT = average score for the tally, after N histories

  STD = standard deviation in the average score, after N histories

# Review: Basic Statistics

- **Average, standard deviation, relative error**
  - Let $x_k$ = the value of a tally for the $k^{th}$ history
    N = number of histories run (so far)
  - **Average tally**, after N histories

$$\bar{x} = \frac{1}{N}\sum_{k=1}^{N} x_k$$

  - **Standard deviation** of average tally, after N histories

$$S_{\bar{x}} = \frac{1}{\sqrt{N-1}}\sqrt{\frac{1}{N}\sum_{k=1}^{N} x_k^2 - \bar{X}^2} \approx \frac{1}{\sqrt{N}}\sqrt{\frac{1}{N}\sum_{k=1}^{N} x_k^2 - \bar{X}^2}$$

  - **Relative error** in average tally, after N histories

$$RELERR = \frac{S_{\bar{x}}}{\bar{X}} \qquad RELERR \propto S_{\bar{x}} \propto \frac{1}{\sqrt{N}}$$

# Review: Basic Statistics

- **Relative error vs number of histories (N)**

$$RELERR \propto S_{\bar{x}} \propto \frac{1}{\sqrt{N}}$$

  - To cut the relative error in **half**, must run **four times** as many histories
  - To reduce relative error by **10x**, must run **100x times** as many histories

- **Precision**

  The RELERR or STD DEV reflect the **precision** of results, ie, the uncertainty in the result caused by statistical fluctuations in the Monte Carlo simulation

- **Accuracy**

  The **accuracy** of a result is how close the average tally is to the true physical quantity being estimated.

  **Accuracy** depends on the geometry approximations, cross-section data realism, material definitions, physics approximations, code approximations, etc.

- **Running more histories will improve the precision of a result, not the accuracy of a result.**

# Tally Normalization - Flux vs Fluence

- **Fluence = $\int \varphi \, dt$**

- **Remember: MCNP tallies are normalized to be <u>per unit source</u>**

- **Given a tally result $X_n \pm R_n$, reactions/source-particle, then**
  - **If the real source is given as S particles/sec, then the normalized result is**

    $X$ = [S src-part/sec] * [$X_n$ reactions/src-part] $\pm$ $R_n$
    = $S*X_n$ reactions/sec $\pm$ $R_n$

    To get fluence or time-integrated result, **multiply X by $\Delta t$**

  - **If the real source is given as S particles (pulse, already time-integrated), then**

    $X$ = [S src-part] * [$X_n$ reactions/src-part] $\pm$ $R_n$
    = $S*X_n$ reactions $\pm$ $R_n$

    **Don't multiply X by $\Delta t$** to get fluence or time-integrated result

# Cards Associated With Tallies

- Comment card
- Energy bin card
- Time bin card
- Cosine bin card (tally type 1 only)
- Tally print control cards
- Multiplier card
- Segment divisor card
- Dose cards

# Tally Comment Card

- **Form:** **FCn Any desired title**
  - **n is tally number (e.g., FC432)**
  - **Remaining contents of card (here, Any desired title) will appear in the output as the heading for tally n**
  - **Example**

    **FC24 tally of flux in fuel pin**

# Energy Bin Card

- **Form:** **En** $E_1 \ldots E_m$
  - n is tally number (e.g., E104)
  - $E_k$ is upper bound of kth energy bin in MeV, in strictly increasing order
    - $0 < E_1 < E_2 < E_3 \ldots < E_m$
    - A lower bound of 0 is assumed for Bin 1
  - E0 card specifies energy structure for all tallies except those which have been a specific energy bin card associated with them
  - If no energy bin card is specified, each tally is over the entire energy range
  - Example: Fast & thermal fluxes for F24:n tally

```
E24     .625e-6   20.
```

[Note: EMn card may be used to multiply tallies for energy bins]

# Tally Time Card

- **Form:** Tn $t_1 \ldots t_m$
  - n is tally number (e.g., T114)
  - $t_k$ is upper bound of $k^{th}$ time bin in shakes, in strictly increasing order
    - $-\infty < t_1 < t_2 < t_3 \ldots < t_m$
    - A lower bound of $-\infty$ is assumed for Bin 1
    - 1 shake = 1e-8 seconds
  - T0 card specifies time structure for all tallies except those which have a specific tally time card associated with them
  - If no tally time card is specified, there will be one bin over all times
  - Example: 3 time bins for F22:n tally, plus total over all times

```
T22     -1    100    1.0E+11
```

[Note: TMn card may be used to multiply time bins]

# Cosine Bin Card (Tally Type 1 Only)

- **Form:** Cn $C_1 \ldots C_m$
  - n is tally number (e.g., C221)
  - $C_k$ is upper bound of kth cosine bin, in strictly increasing order
    - $-1 < C_1 < C_2 \ldots < C_m \leq +1$
    - A lower bound of -1 is assumed for Bin 1
  - C0 card specifies cosine structure for all tallies except those which have a specific cosine bin card associated with them
  - If no cosine bin card is specified, each tally is over the entire angular range from -1 to 1
  - Example: inward & outward partial currents for F11 tally

    C11 0. 1.

[Note: CMn card may be used to multiply cosine bins]

# Tally Print Control Cards

**FQn** Print Hierarchy.

Default: F D U S M C E T

F cell, surface, or detector tally
D direct or flagged tally
U user bins
S segment tallies
M multiplier bins
C cosine bins
E energy bins
T time bins

Example: FQ4 M F U

user bins printed across top
cell bins printed vertically
user-cell blocks printed by multiplier bins

# Tally Multiplier Card

**Note: Discussion here will be limited to the multiplier form of the tally multiplication card**

- **Form:** **FMn C m $B_1$ ($B_2$ . . . $B_i$) . . . ($B_j$ . . . $B_m$) $B_{last}$**
  - **n is tally number (e.g., FM24)**
  - **C is a multiplicative constant**
    - **C > 0 means multiply tally by C**
    - **C < 0 means multiply tally by |C|, and by atom density in the tally cell**
  - **m is a material number (from an Mm card)**
  - **$B_k$ is a reaction type identifier**
  - **$B_{last}$ is either blank or T (T sums over previous tallies)**

# Combinations of Reaction Type Identifiers

- Reaction identifiers can be combined, either additively or multiplicatively, to form a single tally
- All component identifiers must be enclosed in a single set of parentheses
- Colon between two reaction type identifers means they are to be added ($B_i : B_j : B_k$)
- Blank space between two reaction type identifiers means they are to be multiplied ($B_i \; B_j$)
- If no ( ), precedence of operations is multiply first, then add

# Reaction Type Identifiers

- **Reaction type identifiers can be either positive or negative**
  - **Positive values correspond to ENDF reaction types (MT numbers)**
  - **Negative values are specific to the type of library (multipgroup or continuous-energy) and particle (neutron or photon) employed**

| MT | Reaction Type |
|---|---|
| 1 | Total |
| 2 | Elastic scattering |
| 18 | Fission |
| 101 | Capture |
| 102 | (n,$\gamma$) |

Note: The MCNP manual reverses the usual definitions of capture and absorption (physicists versus nuclear engineers). Throughout this presentation, we will use absorption = fission + capture

# Other MCNP Reaction Type Identifiers

# FM Card Examples

- **Fission rate in cell 10, which contains material 100, with continuous-energy neutron data**

```
F14:n    10
FM14     -1.0   100    -6
```

- **Nu-fission rate in cell 10, which contains material 100, with continuous-energy neutron data**

```
F24:n    10
FM24     -1.0   100    ( -6   -7 )
```

# Tallies With Macrobodies

- **Surfaces of most macrobodies are formed by several distinct components (referred to as "facets")**
- **Specific facet(s) must be specified for surface tallies**
- **Facet is identified as S.F, where S is the surface number for the macrobody and F is the facet number**
- **Facet numbers are fixed with respect to the orientation of the Macrobody**
- **Examples**

<u>Rectangular Parallelepiped (RPP)</u>

1 right side
2 left side
3 front
4 back
5 top
6 bottom

<u>Right Circular Cylinder (RCC)</u>

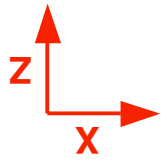
1 side of cylinder
2 top of cylinder
3 bottom of cylinder

# Example - tmac

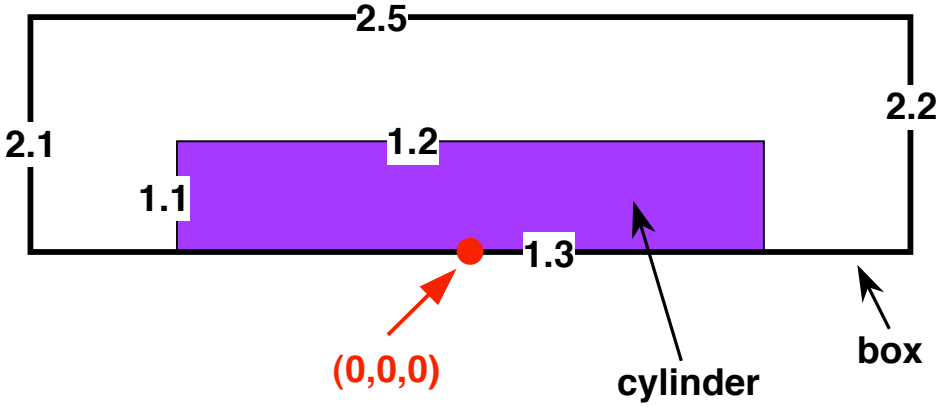
- **Copy file tmac from SOLUTIONS folder to file tmac1**

- **Edit file tmac1, and add tallies for:**
  - All particles crossing top of cylinder
  - All particles crossing top of box
  - All particles leaking out of box

- **Save and run:** **mcnp i = tmac1**

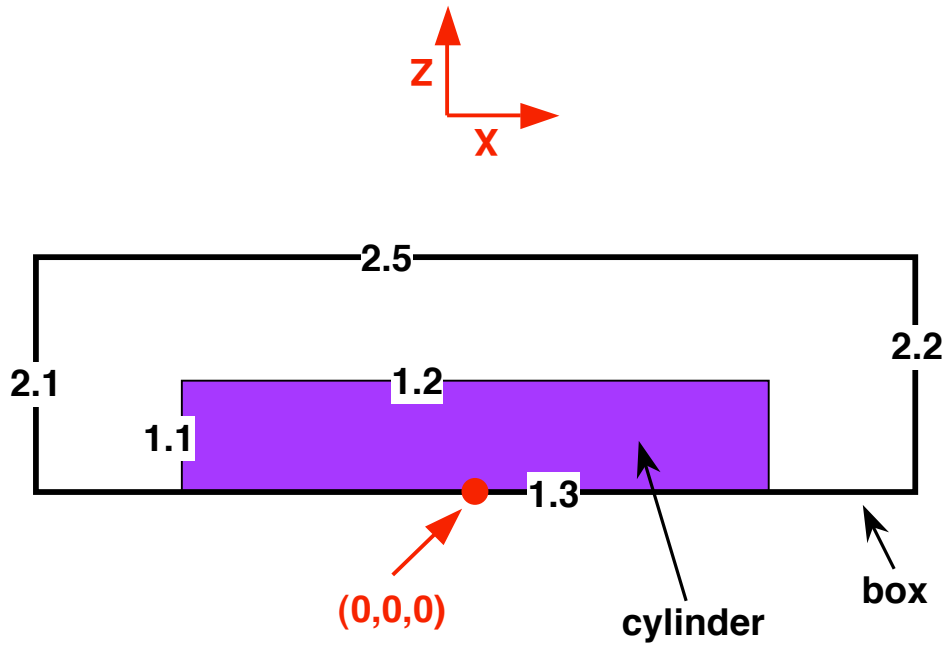



- **Look at the output**



# Tallies With Macrobodies - Example tmac

```
Tally Problem with macrobodies
1 5 -1.0 -1
2 0       1 -2
3 0       2

1 RCC  0 0 0   0 0 1   4
2 RPP -5 5   -5 5   0 2

mode  n  p
nps 100000
imp:n,p  1  1  0
sdef   axs 0 0 1   rad d1
       vec 0 0 1   dir 1   sur=1.3
       erg=10    par=n
si1 5
m5   1001.66c 2.0   8016.66c 1.0
print  110  170
c
c current across cylinder top
f111:n   1.2
f121:p   1.2
c
c current across box top
f211:n  2.5
f221:p  2.5
c
c leakage from box
f311:n  2.1  2.2  2.3  2.4  2.5  1.3  T
f321:p  2.1  2.2  2.3  2.4  2.5  1.3  T
```



# Tallying on Macrobody Surfaces --- Notes

1. **In some cases, eliminated surfaces cannot be used in source.**

   **sdef sur=2.6** *fails*
   *fatal error. source surface 3 is not defined.*

2. **Cannot tally on macrobody surface --- must tally on facet.**

   **f1:n 1 2** *fails because surfaces 1 and 2 do not exist*
   *fatal error. facet 1.0 of tally 1 does not exist.*

3. **In some cases, eliminated surfaces cannot be used in tally.**

   **f11:p 2.6**
   *fatal error. facet 1.0 of tally 11 does not exist.*
   *surface 1.3 is same as 2.6; 2.6 eliminated.*

# Segment Divisor Card (SD)

- **MCNP normalizes flux tallies by dividing by area, volume, or mass**
  - For cell flux tallies (F4), must divide by **volume**
  - For surface flux tallies (F2), must divide by **area**
  - For energy deposition (F6), must divide by **mass**
  - For fission heating (F7), must divide by **mass**
  - **Can use SD card to supply areas, volumes, or masses**
  - MUST do this if they are not calculated by MCNP

- **Form: SDn d1 d2 …**
  - n = tally number
  - d1, d2, … = divisors for each tally bin
  - Must have as many entries as there are tally bins for tally "n"
  - **Can use 1.0 to avoid dividing by volume or area or mass**
    **Note: dividing by 1.0 instead of volume <u>changes units</u>, etc.**

# Dose Cards (DE, DF)

- **Function to convert from particle flux to human biological dose equivalent rate**

$$\text{Dose} = \int_E D(E)\phi(E)dE$$

**DEn A E1 E2 . . . Ek**

- Ei = energy points (MeV)
- A = LOG or LIN energy interpolation method

**DFn B F1 F2 . . . Fk**

- Fi = corresponding value of the dose function at each energy on DEn
- B = LOG or LIN dose interpolation method

**Example: Neutron Flux-to-Dose Factors from NCRP-38, ANSI/ANS-6.1.1-1977**

```
de4  log  2.5E-08   1.0E-07   1.0E-06   1.0E-05   1.0E-04   1.0E-03
          1.0E-02   1.0E-01   5.0E-01   1.0       2.5       5.0
          7.0       10.0      14.0      20.0
df4  log  3.67E-06  3.67E-06  4.46E-06  4.54E-06  4.18E-06  3.76E-06
          3.56E-06  2.17E-05  9.26E-05  1.32E-04  1.25E-04  1.56E-04
          1.47E-04  1.47E-04  2.08E-04  2.27E-04
```

- **In MCNPX, several standard neutron and photon dose functions are available via the DFn card (note that this requires a different card format and keywords)**

# Current & Flux Tallies

# Current Tallies (F1)

- **Current tallies, by default, produce total, not net, current**
  - F1 tally: $J^+ + J^-$
- **Net current can be determined, however, as the difference between inward and outward partial currents**
  - **Net** current: $J^{net} = J^+ - J^-$
- **Inward and outward partial currents can be tallied using the cosine bin card**
  - Use the Cn card with F1 to get inward & outward partial currents
  - Example: partial currents on surface 7

```
f1:n    7          $ current across surface 7
c1    0.  1.       $ use 2 bins:  (-1,0)  &  (0,1)
```

To get **net** current, subtract (-1,0) bin from (0,1) bin

# Example - Flux & Current Tallies - tal2

- **Copy file fv1 to file tal2**

- **Edit file tal2:**
  - For unit cell with reflecting BC (* before surface 3)
  - Add a print card, to print everything: **`print`**
  - Add a cell flux tally for the fuel (Cell 10) & for fission in the fuel
  - Add surface flux tallies at the outside of the fuel (Surface 1) and clad (surface 2)

- **Run the problem & examine the output file**
  - Search for "1tally"

# Example - tal3

- **Copy file tal2 to file tal3**

- **Edit file tal3:**
  - Add clad (Cell 20) and water (cell 30) to cell flux tally
    - Examine previous tal2 output -- volumes calculated for Cells 10 & 20, <u>but not for Cell 30</u> (water outside cylinder & inside box)
    - **Must add SD card to supply volumes (must supply all 3)**
      Volume of cell 30 = 1.4x1.4x420 - (volume cell 10) - (volume cell 20)

- **Run it & examine the tally output**

- **Edit file tal3:**
  - Split all the tallies into **fast** (E > .625 eV) and **thermal** (E < .625 eV) bins, using the **E0 card**

- **Run it & examine the tally output**

# Example - tal4

- **Copy file tal3 to file tal4**

- **Edit file tal4:**

  - **Add an F1 tally to determine inward & outward partial currents at the outer edge of the fuel & cladding**

  - Need to use F1 & C1 cards

    - **Produce two tallies, the first with a cosine $-1 \leq \mu \leq 0$, and the second with a cosine $0 \leq \mu \leq 1$.**

    - **Subtract the first from the second, and the result is the net outward current through the top of the cube.**

- **Run it & examine the tally output**

# Reaction Rate Tallies

# Reaction Rates

- Reaction rate tallies can be obtained using the tally multiplier card
- Tally process produces an
  {effective microscopic cross section} x {flux}, $(\sigma\phi)$
- To obtain reaction rate, result must be multiplied by appropriate number density
- Number density can be supplied as first entry (C) on tally multiplier card
- $C < 0 \Rightarrow$ replace C with |C| times number density in cell where tally is taken

# Example of Reaction Rate Tallies

- **What if we want to determine the absorption rate in the fuel in Problem tal4, for fast & thermal energies?**

  ( absorption = capture + fission )

- **What if we want to determine the neutron production rate from fission in the fuel in Problem tal4 , for fast & thermal energies?**

or

```
e0   .625e-6   20.
c
f14:n 10
fm14 -1.0 100 (-2)(-6) T
c
f24:n 10
fm24 -1.0 100 (-6 -7)
```

```
e0   .625e-6   20.
c
f14:n 10
fm14 -1.0 100 (-2:-6)
c
fm24:n 10
fm24 -1.0 100 (-6 -7)
```

# Isotopic Reaction Rate Tallies

- Isotopic reaction rate edits may be obtained only by defining a new material that contains that isotope
- The isotope may or may not be present in the cell(s) in which the tally occurs
- Recommendation: do not combine tallies over more than one cell unless the cells have the same composition

# Example of Isotopic Reaction Rate Tally

- Suppose we want to find the $^{238}U$ fission rate in our fuel cell problem, tal4

$N_{28}$ = 2.2380e-2 (from atom density on m100 card)

```
F34:n  10
FM34   2.2380e-2   910   -6
C
c dummy material, for isotopic tally
M910  92238.60c   1.0
```

# Tally Plotting

# Tally Plotting with MCPLOT

| | |
|---|---|
| **TALLY n** | Define tally n as the current tally to plot. |
| **FREE q** | Plot variable q along the horizontal axis, where q is F for cell/surface bins, M for FM bins, E for energy bins, etc. |
| **FIXED q n** | Set n as the bin number for fixed variable q. |
| **LINLIN** | Use a linear x and y axis. |
| **LOGLIN** | Use a linear x axis and a logarithmic y axis |
| **TFC x** | Plot the tally fluctuation chart of the current tally, where x is M for mean, E for relative error, F for figure-of-merit, etc. |

Use the MPLOT card to plot during execution:

```
MPLOT  TALLY 4  FREE E  FIXED M 3 LOGLIN
```

# Tally Plotting Exercise

## Exercise tal4a

- Copy file tal4 to tal4a
- Add an mplot card, to show a tally a a function of nps (TFC…)
- Run the problem

- Card added to tal4a

```
mplot  tally 14   tfc m
```

# Mesh Tallies

# Mesh Tallies

- **Mesh tallies cover 3D regions of space independent of the problem geometry**
  - Can be used to tally flux, reaction rates, heating, particle birth, …
  - Rectangular & cylindrical meshes
  - Bin on energy values
  - Unlimited number of meshes
  - Size of mesh limited by computer parameters
  - Rotated by using a TR card
  - Modified by DE/DF or FM cards
  - Plot results in MCNP5/X

MCNP5 & MCNPX:

- Type 1: Pathlength tallies

MCNPX:

- Type 2: Source Mesh Tally
- Type 3: Energy Deposition Mesh Tally
- Type 4: DXTRAN Mesh Tally

# Mesh Tally Card (MCNP5)

**FMESHn:p GEOM= ORIGIN= AXS= VEC= IMESH= IINTS= JMESH= JINTS= KMESH= KINTS= EMESH= EINTS= FACTOR= OUT= TR=**

- Can be used with DEn, DFn, and FMn cards.
- Caution: It is easy to create huge mesh tallies that can overflow computer memory.

| | | DEFAULTS |
|---|---|---|
| GEOM | = mesh geometry: Cartesian (xyz or rec) or cylindrical (rzt or cyl) | xyz |
| ORIGIN | = x,y,z coordinates in MCNP cell geometry superimposed mesh origin | 0. 0. 0. |
| AXS | = direction vector of the cylindrical mesh axis | 0. 0. 1. |
| VEC | = direction vector, along with AXS that defines the plane for angle theta=0 | 1. 0. 0. |
| IMESH | = coarse mesh locations in x (rectangular) or r (cylindrical) direction | --- |
| IINTS | = number of fine meshes within corresponding coarse meshes | 1 |
| JMESH | = coarse mesh locations in y (rectangular) or z (cylindrical) direction | --- |
| JINTS | = number of fine meshes within corresponding coarse meshes | 1 |
| KMESH | = coarse mesh locations in z (rectangular) or theta (cylindrical) direction | --- |
| KINTS | = number of fine meshes within corresponding coarse meshes | 1 |
| EMESH | = values of coarse meshes in energy | all energies |
| EINTS | = number of fine meshes within corresponding coarse energy meshes | 1 |
| FACTOR | = multiplicative factor for each mesh | 1. |
| OUT | = blank: three 2D matrices written to OUTP: ij, ik, and jk, where | col |
| | ij = xy or rz 2D matrices, ik = xz or r θ 2D matrices, jk = yz or zθ 2D matrices | |
| | = col column format for mesh tallies written to OUTP | |
| | = cf col output + mesh volumes and tallies multiplied by mesh volumes | |
| TR | = transformation number to be applied to the tally mesh | |

# Mesh Tally Plotter Commands (MCNP5)

```
mcnp5  run=runfile  z
```

Then:

**FMESH n**

**n = mesh tally number**

**This command invokes the mesh tally plotter.**

**ZLEV $n_1$ $n_2$…**

**$n_i$ = "log", "lin", and/or values in increasing order.**

**Adjusts the scaling of the mesh plot.**

**FMRELERR n**

**n = mesh tally number or blank**

**Plots the relative errors of the mesh bins.**

**EBIN n**

**n = energy bin number (not value)**

**Plots a specific energy bin of the mesh tally**

## Track Averaged Mesh Tally (MCNPX type 1)

**FORM: (R,C,S)MESHn:*<pl>*  keyword = value**

**n = 1, 11, 21, 31,...**

(note, number must not duplicate one used for an ‘F1’ tally)

*<pl>* is a particle type. There is no default.

**Example:**

```
tmesh
    rmesh1:n    flux
    cora1  -15.0 100i 15.0
    corb1  -15.0 15.0
    corc1  -30.5 100i 30.5
endmd
```

## Mesh Tally Keywords (MCNPX Type 1)

| Keyword | Description |
|---|---|
| **TRAKS** | Tally the number of tracks through each mesh volume. No values accompany the keyword. |
| **FLUX** | Tally the average fluence (particle weight times track length divided by volume) in units of number/cm$^2$.<br><br>If the source is considered to be steady state in particles per second, then the value becomes flux in number/cm$^2$-s. |
| **TRANS** | Translate and/or rotate the mesh, according to the specified TR card. Must be followed by a single TR card number. |

Additonal keywords:

DOSE, POPUL, PEDEP, MFACT

## Source & Energy Deposition Mesh Tallies (MCNPX types 2 &3)

**Source Mesh Tally:**

**Form:** **(R,C,S)MESH*n* *<pl$_1$> <pl$_2$>...<pl$_n$>* trans = #**

**n =** 2, 12, 22, 32, ...

(note, number must not duplicate one used for an 'F2' tally)

***<pl>*** = particle type(s) (Up to 10 allowed)

**Energy Deposition Mesh Tally:**

**General Form:** **(R,C,S)MESHn** ***keyword***

***n = 3, 13, 23, 33, ...***

***Example:*** ***Mesh tally of total energy deposited, all sources***

```
tmesh
     RMesh3 total
     cora3  -15.0 100i 15.0
     corb3  -15.0 15.0
     corc3  -30.5 100i 30.5
endmd
```

## Energy Deposition Mesh Tally Keywords (MCNPX)

| Keyword | Description |
|---|---|
| TOTAL | If TOTAL appears on the input line, score energy deposited from any source. (DEFAULT) |
| DE/DX | If DE/DX appears on the input line, score ionization from charged particles. |
| RECOL | If RECOL appears on the input line, score energy transferred to recoil nuclei above tabular limits. |

Additional keywords

TLEST, DELCT, MFACT, NTERG, TRANS (see the manual)

## MCNPX Flux And Source Mesh Tally Exercise

1. Copy **SOLUTIONS/talmh** file to working folder

2. Plot and understand the geometry.

3. Add a rectangular flux mesh tally for protons and neutrons within the water. Use one bin in the "y" direction.

4. Add a rectangular source mesh tally for protons and neutrons within the water.

5. Plot your results with the MCNPX plotter.

# MCNPX Mesh Tally Plotting

- **From MDATA files**
  - Use gridconv and postprocessor (e.g Moritz, Tecplot, PAW, etc.)

**OR,**

- **From MCTAL files make a contour plot**
  - MCNPX z
  - <mcplot> plot
  - rmc = <mctal filename>
  - tal n free ik contour 5 95 10 %
    - this tells MCNPX to plot tally "n", set the plot indices to your mesh tally coordinates (ik=xz), contour colors where blue=5<95=red, with 10 percent interpolates in between.

**OR Superimpose on Geometry:**

- **From INP file during run:**
  - **mplot freq 5000 PLOT ex 40 py 4 la 0 1 tal color on la 0 0**
    **(see manual for mplot command detail)**

**OR,**

- **From runtpe:**
  - **mcnpx z run = <runtpe filename>**
  - **<mcplot> plot $brings up the geometry plotter**
  - **[buttons] tal, N, color**

## FORM: CONTOUR [cmin cmax cstep] [commands]

### All command entries are optional

| | |
|---|---|
| cmin | minimum contour value |
| cmax | maximum contour value |
| cstep | number of contour steps |
| | |
| % or pct | interpret step values as percentages |
| log | step values logarithmic with cstep interpolates |
| All | contours normalized to min and max values of entire tally |
| noall | contours normalized to min and max values of contour slice (FIXED command) |
| line/noline | do/don't draw lines around contours |
| color | make color contour plot |
| nocolor | contour lines only |

# MESH PLOT CONTOUR COMMAND examples (MCNPX)

## EXAMPLES

```
CONTOUR  5 95 10 & line color
```

**There will be 10 contour lines at 5%, 15%,…95% of the maximum value.**

**Lines will be drawn around the colored contours as in Figure 1.**

**Note: this is the default setting**

```
CONTOUR 1e-4  1e-2  12 log
```

**There will be 12 contour lines logarithmically spaced between 1e-4 and 1e-2**

# DXTRAN Mesh Tally (MCNPX type 4)

**General Form:** **(R,C,S)MESHn:*<pl>*** **trans = #**

**n = 4, 14, 24, 34, ...**

(note, number must not duplicate one used for an ‘F4’ tally)

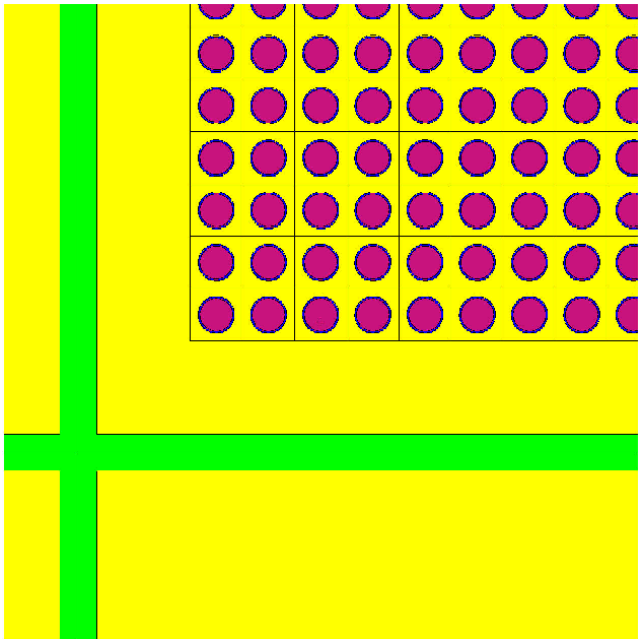
*<pl>* is a particle type. There is no default.

**trans** must be followed by a single reference to a TR card that can be used to translate and/or rotate the entire mesh. Only one TR card is permitted with a mesh card.

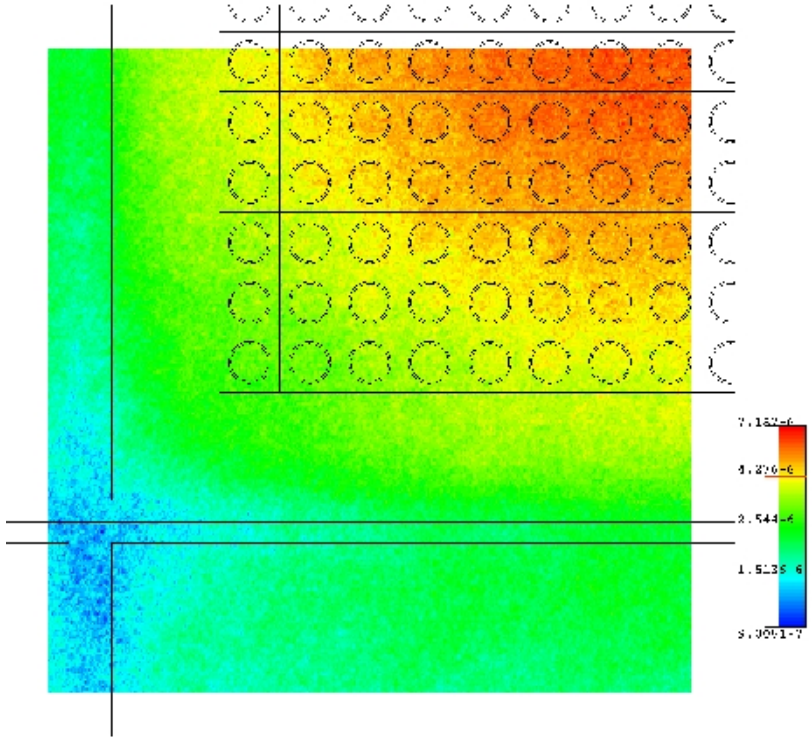
# Specifying materials on the FM card

**Some calculations, such as energy deposition, require the use of the FM card. However, the FM card requires the user to identify a material whose reaction cross sections are used in the tally. This is impractical for a tally that contains several materials.**

Fuel Storage Vault



Neutron Fluence



# Example of the FM card (MCNP5)

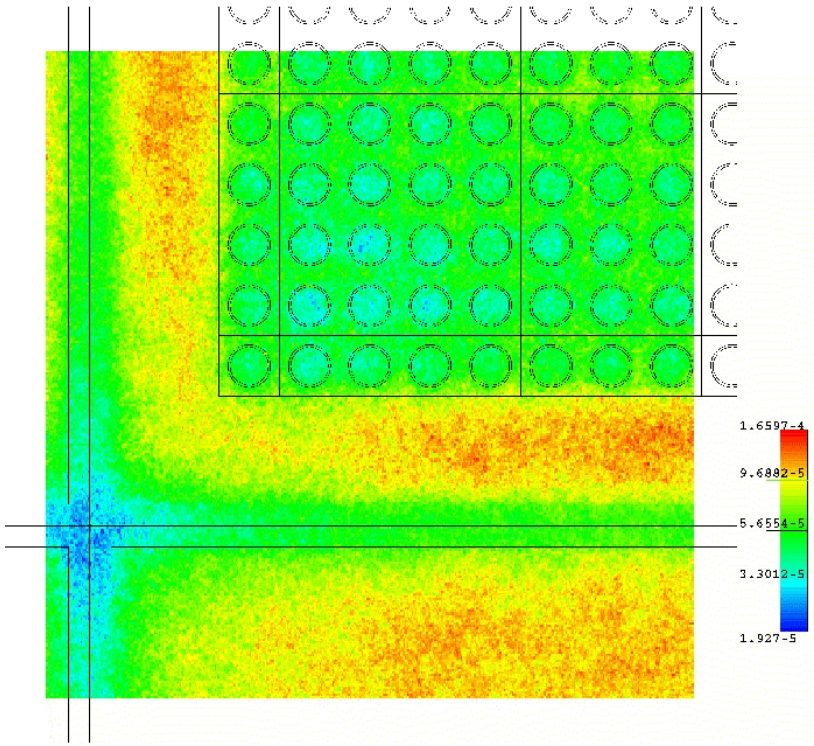
## Calculate the average fission energy deposition

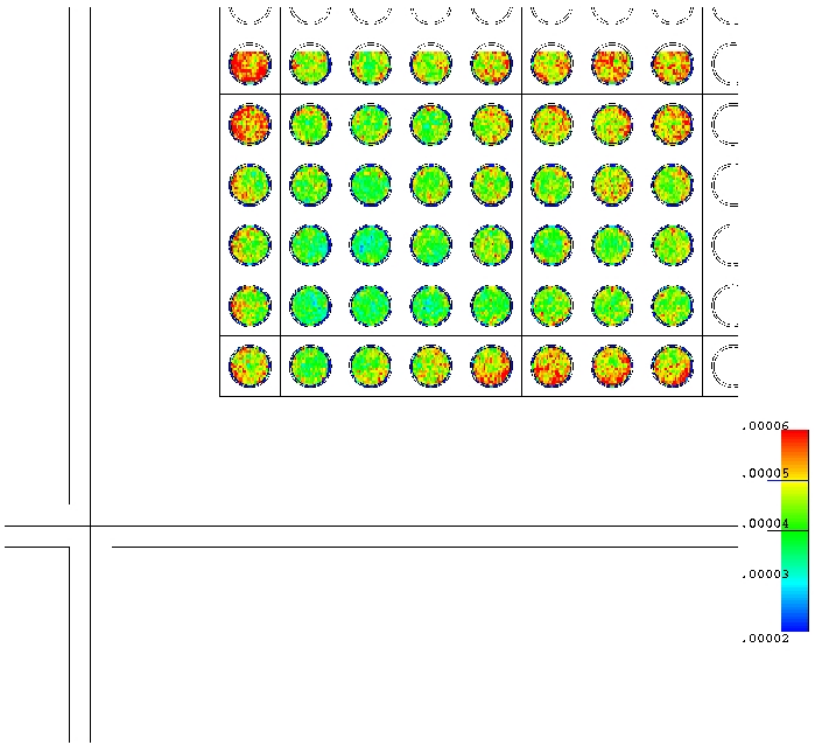
FM card format: FMn C m R1 R2

**Put '0' as the material number**

FM124 0.06925613 1 -6 -8

FM134 -1.0 **0** -6 -8



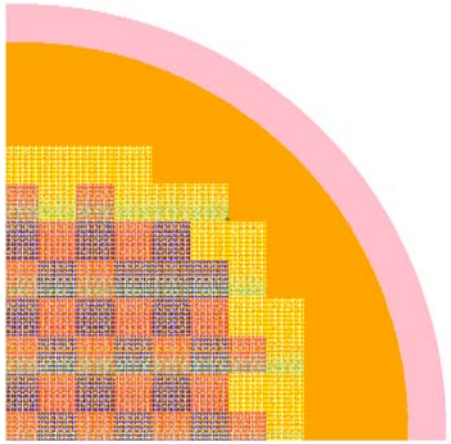
# Example (MCNP5)

- For Watts-Bar Unit #1, 3D whole-core model

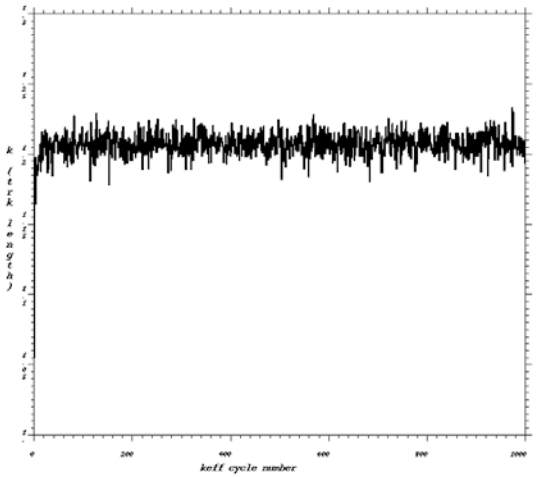
```
c
c   Mesh tally for assembly powers   (flux*fission*Q)
fmesh104:n  geom=xyz  origin= -161.25 -161.25 -194.492
            imesh=161.25 iints=15
            jmesh=161.25 jints=15
            kmesh=170.508 kints=1
fm104 -1.0  0  -6 -8      $ NOTE: 0 = use actual materials
c
c  Mesh tally for detailed fast & thermal flux
fmesh204:n  geom=xyz  origin= -161.25 -161.25 -194.492
            imesh=161.25 iints=150
            jmesh=161.25 jints=150
            kmesh=170.508 kints=1
            emesh  .625e-6  20.
```

# Example (cont'd)

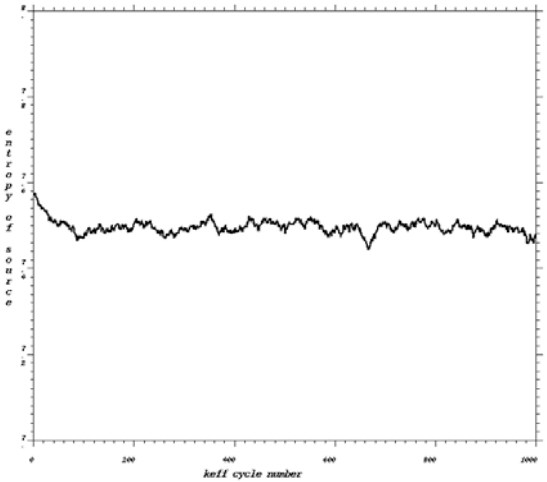
## Geometry Model (1/4)



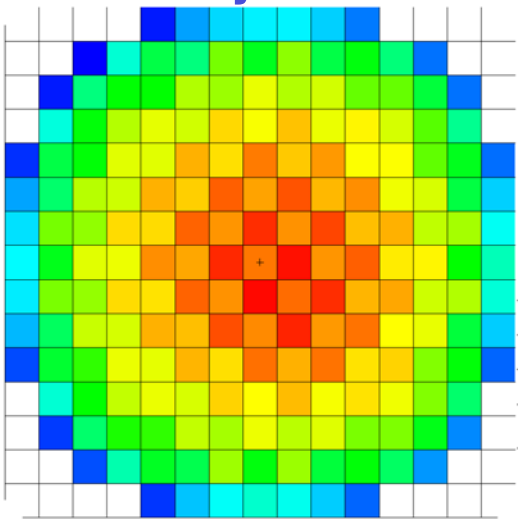
## K vs cycle



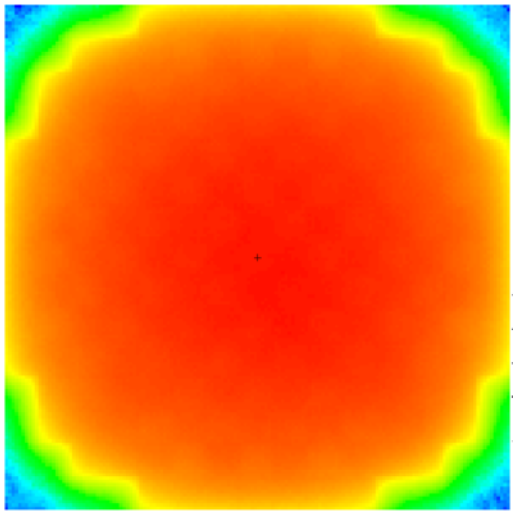

## $H_{src}$ vs cycle



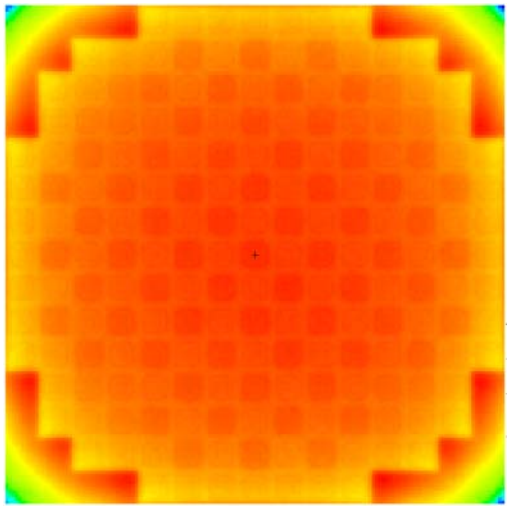

## Assembly Powers



## Fast Flux



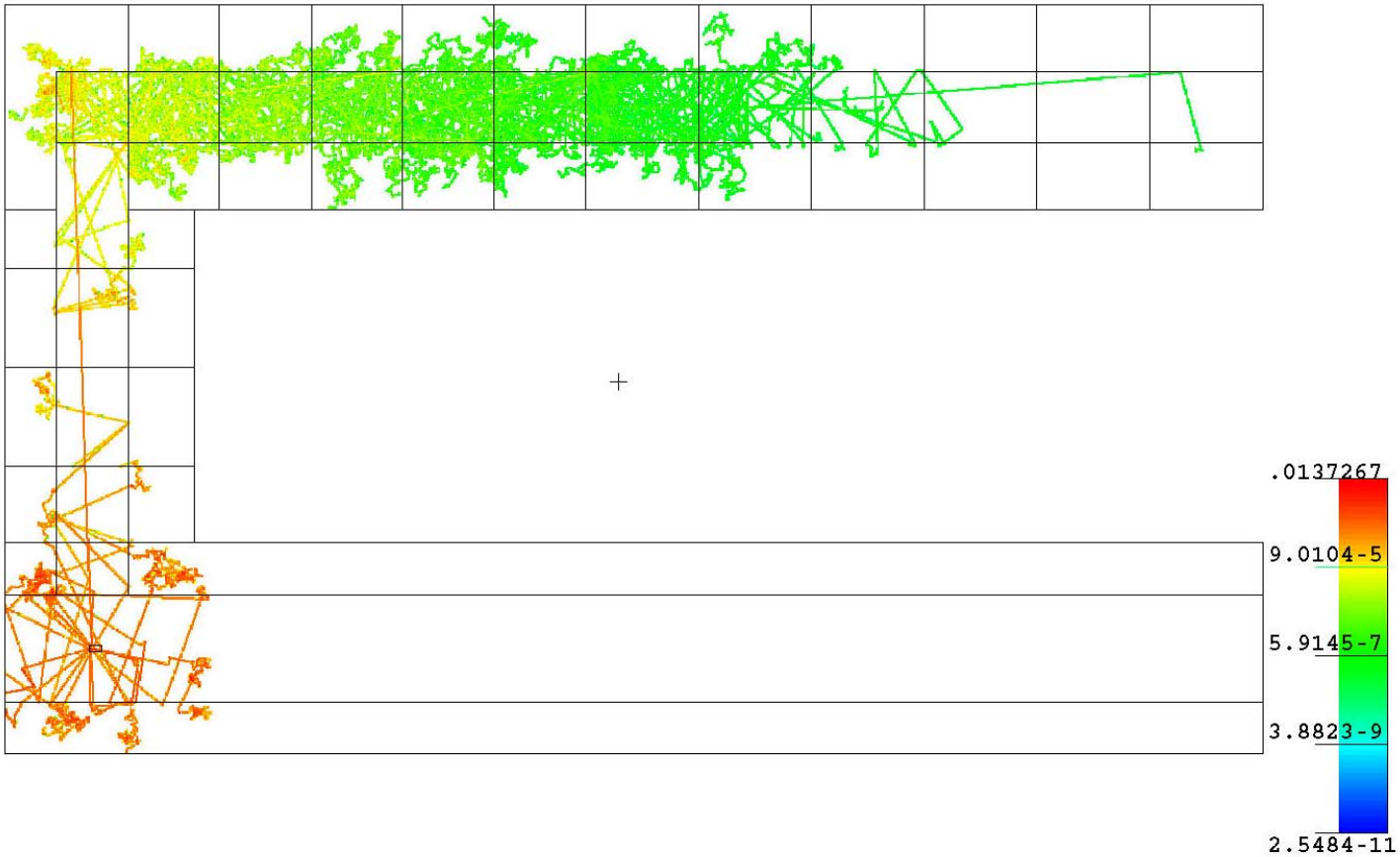
## Thermal Flux



# Plotting Particle Tracks

**By using a very fine mesh, particle tracks from individual histories can be plotted.**

2000 x 1100 x 1 mesh



# Mesh Tally Example - tal6 (MCNP5)

- **Copy fuel assembly problem fv4 to file tal6**

- **Edit tal6**
  - **Define a 15 x 15 x 1 mesh tally that covers all unit fuel cells**
  - Add an FM card to get the fission rate in each mesh cell
  - For fun, add: mplot fmesh 4 freq 5

- **Run the problem, then examine the meshtal file, & then use the mcnp5 plotter to plot the power distribution for the assembly using the mesh tally plots**

  mcnp5 r=runtpe z

  Then

  fmesh 4

  …..

- **To get smooth plots, change to: iints=100 jints=100 and rerun/replot**

# Example - tal6 (MCNP5)

- Add these lines to contents of fv4

```
c
c power distribution using mesh tallies
fmesh4:n   geom=xyz  origin= -10.5 -10.5 -180.
        imesh= 10.5     iints= 15
        jmesh= 10.5     jints= 15
        kmesh= 180.     kints= 1
fm4  -1.0  0   -6
c
mplot  fmesh 4  freq 5
```

# Pulse Height Tallies

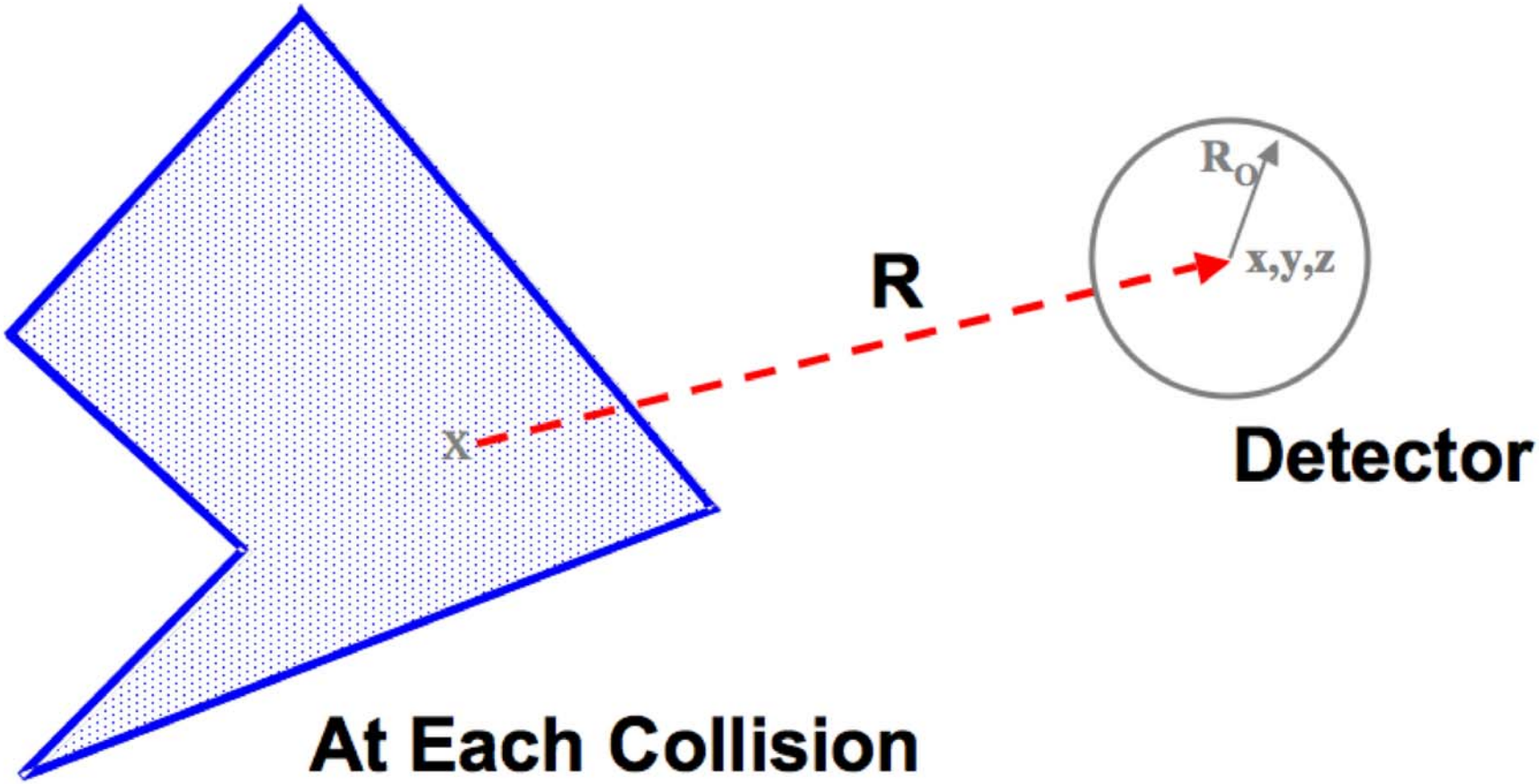
# Pulse Height Tally

## F8:*<pl>* $C_1$ … $C_n$

- **Different from all other tallies**
  - Surface estimator of cell energy deposition
  - Can use variance reduction with F8
  - Energy is accumulated from all tracks of a particle's history
- **Mimics pulse-height detectors: energy bins contain pulses**
  - Energy <0: non-analog negative score balance
  - Energy ~ 0: particles pass through without energy loss
  - Energy > 0: pulse of W put into appropriate energy bin

# Point and Ring Detectors

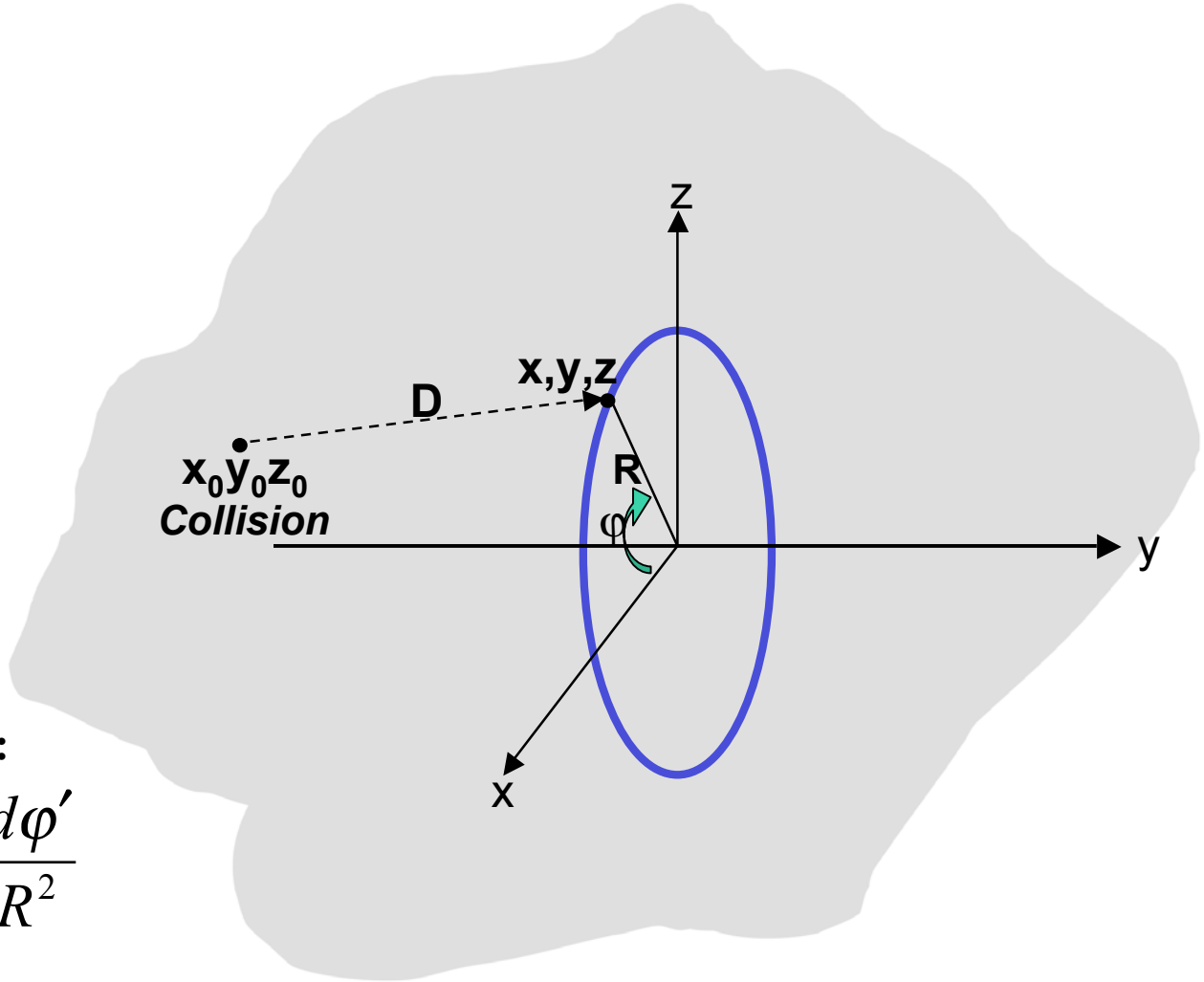
# Point Detectors




**At Each Collision**

$$\Phi = Wp(\mu)e^{-\lambda}/(2\pi R^2)$$

# Ring Detector




**Sample from:**

$$\xi = \frac{C}{2\pi}\int_{-\pi}^{\varphi}\frac{d\varphi'}{R^2}$$

**Adjust weight:**

$$W' = \frac{D^2(\varphi)}{A}W$$

# Detector Cards

- **Point Detectors**

  **F5:<pl> X Y Z $R_O$**

- **Ring Detectors**

  **F5x:<pl> X R $R_O$**

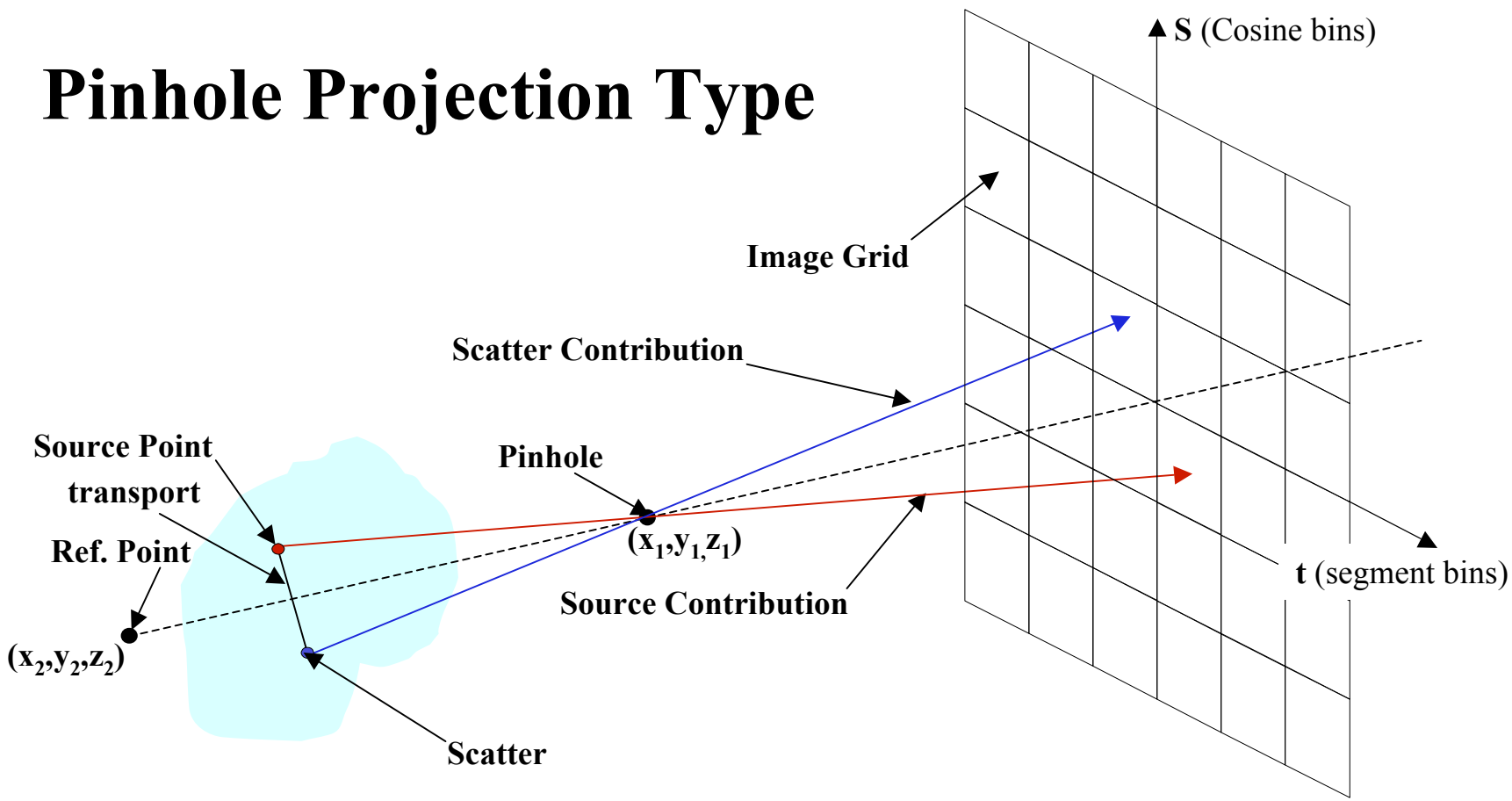
- **Radiography Tallies**
  - **MCNPX**

    **FI5:<pl> $X_1$ $Y_1$ $Z_1$ $R_O$ $X_2$ $Y_2$ $Z_2$ $F_1$ $F_2$ $F_3$**
    **F5y:<pl> Y R $R_O$**
    **F5z:<pl> Z R $R_O$**
  - **MCNP5**

    **FIRn:<pl> $X_1$ $Y_1$ $Z_1$ $R_O$ $X_2$ $Y_2$ $Z_2$ $F_1$ $F_2$ $F_3$**
    **FICn:<pl> $X_1$ $Y_1$ $Z_1$ $R_O$ $X_2$ $Y_2$ $Z_2$ $F_1$ $F_2$ $F_3$**
    **FIPn:<pl> $X_1$ $Y_1$ $Z_1$ $R_O$ $X_2$ $Y_2$ $Z_2$ $F_1$ $F_2$ $F_3$**

# Radiography Tallies

# Pinhole Projection Type



## Radiography Tally (Pinhole Projection type)

**General Form:**

| | | | | | |
|---|---|---|---|---|---|
| **MCNPX:** | **PIn:*<pl>*** | $X_1\ Y_1\ Z_1$ | $R_0$ | $X_2\ Y_2\ Z_2$ | $F_1\ F_2\ F_3$ |
| **MCNP5:** | **FIPn:*<pl>*** | $X_1\ Y_1\ Z_1$ | $R_0$ | $X_2\ Y_2\ Z_2$ | $F_1\ F_2\ F_3$ |

**FS*n* -20. 99i 20.** ***$ establishes an image grid with***

**C*n* -20. 99i 20.** ***$ 100 Seg. x 100 Cos. bins***

**n** is the tally number and must end with a 5 since this is a detector-type tally.

***<pl>*** is the particle type for the tally.
*Neutrons or photons only!*

**(see next slide for explanation of Argument elements)**

## Pinhole Radiography Argument Descriptions

**$X_1$, $Y_1$, $Z_1$** The coordinates of the pinhole.

**$R_0$** Always 0 (zero) for this application.

*Note,* neither the pinhole nor the grid should be located within a highly scattering media.

**$X_2$, $Y_2$, $Z_2$** The reference coordinates that establish the reference direction cosines for the normal to the detector grid. This direction is defined as being from **$X_2$, $Y_2$, $Z_2$** to the pinhole at **$X_1$, $Y_1$, $Z_1$** .

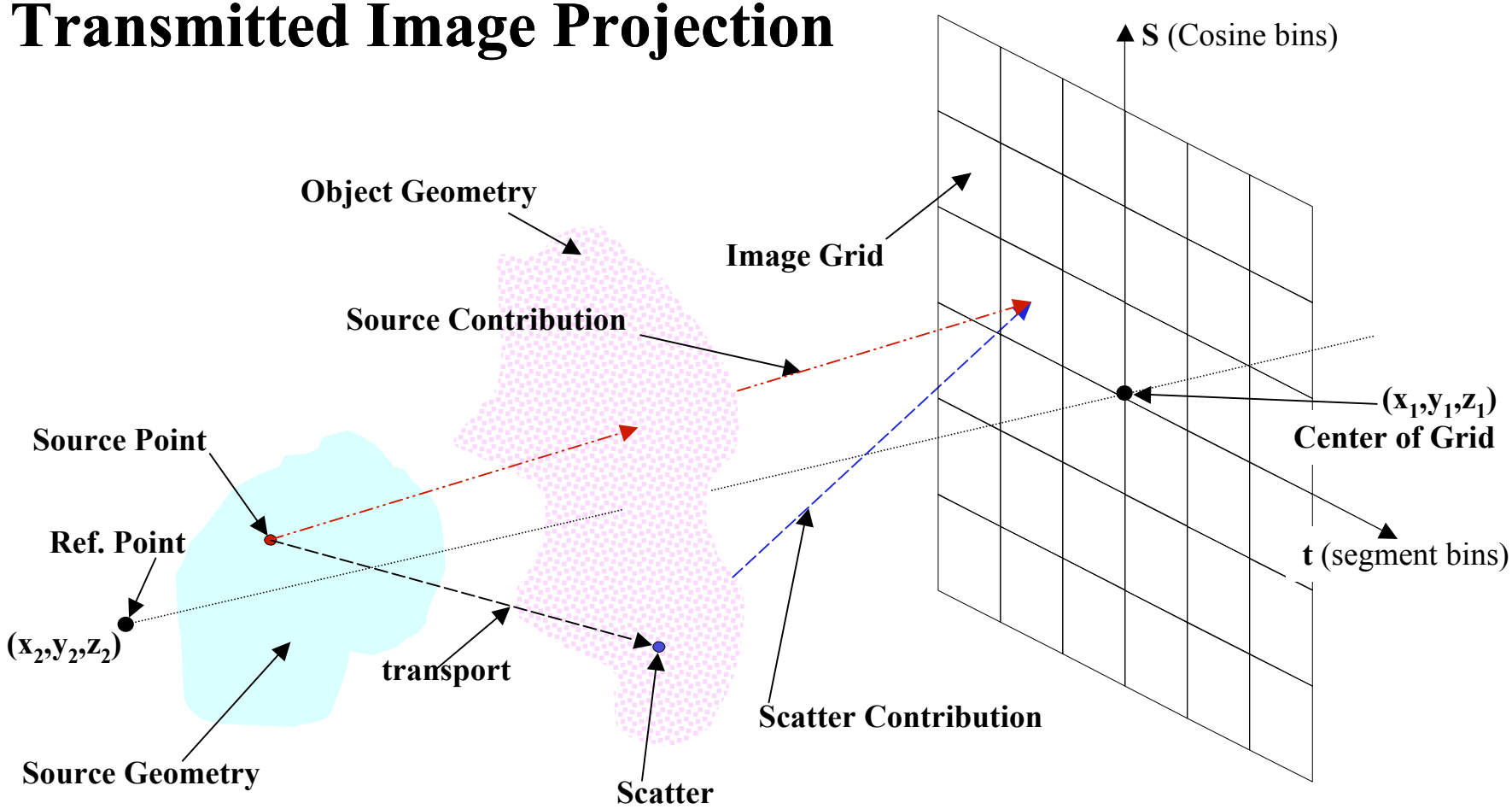
**$F_1$** If **F1**>0, the radius of a cylindrical collimator, centered on and parallel to the reference direction, which establishes a radial field of view through the object.

**$F_2$** The radius of the pinhole perpendicular to the reference direction.

- **F2**=0 represents a perfect pinhole
- **F2**>0 the point through which the particle contribution will pass is picked randomly. This simulates a less-than-perfect pinhole.

**$F_3$** The distance from the pinhole at **$X_1$, $Y_1$, $Z_1$** to the detector grid along the direction established from **$X_2$, $Y_2$, $Z_2$** to **$X_1$, $Y_1$, $Z_1$** , and perpendicular to this reference vector.

# Transmitted Image Projection



# Radiography Tally (Transmitted Image Projection Type)

## General Form:

**MCNPX:**

**TIRn:*<pl>*** $X_1\ Y_1\ Z_1$ $R_0$ $X_2\ Y_2\ Z_2$ $F_1\ F_2\ F_3$

**TICn:*<pl>*** $X_1\ Y_1\ Z_1$ $R_0$ $X_2\ Y_2\ Z_2$ $F_1\ F_2\ F_3$

**MCNP5:**

**FIRn:*<pl>*** $X_1\ Y_1\ Z_1$ $R_0$ $X_2\ Y_2\ Z_2$ $F_1\ F_2\ F_3$

**FICn:*<pl>*** $X_1\ Y_1\ Z_1$ $R_0$ $X_2\ Y_2\ Z_2$ $F_1\ F_2\ F_3$

**..*R* is used to establish a grid on a *plane surface***

..*C* is used to establish a grid on a ***cylindrical surface***.

| | | |
|---|---|---|
| n | = | the tally number and must end with a 5 since this is a detector type tally. |
| *<pl>* | = | the particle type for the tally. (N or P only) |
| $X_1\ Y_1\ Z_1$ | = | Center of rect. or cyl. grid defined with FSn and Cn |
| $R_0$ | = | 0.00 |
| $X_2\ Y_2\ Z_2$ | = | reference point defining rectangluar grid outward normal of cylindrical grid axis |
| $F_1$ | = | -1/0 Scattered contribution only/Source + scattered contributions |
| $F_2$ | = | radial field of view |
| $F_3$ | = | 0 /1 Contributions to grid bin centers/random **positions** |

## Transmitted Image Projection (NPSMG on the NPS card)

```
NPS        NPP NPSMG
```

NPP = number of histories requested

NPSMG = number of ***direct source contributions*** requested

**Example:**

```
NPS 100000 60000
```

# More Statistics

**Review: Statistics & Confidence Intervals**

**Statistical Checks in MCNP**

# Central Limit Theorem

- **Monte Carlo computes overall results by averaging together the tallies from many independent trials (histories)**
  - Histories must be <u>independent</u> of each other
  - Histories must all sample random variables from the same probability densities (ie, <u>identically distributed</u>)

- **Central Limit Theorem**

  The distribution of the **sum** of N independent, identically-distributed random variables **approaches a normal distribution** as N becomes large

  For tally X, let $\bar{x} = \frac{1}{N}\sum_{k=1}^{N} x_k$ and $\mu$ = exact mean, $\sigma$ = exact STD DEV

  Then the central limit theorem is:

$$\text{Prob}\left\{\mu - \alpha\frac{\sigma}{\sqrt{N}} \le \bar{x} \le \mu + \alpha\frac{\sigma}{\sqrt{N}}\right\} \xrightarrow[\text{large N}]{} \frac{1}{\sqrt{2\pi}}\int_{-\alpha}^{+\alpha} e^{-t^2/2}dt$$

$\alpha = 1$ gives 68% Confidence Interval

$\alpha = 2$ gives 95% CI

$\alpha = 2.6$ gives 99% CI

# Confidence Intervals

- **Confidence interval**
  - Using the computed STD DEV as an estimate of $\sigma$, we can estimate the probability that the true mean lies with an interval:

$$\text{Prob}\left\{ \bar{x} - 1 \cdot s_{\bar{x}} \le \mu \le \bar{x} + 1 \cdot s_{\bar{x}} \right\} = 68\%$$

$$\text{Prob}\left\{ \bar{x} - 2 \cdot s_{\bar{x}} \le \mu \le \bar{x} + 2 \cdot s_{\bar{x}} \right\} = 95\%$$

$$\text{Prob}\left\{ \bar{x} - 2.6 \cdot s_{\bar{x}} \le \mu \le \bar{x} + 2.6 \cdot s_{\bar{x}} \right\} = 99\%$$

  - Think about what this means …..
    - If you repeat a calculation many times, it is likely that 1/3 of the time the true result will lie outside of the computed 1-$\sigma$ confidence interval

# Practical Guidelines for Tally Results

- **Check the relative error (R):**
  - For a point detector tally (F5), should have $R < .05$
  - For other tallies, should have $R < .10$

- **Check the tally fluctuation charts**
  - Should have a stable Figure-Of-Merit, $FOM = 1 / (R^2T)$
  - Should have a stable value for the mean (ie, no obvious trends)

- **Form a confidence interval**
  - 68% chance that true mean is within X-S to X+S
  - 95% chance that true mean is within X-2S to Xn+2S
  - 99% chance that true mean is within X-2.6S to X+2.6S

# MCNP - Ten Statistical Checks

**MCNP performs 10 statistical checks on tallies to try & assess whether they are valid results**

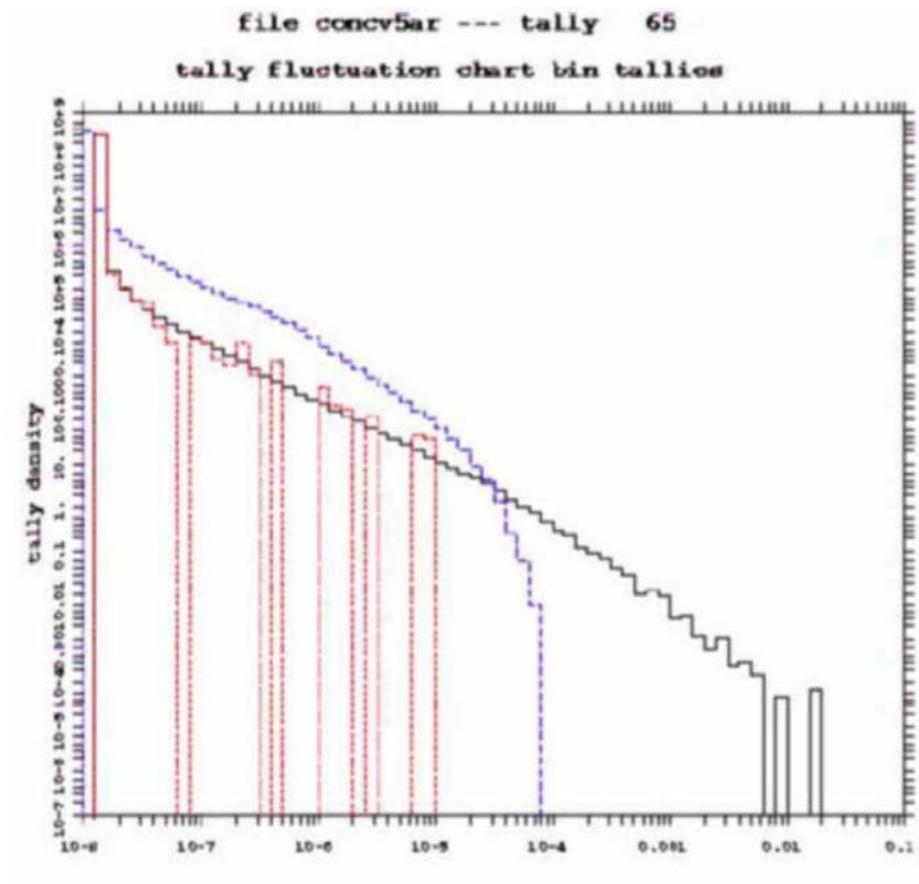
1. Estimated **mean** should have random behavior for last half of problem
2. Estimated **relative error** should be <.05 for point detector, <.10 for others
3. Estimated **RELERR** should monotonically decrease in last half of problem
4. Estimated **RELERR** should decrease as $1/N^{1/2}$ in last half of problem
5. Estimated **variance of the variance** (VOV) should be < .10
6. Estimated **VOV** should monotonically decrease in last half of problem
7. Estimated **VOV** should decrease as 1/N in last half of problem
8. Estimated **FOM** should not have obvious trends in last half of problem
9. Estimated **FOM** should show random behavior in last half of problem
10. Tail of **tally probability density** should fall off as $1/x^m$, with $m>3$

# History Score Tally Probability Density

- **f(x)dx is the probability that a given Monte Carlo history will score a tally contribution between x and x+dx per unit tally score x**

- **MCNP computes empirical f(x) automatically:**
  - 600 bins (60 decades with 10 log bins/decade)
  - $f_J(x)$ = (number of scores in bin J) / (N**Δ**x)

- **At least the first 2 moments of f(x) must exist (be finite) in order to satisfy the Central Limit Theorem**
  - A condition for this is that the tails of f(x) must fall off faster than $1/x^3$
  - The "slope m" (on a log-log plot of $f_J(x)$ vs x) must be > 3



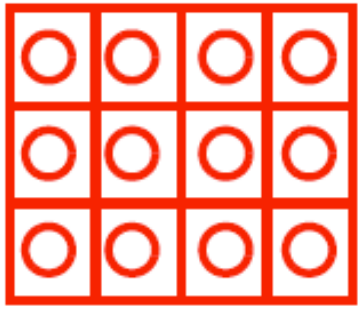
# MCNP - Ten Statistical Checks

```
==========================================================

     results of 10 statistical checks for the estimated
     answer for the tally fluctuation chart (tfc) bin of tally   2

tfc bin     --mean--      ---------relative error---------
behavior    behavior      value   decrease   decrease rate

desired      random       <0.10      yes      1/sqrt(nps)
observed     random        0.00      yes          yes              . . .
passed?        yes          yes      yes          yes
==========================================================


        ==================================================================
        ----variance of the variance----      --figure of merit--     -pdf-
        value   decrease   decrease rate       value     behavior     slope
        <0.10      yes        1/nps           constant    random      >3.00
. . .    0.00      yes         yes            constant   increase     10.00
          yes      yes         yes               yes        no         yes
        ==================================================================

warning.  the tally in the tally fluctuation chart bin did not pass
           1 of the 10 statistical checks.
```

# Tallies & Sources in Repeated Structures

# Tallies in Repeated Structures

- **What' s special about a tally or source in a repeated structure?**
  - Must provide the path to tally or source cell
  - Which pin cell ??

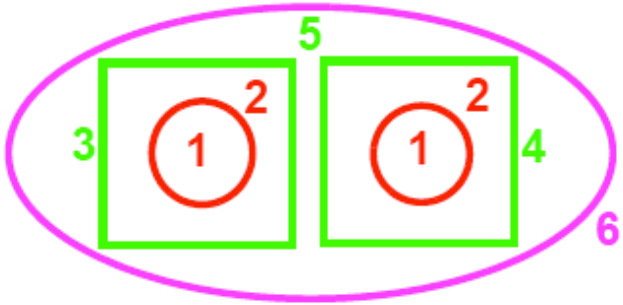


- **Enables tallies or sources in specific cells of repeated structures.**

# Purpose and Definitions

- **Example - levels**

```
c cell cards
 1   0      -11   u=1
 2   0       11   u=1
 3   0      -22    fill=1                    u=2
 4   0      -33    fill=1   trcl(1 0 0)      u=2
 5   0       22 33                           u=2
 6   0      -44    fill=2
```




– **Cell 6 is in the "real world", not a universe - called "Level 0"**
**Cells 3,4,5 are in Universe 2, which fills Cell 6 - called "Level 1"**
**Cells 1,2 are in Universe 1, which fills Cell 3 and Cell 4 - called "Level 2"**

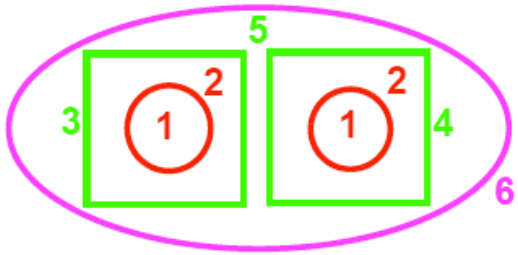
– **The left pin (Cell 1) is contained in Cell 3, that is contained in Cell 6.**

– **Use the symbol "<" to mean "is contained in":**

**1 < 3 < 6 <-- This uniquely identifies the left pin**
**1 < 4 < 6 <-- This uniquely identifies the right pin**

# Geometric Paths & Tally Paths

- **Geometric chain**
  - A list of cells, one in each level of geometry, that uniquely specifies a particular cell in a repeated structure
  - Start at the deepest level of geometry, finish in the real world
- **Tally path**
  - A subset of a geometric chain




- **Example**

  **C1 < C2 == "Cell C1 is contained in Cell C2"**

  - All geometric paths: 1<3<6 2<3<6 1<4<6 2<4<6
    3<6 4<6 5<6 6
  - Possible tally paths:

    1<3<6 <-- the left pin
    1<4<6 <-- the right pin
    1<6 <-- BOTH pins

# Tally Paths

**FORM:** **Fn:p ( E1 < C1 < C2 ) ...**

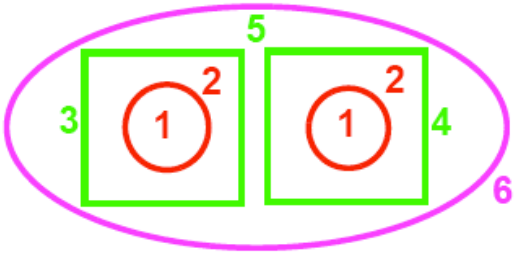
- Left arrow (<) identifies levels within a tally chain, translate it to: **"is contained in"**

- <u>Requires</u> an outer set of parentheses.

- First level entries (Ei) are either:
  - **Tally <u>surfaces</u> if tally type 1 or 2.**
  - **Tally <u>cells</u> if tally type 4, 6, 7, or 8**

- Upper level entries (Ci) must be:
  - **<u>Cells</u> with a FILL entry that is nonzero.**
  - **Produces a tally only when a particle is in a geometric chain that corresponds to an input tally chain.**

# Tally Paths




```
c cell cards
 1   0      -11   u=1
 2   0       11   u=1
c
 3   0      -22    fill=1                  u=2
 4   0      -33    fill=1   trcl(1 0 0)   u=2
 5   0       22 33                        u=2
 6   0      -44    fill=2

C surface cards
. . .

C data cards -- flux tallies
F4:n        (1<3<6)        <-- the left pin
            (1<4<6)        <-- the right pin
            (1<6)          <-- BOTH pins
```

# Tally Paths - Lattices

**Form: Fn:p ( (E2 E3) < ( C3 C4 [I1 ... I2] ) < (C5 C6) ) ...**

- **Brackets, [ ], identify one or more elements of a lattice:**
  - Must follow a <u>filled</u> <u>lattice cell</u> (C4 ).
  - A lattice cell listed without brackets gives the union

- **Three possible formats: over all lattice elements.**

[ I ]

Indicating the I-th lattice element in fully specified FILL array.

[ I1:I2 J1:J2 K1:K2 ]

Indicating one or more lattice elements (see FILL card).

[ I1 J1 K1, I2 J2 K2, ... ]

Indicating lattice element (I1, J1, K1), (I2, J2, K2), etc.

# Tallies - Multiple Bin Format

- **Automatically invoked for levels with multiple entries.**
- **Can be disabled at any level by ( ) around all entries.**
- **Number of total bins generated is given by the product of the number of bins at each level.**
- **The order of generated tally bins can be important,**
  - ( E4 E5 < (C1 C2) < C3 C4 ) becomes:

    **(E4 < (C1 C2) < C3)** **(E5 < (C1 C2) < C3)**
    **(E4 < (C1 C2) < C4)** **(E5 < (C1 C2) < C4)**

  - A segment divisor entry (SD card) may be input for each generated tally bin.

# Example - tal7

- **Modify Problem fv5 to get tallies in every fuel pin**
  - **Copy fv4 to tal7**
    - **Note: fv4 was a 15x15 array of pins, with vacuum BC**
    - **The center pin is [0 0 0], so lattice indices are in the range:**

      **-7:7 -7:7 0:0**
  - **Edit tal7:**
    - **Add F4 tallies of fission heating in Cell 10 in each lattice location**
    - **Add FM4 card, with reactions -6 -8**
    - **Use the SD4 card to enter volumes of 1 for all 225 pins, so that units are MeV, not Mev/gram (note: 1 224r )**
  - **Run the problem & examine the output**

# Sources in Repeated Structures

- KSRC Format
- SDEF CEL Format

# KSRC Format

- **No special format for repeated structures**
- **Enter x,y,z locations in the coordinate system of the highest level (the real world)**
- **All source points are absolute coordinates, ie, "real world" coordinates**

Ksrc 1. 1. 1. 2. 3. 4. 5. 6. 7. . . .

# SDEF CEL Format

## From the manual, pages 3-59 - 3-61:

- The coordinate system for position and direction sampling (pds) is the coordinate system of the first negative or zero Ci in the geometric path starting from the right and proceeding left.

- Each entry in the source path represents a geometry level, where level zero is the last source path entry, level one the second to the left, etc., and level zero is above level one, level two is below level one.

- The pds level is the level associated with the pds cell or pds coordinate system. All levels above the pds level must be included in the source path. Levels below the pds level need not be specified, and when given, may include one or more zero entries.

- The default pds level is the first entry in the source path when the path has no negative or zero entry.

- Position rejection is done in cells at all levels where $Ci \neq 0$, but if any Ci has a negative universe number on its cell card and is at or above the pds level, higher level cells are not checked.

# SDEF CEL Format

- To use the SDEF card to select a particular cell inside a lattice,

```
SDEF     x=d1   y=d2   z=d3    cel=d4
… (xyz position disributions) …
SI4   L   ( geometric-path-to-cell )
SP4     1
```

- Example - center cell in Problem tal7

```
SDEF     x=d1  y=d2  z=d3   cel=d4
SI1        -.7   .7
SP1         0    1
SI2        -.7   .7
SP2         0    1
SI3        -180. 180.
SP3         0    1
SI4   L    ( -10 < 40[ 0 0 0 ] < 50 )        <-- xyz coordinates in "Level 2"
SP4         1
```

# SDEF CEL Format

- **Example - all fuel pins in Problem fv5**

```
SDEF     x=d1  y=d2  z=d3   cel=d4
SI1      -.7   .7
SP1       0    1
SI2      -.7   .7
SP2       0    1
SI3      -180. 180.
SP3       0    1
SI4   L  ( -10 < 40[-7:7  -7:7  0:0]< 50 )        $  xyz coordinates in "Level 2"
SP4        1   224r
```

**Notes:**

(1) 15 x 15 x 1 = 225 lattice elements listed on SI4

(2) Must give (relative) probability for each on SP4

(3) The minus sign in SI4 indicates which coordinate system the x,y,z sampling is done in (Level 2)

(3) Despite the order on the SDEF card, the CEL parameter is chosen first, then x, then y, then z