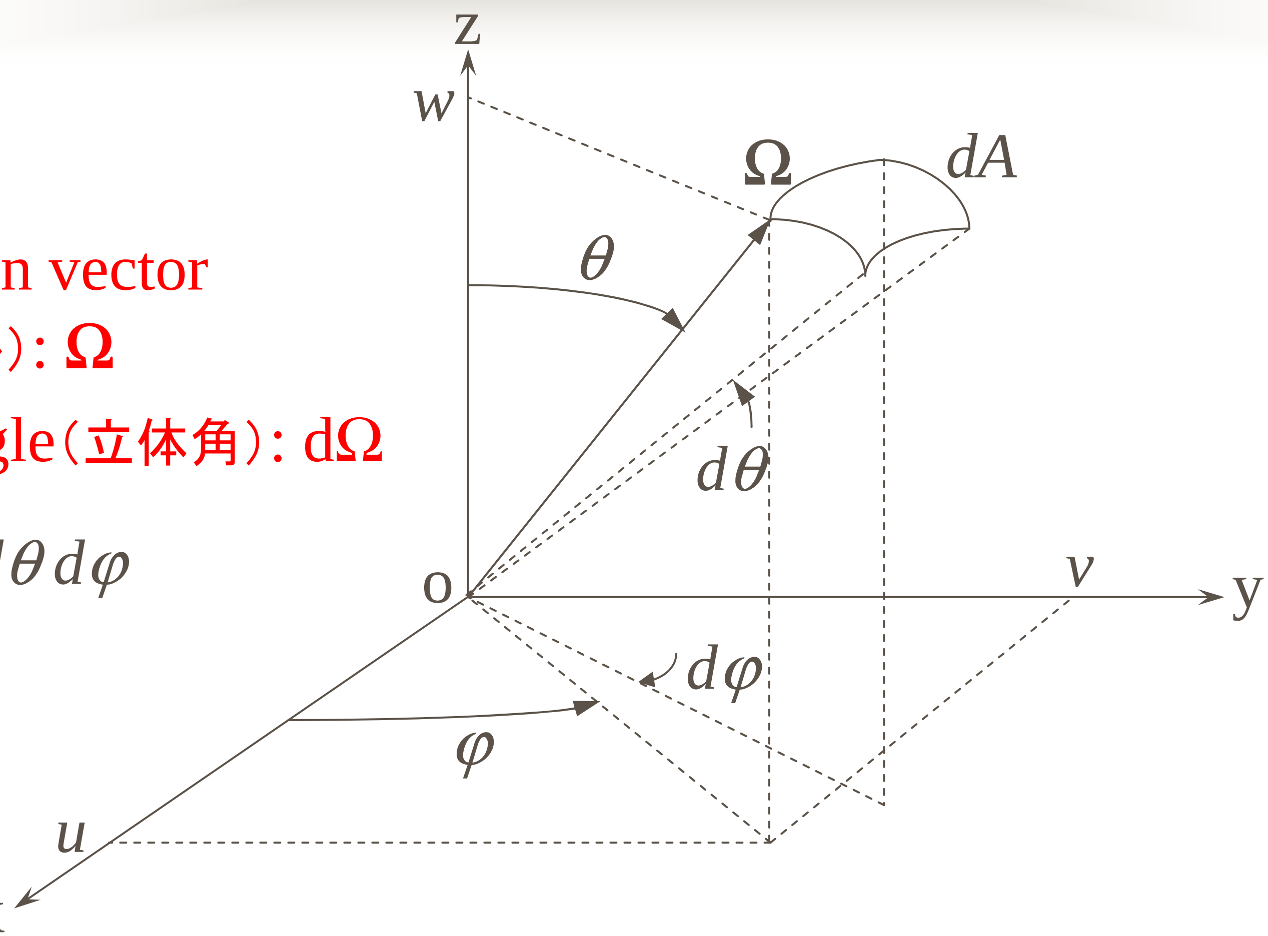
Directions and Solid Angles

The direction vector
（方向ベクトル）: $\mathbf{\Omega}$

The solid angle（立体角）: $d\Omega$

$$dA = \sin\theta\, d\theta\, d\varphi$$



$$\mathbf{\Omega} = \mathbf{i}u + \mathbf{j}v + \mathbf{k}w = \mathbf{i}\sin\theta\cos\varphi + \mathbf{j}\sin\theta\sin\varphi + \mathbf{k}\cos\theta$$

$d\Omega = \sin\theta\, d\theta\, d\varphi$: dimensionless but assigned *steradians*, *sr*

$$\int d\Omega = \int\int \sin\theta\, d\theta\, d\varphi = 4\pi$$

Measures of Radiation Intensity

- Particle Densities（粒子密度）

$$n\,(\mathbf{r}, E, \mathbf{\Omega}, t)\;\; dE\, d\Omega$$

the number of particles per unit volume at space point **r** and time *t* having energies in *dE* about energy *E* and directions in *d*Ω about the unit direction vector **Ω**.

steady-state, or time-independent definition

$$n\,(\mathbf{r}, E)\, dE = \int_{4\pi} n\,(\mathbf{r}, E, \mathbf{\Omega})\, d\Omega\, dE$$

total particle density

$$n\,(\mathbf{r}) \;=\; \int_{4\pi} \int_0^\infty n\,(\mathbf{r}, E, \mathbf{\Omega})\, d\Omega\, dE$$

Measures of Radiation Intensity

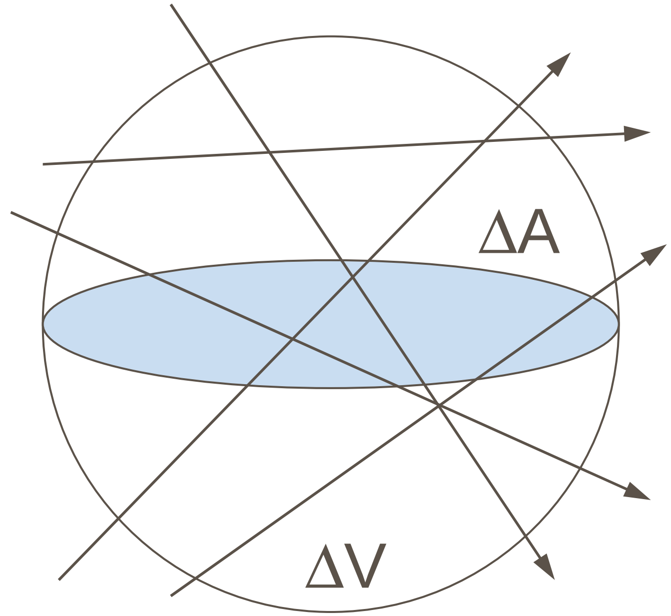
- Flux Densities（線束（密度））

$$\phi(\mathbf{r}, E, \mathbf{\Omega}, t) = v\, n(\mathbf{r}, E, \mathbf{\Omega}, t)$$

where, v is the particle's speed and corresponds to the energy E. (The speed is the scalar magnitude of the particle's velocity vector $\boldsymbol{v}$)

- Fluence（フルエンス）

$$\Phi \equiv dNp/dA = \lim_{\Delta A \to 0} [\Delta Np / \Delta A]$$



Measures of Radiation Intensity

- Flux Densities（線束(密度)）

$$\phi(\mathbf{r}, E, \mathbf{\Omega}, t) = v\, n(\mathbf{r}, E, \mathbf{\Omega}, t)$$

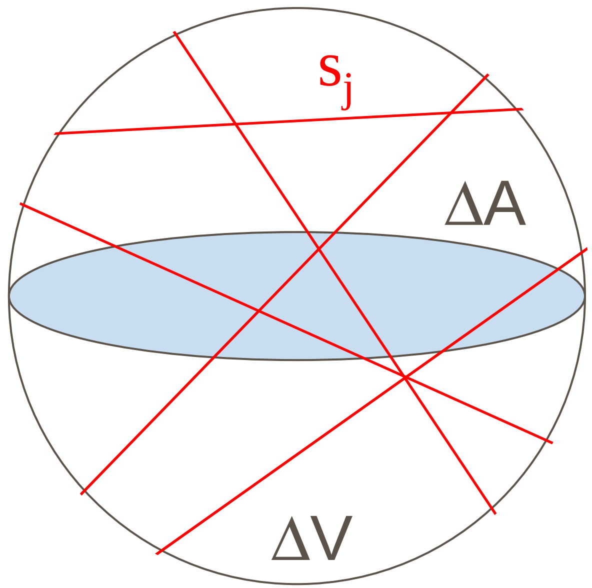
where, v is the particle's speed and corresponds to the energy E. (The speed is the scalar magnitude of the particle's velocity vector $\mathbf{v}$)

- Fluence（フルエンス）

$$\Phi \equiv dNp/dA = \lim_{\Delta A \to 0} [\Delta Np / \Delta A]$$

$$= \lim_{\Delta V \to 0} [\Sigma_i s_i / \Delta V]$$

$$\phi \equiv d\Phi/dt = d^2Np/dAdt$$



Measures of Radiation Intensity

- Current Densities

$\mathbf{J}$ ($\mathbf{r}$, v, $\mathbf{\Omega}$, t) is *angular current*, $\mathbf{J} = n\mathbf{v}$, and is defined as the directed flow per unit area (normal to the $\mathbf{\Omega}$ direction) and time at the space point $\mathbf{r}$ and time t of particles having speeds in dv about v and direction in dΩ about $\mathbf{\Omega}$.

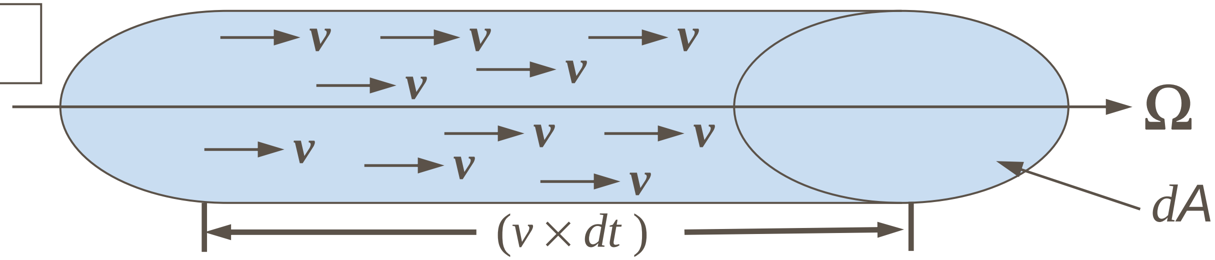
$$\mathbf{J}(\mathbf{r}, v, \mathbf{\Omega}, t)\, dv\, d\Omega = \mathbf{\Omega} v\, n(\mathbf{r}, v, \mathbf{\Omega}, t)\, dv\, d\Omega$$

where, $\mathbf{v} = v\,\mathbf{\Omega}$, $\mathbf{J}(\mathbf{r}, E, \mathbf{\Omega}, t)\, dE = \mathbf{J}(\mathbf{r}, v, \mathbf{\Omega}, t)\, dv$ and $n(\mathbf{r}, E, \mathbf{\Omega}, t)\, dE = n(\mathbf{r}, v, \mathbf{\Omega}, t)\, dv$.

$$\mathbf{J}(\mathbf{r}, E, \mathbf{\Omega}, t)\, dE\, d\Omega = \mathbf{\Omega} v\, n(\mathbf{r}, E, \mathbf{\Omega}, t)\, dE\, d\Omega$$

$$\mathbf{J}(\mathbf{r}, E, \mathbf{\Omega}, t)\, dE\, d\Omega = \mathbf{\Omega}\, \phi(\mathbf{r}, E, \mathbf{\Omega}, t)\, dE\, d\Omega$$

$v \times dt \times dA \times N$



Measures of Radiation Intensity

- Scalar Current

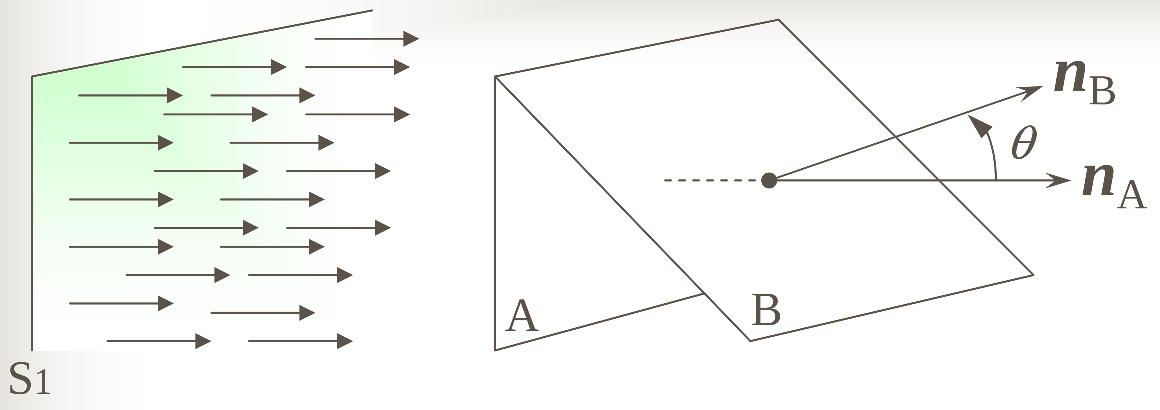
$$J_n(\boldsymbol{\Omega})\, dA \equiv \boldsymbol{J}(\boldsymbol{\Omega}) \cdot (\boldsymbol{n}\, dA)$$

A scalar current, $J_n(\boldsymbol{\Omega})$, that describes the flow of the $\boldsymbol{\Omega}$-directed particles per unit area normal to the direction $\boldsymbol{n}$.

$$J_n(\boldsymbol{\Omega}) = \boldsymbol{n} \cdot \boldsymbol{J}(\boldsymbol{\Omega}) = \boldsymbol{\Omega} \cdot \phi(\boldsymbol{\Omega}) = \cos\theta\, \phi(\boldsymbol{\Omega})$$

where $\boldsymbol{n}$ = the unit vector corresponding to an arbitrary direction,
n = a coordinate-identifying subscript, for example, $n \equiv x$ when $\boldsymbol{n} \equiv \boldsymbol{i}$,
$J_n(\boldsymbol{\Omega})$ = the flow of $\boldsymbol{\Omega}$-directed particles per unit area normal to the direction $\boldsymbol{n}$,
$\phi(\boldsymbol{\Omega})$ = the angular flux variable corresponding to the angular current $\boldsymbol{J}(\boldsymbol{\Omega})$

Measures of Radiation Intensity



A plane source S1 emits monodirectional and monoenergetic particles at the rate of 10^{10} particles cm^{-2} sec^{-1} in a direction normal to the surface.

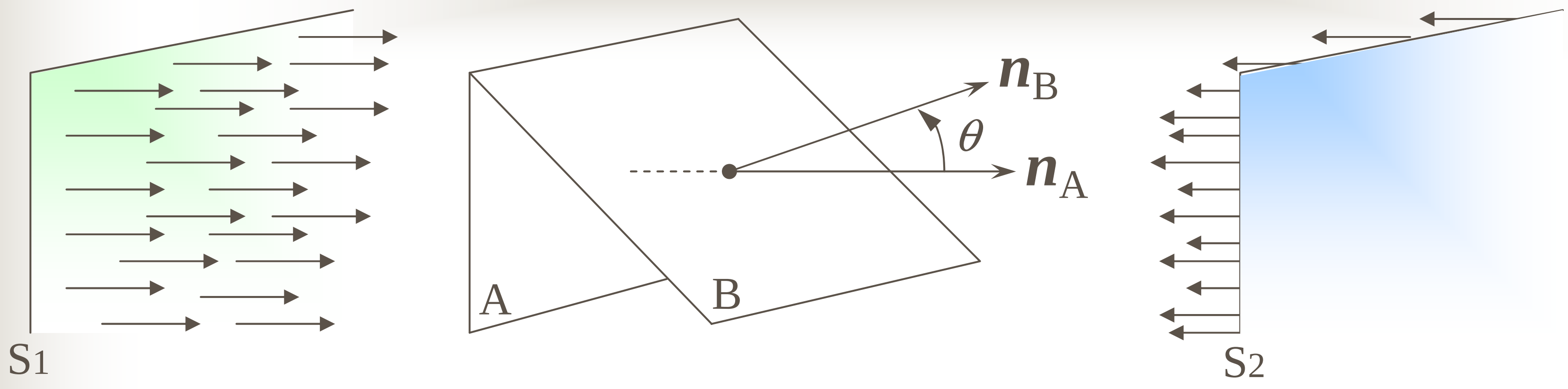
(i) At a point in a plane A whose normal $\boldsymbol{n}_A$ is parallel to that of S1.

Net current density, $\boldsymbol{J}$ in the direction $\boldsymbol{n}_A$ is 10^{10} particles cm^{-2} sec^{-1}
Flux density is 10^{10} particles cm^{-2} sec^{-1} (track length of 10^{10} cm/sec $\div$ 1 cm^3)

(ii) At a point in a plane B whose normal $\boldsymbol{n}_B$ is at angle θ with respect to $\boldsymbol{n}_A$.

Net current density, $\boldsymbol{J}$ in the direction $\boldsymbol{n}_B$ is $\boldsymbol{J}\cdot\boldsymbol{n}_B$, and is $\cos\theta \times 10^{10}$ particles cm^{-2} sec^{-1}. The flux density is not a function of the direction $\boldsymbol{n}_B$ and is 10^{10} particles cm^{-2} sec^{-1}.

Measures of Radiation Intensity



A plane source S1 emits monodirectional and monoenergetic particles at the rate of 10^{10} particles cm^{-2} sec^{-1} in a direction normal to the surface.

A plane source S2 emits monodirectional and monoenergetic particles but in a direction opposite to that of S1 at the rate of 6×10^9 particles cm^{-2} sec^{-1}.

Net current density, $\boldsymbol{J}$ in the direction $\boldsymbol{n}_A$ is 4×10^9 particles cm^{-2} sec^{-1}.
Flux density is 1.6×10^{10} particles cm^{-2} sec^{-1}.

Measures of Radiation Intensity

- **Point Source（点線源）**
  A source emitting radiation from a single point in space.
  Point-source strengths are measured in units of particles or MeV per second.
  When point sources have distributions in direction, energy, and time, the strengths have typical units of particles sec^{-1} MeV^{-1} $steradian^{-1}$.
- **Line Source（線線源）**
  A source with emission confined to a line is a line source.
  Line-source strengths are particles or MeV per second per unit length of the source. Typical units of differential distribution are particles sec^{-1} MeV^{-1} $steradian^{-1}$ cm^{-1}.
- **Surface Source（面線源）**
  A surface source is one in which radiation emanates from a plane or other two-dimensional surface. The units of source strength are particles or MeV per unit time per unit source areas (ie., particles sec^{-1} cm^{-2})
  Since surface source strengths have the same units as flux and current density, care must be taken not to confuse the three quantities.
- **Volume Source（体積線源）**
  A radiation source distributed throughout a closed surface constitutes a volume source. The source strengths are expressed in particles or MeV per unit volume per second. (ie., particles sec^{-1} cm^{-3})

Measures of Radiation Intensity

- Isotropic Distribution（等方分布）

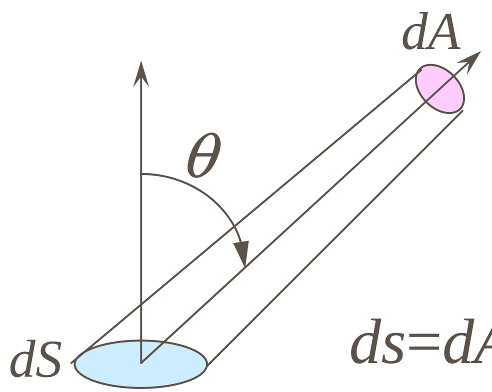
  An *isotropic* direction distribution is one in which all directions of travel are equally likely. Since there are 4π steradians of solid angle surrounding a point, the normalized isotropic differential distribution function is a constant function, $1/4\pi$ per steradian. Many nuclear reactions that cause the emission of radiations are considered to be isotropic in nature. Thus neutrons and gamma rays emitted from a fissioning nucleus, fission products, activated nuclei, and electron-positron annihilations may be assumed to be isotropic. However, certain reactions, most notably those which involve a scattering process, are not isotropic, and assuming so can lead to significant errors.

Measures of Radiation Intensity

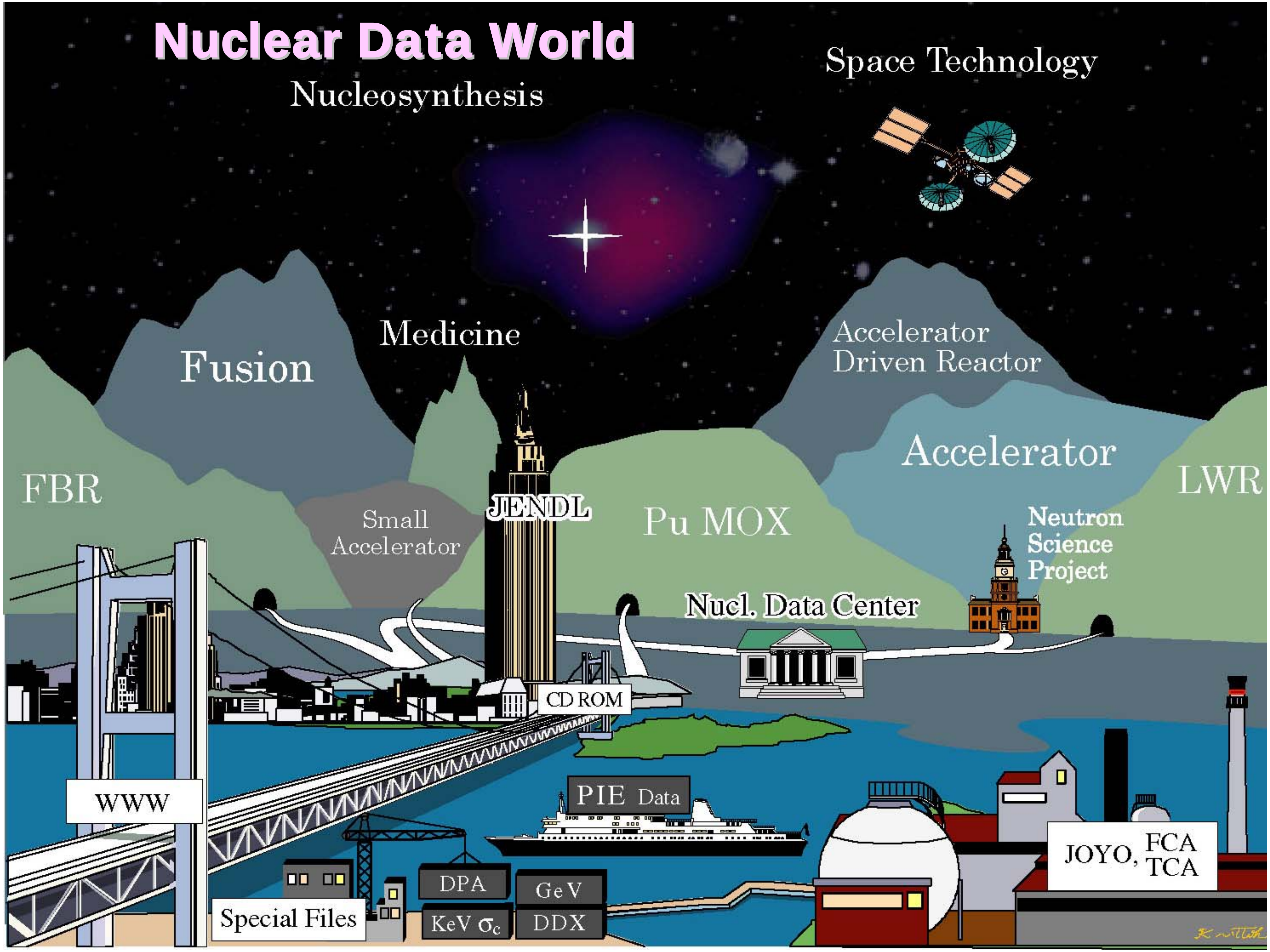
- Cosine Distribution（余弦分布）

Radiation emerging from the surface of a volume-distributed source often depends on the cosine of the angle between the normal to the surface and the direction of emergence. In many cases the dependence closely approximates or is exactly a cosine distribution of that angle. For such a source, if S_a particles sec^{-1} cm^{-2} is the source strength, the differential source angle distribution function is $(1/2\pi)S_a\cos\theta$ particles sec^{-1} $steradian^{-1}$ cm^{-2} emitted along a direction inclined at an angle θ to the normal. We must divide by $\cos\theta$ to obtain the flux density. The flux density at r is then

$$S_a/2\pi r^2 \text{ particles cm}^{-2}\text{ sec}^{-1}\text{ steradian}^{-1}$$



$$ds = dA/\cos\theta$$



Nuclear Data World
Nucleosynthesis
Space Technology
Medicine
Fusion
Accelerator
Driven Reactor
Accelerator
FBR
LWR
Small
Accelerator
JENDL
Pu MOX
Neutron
Science
Project
Nucl. Data Center
CD ROM
WWW
PIE Data
JOYO, FCA
TCA
DPA
GeV
KeV σ_c
DDX
Special Files

Nuclear Data

Overview of Evaluated Nuclear Data (1/2)

Nuclear Data : Quantitative values for nuclear reaction, decay and structure in various application fields not only nuclear engineering but also natural science, engineering, medical science, etc.

Nuclear Reaction : Cross Sections for various reactions; (n, n), (n, n'), (n, p), (n, d), (n, α),...

Nuclear Decay : Half-life, Decay mode(α, β, γ, sf), Emitted particle energy, Fission Product Yields, ...

Nuclear Structure : Level structure and energy, spin, parity, High-spin structure, ...

Overview of Evaluated Nuclear Data (2/2)

Evaluated Nuclear Data : Tabulated data evaluated (recommended) through review process by using measured data and nuclear model calculations

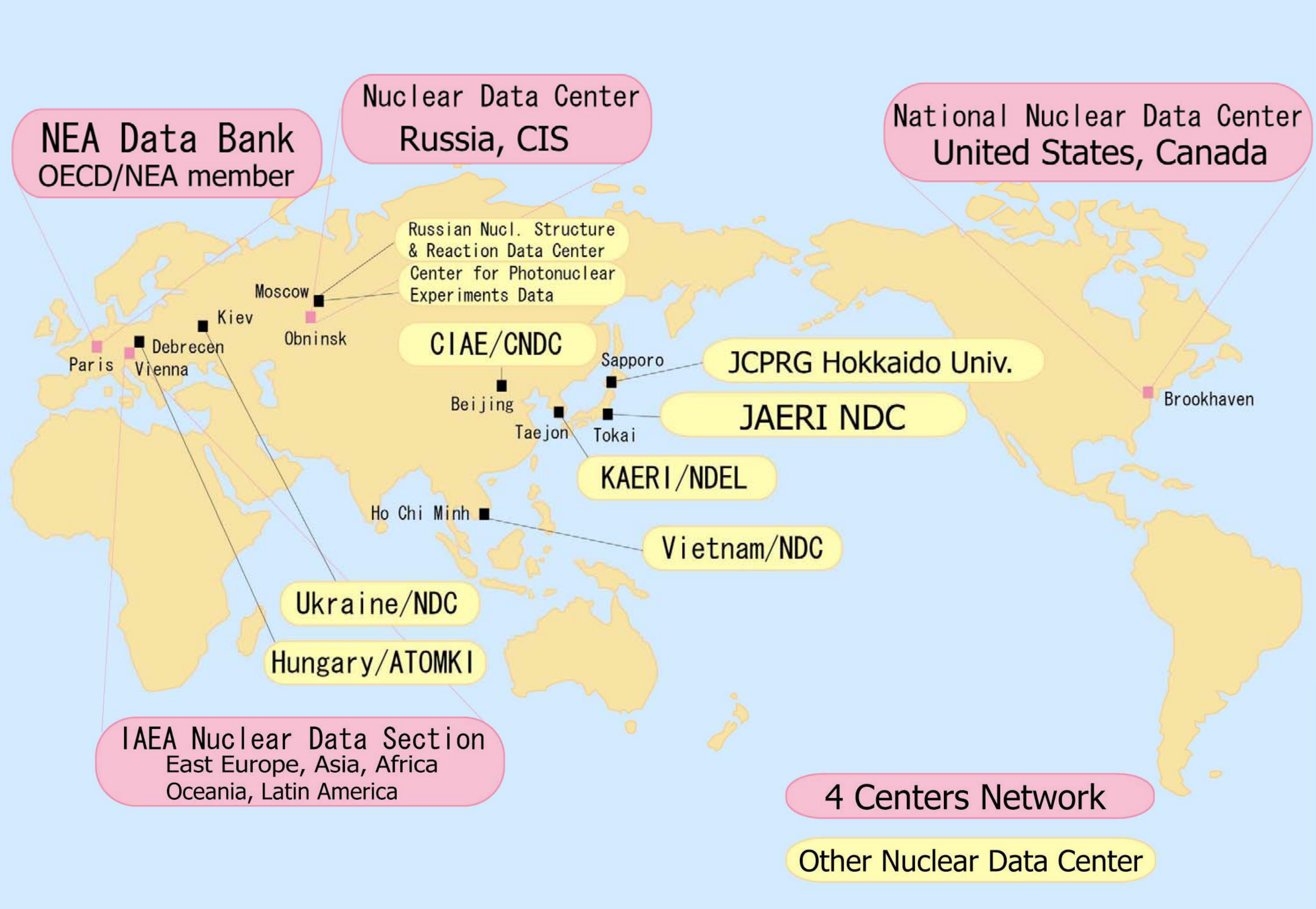
Evaluated nuclear data have been especially developed in nuclear engineering fields such as nuclear reactors and accelerators by United States, Japan, EU, Russia

ENDF/B : United States, JENDL : Japan,

JEFF : EU, Russia: BROND.

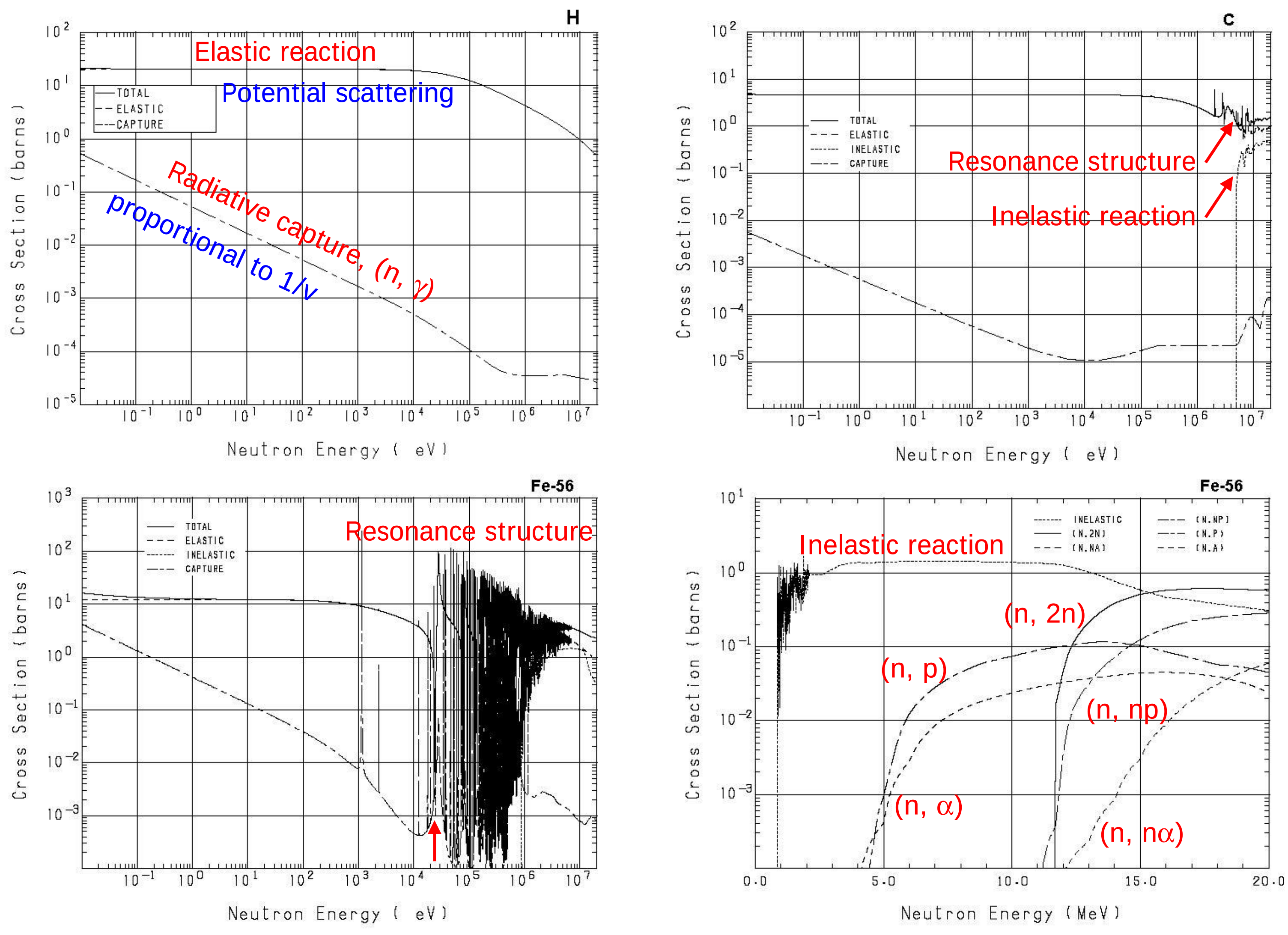
Evaluation of nuclear data is now performing in cooperation with US, Japan, EU, Russia, Korea, China, etc.

Nuclear Data Centers Network



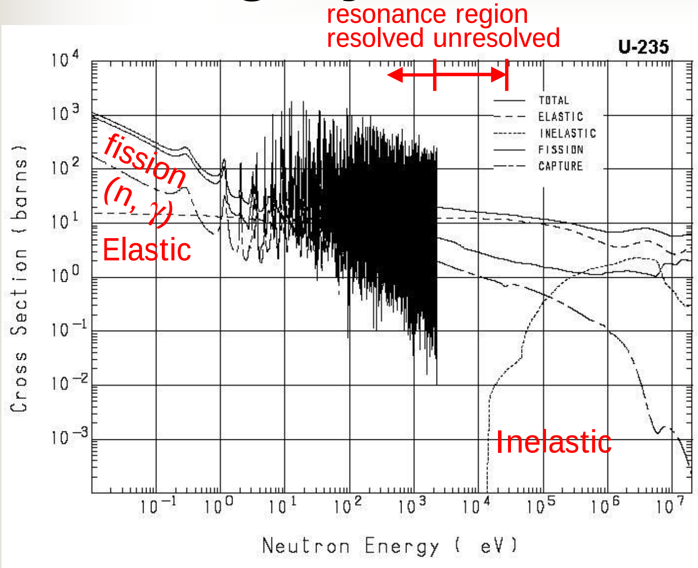
Nuclear Data

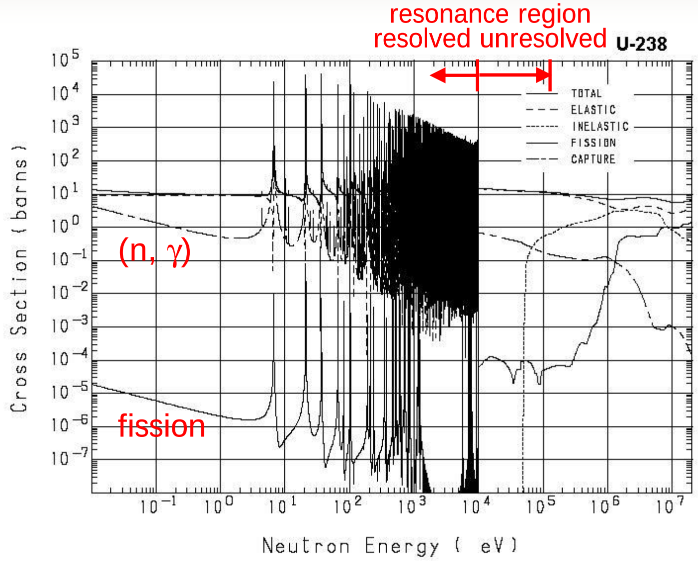
Category of nuclear reaction JENDL-3.3

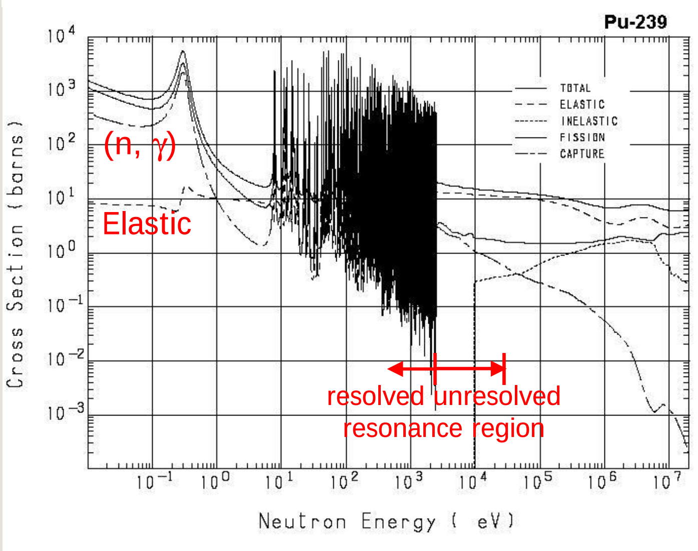


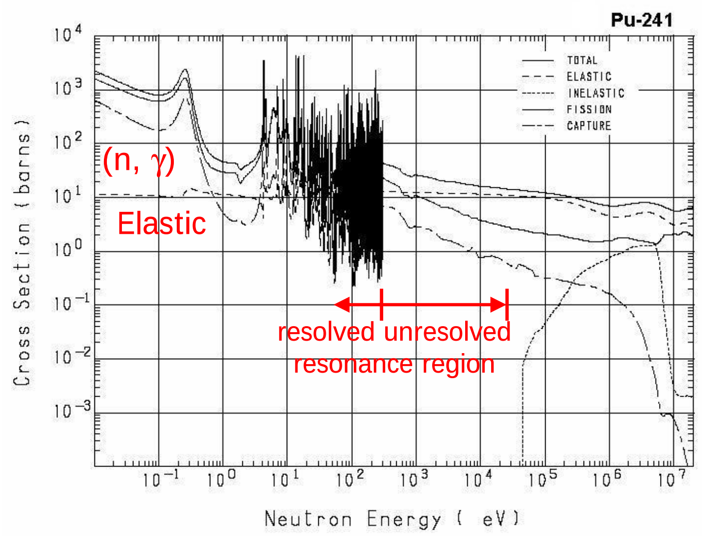
Nuclear Data

Category of nuclear reaction









Nuclear Data

Category of nuclear reaction

Elastic/Inelastic scattering, Radiative Capture (n, γ),
(n, 2n), (n, 3n),... (n, Xn),
(n, p), (n, d), (n, α), (n, np), (n, n α), (n, XnYp),...
Fission

These reaction cross sections in the evaluated nuclear data file are expressed by;

Resolved/Unresolved Resonance Parameters,

Smooth cross sections, and

Energy-Angle distributions

in ENDF-6 format.

Nuclear Data

What is the ENDF-6 format ?

ENDF is an abbreviation for Evaluated Nuclear Data File of United States. The ENDF-6 format is a computer storage format adopted in ENDF/B-VI for international use of nuclear data representation.

The format specification is determined by CSEWG*, and distributed from Brookhaven National Laboratory.

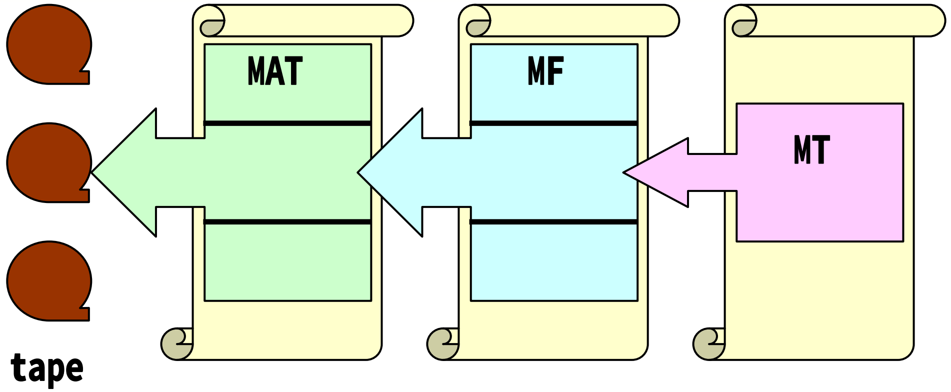
The ENDF-6 format is an ASCII text file format and has hierarchical structure consists of TAPE, MAT, MF, MT.

```
 1.400000+4 2.784400+1          0          0          0          01400 3 16
-8.473800+6-8.473800+6          0          0          1         121400 3 16
         12          2                                            1400 3 16
 8.778100+6 0.000000+0 1.000000+7 6.166000-3 1.100000+7 1.564000-21400 3 16
 1.200000+7 2.589000-2 1.300000+7 3.650000-2 1.400000+7 4.663000-21400 3 16
 1.500000+7 5.400000-2 1.600000+7 5.620000-2 1.700000+7 5.734000-21400 3 16
 1.800000+7 5.830000-2 1.900000+7 5.870000-2 2.000000+7 5.892000-21400 3 16
 0.000000+0 0.000000+0          0          0          0          01400 3  0
```

* Cross Section Evaluation Working Group

Nuclear Data

What is the ENDF-6 format ?



TAPE: collection file number which includes several materials in a magnetic tape image. (ex. Tape 511 means the standard file of ENDF/B-V)

MAT: material number (ex. 125 means H-1, 128 is H-2, 2625 is Fe-54),

MF: file number (ex. MF=3 means smooth reaction cross sections are stored),

MT: reaction identification number (ex. MT=1 means total).

What is the ENDF-6 format ?

	MAT	MF	MT	
tape id	7777	0	0	
start of MF1, MT451 (description)	1111	1	451	
...				
SEND record	1111	1	0	MT451 section end
FEND record	1111	0	0	MF1 section end
start of MF2, MT151 (resonances)	1111	2	151	
...				
SEND record	1111	2	0	
FEND record	1111	0	0	MF2 section end
start of MF3, MT1 (total cross section)	1111	3	1	
...				
SEND record	1111	3	0	MF3 section end
start of MF3, MT2 (elastic cross section)	1111	3	2	
...				
FEND record	1111	0	0	MF section end
MEND record	0	0	0	MAT section end
TEND record	-1	0	0	tape end

What is the ENDF-6 format ?

Types of records are TPID, END records(SEND, FEND, MEND, TEND), TEXT, CONT, HEAD, DIR, LIST, TAB1, and TAB2 records

Numerical data are defined in a number of 11 figures (E11.0)

Units: energy(eV), cross section(barns), temperature(Kelvin), angle($\mu=\cos\theta$), mass(neutron mass), angular distribution(μ^{-1}), energy distribution(eV^{-1}), energy-angle distribution($\mu^{-1}\,eV^{-1}$), half-lives(seconds)

ZA number is ZA =1000.0*Z+A、AWR is nuclear mass ratio (ratio of target mass to neutron mass)

What is the ENDF-6 format ?

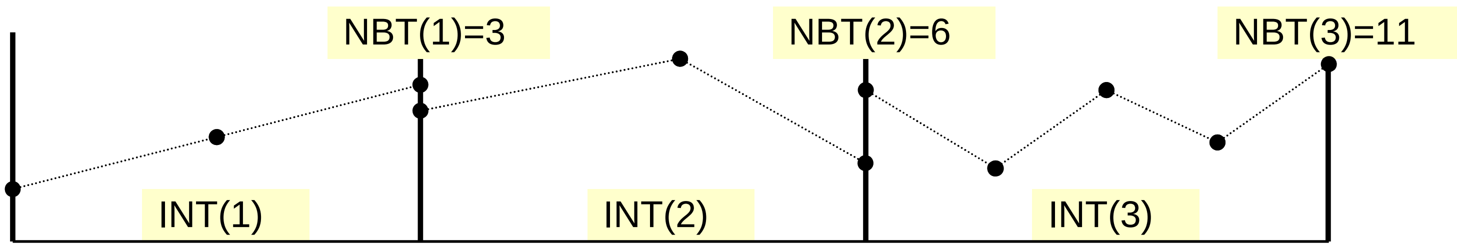
Numerical data in the ENDF-6 format are defined as discrete values, and the interpolation method is defined as a set of NBT(m) and INT(m) at the second record of TAB1 and/or TAB2 records

NBT(m): pair index separating the m^{th} and $(m+1)^{th}$ ranges

INT(m): interpolation scheme used in the m^{th} range

NP=11

NR=3



INT: 1(y is constant in x), 2(y is linear in x), 3(y is linear in ln(x), 4(ln(y) is linear in x), 5(ln(y) is linear in ln(y), 6(y obeys a Gamow charged-particle penetrability law), 11-15, 21-25(unit base interpolation (follow interpolation laws of 1-5)

Nuclear Data

MF=1 contains descriptive and miscellaneous data,
MF=2 contains resonance parameter data,
MF=3 contains reaction cross sections vs. energy,
MF=4 contains angular distributions,
MF=5 contains energy distributions,
MF=6 contains energy-angle distributions,
MF=7 contains thermal scattering data,
MF=8 contains radioactivity data
MF=9-10 contain nuclide production data,
MF=12-15 contain photon production data, and
MF=30-40 contain covariance data

MT is defined by each reaction. (ex. MT=1: total cross section, MT=2: elastic scattering, MT=16: (n,2n) reaction, MT=18: fission, MT=102: radiative capture)
MT=1 total, (Sum of MT=2, 4, 5, 11, 16-18, 22-26, 28-37, 41-42, 44-45, and 102-117)
MT=3 non-elastic, (Sum of MT=4, 5, 11, 16-18, 22-26, 28-37, 41-42, 44-45, 102-117)
MT=4 inelastic, (Sum of MT=50-91)
MT=18 fission, (Sum of MT=19-21, 38)

Nuclear Data

As an example, here is a TAB1 record for the (n,2n) reaction in natural silicon from ENDF/B-VI (MF=3, MT=16):

```
 1.400000+4 2.784400+1          0          0          0          01400 3 16
-8.473800+6-8.473800+6          0          0          1        121400 3 16
         12          2                                          1400 3 16
 8.778100+6 0.000000+0 1.000000+7 6.166000-3 1.100000+7 1.564000-21400 3 16
 1.200000+7 2.589000-2 1.300000+7 3.650000-2 1.400000+7 4.663000-21400 3 16
 1.500000+7 5.400000-2 1.600000+7 5.620000-2 1.700000+7 5.734000-21400 3 16
 1.800000+7 5.830000-2 1.900000+7 5.870000-2 2.000000+7 5.892000-21400 3 16
 0.000000+0 0.000000+0          0          0          0          01400 3  0
```

The first line is the HEAD record; it contains the ZA value (1000*Z+A) and the AWR value (ratio of target mass to neutron mass). The second card starts the TAB1 record and contains the reaction Q value (-8.4738 MeV) and some counts. The third line contains some interpolation information. Finally, the rest of the record contains the tabulation given as energy, cross section pairs with energies in eV and cross sections in barns. Therefore, we can immediately read off the 14 MeV cross section of 0.04663 barns. The last line in the section is just the SEND record.

Note that this is an endothermic reaction (negative Q value), and it has a threshold energy of 8.7781 MeV. We can also compute the threshold from the Q value using

$$-Q*(AWR+1)/AWR = 8.778131 \text{ MeV}$$

Nuclear Data

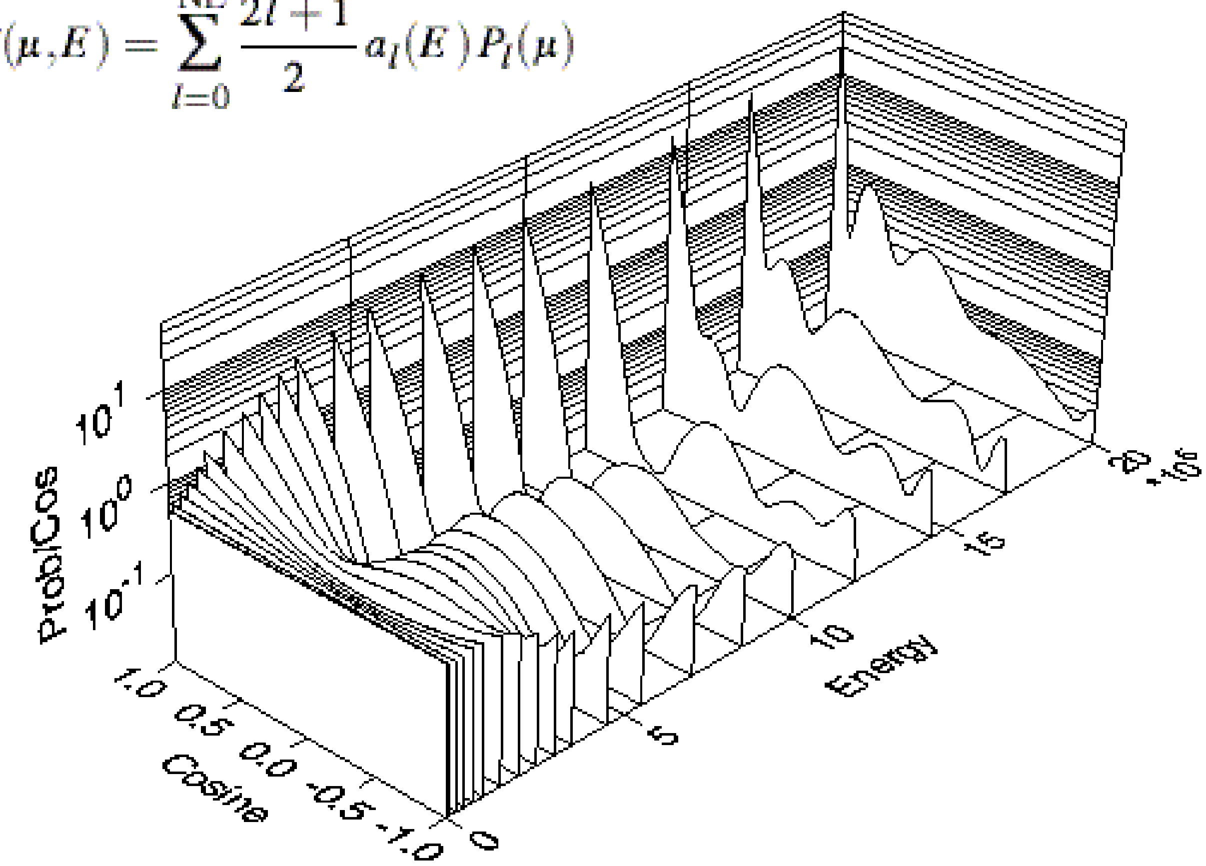
As an example, here is a TAB2 record for the angular distribution of elastic reaction in Cu-63 from ENDF/B-VI (MF=4, MT=2):

```
 2.906300+4 6.238900+1          0          1          0          02925 4  2
 0.000000+0 6.238900+1          0          2          0          02925 4  2
 0.000000+0 0.000000+0          0          0          1         222925 4  2
         22          2                                            2925 4  2
 0.000000+0 1.000000-5          0          0          1          02925 4  2
 0.000000+0                                                       2925 4  2
 0.000000+0 2.530000-2          0          0          1          02925 4  2
 0.000000+0                                                       2925 4  2
 0.000000+0 1.000000+4          0          0          2          02925 4  2
 3.214700-3 1.190800-4                                            2925 4  2
 0.000000+0 1.000000+5          0          0          4          02925 4  2
 3.619500-2 3.845600-3 3.661300-5 0.000000+0                      2925 4  2
 0.000000+0 3.000000+5          0          0          4          02925 4  2
 7.500000-2 1.800000-2 4.000000-4 0.000000+0                      2925 4  2
 0.000000+0 5.000000+5          0          0          4          02925 4  2
 1.200000-1 5.500000-2 2.550000-3 1.200000-4                      2925 4  2
 0.000000+0 7.500000+5          0          0          4          02925 4  2
 1.730000-1 1.070000-1 1.300000-2 2.730000-3                      2925 4  2
 0.000000+0 1.000000+6          0          0          6          02925 4  2
 2.258400-1 1.602700-1 3.980500-2 1.286300-2 1.560800-5 0.000000+02925 4  2
 0.000000+0 1.500000+6          0          0          6          02925 4  2
 2.738500-1 2.188700-1 9.602200-2 3.370000-2 1.499300-4 0.000000+02925 4  2
 ...(省略)
 0.000000+0 2.000000+7          0          0         14          02925 4  2
 8.105400-1 6.500300-1 5.507300-1 4.828500-1 4.177800-1 3.523300-12925 4  2
 2.899200-1 2.378900-1 1.840200-1 1.207600-1 6.096100-2 2.102800-22925 4  2
 4.210300-3 0.000000+0                                            2925 4  2
                                                                  2925 4  0
```

Nuclear Data

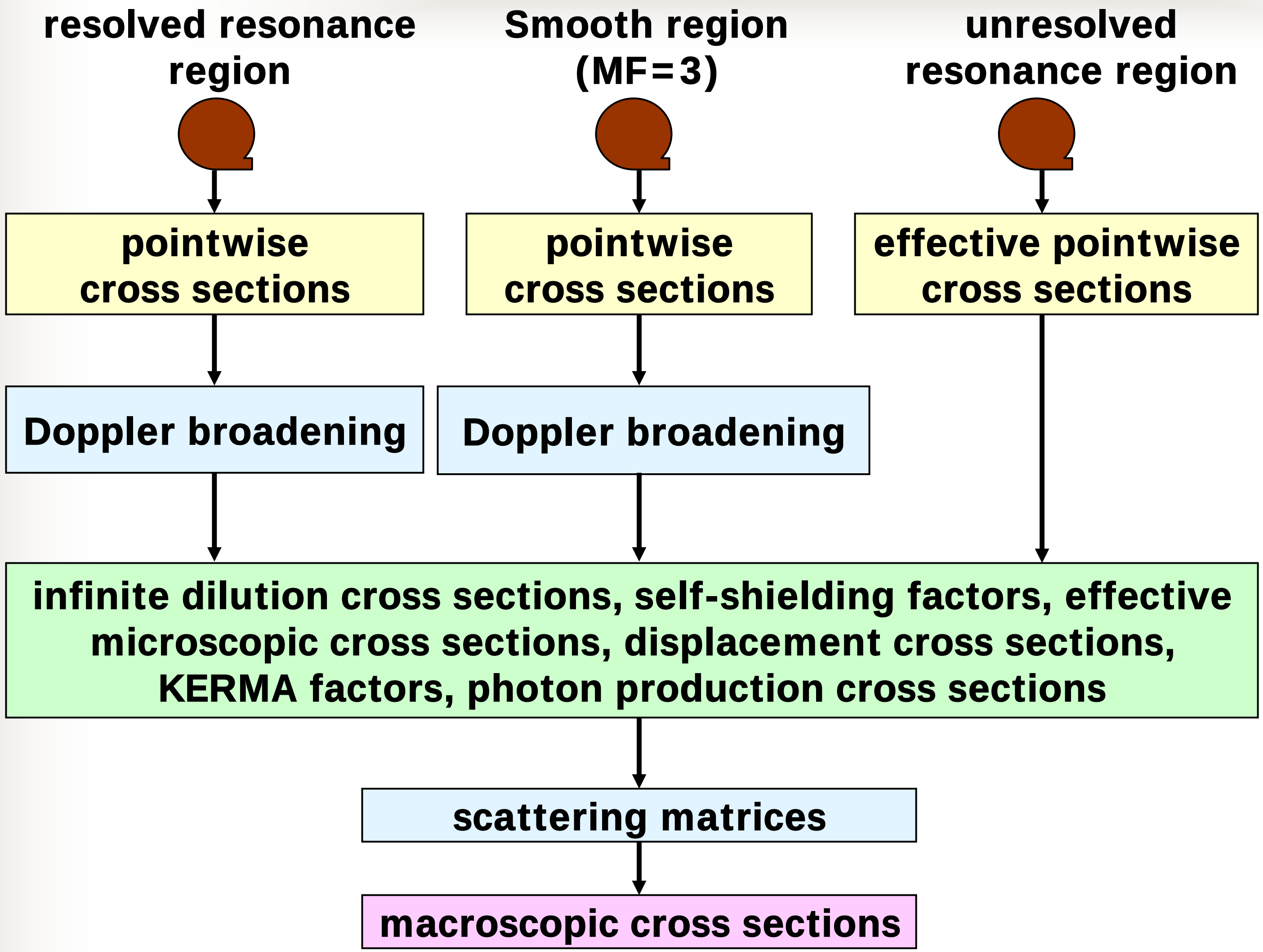
As an example, here is a TAB2 record for the angular distribution of elastic reaction in Cu-63 from ENDF/B-VI (MF=4, MT=2):

$$f(\mu, E) = \sum_{l=0}^{NL} \frac{2l+1}{2} a_l(E) P_l(\mu)$$



V. McLane (Ed.), “ENDF-102 Data Format and Procedures for the Evaluated Nuclear Data File ENDF-6,” BNL-NCS-44945-01/04-Rev. (2001)

Cross-Section Processing



Resolved resonance process

Data in MF=2, MT=151 are processed.

LRF=1: SLBW (Single-level Breit-Wigner) no r-r interference,

LRF=2: MLBW (Multi-level Breit-Wigner) r –r interference(elastic,total),

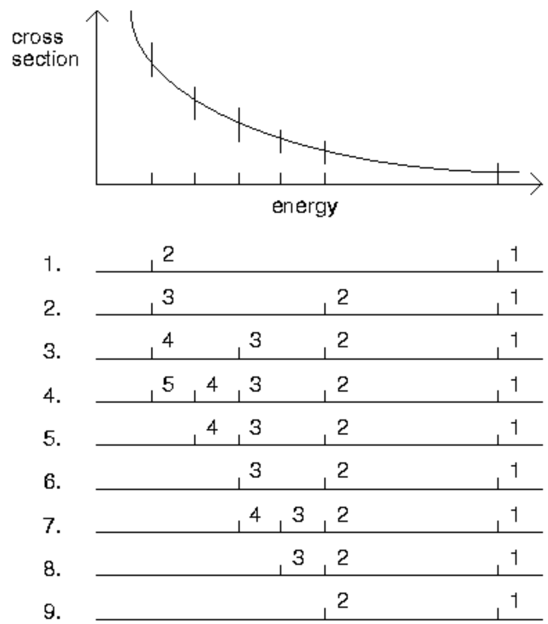
LRF=3: Reich-Moore (multilevel multichannel R-matrix) no comp,

LRF=4: Adler-Adler (level-level, channel-channel interference effects) no comp,

LRF=5: General R-Matrix (multilevel multichannel R-matrix)

LRF=6: Hybrid R-functions (level-level interference effects

Resonance parameters are applied to the resonance formula to generate point-wise cross section automatically in order to satisfy a given accuracy (~0.1%) for linear interpolation scheme such as RESENT, RECONR codes



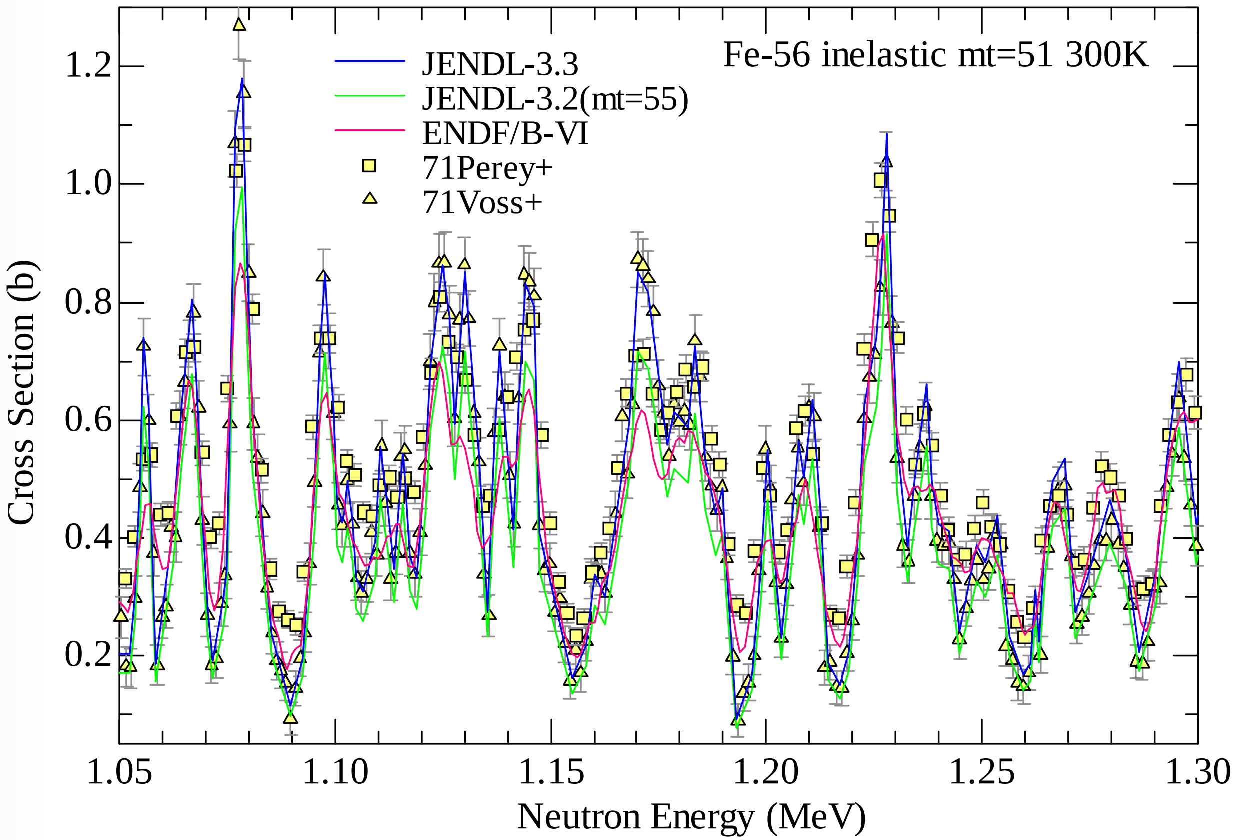
Unresolved resonance process

In the ENDF-6 evaluations, this "unresolved range" is handled by giving average values for the resonance spacing and the various partial widths, together with probability distributions for the spacing and partial widths. These unresolved resonance parameters are used three ways in practice:

- **Infinitely-dilute cross sections:** the cross sections that would be measured for a thin sample (which are equivalent to the cross sections that would act in a very dilute mixture) can be calculated using direct integrals over the probability distributions.
- **Self-shielded effective cross sections:** effective cross sections for thicker targets or less dilute mixtures show self-shielding effects that can be computed *vs.* temperature and background cross section by using the MC^2 scheme.
- **Probability tables:** probability tables for the total cross section and the dependent elastic, fission, and capture cross sections can be used to sample for cross sections in continuous-energy Monte Carlo codes like MCNP. The probability table can be generated by using the PURR module of NJOY.

Smooth cross section process

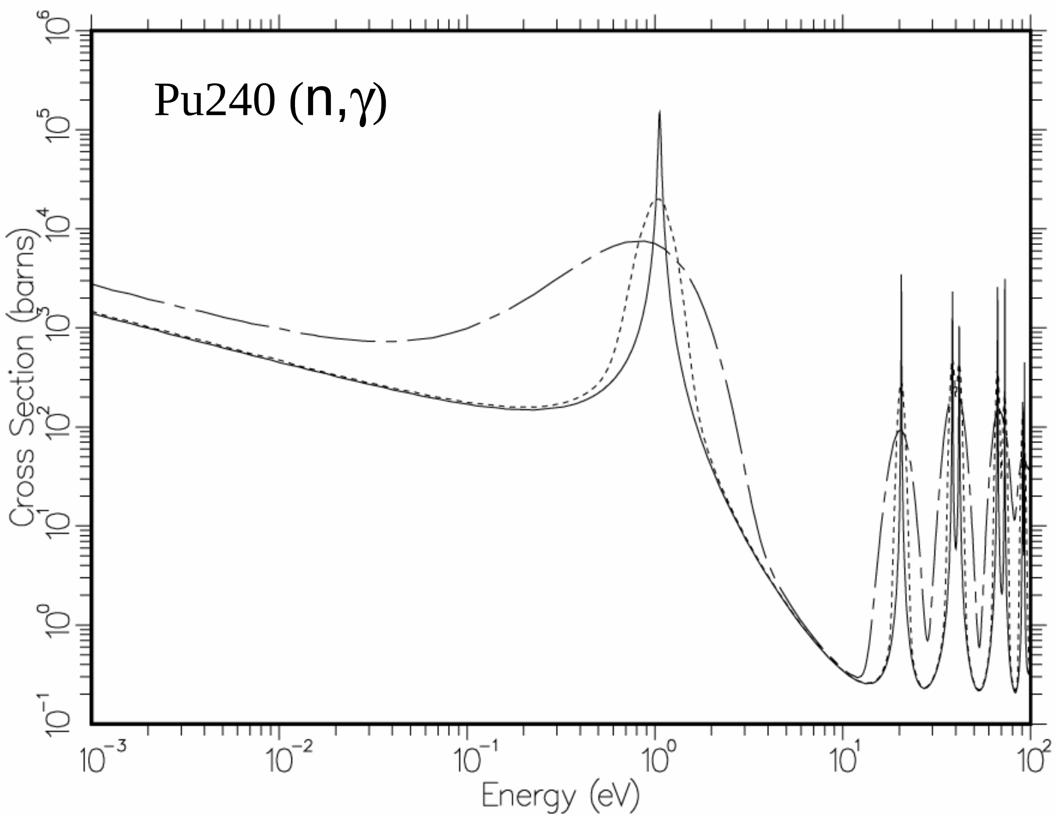
Reaction cross sections stored in MF3 are processed by using the same method to generate point-wise cross sections because the inelastic cross sections from discrete levels are evaluated by tracing the experimental data.



Doppler broadening process

Temperature dependence to the point-wise cross sections is computed in the Doppler broadening process. The kernel-broadening method adopted in the SIGMA1 code is widely used in the cross-section processing codes such as BROADR module in NJOY.

The (n,γ) cross section for Pu-240 is shown for temperatures of 0K, 30000K, and 300000K. Resonances with energies larger than *kT/A* broaden symmetrically (and their areas tend to remain constant). Low energy resonances develop an additional 1/*v* tail, and their areas do not remain constant under Doppler broadening.



Self-shielding factor process

When neutrons slow down in a medium with resonance absorption present, dips will appear in the flux corresponding to resonance peaks, and sometimes sharp peaks will occur in the flux corresponding to deep minima in the cross section, or "windows." The products of cross section and flux that appear in the definitions of the multigroup constants will clearly be reduced (self shielded) when the dips are large. The classical method for handling self-shielding in multigroup codes is the Bondarenko model. For narrow resonances in large systems, the flux takes the form:

$$\phi(E,\sigma_0,T) = \phi_s(E,T)/(\sigma_t(E,T)+\sigma_0)$$

where background cross section σ_0 is defined as

$$\sigma_0 = \sum_{m\neq k} N_k \sigma_{kt} / N_m$$

$$\sigma_x(\sigma_0,T) = \int \sigma_x(E,T)\phi(E,\sigma_0,T)dE \Big/ \int \phi(E,\sigma_0,T)dE$$

self-shielding factor f_x corresponding to effective cross section σ_0 is defined as

$$f_x(\sigma_0,T) = \sigma_x(\sigma_0,T)/\sigma_x(\sigma_0=\infty,T_0)$$

Photon production cross-section process

When neutrons penetrate in medium, photons are generated by radiative capture reaction, (n, 2n), (n, p), (n, α) and inelastic reactions. These photons become secondary source in the medium.

Photon production yields and the energies are calculated by using MF12-15 data in the evaluated nuclear data file, and merged into scattering matrix.

Scattering matrix process

In the multigroup transport calculation, scattering matrices from incident neutron energy E' to outgoing energy E are required for each reaction such as elastic, inelastic and (n, 2n) reactions. The angular distributions for each reaction are retrieved from MF4 or MF6. The angular distribution is expressed by the Legendre coefficients or tabular forms. In the higher energy region, energy-angle correlation is important so that the MF6 is ordinarily used to express the energy-angle data. Kalbach systematics is also adopted in the intermediate energy region.

Scattering matrix process (cont.)

Upscattering is important below 5 eV for thermal neutron calculation. The upscattering matrix is usually produced by using S(α, β, T) function in MF7. For the energy beyond the limit of α or β, the short-collision-time (SCT) approximation is adopted.

Scattering matrices for elastic, inelastic and (n, 2n) reactions are usually considered in the present cross-section processing codes. For higher energy region, the other reactions such as (n, Xn), (n, XnYp) should be considered. In the high energy accelerator applications, the scattering matrix should be directly treated by the form of energy-angle double-differential cross section (DDX).

Effective cross-section process

Effective macroscopic/microscopic cross sections for target application are produced with the given library formats. The "effective" means the cross sections with self-shielding effect are considered. When the neutron-photon coupled group library is required, the neutron and photon coupled data are generated. Photon interaction cross sections are usually generated by using PHOTX-V2 library.

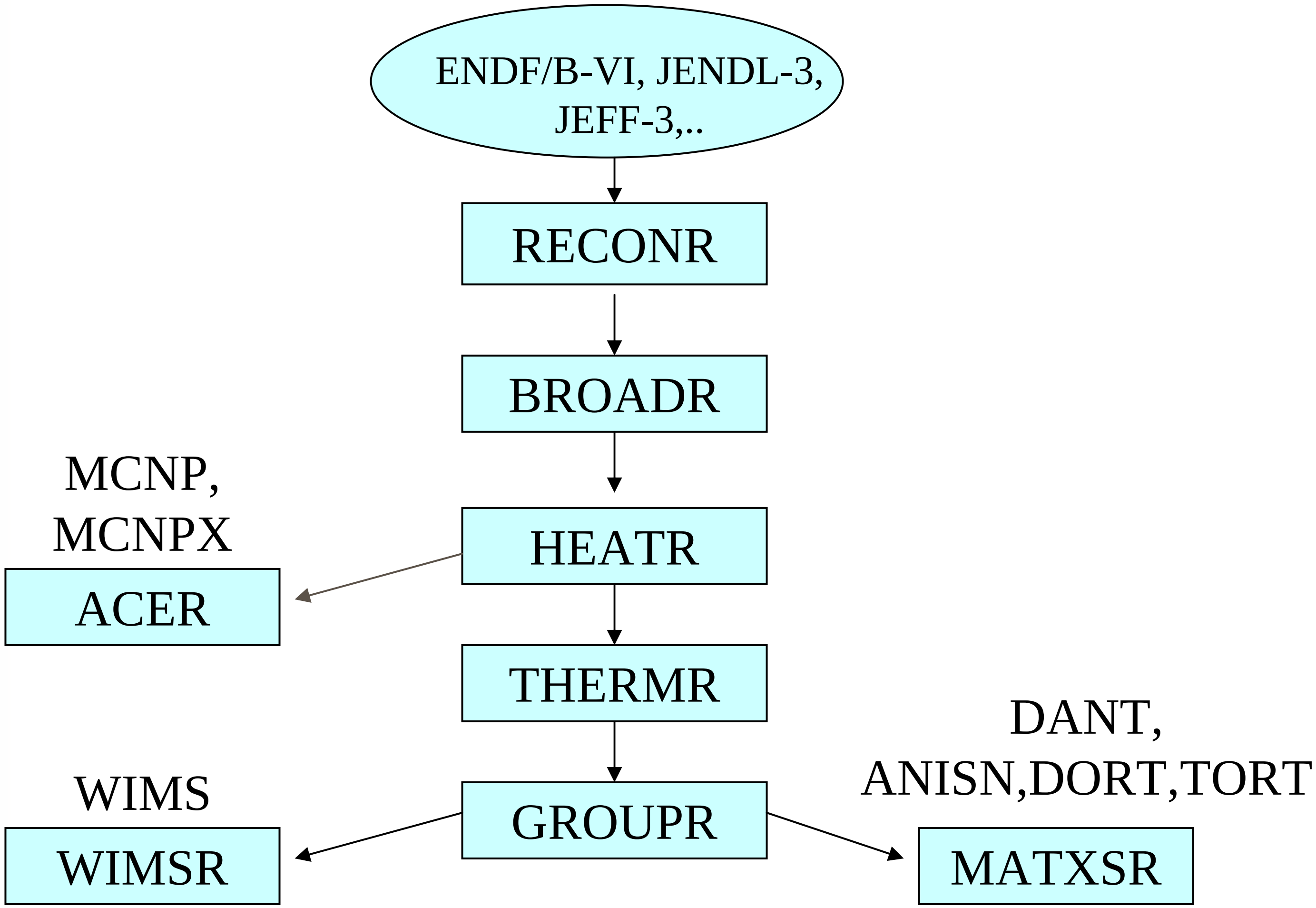
NJOY Code

- The NJOY Nuclear Data Processing System is used to convert evaluated nuclear data in ENDF format into forms useful for applications. As a bridge between physics and engineering, it is best used by people with some knowledge of things like nuclear reaction theory, resonance theory, or scattering theory on one side, and some knowledge of things like particle transport codes, reactor core calculations, or radiation medicine on the other.

- The NJOY code is a modular computer code designed to read evaluated data in ENDF format, transform the data in various ways, and output the results as libraries designed to be used in various applications. Each module performs a well defined processing task.

NJOY Code (cont.)

Generation of Cross Section Library



NJOY Code (cont.)

- **NJOY** directs the flow of data through the other modules and contains a library of common functions and subroutines used by the other modules.
- **RECONR** reconstructs pointwise (energy-dependent) cross sections from ENDF resonance parameters and interpolation schemes.
- **BROADR** Doppler broadens and thins pointwise cross sections.
- **UNRESR** computes effective self-shielded pointwise cross sections in the unresolved energy range.
- **HEATR** generates pointwise heat production cross sections (KERMA coefficients) and radiation-damage cross sections.

NJOY Code (cont.)

- **THERMR** produces cross sections and energy-to-energy matrices for free or bound scatterers in the thermal energy range.
- **GROUPR** generates self-shielded multigroup cross sections, group-to-group scattering matrices, photon-production matrices, and charged-particle cross sections from pointwise input.
- **GAMINR** calculates multigroup photoatomic cross sections, KERMA coefficients, and group-to-group photon scattering matrices.
- **ERRORR** computes multigroup covariance matrices from ENDF uncertainties.
- **COVR** reads the output of ERRORR and performs covariance plotting and output formatting operations.

NJOY Code (cont.)

- **MODER** converts ENDF "tapes" back and forth between ASCII format and the special NJOY blocked-binary format.
- **DTFR** formats multigroup data for transport codes that accept formats based in the DTF-IV code.
- **CCCCR** formats multigroup data for the CCCC standard interface files ISOTXS, BRKOXS, and DLAYXS.
- **MATXSR** formats multigroup data for the newer MATXS material cross-section interface file, which works with the TRANSX code to make libraries for many particle transport codes.
- **RESXSR** prepares pointwise cross sections in a CCCC-like form for thermal flux calculators.

NJOY Code (cont.)

- **ACER** prepares libraries in ACE format for the Los Alamos continuous-energy Monte Carlo code MCNP.
- **POWR** prepares libraries for the EPRI-CELL and EPRI-CPM codes.
- **WIMSR** prepares libraries for the thermal reactor assembly codes WIMS-D and WIMS-E.
- **PLOTR** reads ENDF-format files and prepares plots of cross sections or perspective views of distributions for output using VIEWR.
- **VIEWR** takes the output of PLOTR, or special graphics from HEATR, COVR, DTFR, or ACER, and converts the plots into Postscript format for printing or screen display.

NJOY Code (cont.)

- **MIXR** is used to combine cross sections into elements or other mixtures, mainly for plotting.
- **PURR** generates unresolved-resonance probability tables for use in representing resonance self-shielding effects in the MCNP Monte Carlo code.
- **LEAPR** generates ENDF scattering-law files (File 7) for moderator materials in the thermal range. These scattering-law files can be used by THERMR to produce the corresponding cross sections.
- **GASPR** generates gas-production cross sections in pointwise format from basic reaction data in an ENDF evaluation. These results can be converted to multigroup form using GROUPR, passed to ACER, or displayed using PLOTR.