”Total Monte Carlo” applied to Phase I-1: burn-up calculation

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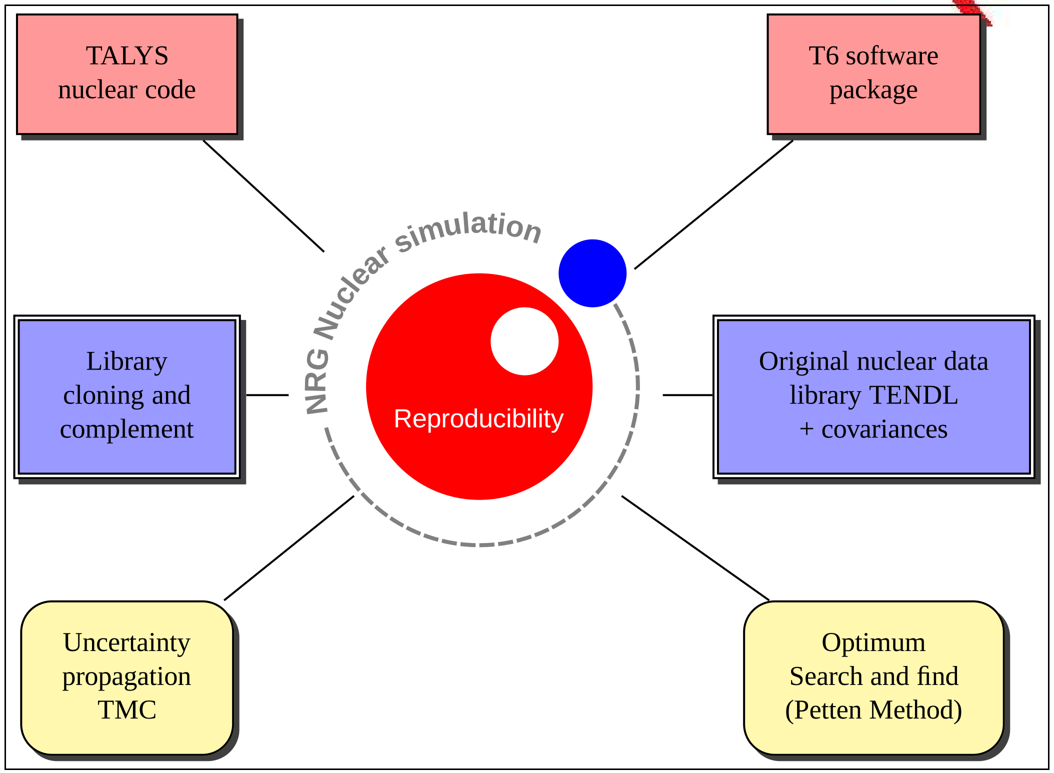
Overview

- Method: Total Monte Carlo (TMC)
- Description of the SERPENT model (Fuel pin-cell)
- Considered data in TMC
- Results



The complete report (NRG-113696) can be found at

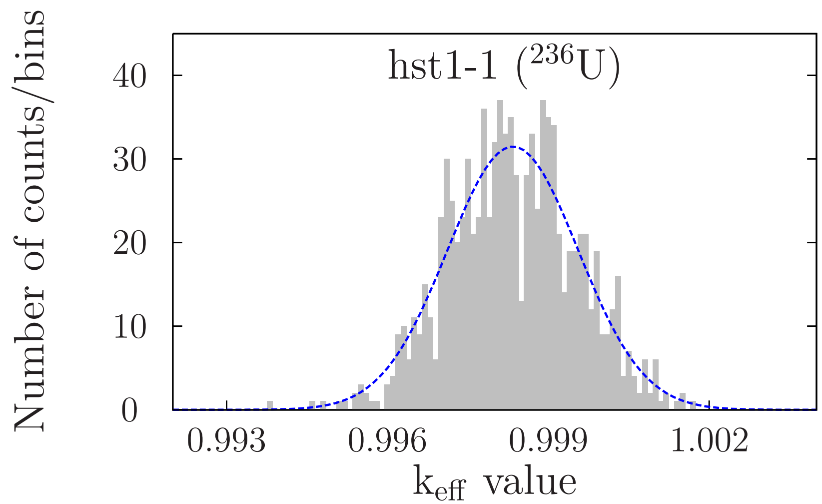
ftp://ftp.nrg.eu/pub/www/talys/bib_rochman/tmc.nrg.pdf

Method: Total Monte Carlo (TMC)

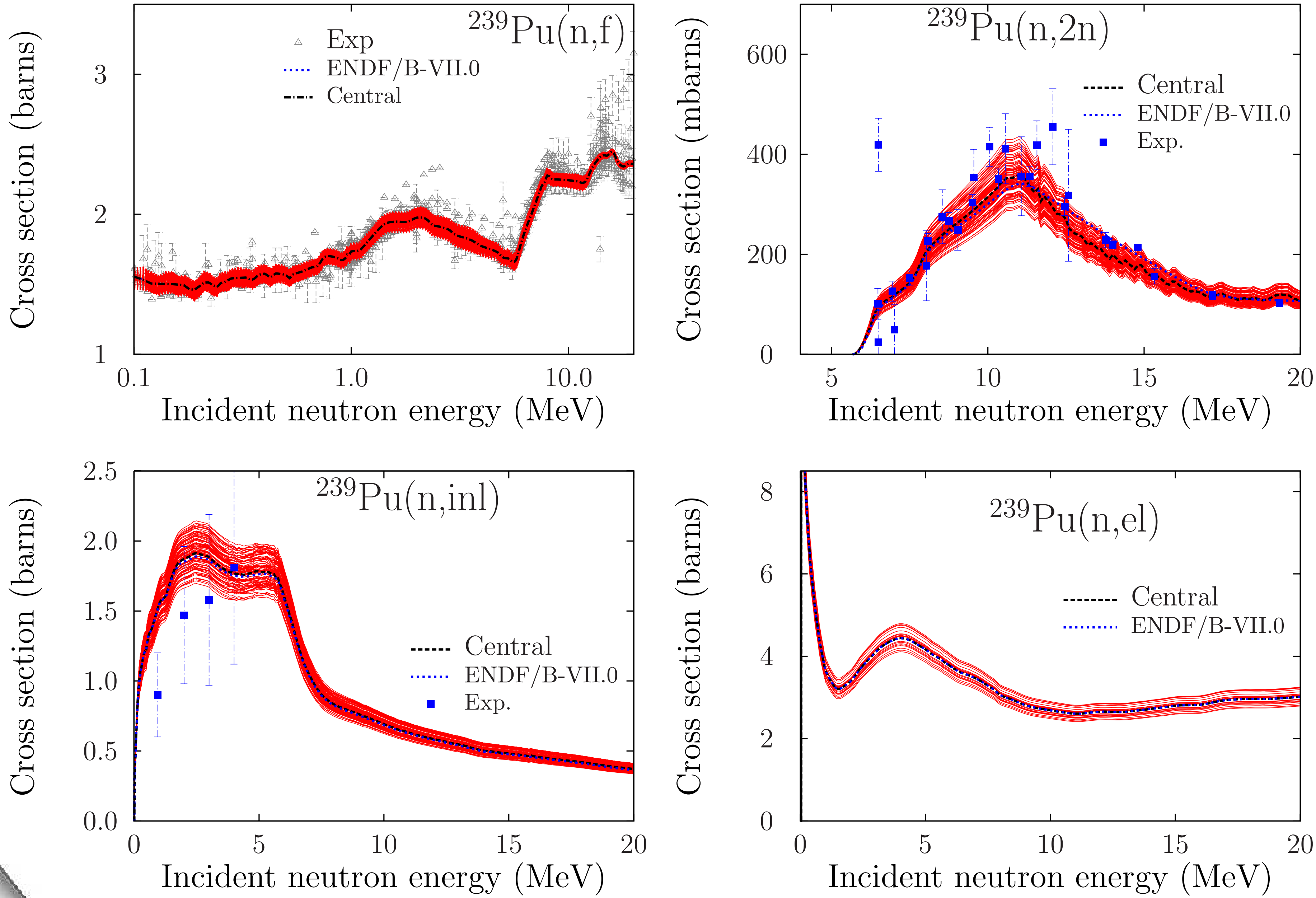
Control of nuclear data (TALYS)
+ simple processing (NJOY)
+ system simulation (MCNP/ERANOS/CASMO...)

1000 times

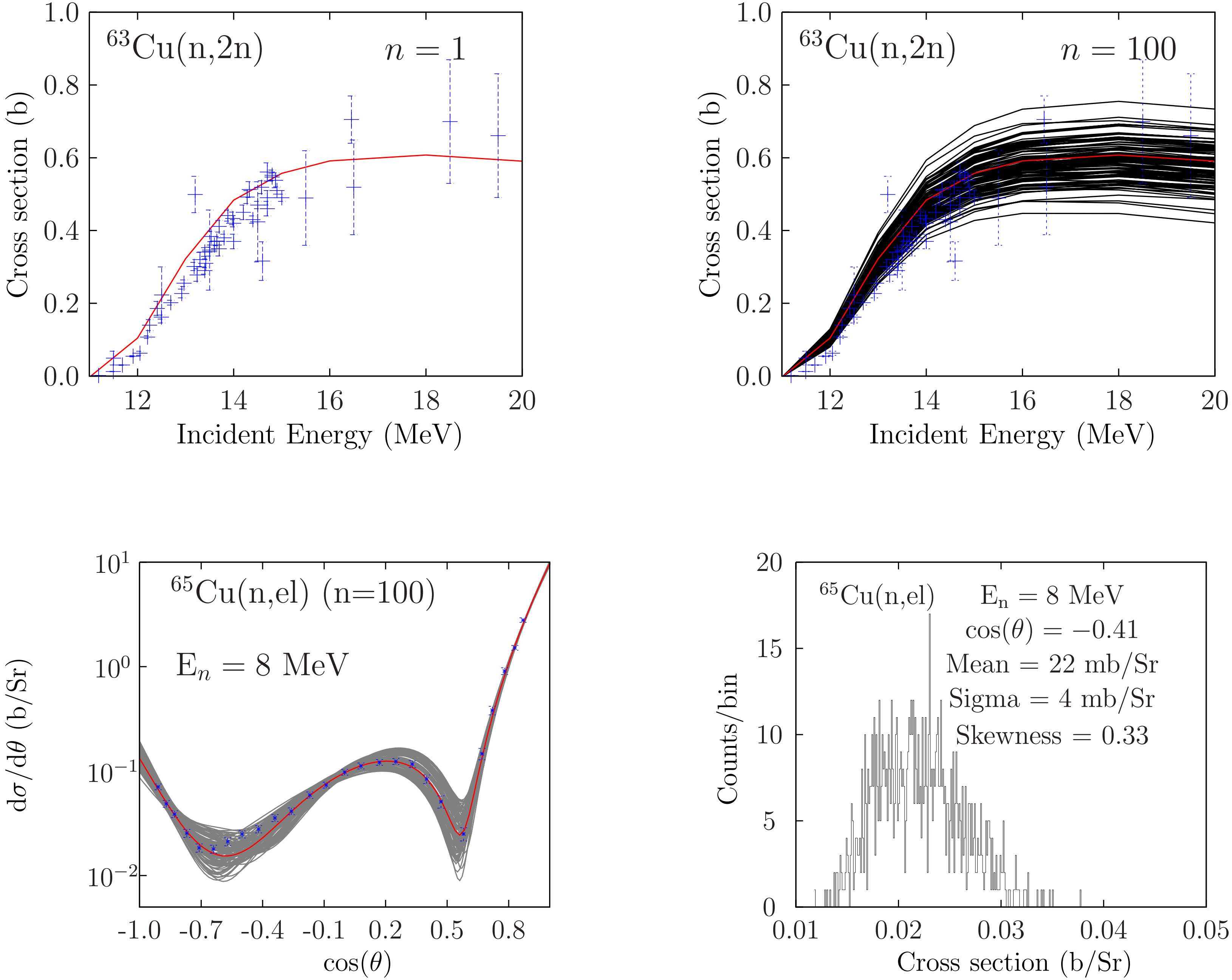
For each random ENDF file, the benchmark calculation is performed with MCNP. At the end of the n calculations, n different k_{eff} values are obtained. In the obtained probability distribution of k_{eff}, the standard deviation σ_{total} reflects two different effects: $\sigma^2_{total} = \sigma^2_{statistics} + \sigma^2_{nuclear\ data}$.



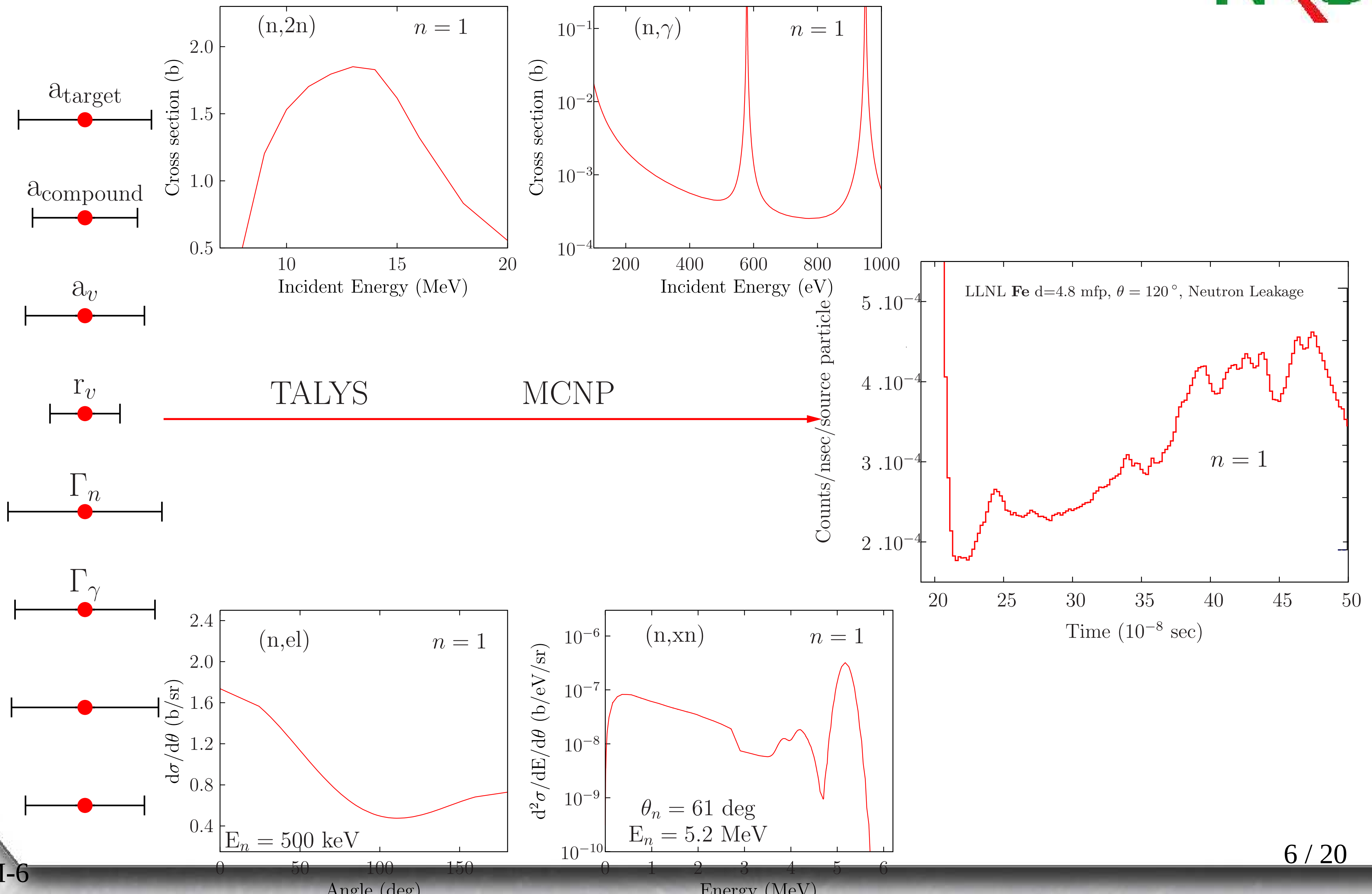
Example of random nuclear data for ^{239}Pu



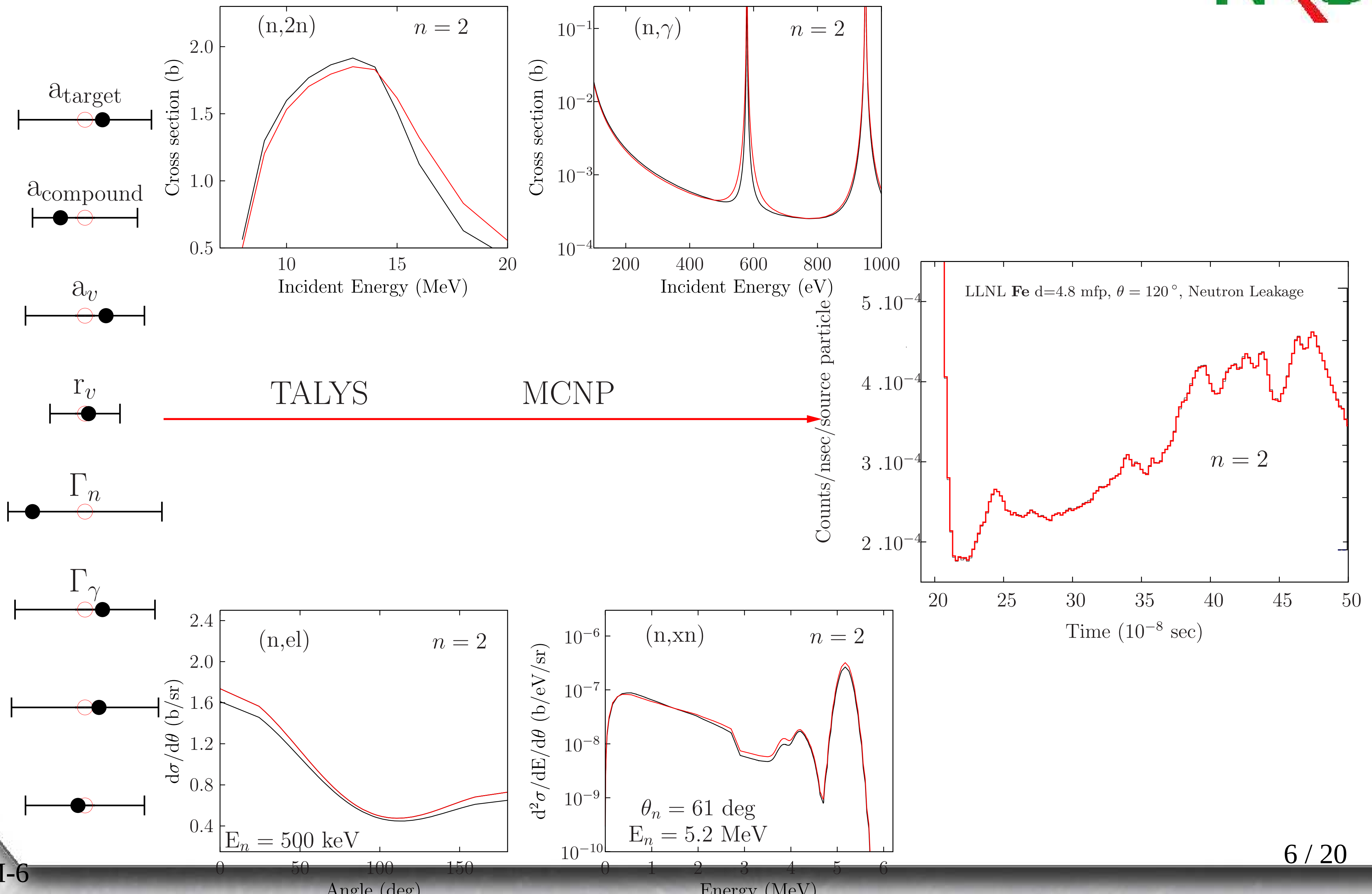
Examples with ^{63}Cu(n,2n) and ^{65}Cu(n,el)



Hands on "1000 ×(Talys + ENDF + NJOY + MCNP) calculations"



Hands on "1000 $\times$(Talys + ENDF + NJOY + MCNP) calculations"



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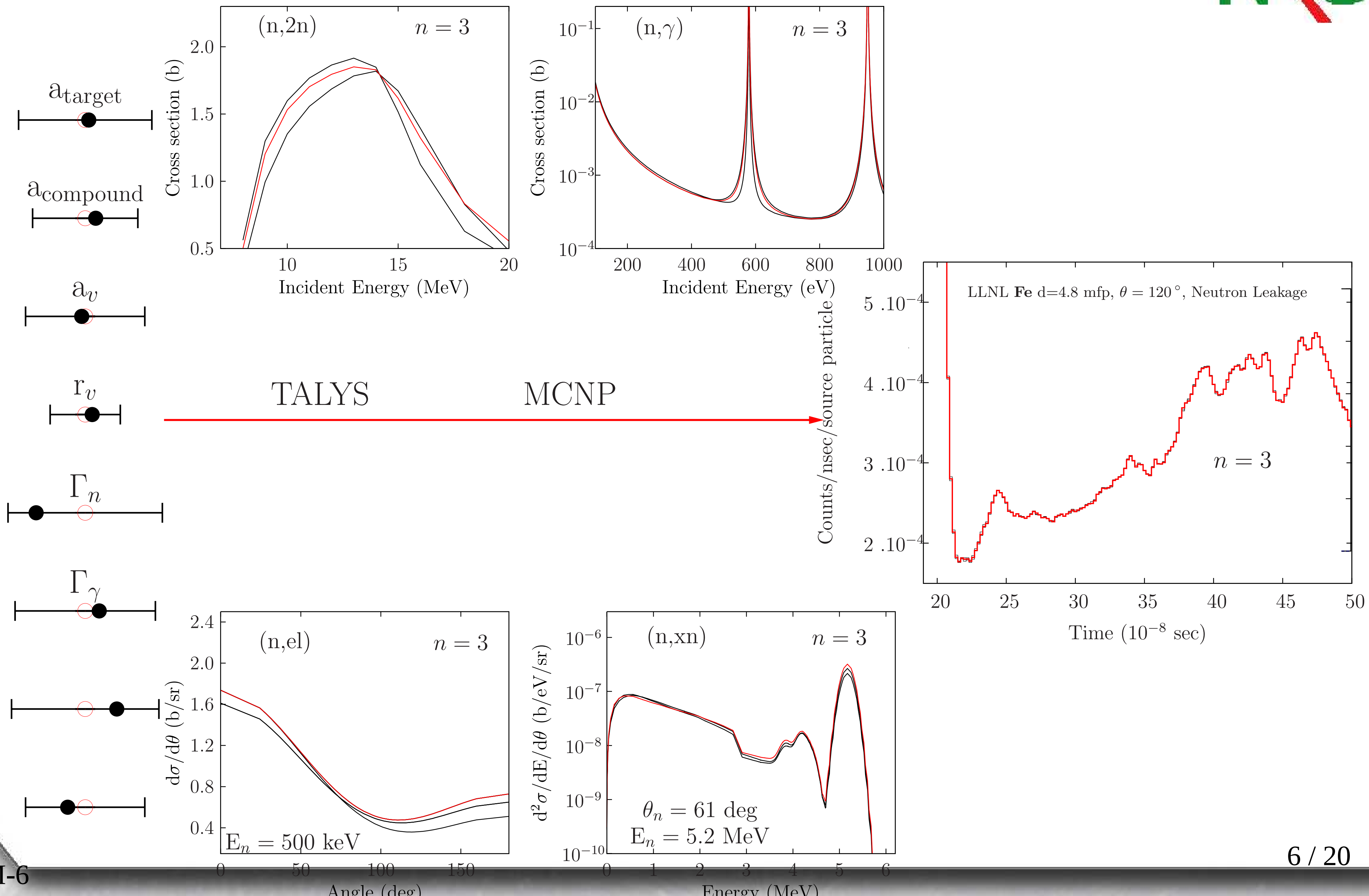
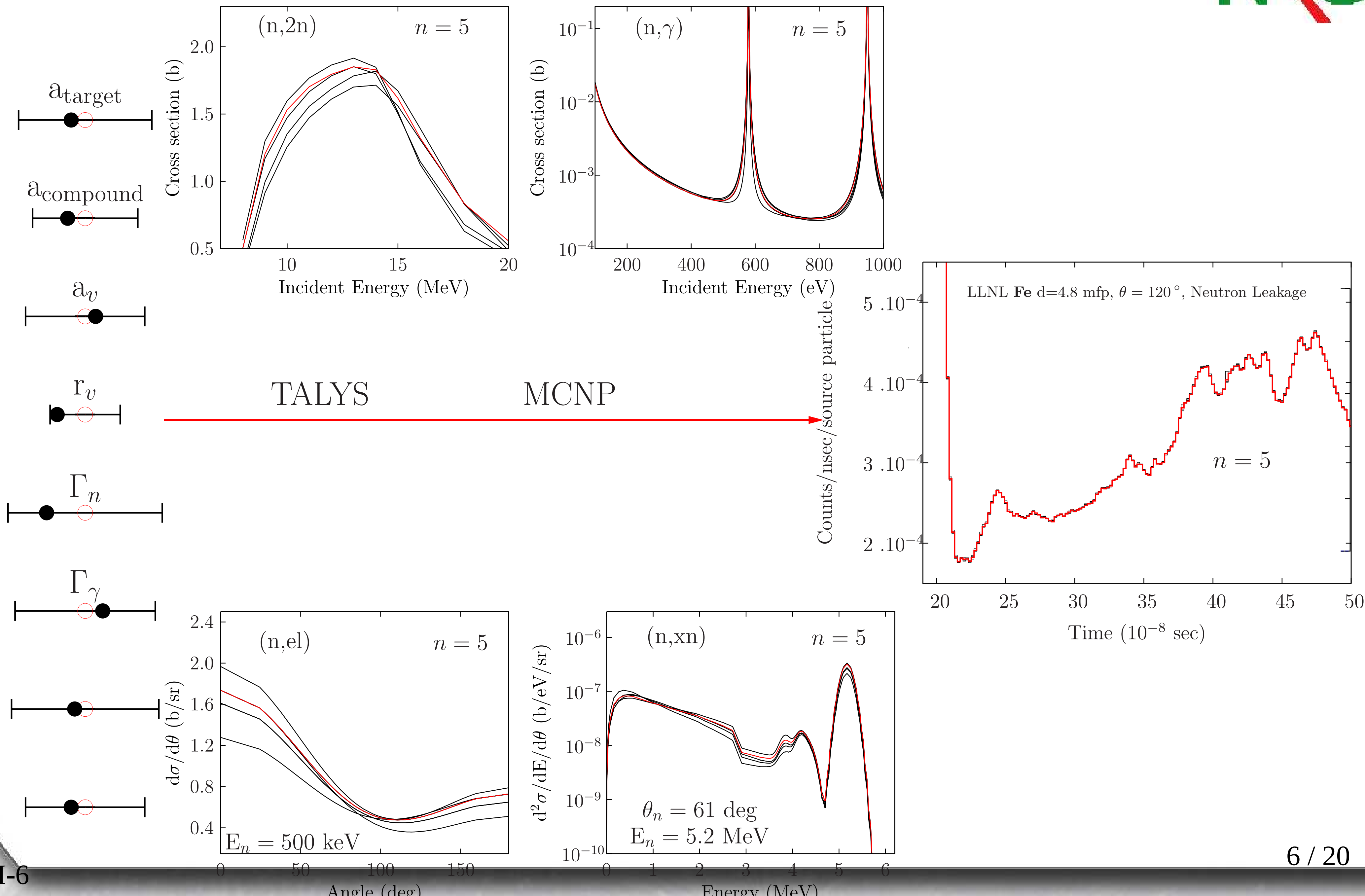
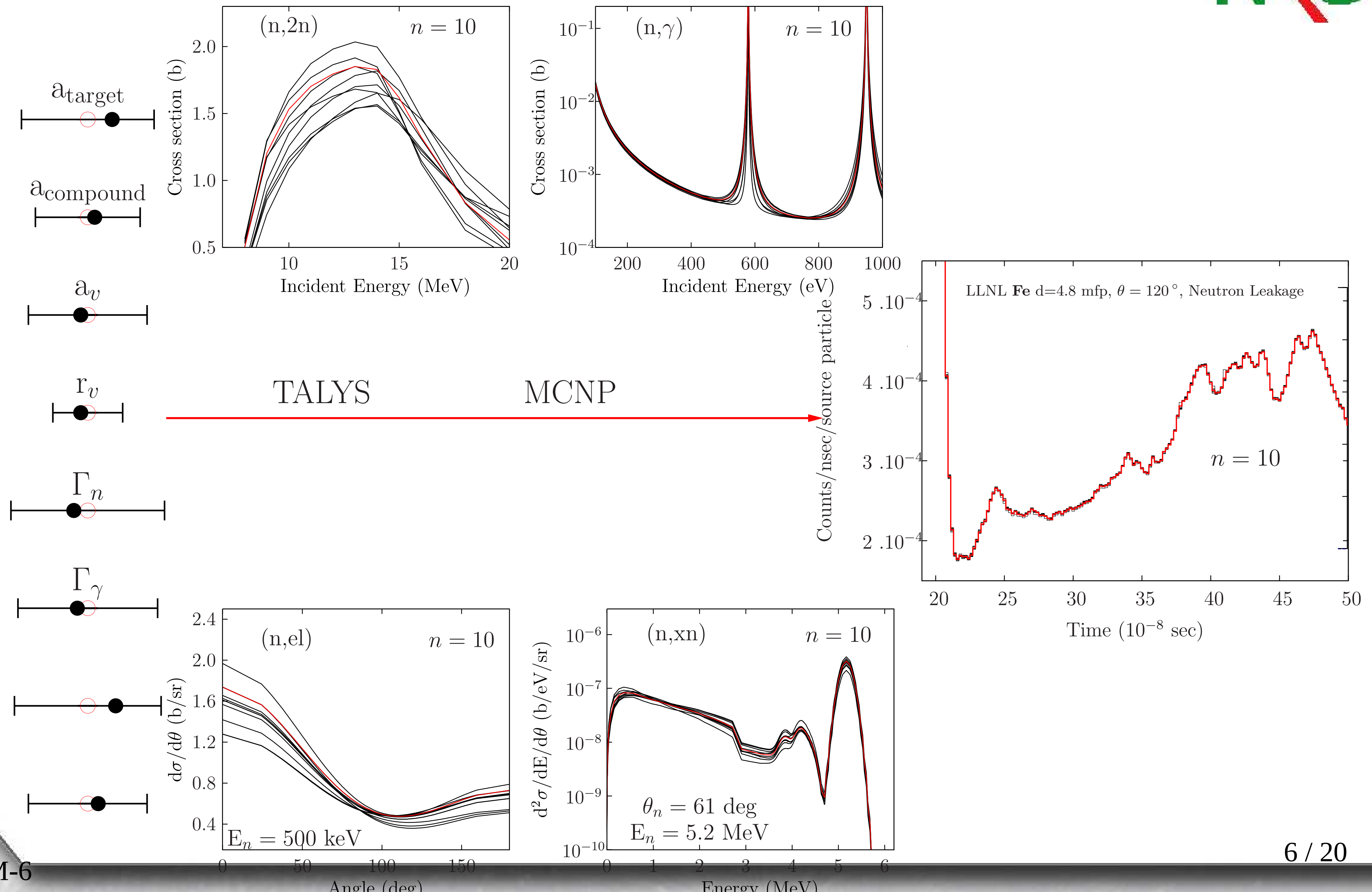


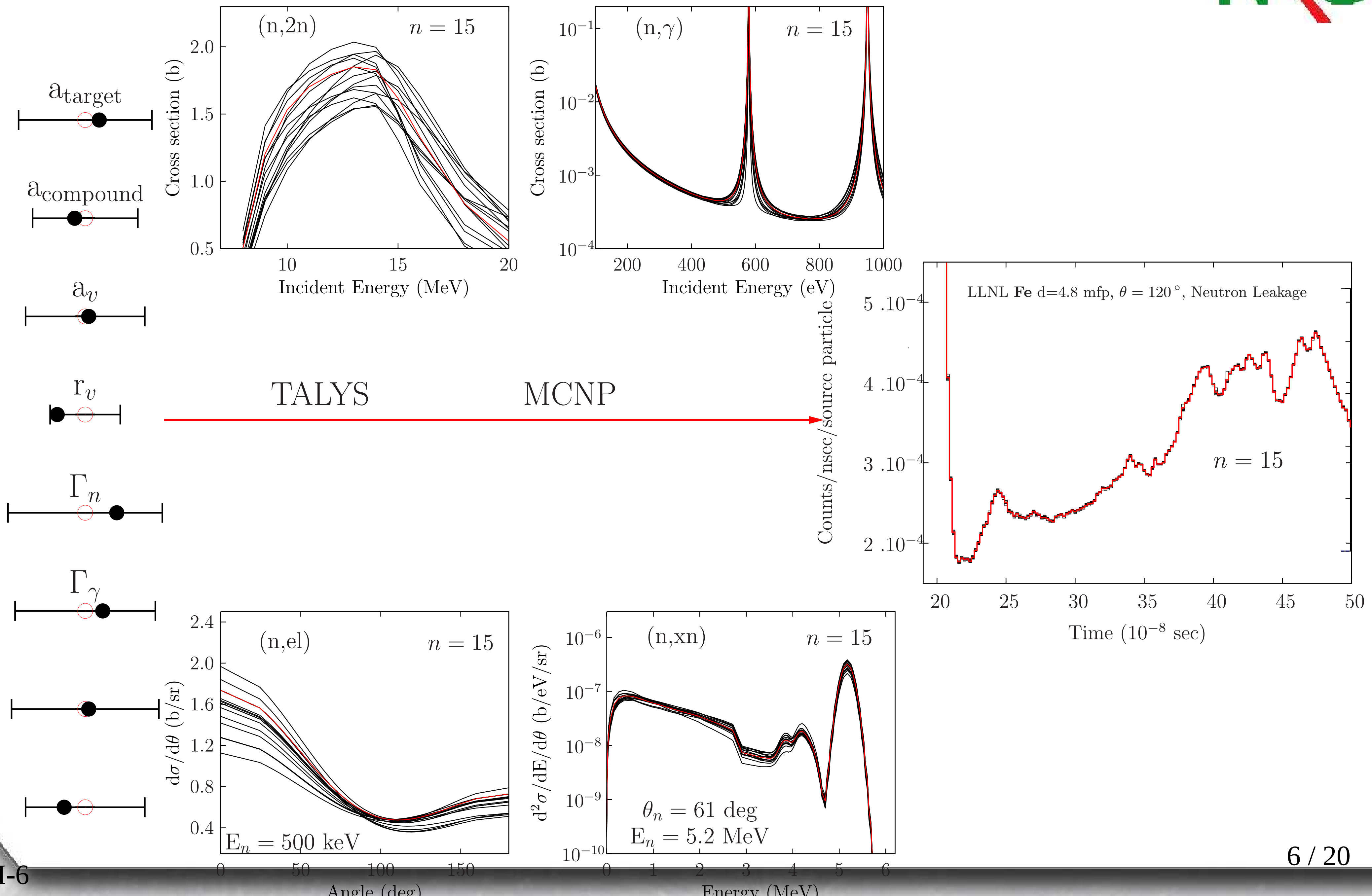
Hands on "1000 ×(Talys + ENDF + NJOY + MCNP) calculations"



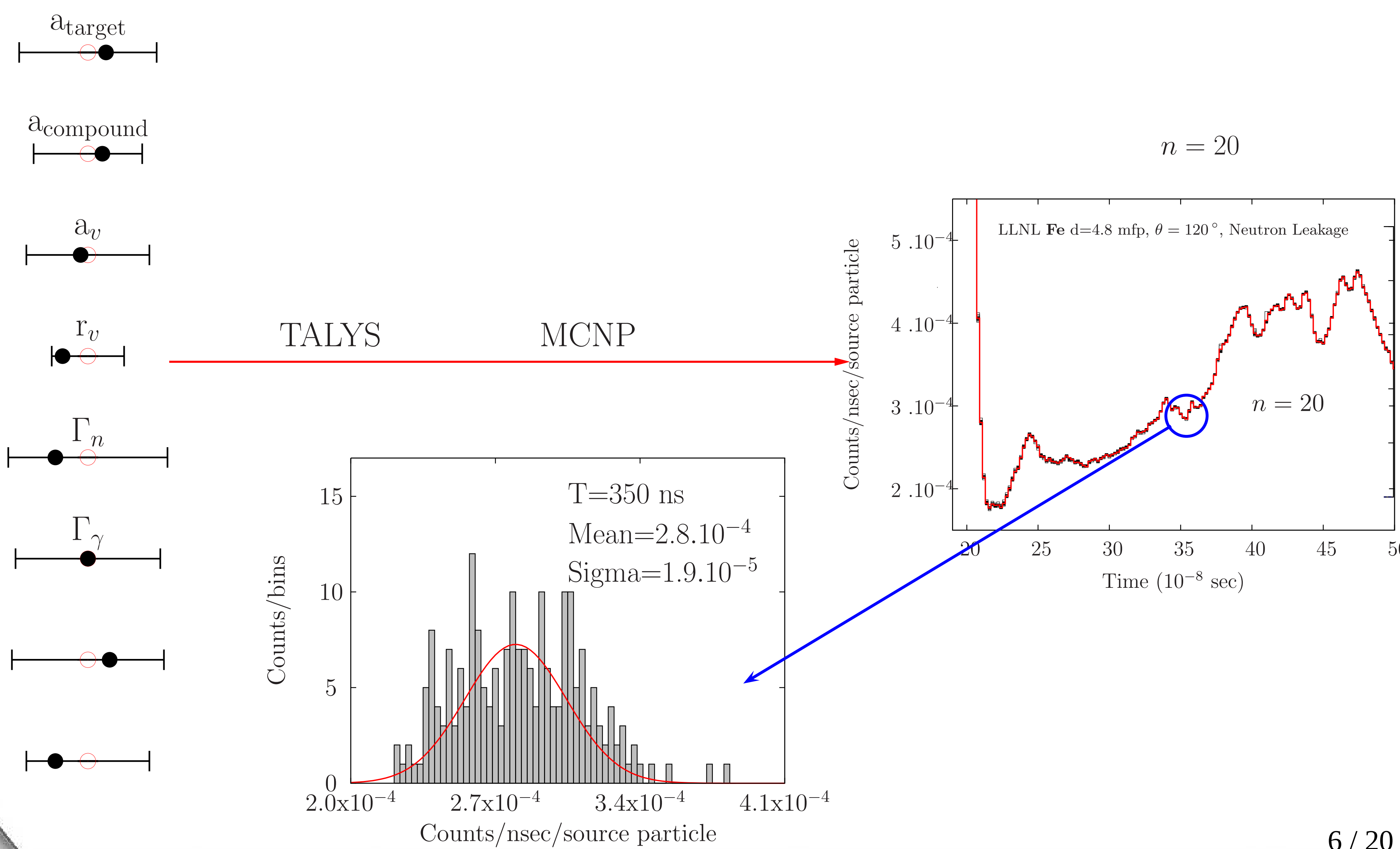
Hands on “1000 ×(Talys + ENDF + NJOY + MCNP) calculations”



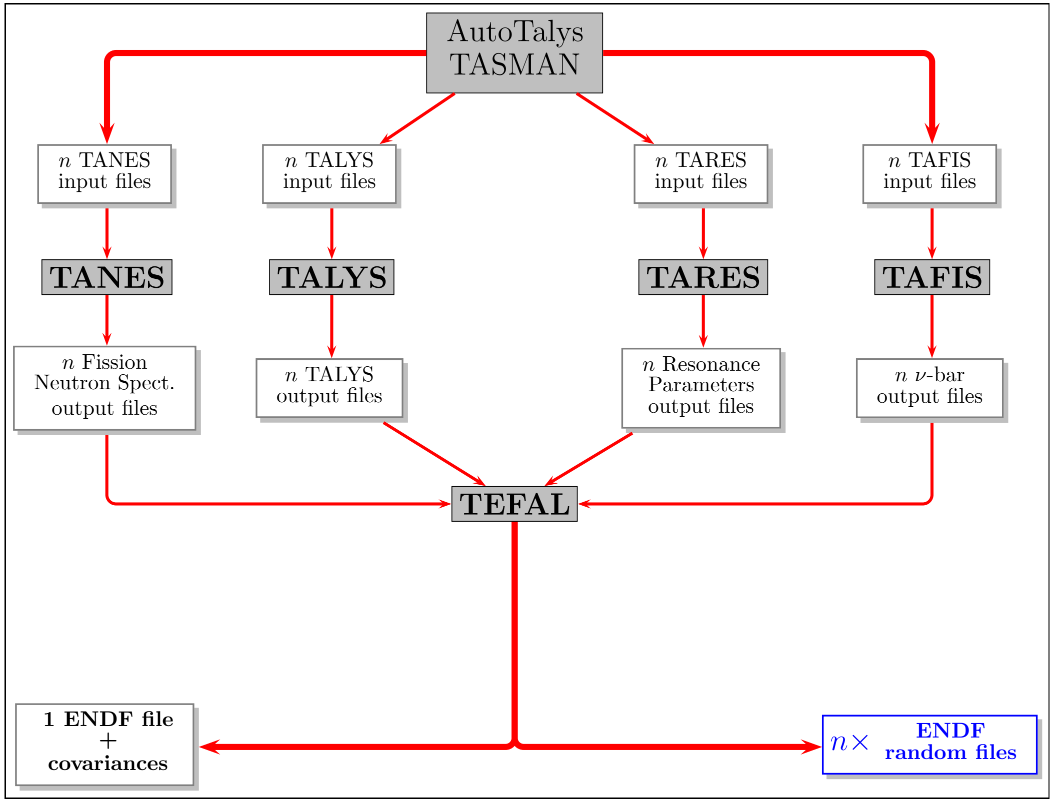
Hands on “1000 $\times$(Talys + ENDF + NJOY + MCNP) calculations”

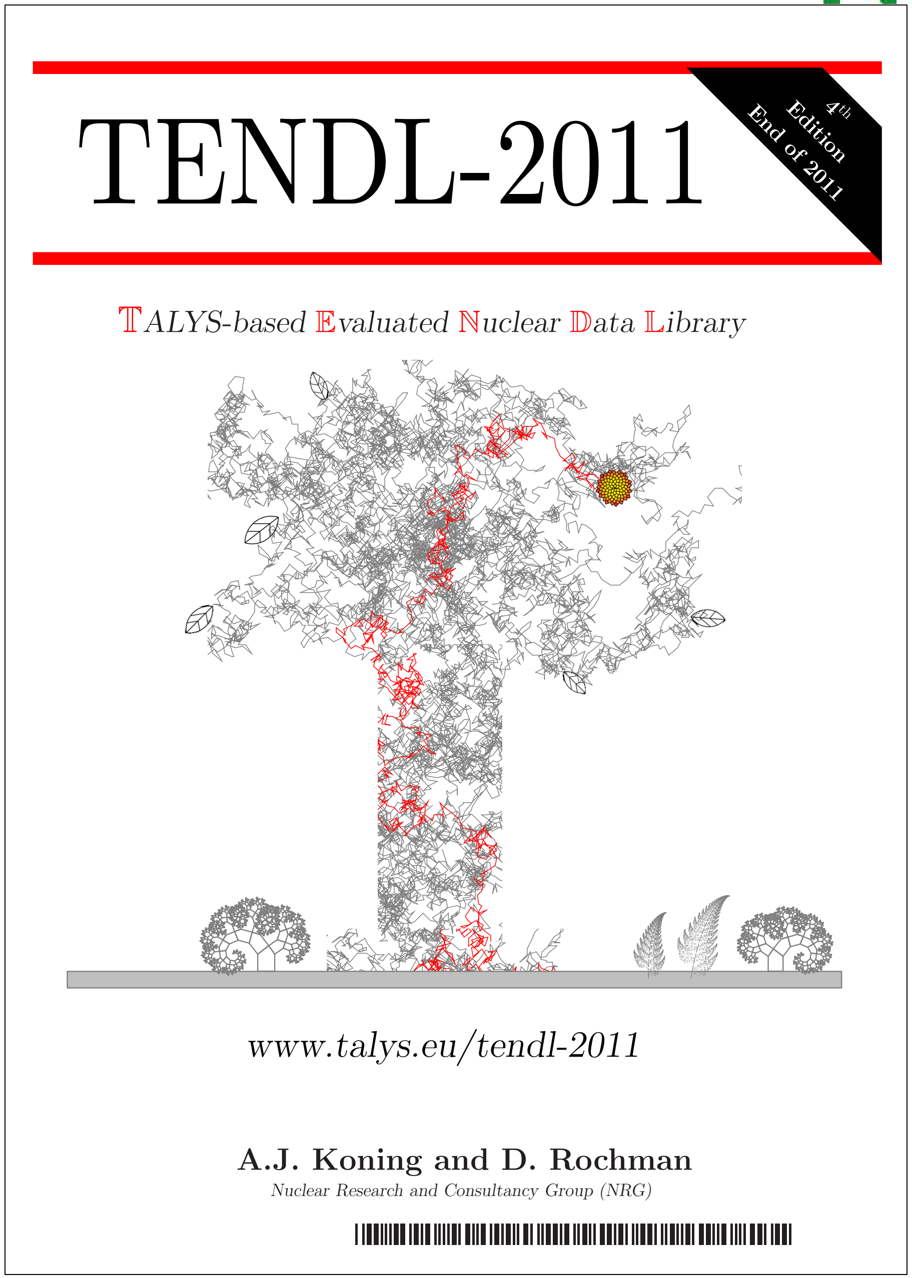


Hands on “1000 ×(Talys + ENDF + NJOY + MCNP) calculations”



Random nuclear data come from TENDL-2011.





Description of the SERPENT model (Fuel pin-cell)

The fuel test is a typical fuel rod from TMI-1 PWR, 15x15 assembly design.

Hot Full Power conditions		Configuration	
Fuel temperature (K)	900	Unit cell pitch (mm)	14.427
Cladding temperature (K)	600	Fuel pellet diameter (mm)	9.391
Moderator (coolant) temperature (K)	562	Fuel pellet material	UO_2
Moderator (coolant) density (g/cm^3)	0.7484	Fuel density (g/cm^3)	10.283
Reactor power (MWt)	2772	Fuel enrichment (w/o)	4.85
Number of assembly in reactor core	177	Cladding outside diameter (mm)	10.928
Number of fuel rods/fuel assembly	208	Cladding thickness (mm)	0.673
Active core length (mm)	3571.20	Cladding material	Zircaloy-4
		Cladding density (g/cm^3)	6.55
		Gap material	He
		Moderator material	H_2O

The fuel sample is burned for a unique complete cycle and the lengths of the burn time and cooling time:

Operating cycle	1
Burn time (days)	1825
Final burnup (GWd/MTU)	61.28
Downtime (days)	1870
Specific power (kW/kgU)	33.58

Considered data in TMC

pellet diameter	2 %
fuel enrichment	3 %
fuel density	4 %
moderator density	5 %

Nuclear data	ENDF-6 name	^{235}U	^{238}U	^{237}Np	^{239}Pu	^{241}Pu	Lumped fiss. prod.
complete ENDF file	MF1-6,10,12,14	X	X		X		X
fission yields	MF8	X	X	X	X	X	
ν-bar	MF1	X	X		X		
Resonance range	MF2	X	X		X		
Fast range	MF3	X	X		X		
Angular distr.	MF4	X	X		X		
Fission neut. spec.	MF5	X	X		X		
(n,γ)		X	X		X		
(n,f)		X	X		X		
(n,el)		X	X		X		

Lumped (138) fission products: $^{72-74,76}$Ge, ^{75}As, $^{76-80,82}$Se, ^{79,81}Br, $^{80-84,86}$Kr, ^{85,87}Rb, $^{86-88,92}$Sr, ^{89}Y, ^{93,95}Zr, ^{94,95}Nb, $^{95-97}$Mo, ^{99}Tc, $^{99-104,106}$Ru, ^{103,105,106}Rh, $^{104-108,110}$Pd, ^{109}Ag, $^{111-114,116}$Cd, ^{113,115}In, $^{115,117-119,126}$Sn, ^{121,123,125}Sb, $^{122-128,130}$Te, ^{127,129,135}I, $^{128,130-132,134-136}$Xe, $^{133-137}$Cs, $^{134-138}$Ba, ^{140}La, ^{140,142}Ce, ^{141,144}Pr, $^{142-146,148,150}$Nd, $^{147-149}$Pm, $^{147,149-152,154}$Sm, $^{151-156}$Eu, $^{152,154-158,160}$Gd, ^{159,160}Tb, $^{160-164}$Dy, ^{165}Ho, ^{166,167}Er.

Results on k_∞

	Burn-up (GWd/MTU)						
	0	10	20	30	40	50	60
k_∞	1.41e+00	1.25e+00	1.16e+00	1.08e+00	1.02e+00	9.55e-01	9.01e-01
Order							
1.	^{235}U	^{238}U	^{238}U	^{238}U	^{238}U	^{239}Pu	^{239}Pu
2.	^{238}U	^{235}U	^{235}U	^{235}U	^{235}U	Fiss. Yields	Fiss. Yields
3.		Fiss. Prod.	Fiss. Prod.	^{239}Pu	^{239}Pu	^{238}U	^{238}U
4.		^{239}Pu	^{239}Pu	Fiss. Prod.	Fiss. Yields	Fiss. Prod.	Fiss. Prod.
5.		Fiss. Yields	Fiss. Yields	Fiss. Yields	Fiss. Prod.	^{235}U	^{235}U

	Uncertainties (in %) coming from						
^{235}U	0.50	0.43	0.39	0.35	0.32	0.28	0.24
^{238}U	0.46	0.47	0.44	0.40	0.35	0.33	0.36
^{239}Pu	0.05	0.15	0.26	0.33	0.39	0.44	0.47
Fiss. Yiel.	0.00	0.21	0.25	0.29	0.32	0.35	0.36
Lumped F.P.	0.00	0.37	0.36	0.31	0.31	0.29	0.28
Total	0.68	0.79	0.78	0.76	0.76	0.76	0.79

Results on reaction rates

	Burn-up (GWd/MTU)						
	0	10	20	30	40	50	60
rr ^{235}U(n,γ)	^{235}U(n,f)	^{235}U(n,f)	^{235}U(n,f)	^{235}U(n,f)	^{235}U(n,f)	^{235}U(n,γ)	^{235}U(n,γ)
rr ^{238}U(n,γ)	^{238}U(n,el)	^{238}U(n,γ)	^{238}U(n,γ)	^{238}U(n,γ)	^{238}U(n,γ)	^{238}U(n,γ)	^{238}U(n,γ)
rr ^{239}Pu(n,γ)	^{235}U(n,f)	^{235}U(n,f)	^{235}U(n,f)	^{235}U(n,f)	^{238}U(n,γ)	^{235}U(n,γ)	^{238}U(n,γ)
rr ^{240}Pu(n,γ)	^{235}U(n,f)	^{235}U(n,f)	^{235}U(n,f)	^{238}U(n,γ)	^{238}U(n,γ)	^{238}U(n,γ)	^{238}U(n,γ)
rr ^{241}Pu(n,γ)	^{235}U(n,f)	^{235}U(n,f)	^{235}U(n,f)	^{235}U(n,f)	^{238}U(n,γ)	^{238}U(n,γ)	^{238}U(n,γ)
rr ^{235}U(n,f)	^{235}U MF5	^{235}U(n,f)	^{235}U(n,f)	^{235}U MF5	^{238}U(n,γ)	^{238}U(n,γ)	^{238}U(n,γ)
rr ^{238}U(n,f)	^{235}U MF5	^{235}U MF5	^{235}U MF5	^{235}U MF5	^{235}U MF5	^{235}U MF5	^{239}Pu MF5
rr ^{239}Pu(n,f)	^{239}Pu(n,el)	^{239}Pu(n,el)	^{239}Pu(n,el)	^{239}Pu(n,f)	^{239}Pu(n,f)	^{239}Pu(n,f)	^{239}Pu(n,f)
rr ^{240}Pu(n,f)	^{235}U MF5	^{235}U MF5	^{235}U MF5	^{235}U MF5	^{235}U MF5	^{235}U MF5	^{239}Pu MF5
rr ^{241}Pu(n,f)	^{235}U(n,f)	^{235}U(n,f)	^{235}U(n,f)	^{235}U(n,f)	^{238}U(n,γ)	^{238}U(n,γ)	^{238}U(n,γ)
Total uncertainties (due to transport data and fission yields, in %) for							
rr ^{235}U(n,γ)	2.05	2.04	2.07	2.14	2.31	2.54	2.79
rr ^{238}U(n,γ)	1.75	1.74	1.69	1.65	1.55	1.37	1.25
rr ^{239}Pu(n,γ)	1.22	1.12	1.09	1.13	1.36	1.68	2.05
rr ^{240}Pu(n,γ)	0.64	0.98	0.64	0.72	0.96	1.27	1.61
rr ^{241}Pu(n,γ)	1.35	1.20	1.16	1.17	1.38	1.69	2.09
rr ^{235}U(n,f)	0.52	0.56	0.69	0.87	1.21	1.61	2.07
rr ^{238}U(n,f)	6.61	5.91	5.29	4.84	4.31	3.91	3.70
rr ^{239}Pu(n,f)	1.99	1.84	1.77	1.77	1.92	2.17	2.53
rr ^{240}Pu(n,f)	2.68	2.45	2.27	2.18	2.14	2.22	2.49
rr ^{241}Pu(n,f)	1.34	1.21	1.15	1.17	1.36	1.67	2.06

Results on macroscopic cross sections

	Burn-up (GWd/MTU)						
	0	10	20	30	40	50	60
Σ_{abs1}	^{235}U(n,f)	^{235}U(n,f)	^{235}U(n,f)	^{239}Pu MF1	^{239}Pu MF1	^{238}U(n,γ)	^{238}U(n,γ)
Σ_{abs2}	^{235}U(n,el)	^{235}U(n,f)	^{235}U(n,f)	^{235}U(n,γ)	^{238}U(n,γ)	^{238}U(n,γ)	^{238}U(n,γ)
Σ_{fiss1}	^{235}U(n,f)	^{235}U(n,f)	^{235}U(n,f)	^{235}U(n,f)	^{235}U(n,f)	^{239}Pu MF5	^{239}Pu MF5
Σ_{fiss2}	^{235}U(n,el)	^{235}U(n,f)	^{235}U(n,f)	^{235}U(n,f)	^{238}U(n,γ)	^{238}U(n,γ)	^{238}U(n,γ)
$\nu\Sigma_{fiss1}$	^{235}U(n,f)	^{235}U(n,f)	^{235}U(n,f)	^{235}U(n,f)	^{238}U ν-bar	^{238}U MF1	^{238}U MF1
$\nu\Sigma_{fiss2}$	^{235}U(n,el)	^{235}U(n,f)	^{235}U(n,f)	^{238}U MF1	^{238}U(n,γ)	^{238}U(n,γ)	^{238}U(n,γ)
D_{iff1}	^{235}U MF5	^{235}U MF5	^{235}U MF5	^{235}U MF5	^{235}U MF5	^{239}Pu MF5	^{239}Pu MF5
D_{iff2}	^{238}U MF4	F. P.	F. P.	F. P.	F. P.	^{235}U MF5	^{239}Pu MF5

Total uncertainties (due to transport data and fission yields, in %) for

	0	10	20	30	40	50	60
Σ_{abs1}	1.08	1.11	1.09	1.04	1.07	1.06	1.08
Σ_{abs2}	1.12	1.06	1.13	1.28	1.50	1.74	2.00
Σ_{fiss1}	1.71	1.75	1.74	1.73	1.76	1.83	2.00
Σ_{fiss2}	1.63	1.44	1.40	1.52	1.73	2.03	2.36
$\nu\Sigma_{fiss1}$	1.98	2.01	2.02	2.07	2.15	2.28	2.46
$\nu\Sigma_{fiss2}$	1.63	1.40	1.39	1.51	1.74	2.03	2.37
D_{iff1}	1.88	1.49	1.32	1.19	1.10	1.06	1.01
D_{iff2}	1.22	1.63	1.62	1.62	1.56	1.56	1.56

Results on number densities for actinides

	Burn-up (GWd/MTU)						Cooling time (years)	
	0	20	30	40	50	60	0	300
^{234}U	-	^{235}U(n,f)	^{235}U(n,f)	^{235}U(n,f)	^{235}U(n,f)	^{235}U(n,f)	^{235}U(n,f)	^{235}U(n,γ)
^{235}U	-	^{235}U(n,f)	^{235}U(n,f)	^{238}U(n,γ)	^{238}U(n,γ)	^{238}U(n,γ)	^{238}U(n,γ)	^{238}U(n,γ)
^{236}U	-	^{235}U(n,γ)	^{235}U(n,γ)	^{235}U(n,γ)	^{235}U(n,γ)	^{235}U(n,γ)	^{235}U(n,γ)	^{235}U(n,γ)
^{238}U	-	^{238}U(n,γ)	^{238}U(n,γ)	^{238}U(n,γ)	^{238}U(n,γ)	^{238}U(n,γ)	^{238}U(n,γ)	^{238}U(n,γ)
^{237}Np	-	^{235}U MF5	^{235}U MF5	^{235}U MF5	^{235}U MF5	^{235}U MF5	^{235}U(n,γ)	^{235}U(n,γ)
^{238}Pu	-	^{235}U MF5	^{235}U MF5	^{235}U MF5	^{235}U MF5	^{235}U(n,γ)	^{235}U(n,γ)	^{235}U(n,γ)
^{239}Pu	-	^{238}U(n,γ)	^{238}U(n,γ)	^{238}U(n,γ)	^{238}U(n,γ)	^{238}U(n,γ)	^{238}U(n,γ)	^{238}U(n,γ)
^{240}Pu	-	^{235}U(n,f)	^{239}Pu(n,f)	^{239}Pu(n,f)	^{239}Pu(n,f)	^{239}Pu(n,f)	^{239}Pu(n,f)	^{239}Pu(n,f)
^{241}Pu	-	^{235}U(n,f)	^{235}U(n,f)	^{239}Pu(n,f)	^{239}Pu(n,f)	^{238}U(n,γ)	^{238}U(n,γ)	^{239}Pu(n,f)
^{241}Am	-	^{235}U(n,f)	^{238}U(n,γ)	^{238}U(n,γ)	^{238}U(n,γ)	^{238}U(n,γ)	^{238}U(n,γ)	^{238}U(n,γ)

Total uncertainties (due to transport data and fission yields, in %) for

^{234}U	-	0.12	0.41	0.55	0.69	0.93	0.97	1.90
^{235}U	-	0.17	0.72	1.21	1.88	2.79	2.93	2.92
^{236}U	-	1.98	1.96	1.95	1.93	1.91	1.90	1.88
^{238}U	-	0.01	0.02	0.03	0.04	0.04	0.04	0.04
^{237}Np	-	9.50	4.13	3.39	2.98	2.74	2.72	1.83
^{238}Pu	-	12.1	4.98	3.83	3.16	2.74	2.71	2.53
^{239}Pu	-	1.78	2.30	2.60	2.91	3.22	3.26	3.22
^{240}Pu	-	1.93	1.95	2.05	2.22	2.41	2.43	2.34
^{241}Pu	-	2.04	1.52	1.62	1.88	2.19	2.23	2.47
^{241}Am	-	2.11	1.63	1.90	2.44	3.14	3.24	2.26

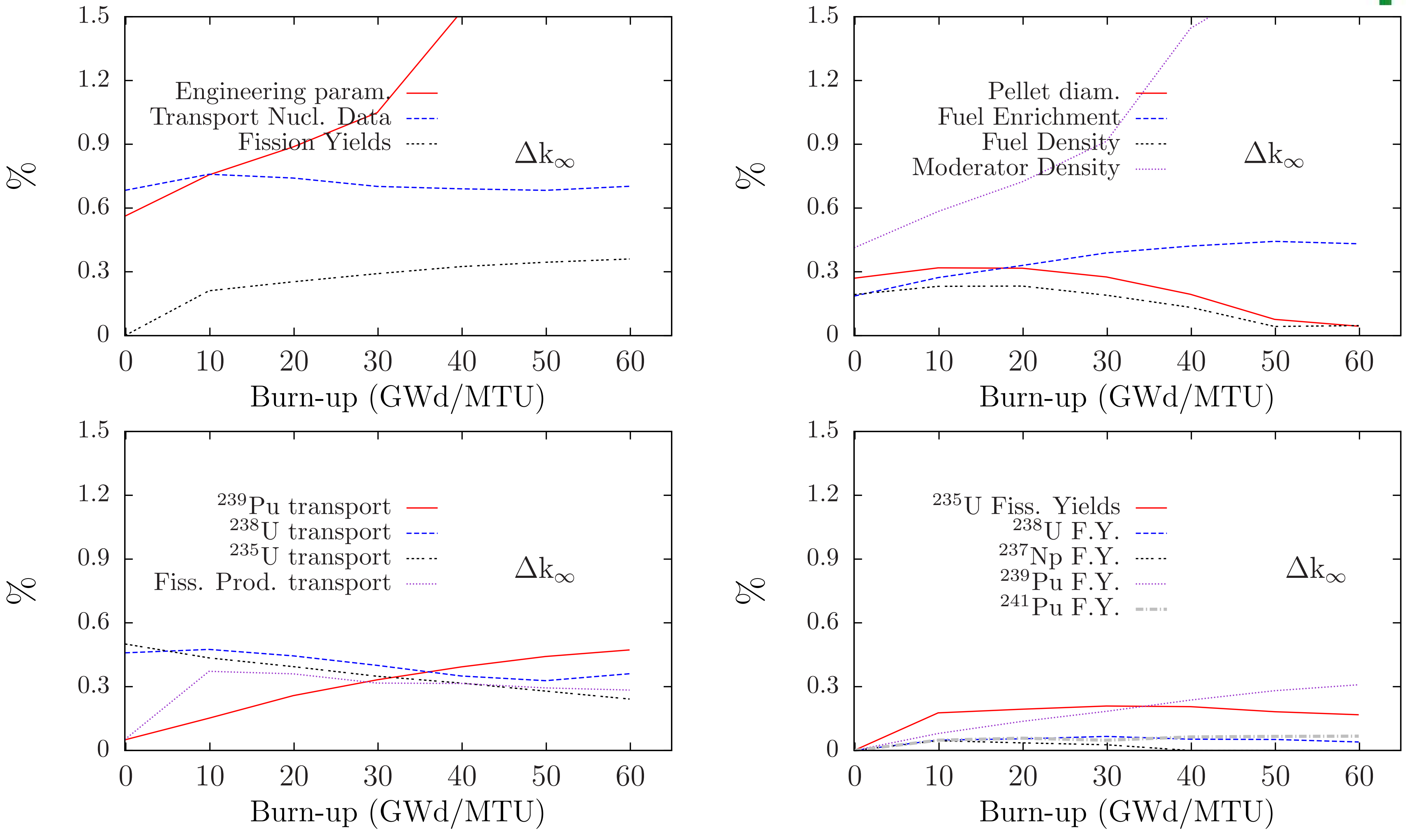
Results on number densities for fission products

	Burn-up (GWd/MTU)						Cooling time (years)	
	0	10	30	40	50	60	0	300
^{99}Tc	-	^{235}U F.Y.	^{235}U F.Y.	^{235}U F.Y.	^{235}U F.Y.	^{235}U F.Y.	^{235}U F.Y.	^{235}U F.Y.
^{133}Cs	-	^{235}U F.Y.	^{235}U F.Y.	^{235}U F.Y.	^{235}U F.Y.	^{239}Pu F.Y.	^{239}Pu F.Y.	^{239}Pu F.Y.
^{140}Ce	-	^{235}U F.Y.	^{239}Pu F.Y.	^{239}Pu F.Y.	^{239}Pu F.Y.	^{239}Pu F.Y.	^{239}Pu F.Y.	^{239}Pu F.Y.
^{143}Nd	-	^{235}U F.Y.	^{235}U F.Y.	^{235}U F.Y.	^{239}Pu F.Y.	^{239}Pu F.Y.	^{239}Pu F.Y.	^{239}Pu F.Y.
^{147}Sm	-	^{235}U F.Y.	^{235}U F.Y.	F.P.	F.P.	F.P.	F.P.	F.P.
^{149}Sm	-	^{235}U F.Y.	^{239}Pu F.Y.	^{239}Pu F.Y.	^{239}Pu F.Y.	^{239}Pu F.Y.	^{239}Pu F.Y.	^{239}Pu F.Y.
^{151}Sm	-	^{235}U F.Y.	F.P.	F.P.	F.P.	F.P.	F.P.	F.P.
^{154}Sm	-	^{235}U F.Y.	^{235}U F.Y.	^{235}U F.Y.	^{239}Pu F.Y.	^{239}Pu F.Y.	^{239}Pu F.Y.	^{239}Pu F.Y.
^{153}Eu	-	^{235}U F.Y.	^{235}U F.Y.	^{235}U F.Y.	^{235}U F.Y.	F.P.	F.P.	F.P.
^{155}Gd	-	F.P.	F.P.	F.P.	F.P.	^{235}U F.Y.	^{235}U F.Y.	^{235}U F.Y.

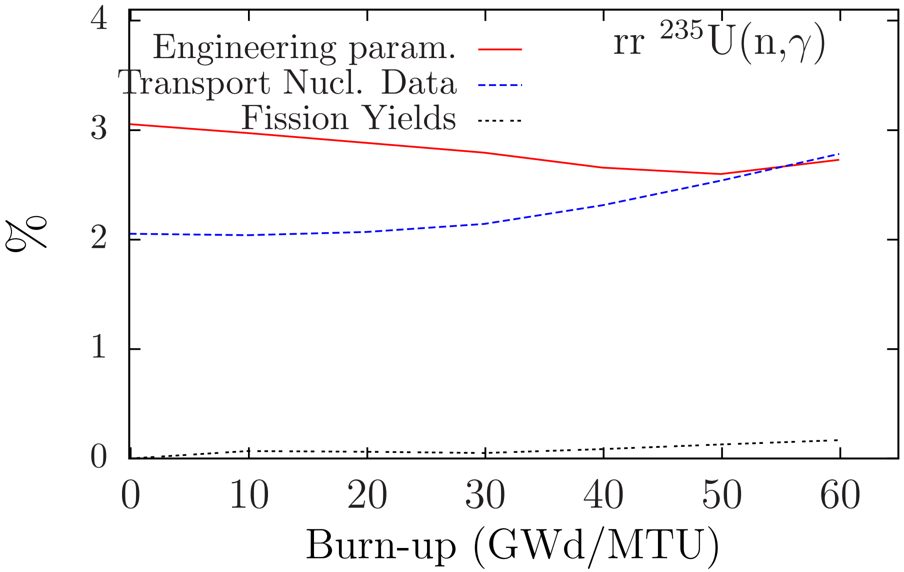
Total uncertainties (due to transport data and fission yields, in %) for

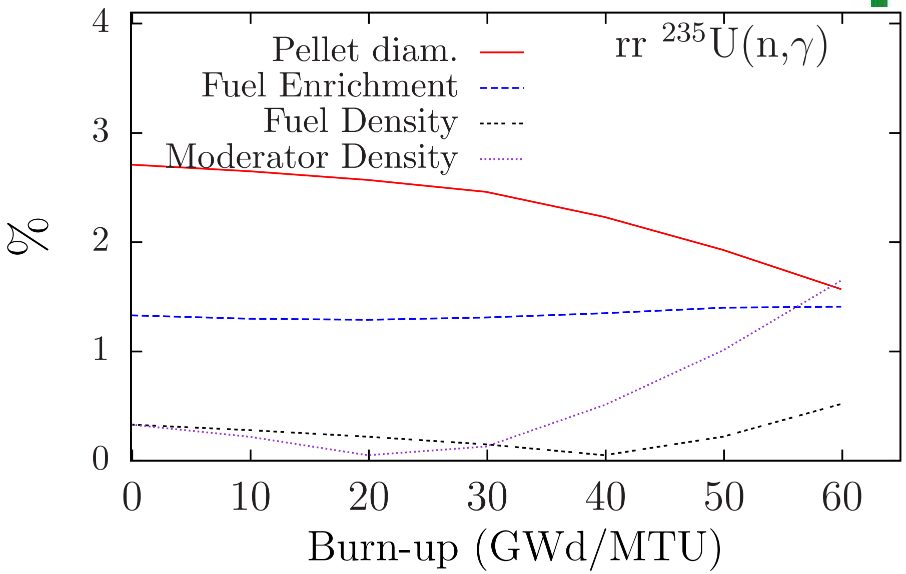
^{99}Tc	-	10.4	9.41	9.19	9.10	9.11	9.12	9.12
^{133}Cs	-	3.50	3.74	4.16	4.72	5.39	5.47	5.45
^{140}Ce	-	2.55	2.78	2.95	3.14	3.34	3.37	3.38
^{143}Nd	-	4.35	4.93	5.42	5.98	6.59	6.67	6.65
^{147}Sm	-	11.4	20.6	25.0	28.7	31.7	32.0	23.8
^{149}Sm	-	11.4	10.8	10.7	11.0	11.3	11.4	10.9
^{151}Sm	-	26.6	22.5	21.5	20.9	20.5	20.5	20.1
^{154}Sm	-	26.2	20.6	19.4	18.6	18.1	18.1	18.1
^{153}Eu	-	13.8	12.1	12.5	12.9	13.3	13.4	13.3
^{155}Gd	-	27.0	22.4	22.4	22.5	22.8	23.0	11.0

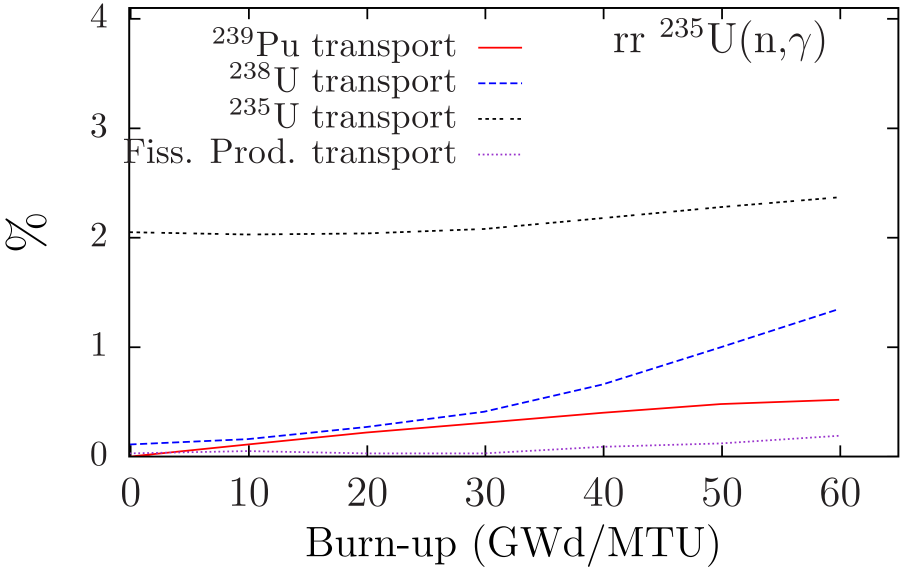
Example for k_∞

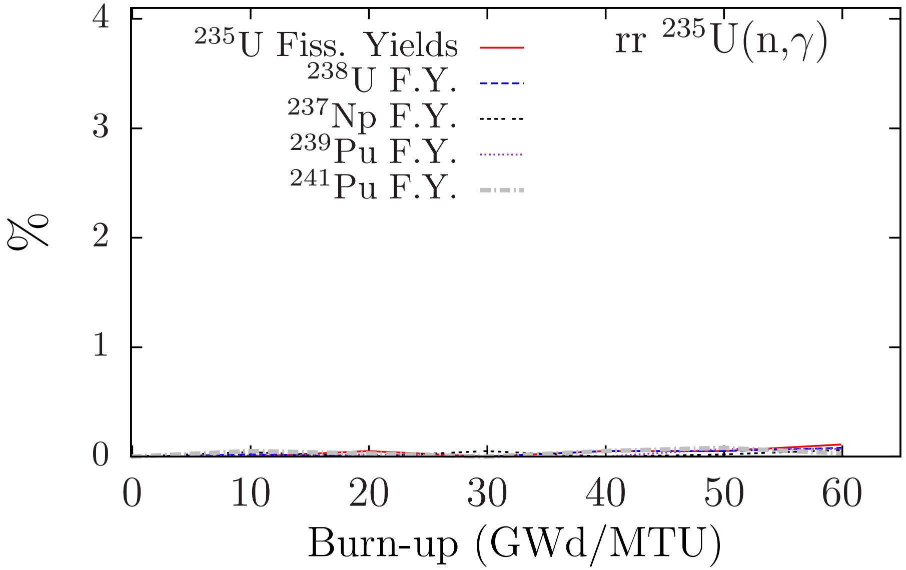


Example for reaction rates

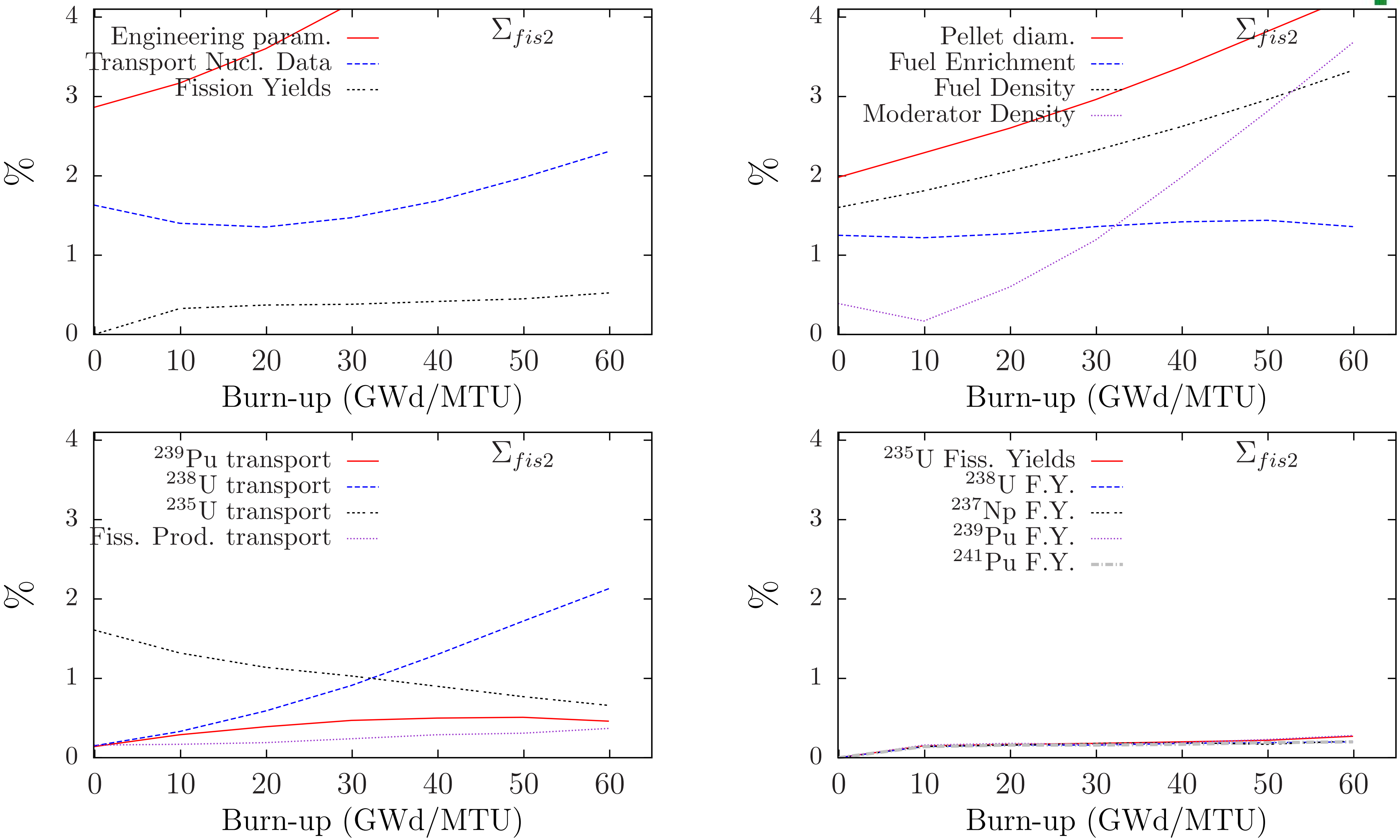




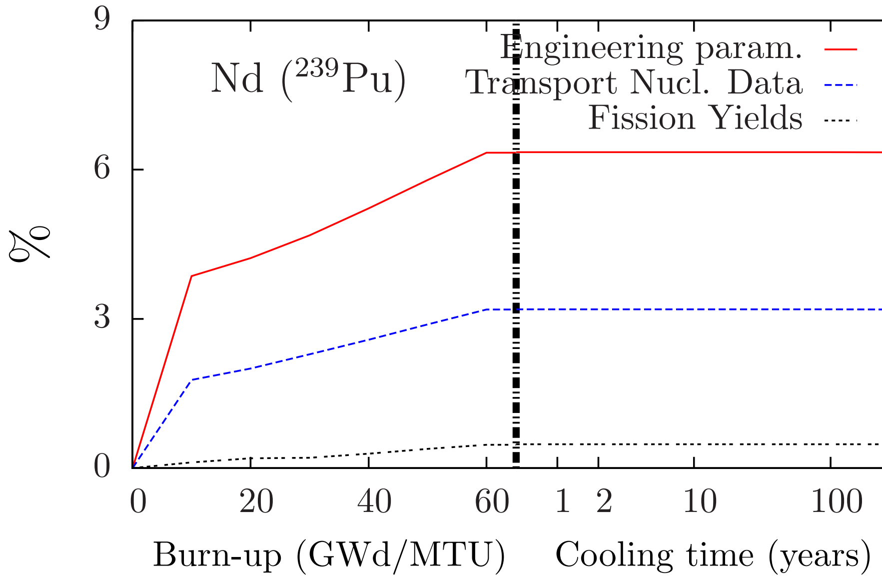


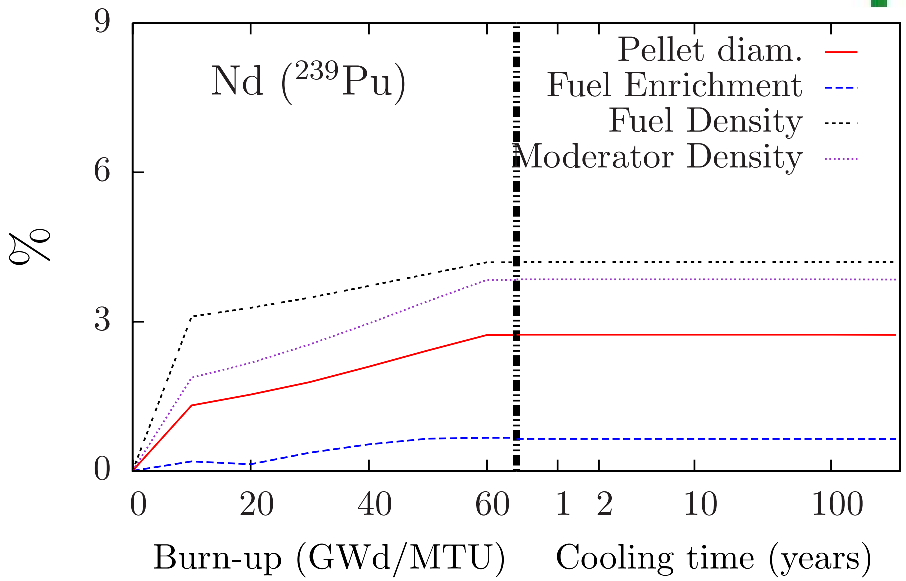


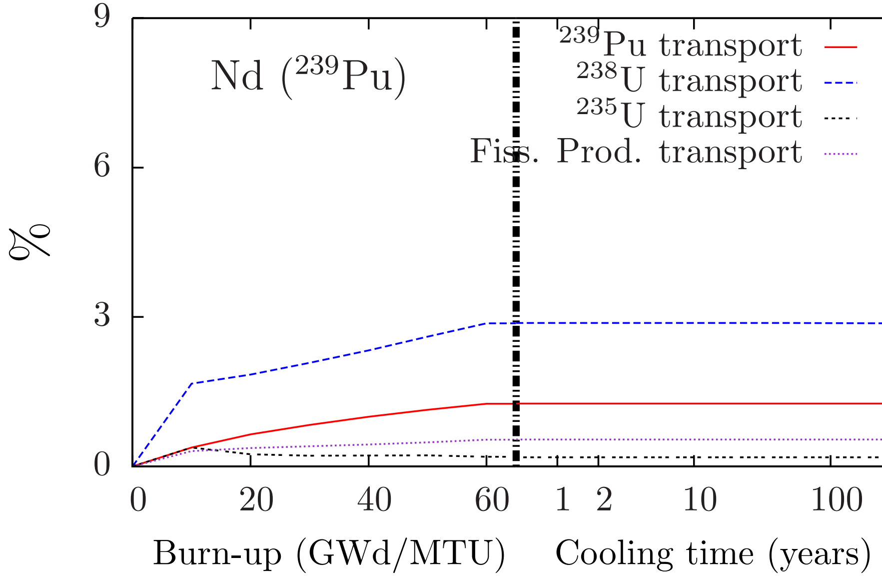
Example for macroscopic cross sections

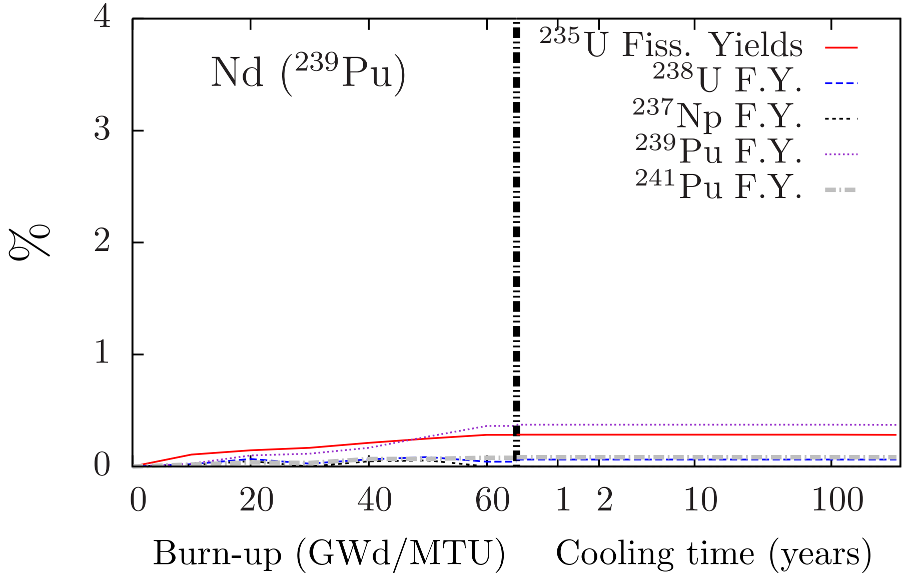


Example for number densities

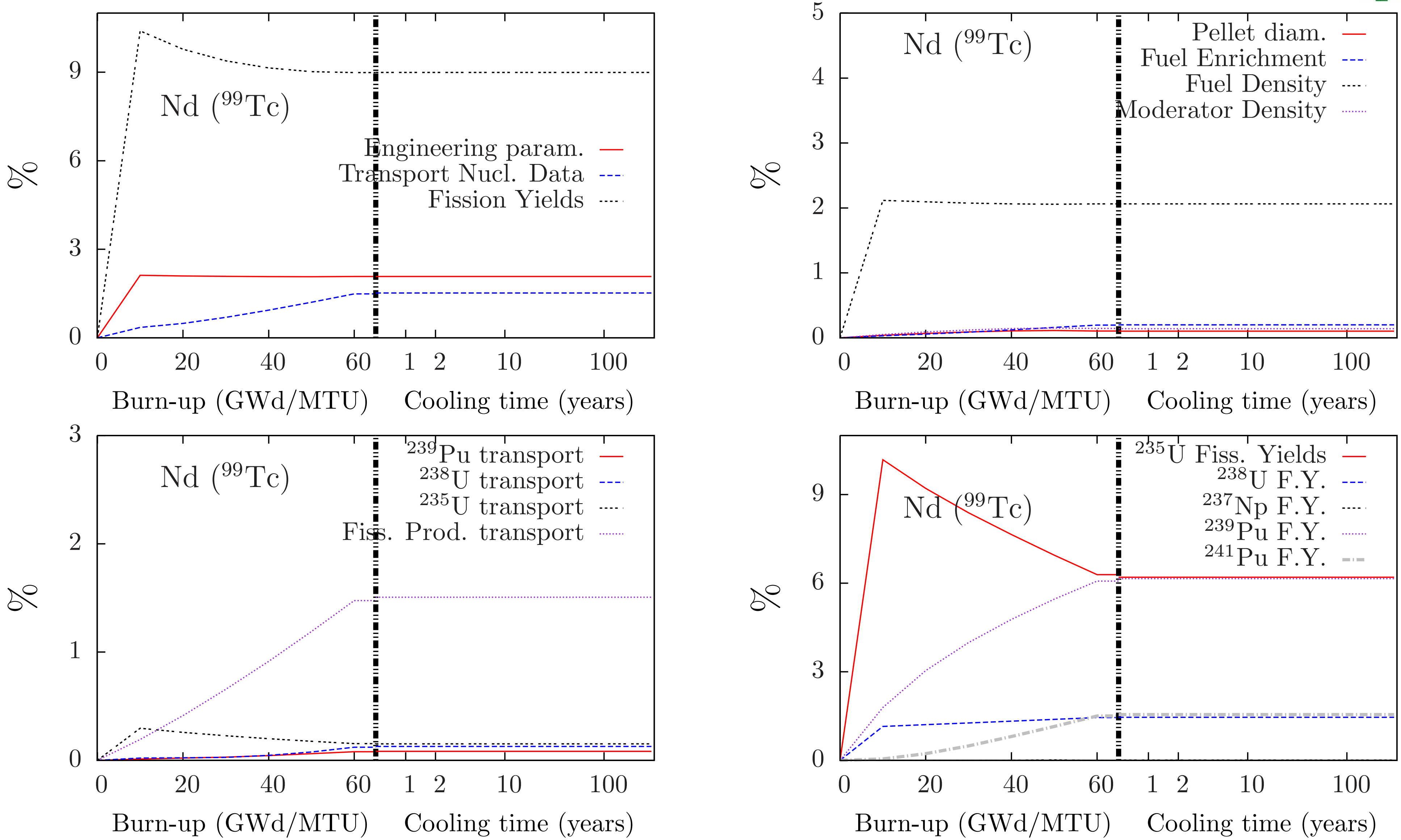








Example for number densities



Conclusions

- TMC successfully applied for this PWR benchmark,
- Transport nuclear data, fission yields and engineering quantities were considered,
- with ^{235,238}U, ^{239,241}Pu, ^{237}Np and lumped fission products.

If we can do a calculation once, we can also do it a 1000 times, each time with a varying data library.