Nuclear data, uncertainties and their applications

Part 3: nuclear data uncertainties for reactors and fuels

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Contents

All slides can be found at:
ftp://ftp.nrg.eu/pub/www/talys/bib_rochman/presentation.html).

Nuclear data uncertainties

General comments:

- uncertainties are not errors (and vice versa),
- uncertainties should now be given with every data (seems obvious ?),
- they are related to risks, quality of work, perception, fear, safety...

- Uncertainty $\leftrightharpoons$ safety $\leftrightharpoons$ professionalism

- True uncertainties do not exist ! They are the reflection of our knowledge and methods.

- All the above for covariances
- The importance of nuclear data uncertainties should be checked. If believed negligible, please prove it !

Nuclear data Uncertainty propagation

- reactors (k_{eff}, β_{eff}, void, Doppler, reaction rates, you name it),
- fuel storage (criticality),
- burn-up (inventory, radiotoxicity),
- transient behaviour...

TMC for nuclear data uncertainty propagation, what else ?

☺ + No MF 32-35 (no 2 Gb files) but every possible cross correlation included
☺ + No approximation but true probability distribution
☺ + Only essential info for an evaluation is stored
☺ + No perturbation code necessary, only "essential" codes
☺ + Feedback to model parameters
☺ + Fission yields and decay data included
☺ + QA
☹ - Needs discipline to reproduce
☹ - Memory and computer time
☹ - Complexity for full reactor core calculation not fully investigated
☹ - Role of data centers would change

TMC for nuclear data uncertainty propagation

"In general, this paper will or will not be a breakthrough in methodology if the [practicality and robustness] can or can not be demonstrated.",

ANE Referee, May 2008

"What about actinides, what about real systems ?",

JEFF &WPEC meetings, May-June 2008

TMC for nuclear data uncertainty propagation

“In general, this paper will or will not be a breakthrough in methodology if the [practicality and robustness] can or can not be demonstrated.”,

ANE Referee, May 2008

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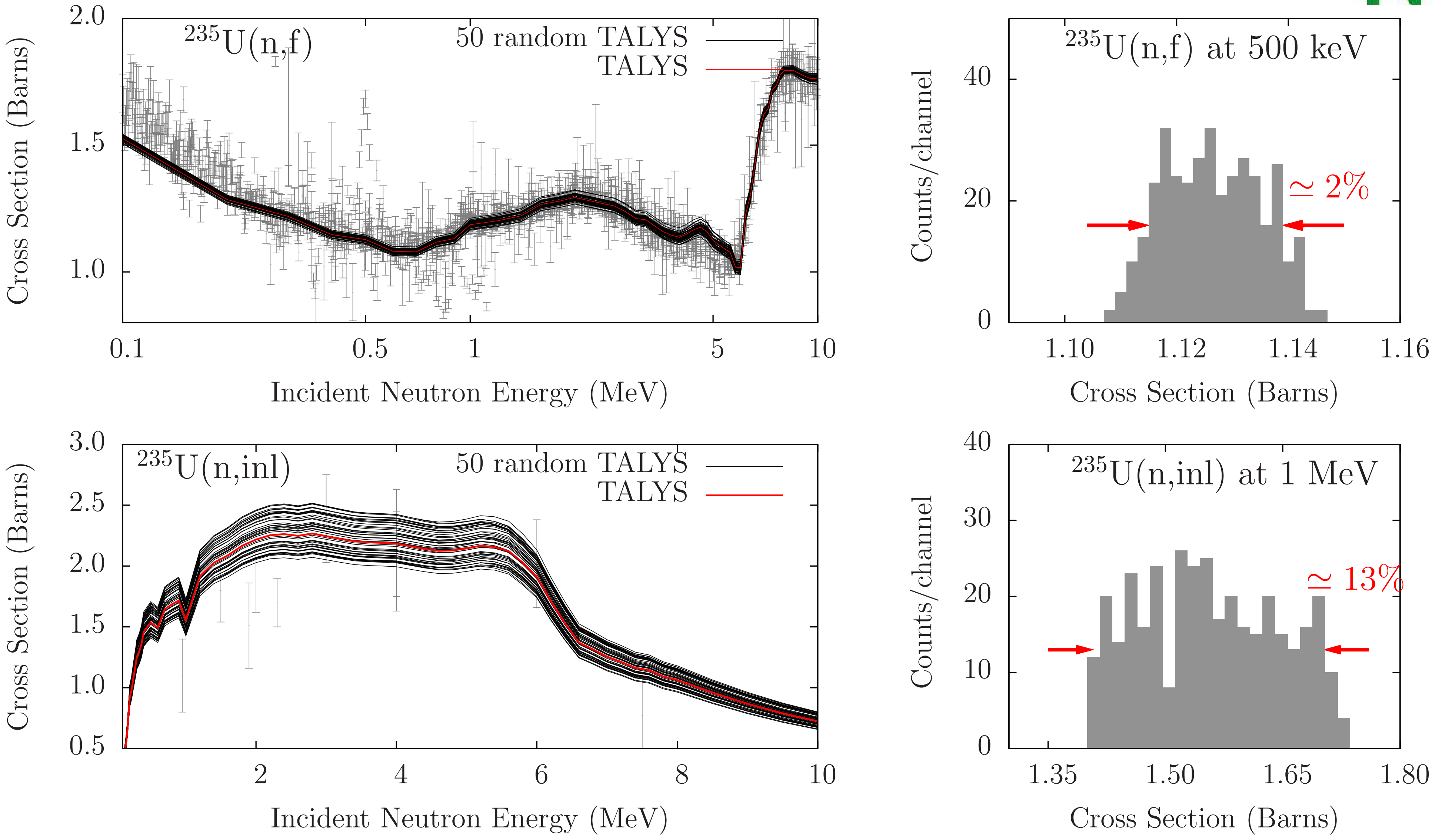
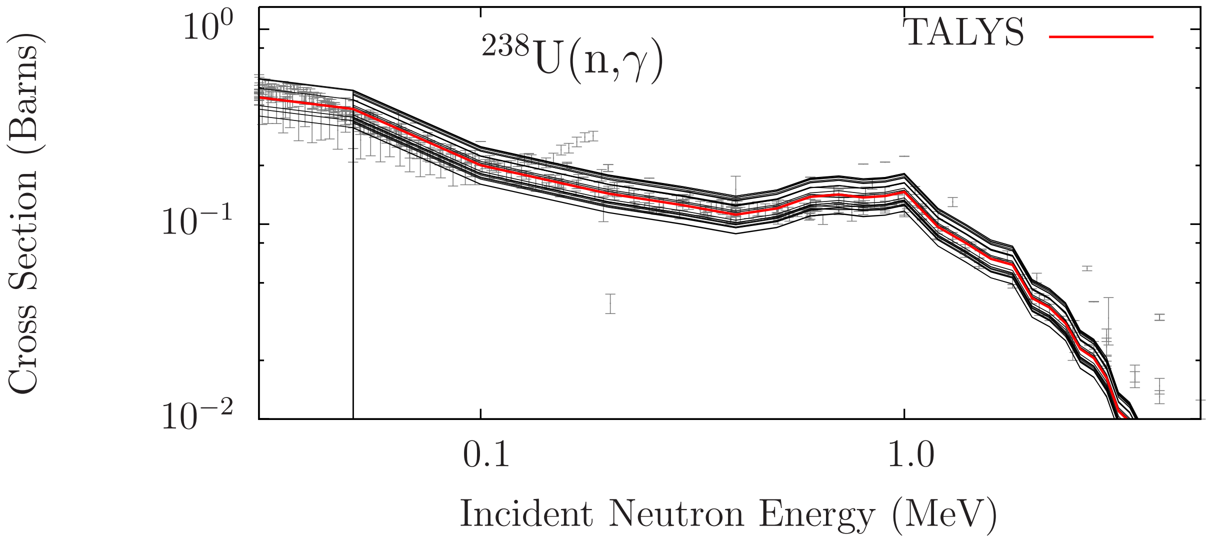
Okay, let’s go from academic solutions (☕) to mass/applied production (🏭) !

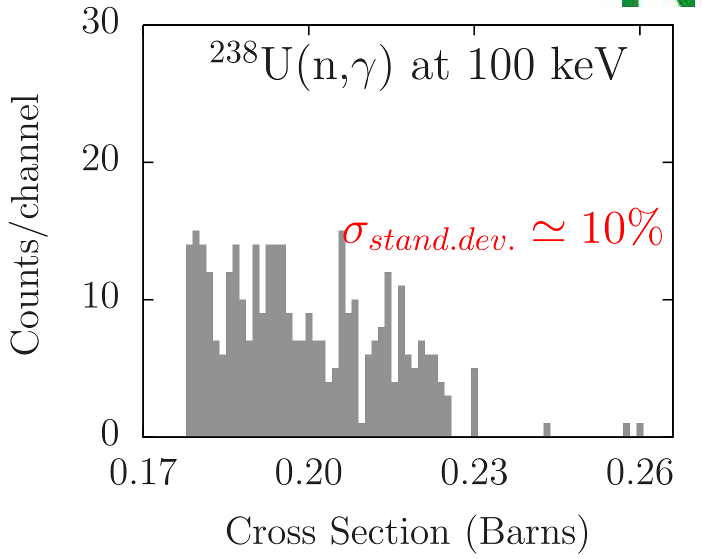
- TALYS calculation + Resonance parameters (RP) + uncertainties
- 100 to 2000 ENDF files per isotope from ^{6}Li to ^{252}Cf
- 190 criticality-safety benchmarks ($>$ 60 000 calculations) from the ICSBP
- All Oktavian shielding benchmarks (neutrons and gammas)
- Reactivity swing for a LWR using an “Inert Matrix Fuel” (Pu and Mo)
- k_{eff} for a HTR (PBMR), ESKOM specifications
- Inventory

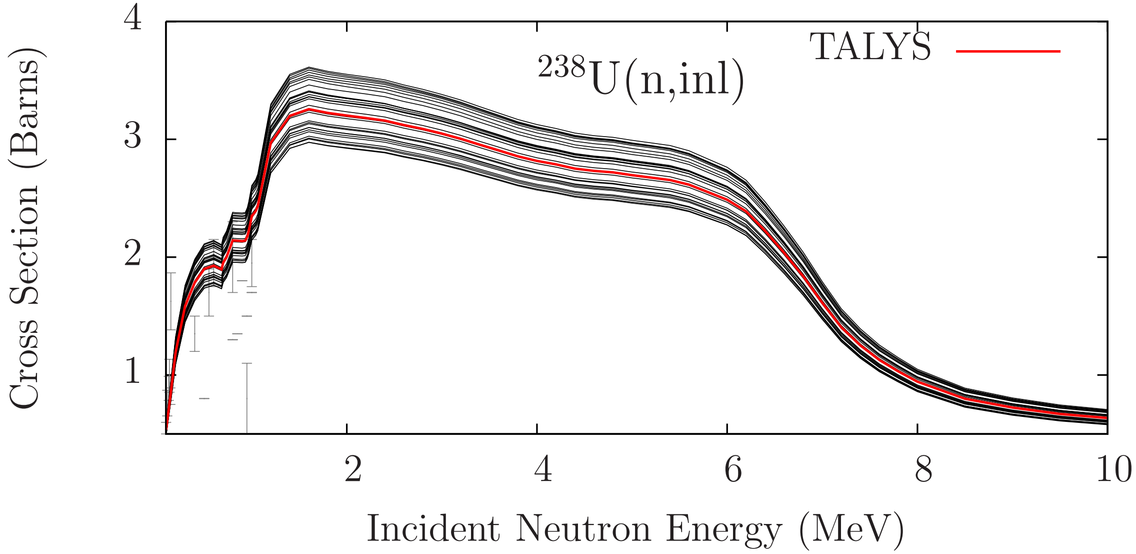
Example with ^{235}U: Monte Carlo calculations

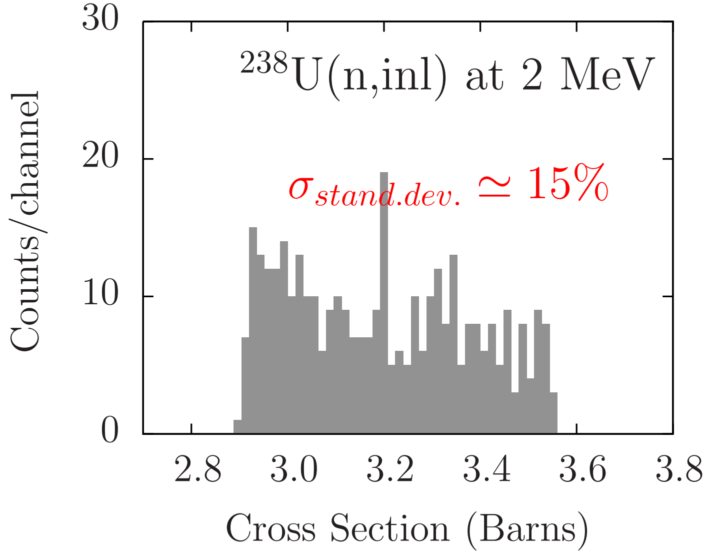


Example with ^{238}U: Monte Carlo calculations

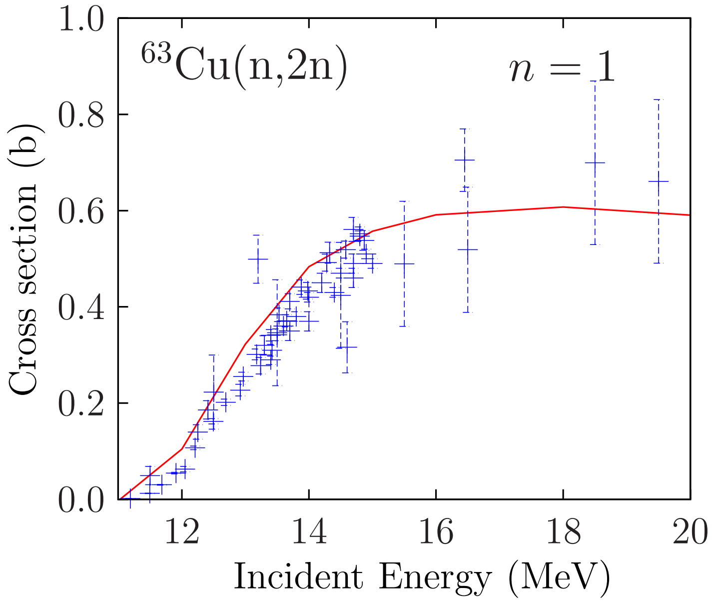


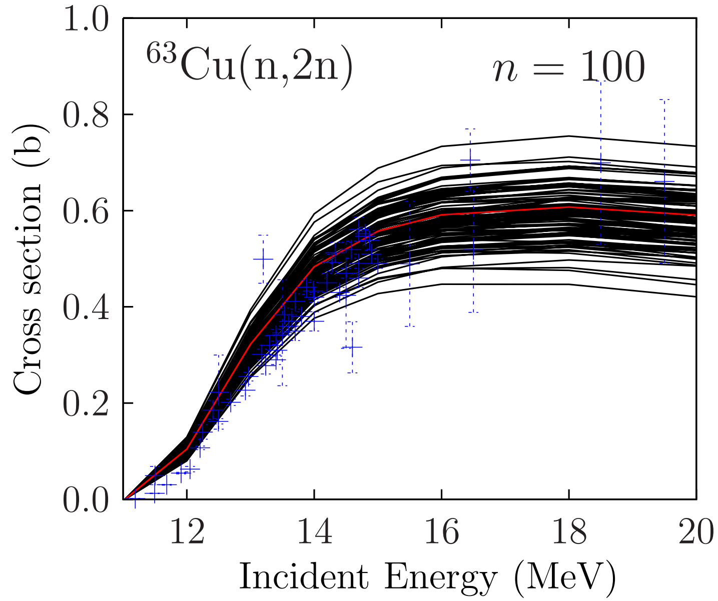


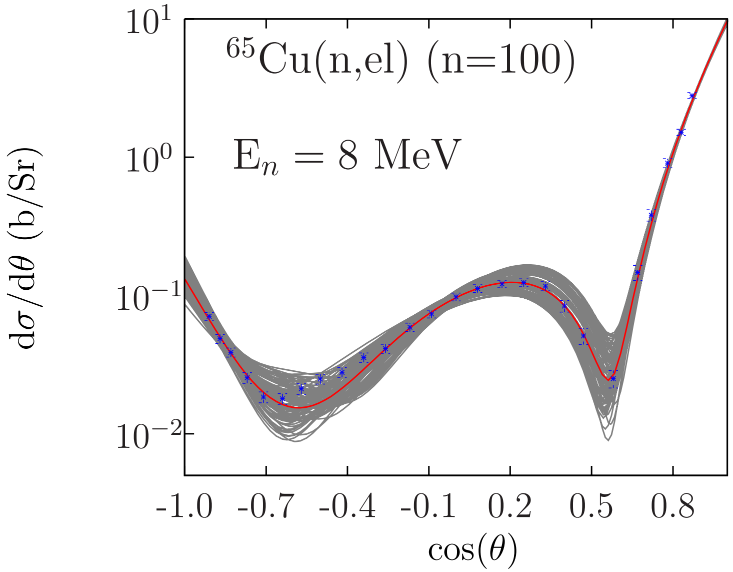


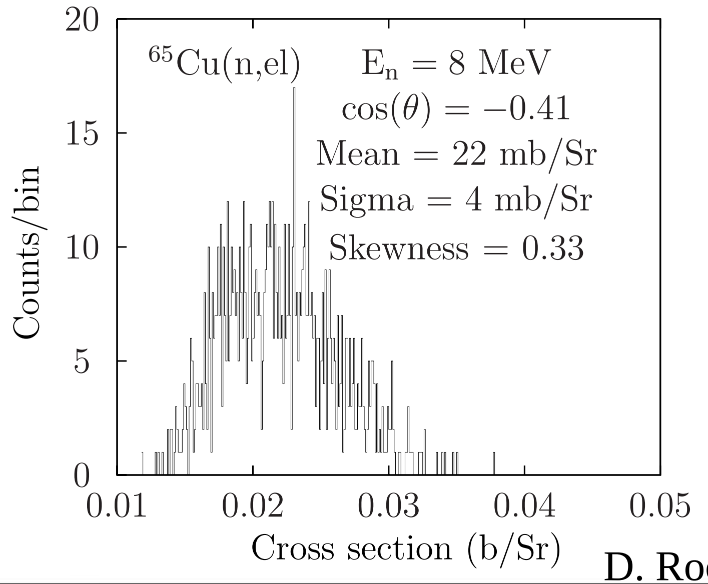


Examples with ^{63}Cu(n,2n) and ^{65}Cu(n,el)

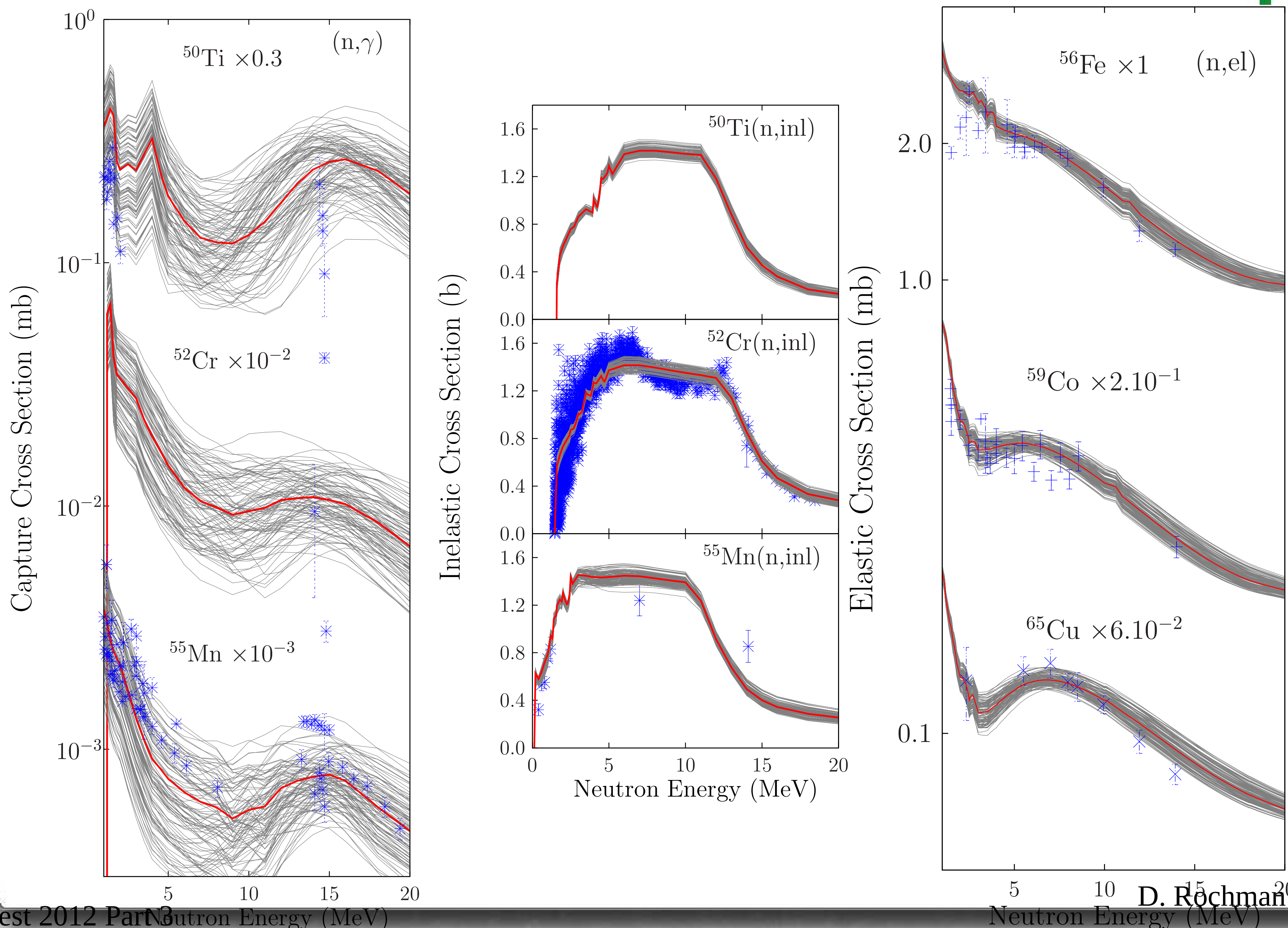




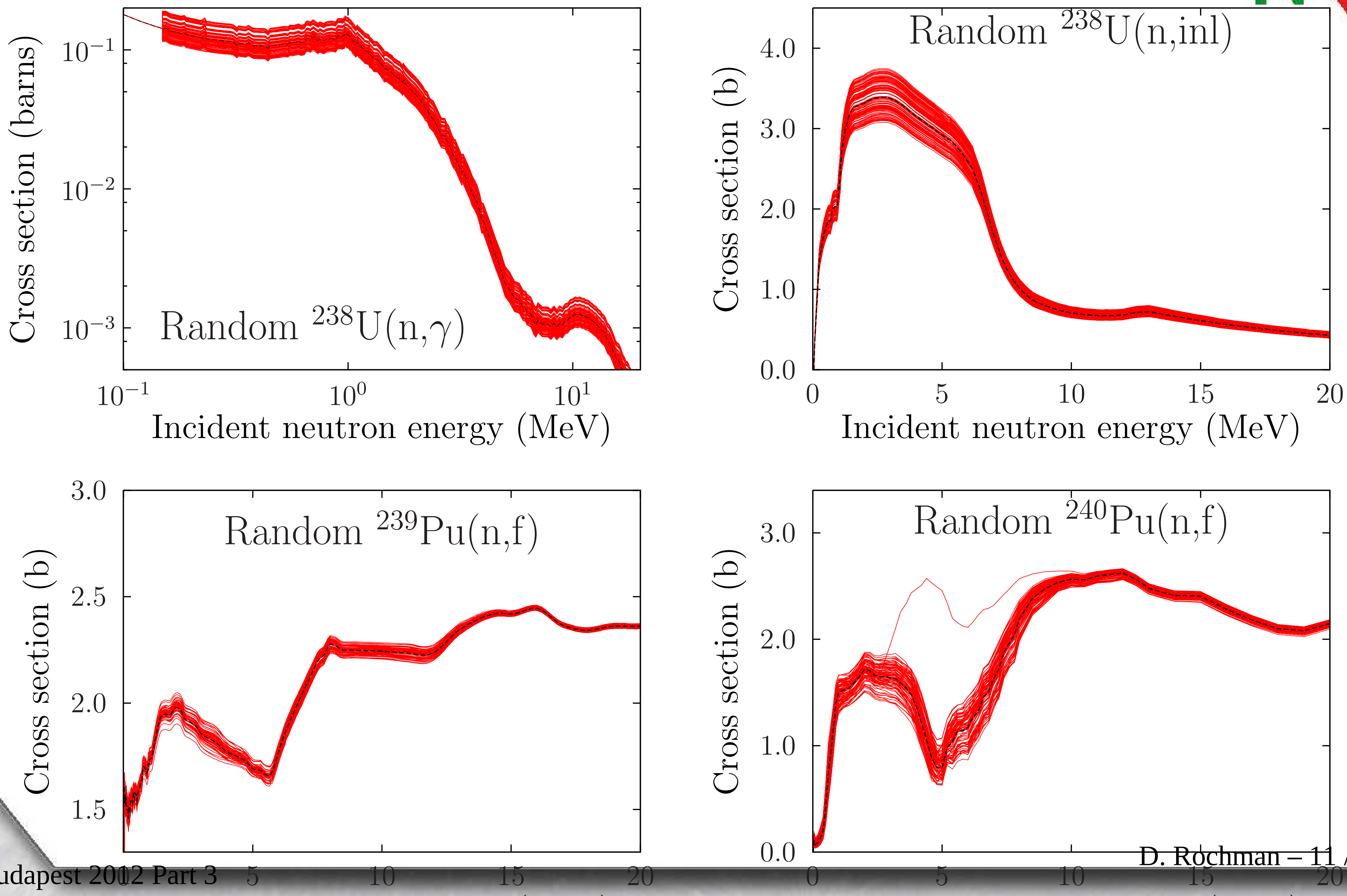




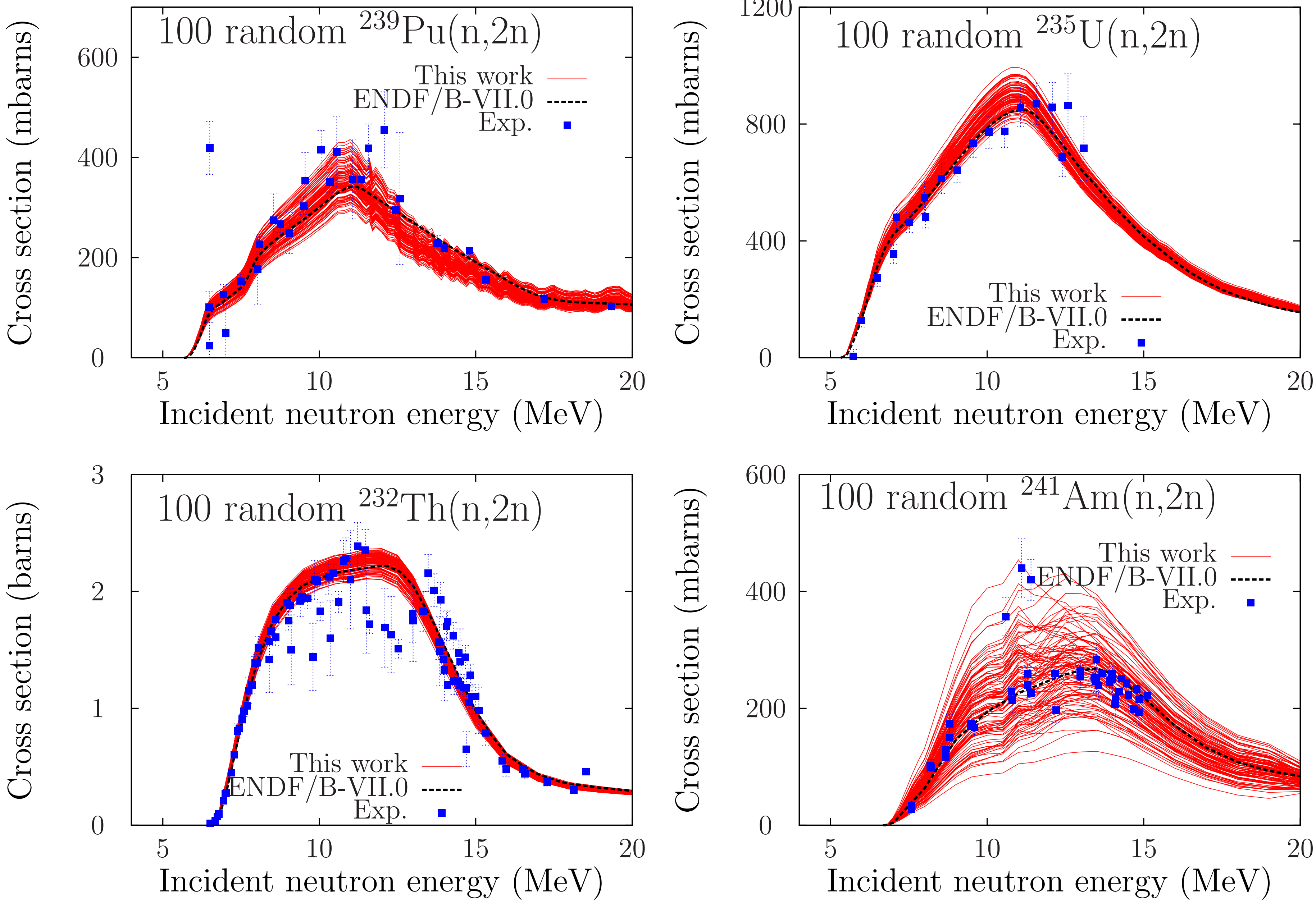
Examples with (n,γ), (n,inl) and (n,el)



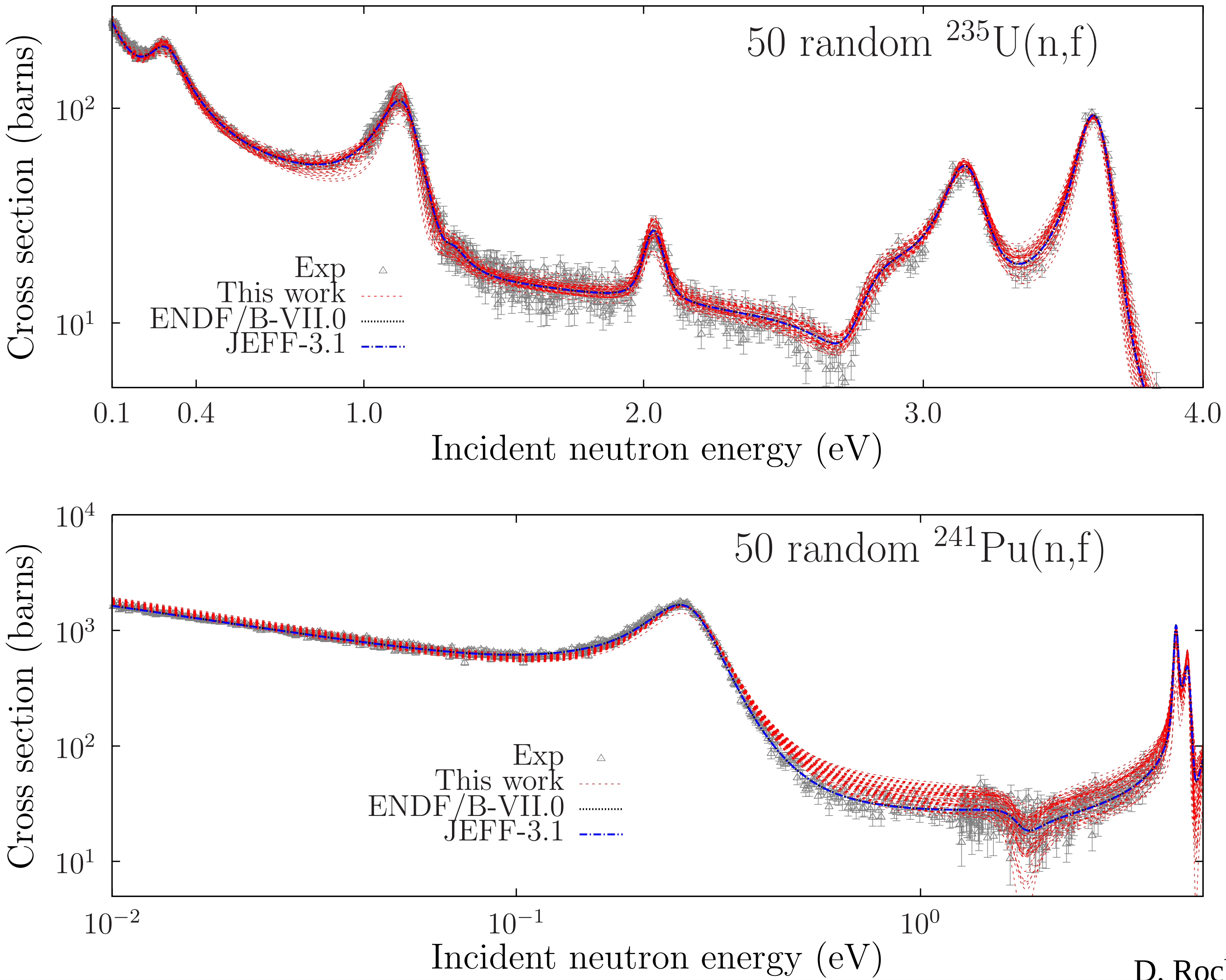
Nuclear data: ^{239}Pu and ^{238}U



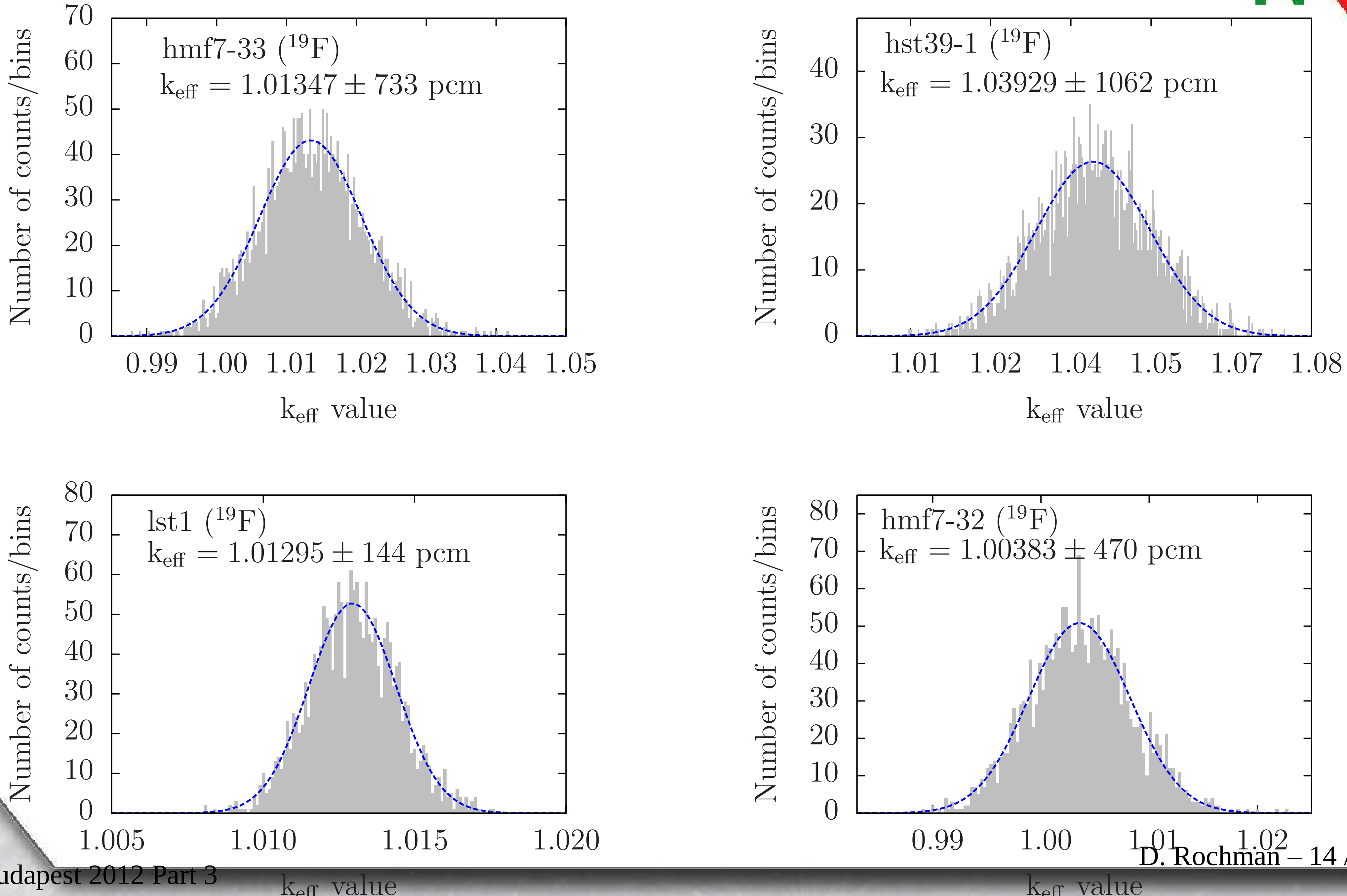
Nuclear data: examples on (n,2n) cross sections



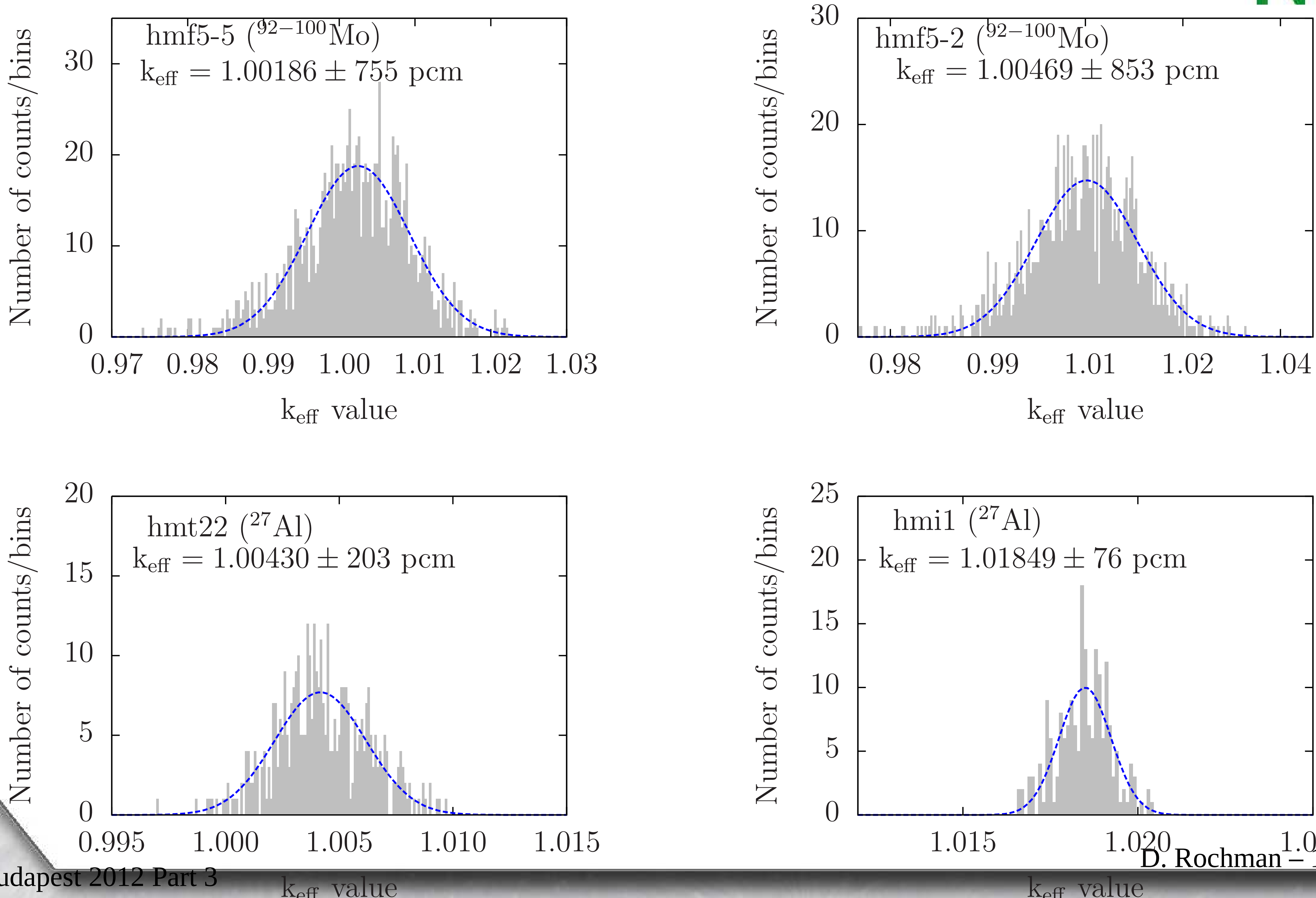
Nuclear data: examples in the resonance region



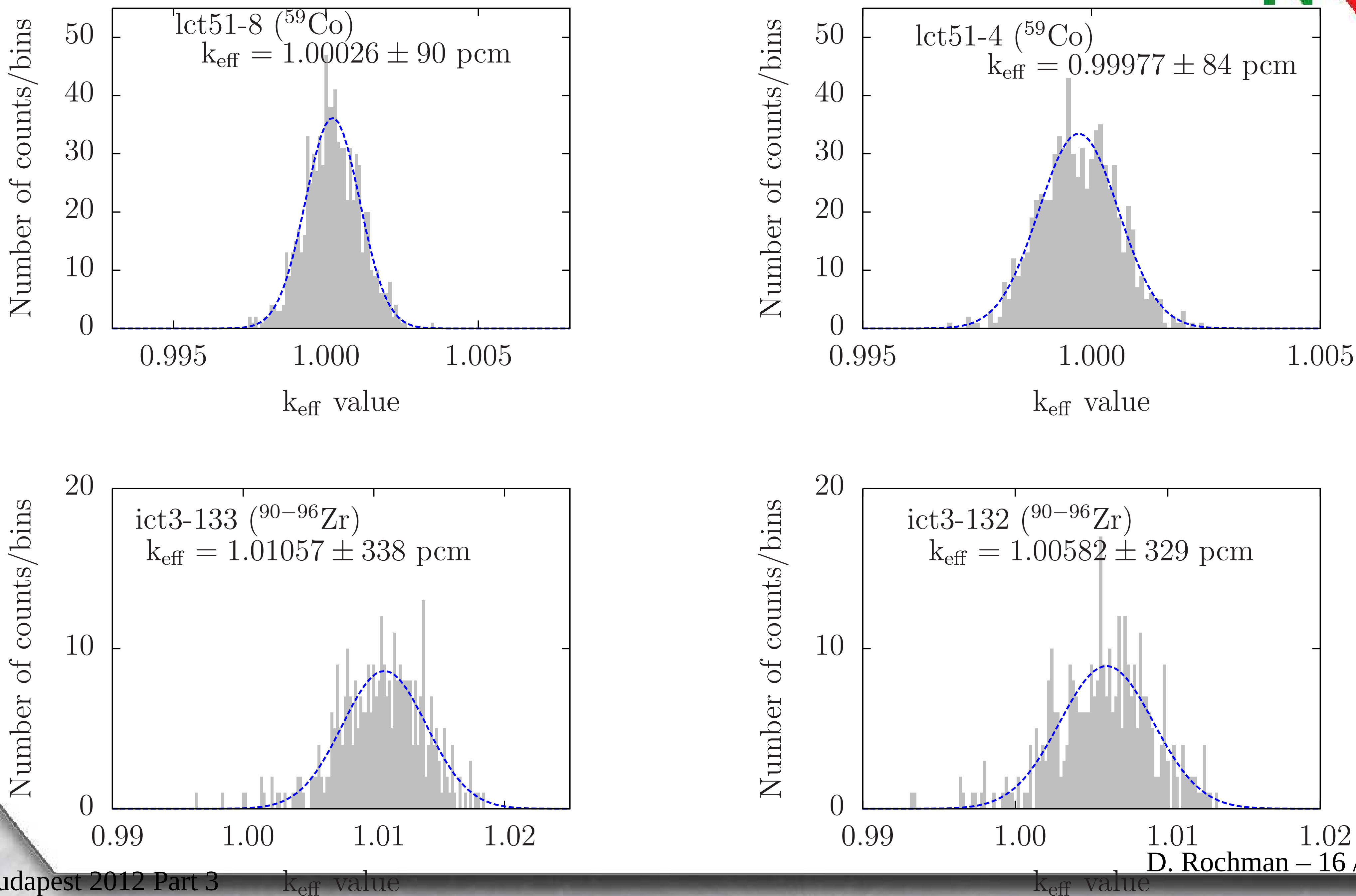
Examples of criticality benchmarks for ^{19}F



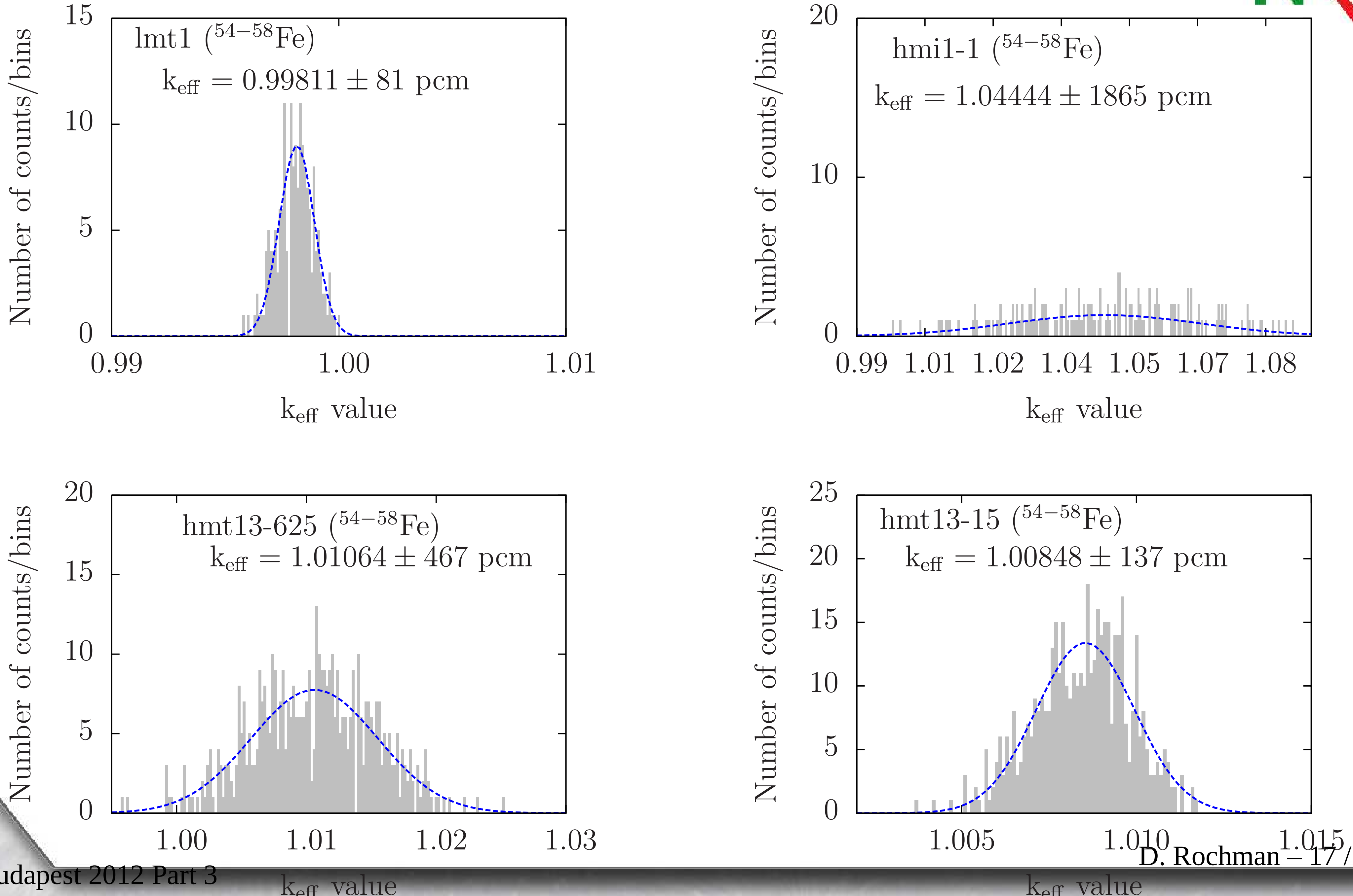
Examples of criticality benchmarks for ^{27}Al and $^{92-100}$Mo



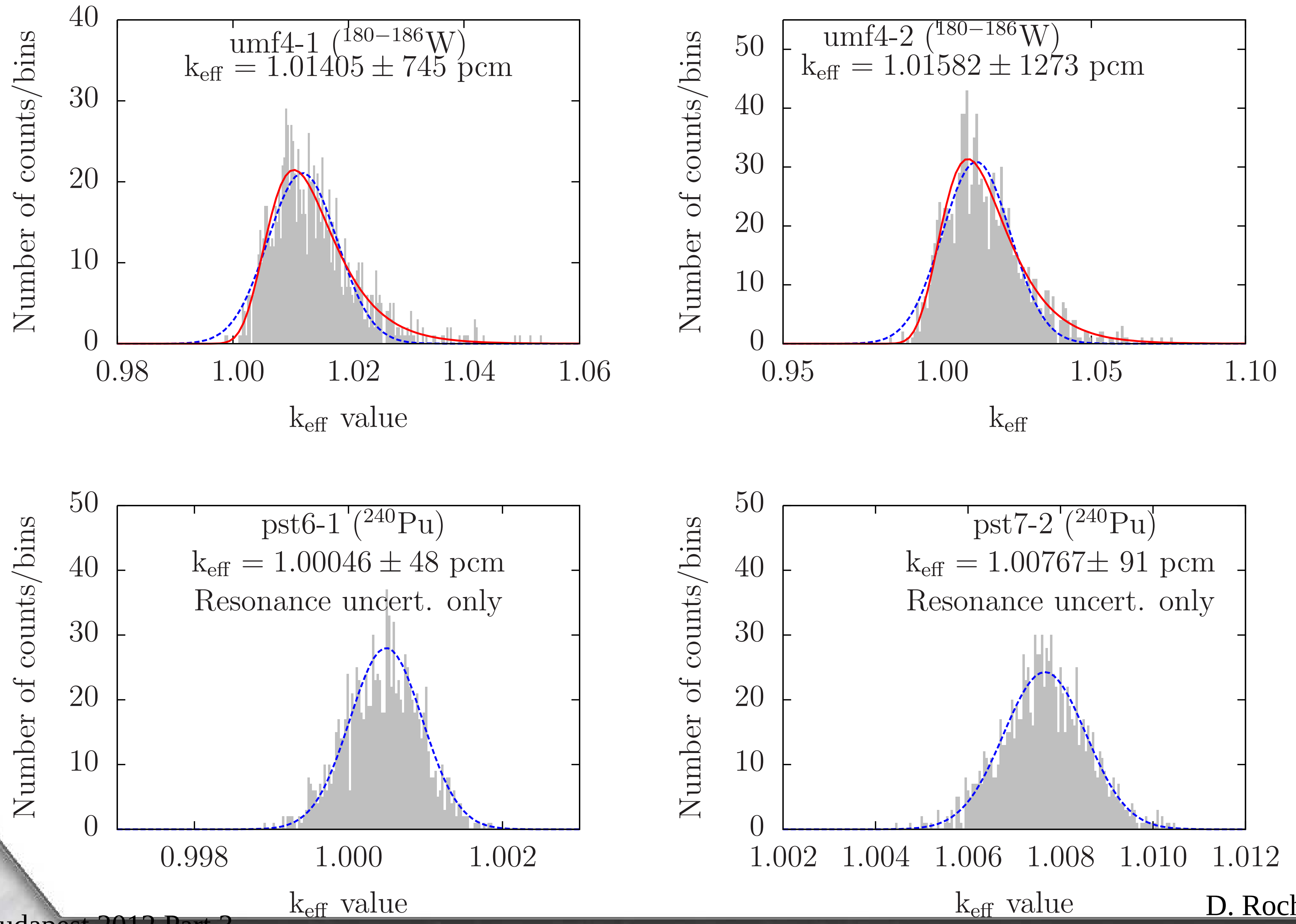
Examples of criticality benchmarks for ^{59}Co and $^{90-96}$Zr



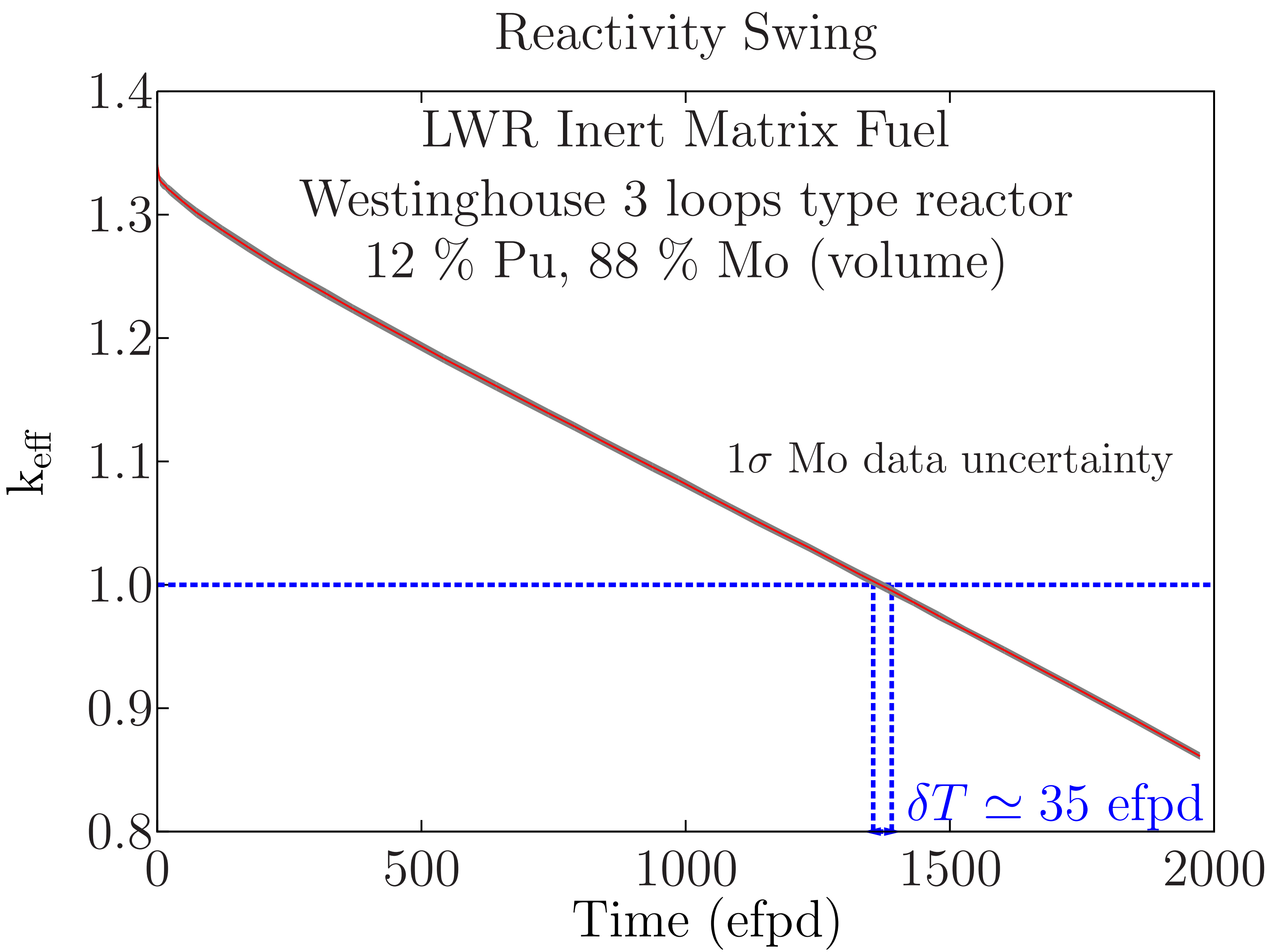
Examples of criticality benchmarks for $^{54-58}Fe$



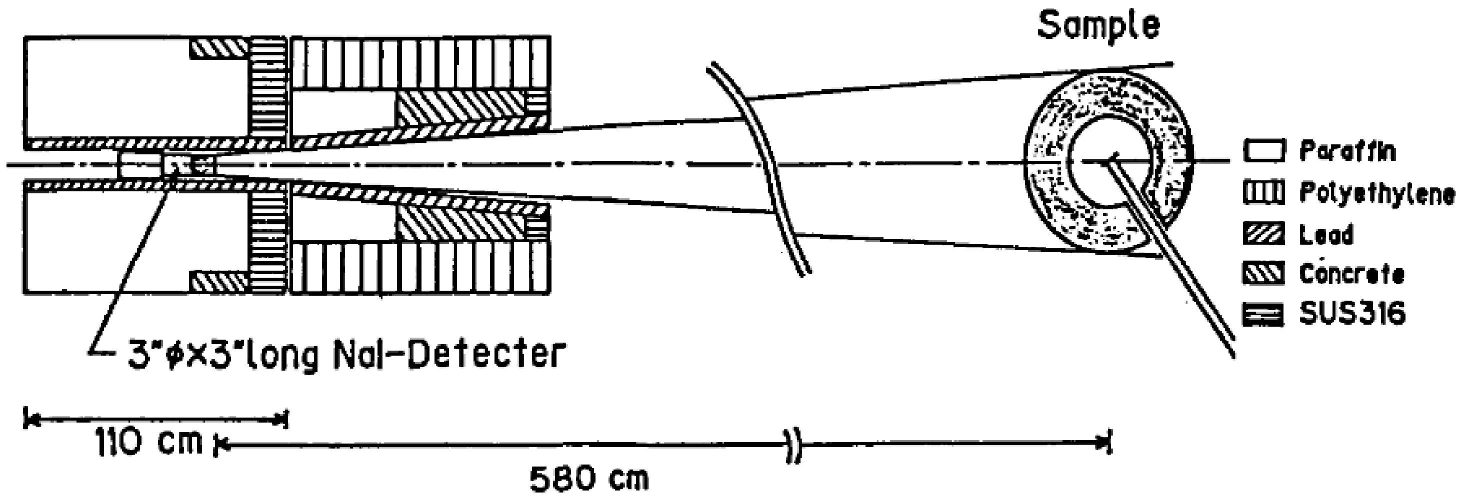
Examples of criticality benchmarks for $^{180-186}$W and ^{240}Pu



Examples of reactivity swing

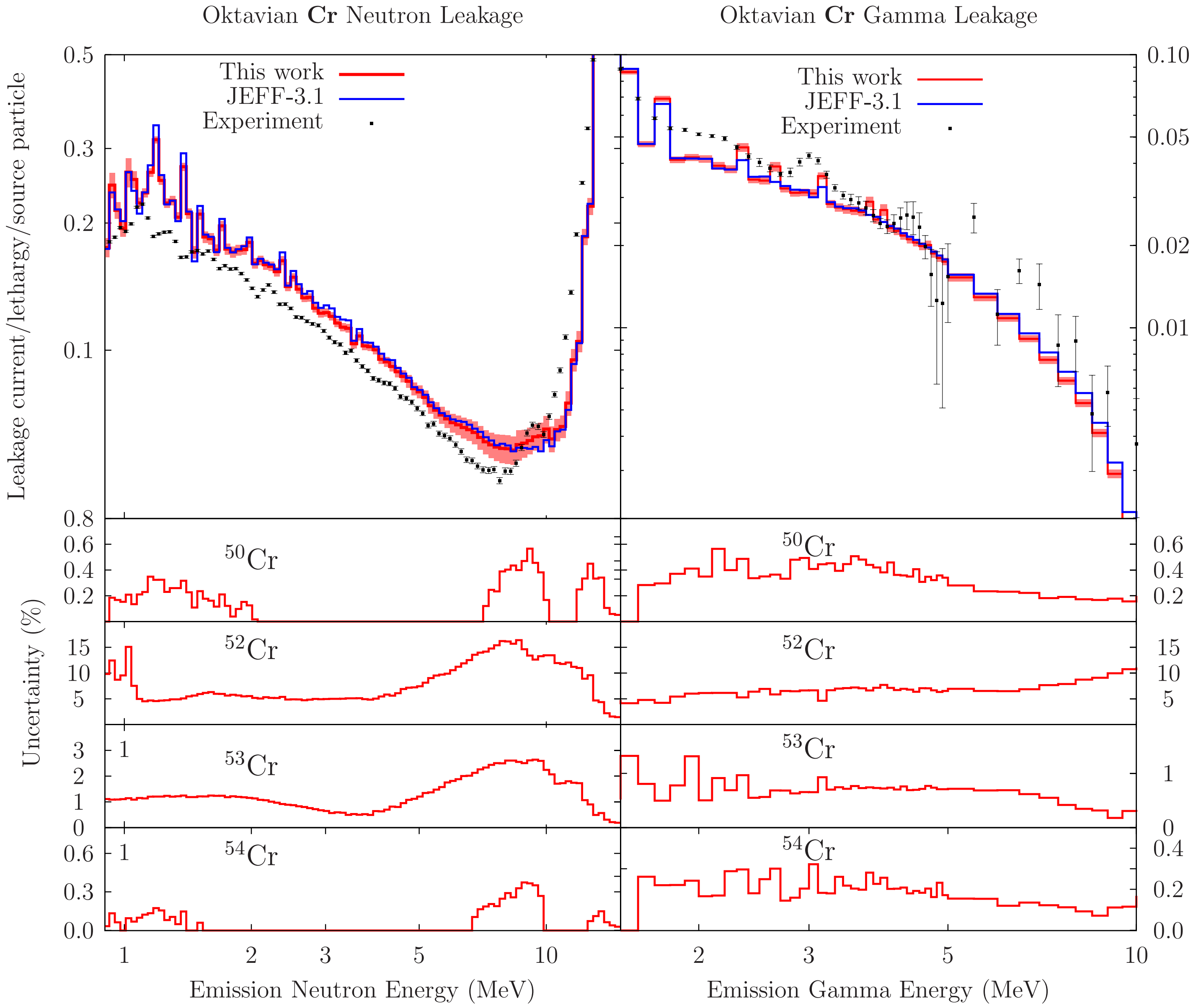


Examples on shielding benchmarks

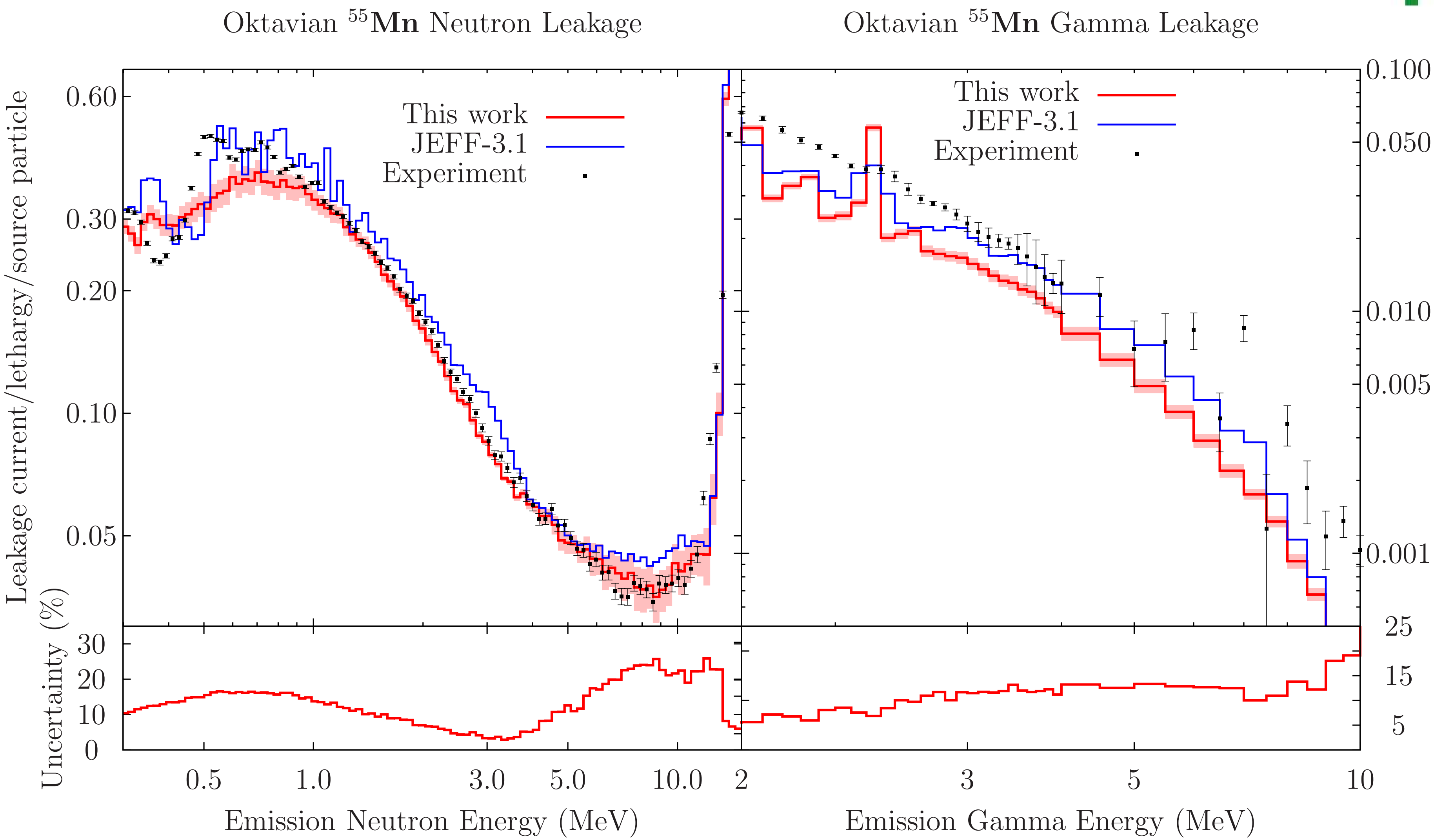


- Oktavian: Leakage current spectrum from the outer surface of a spherical pile of material, 14 MeV D-T neutron source at the center of the pile. (Al, Cu, Si, Ti, Cr, Mn, Co...)
- FNS: Slabs of material of varying thickness, at five different angles, 20 cm from a 14 MeV D-T neutron source. (Fe, W).
- LLNL Pulsed Spheres: Time-of-Flight measurements through spherical shells of varying thickness, 14 MeV D-T neutron source. (Al, Mg, Fe).

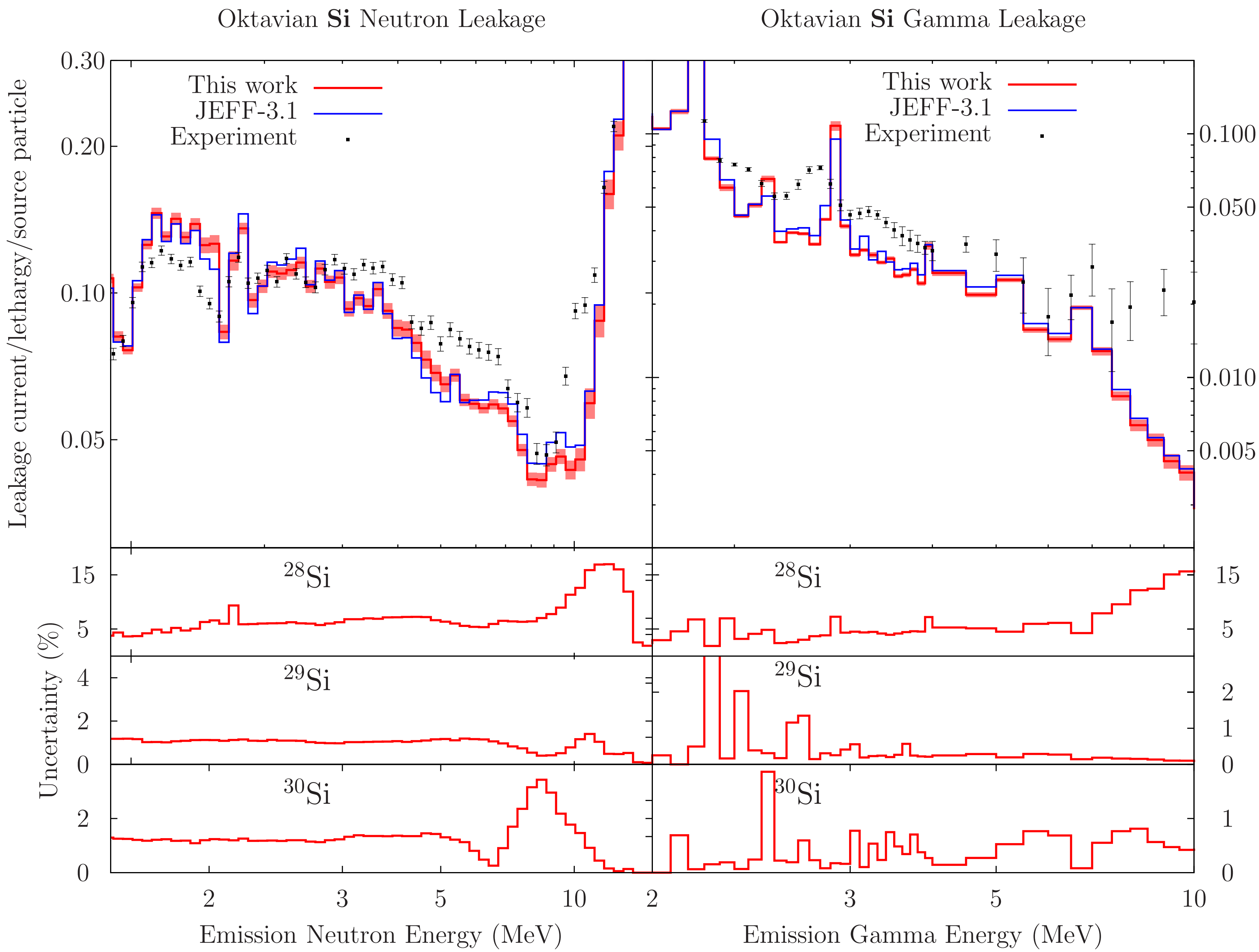
Application for Cr Oktavian benchmark



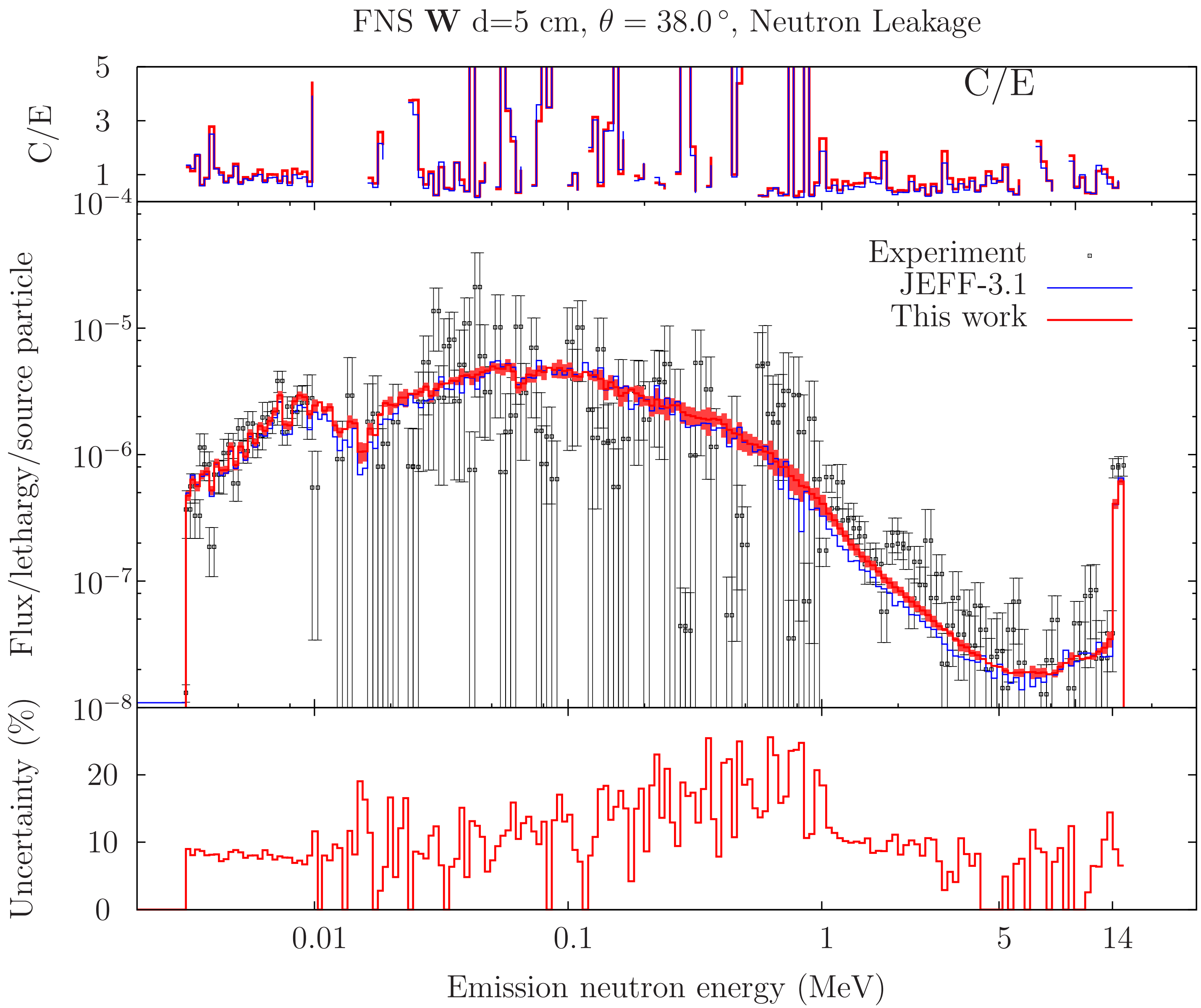
Application for Mn Oktavian benchmark



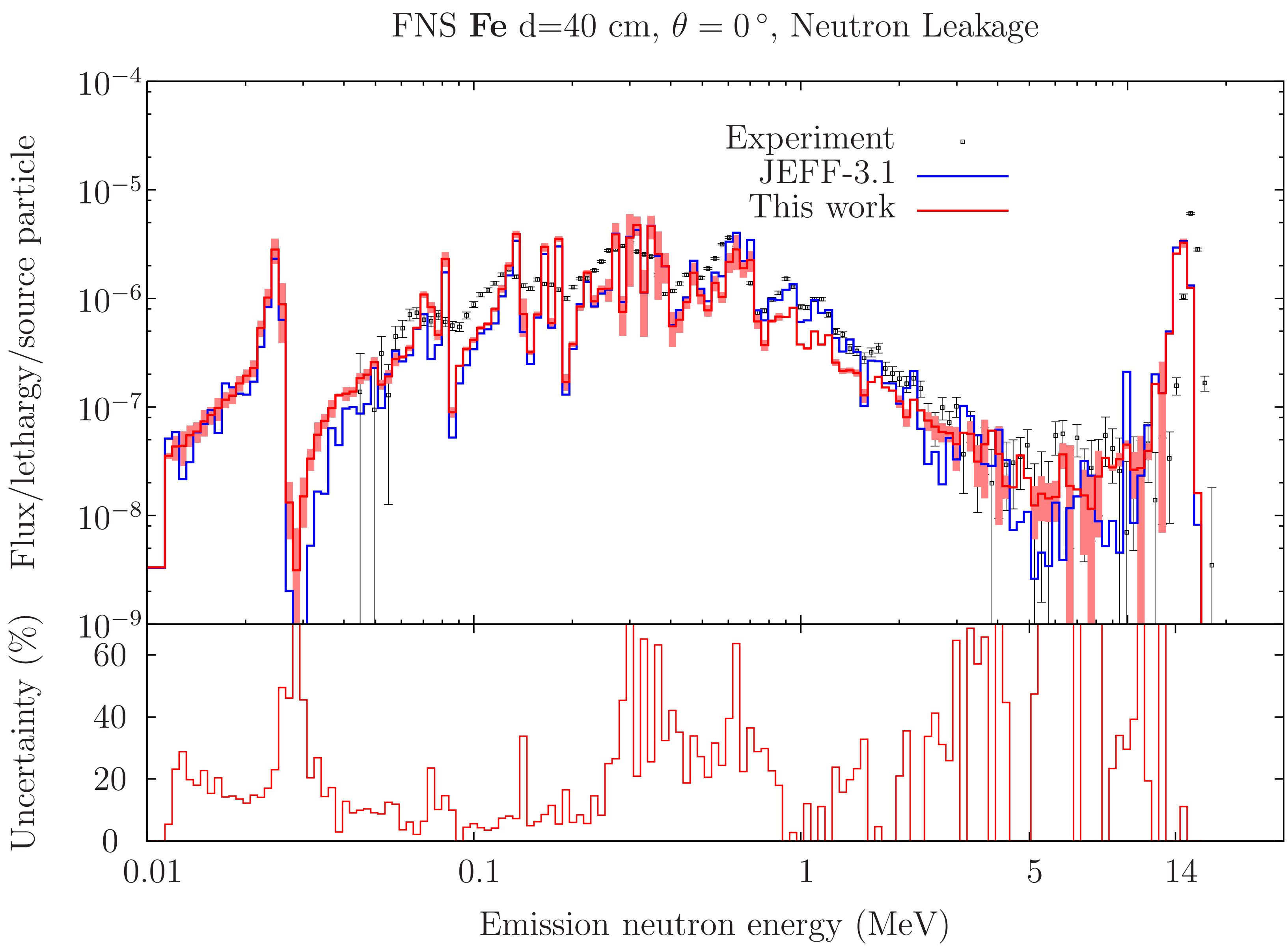
Application for Si Oktavian benchmark



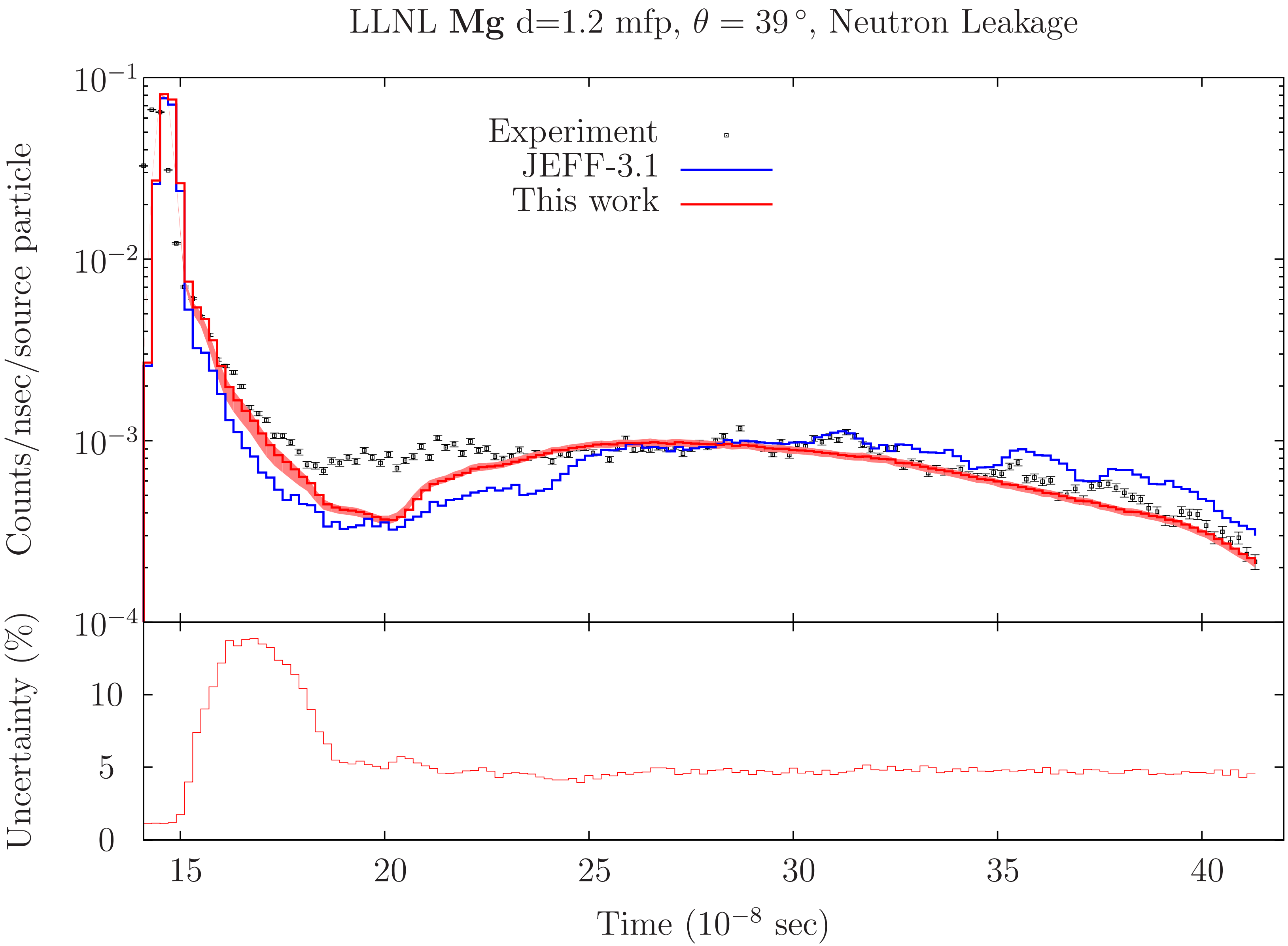
Application for W FNS benchmarks



Application for Fe FNS benchmarks



Application for Mg LLNL benchmarks



Results for the ESFR parameters

The sodium void reactivity (SVR) in units of dollars ($) can be obtained with the following equation :

$$\mathrm{SVR} = \frac{k_2 - k_1}{k_1 k_2} \frac{1}{\beta_{\mathrm{eff}}} \times 10^5, \tag{1}$$

where the number of delayed neutron β_{eff} (in units of pcm) and the k_{eff} values are obtained from the MCNP calculations.

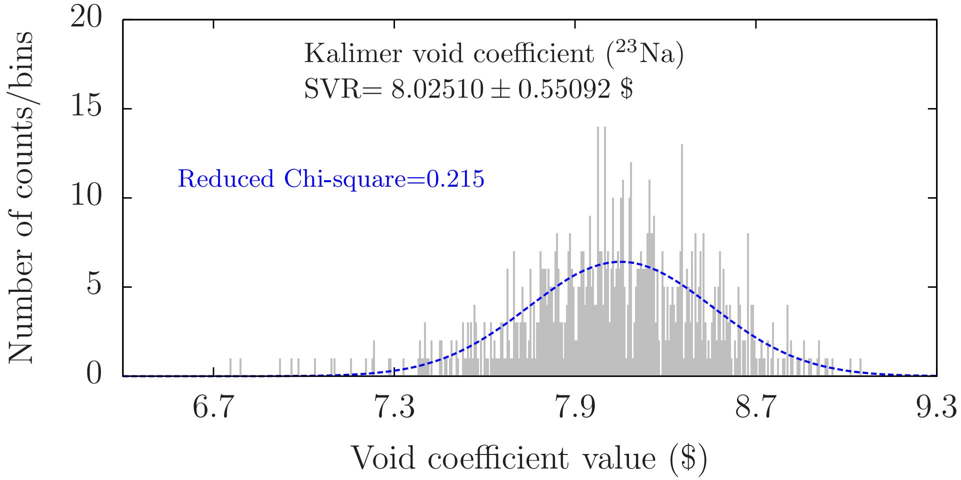
k_1 corresponds to the core flooded with Na coolant, and k_2 to the same core voided of Na coolant.

In both cases the Na coolant present in the axial and radial reflectors is supposed to remain unchanged.

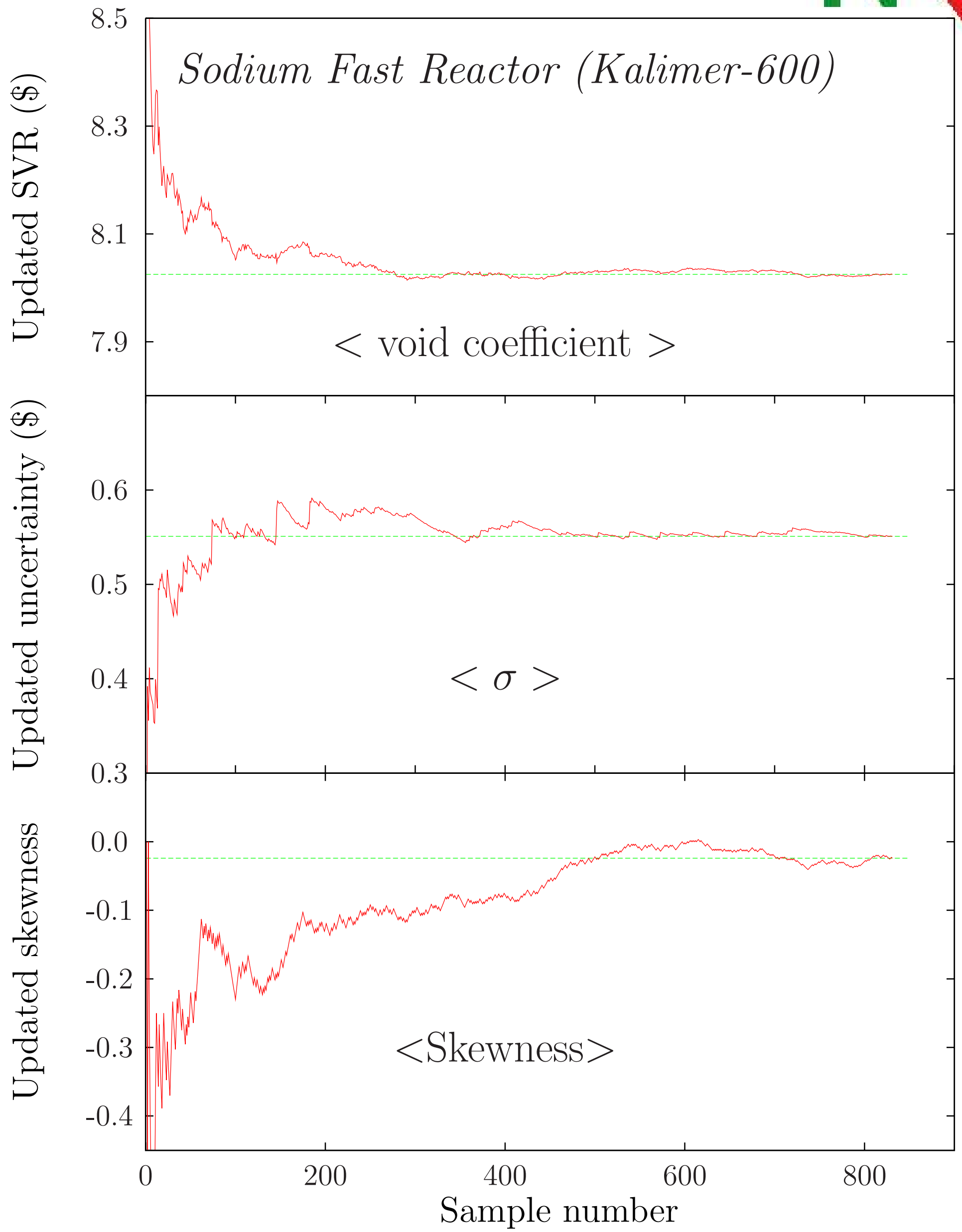
Main components: ^{23}Na, ^{56}Fe, Zr, ^{238}U, ^{239,240}Pu

Most sensitive reactions: ^{239}Pu(n,f) and ^{238}U(n,γ)

Application on void coefficient $\text{SVR} = \frac{k_2 - k_1}{k_1 k_2} \frac{1}{\beta_{eff}}$



Coefficient	Uncertainty due to nuclear data
SVR	$\simeq$ 6-7 %
Non Voided k_{eff}	$\simeq$ 100 pcm
Voided k_{eff}	$\simeq$ 15 pcm
β_{eff}	$<$ 1 pcm
Doppler	$\simeq$ 7-13 %

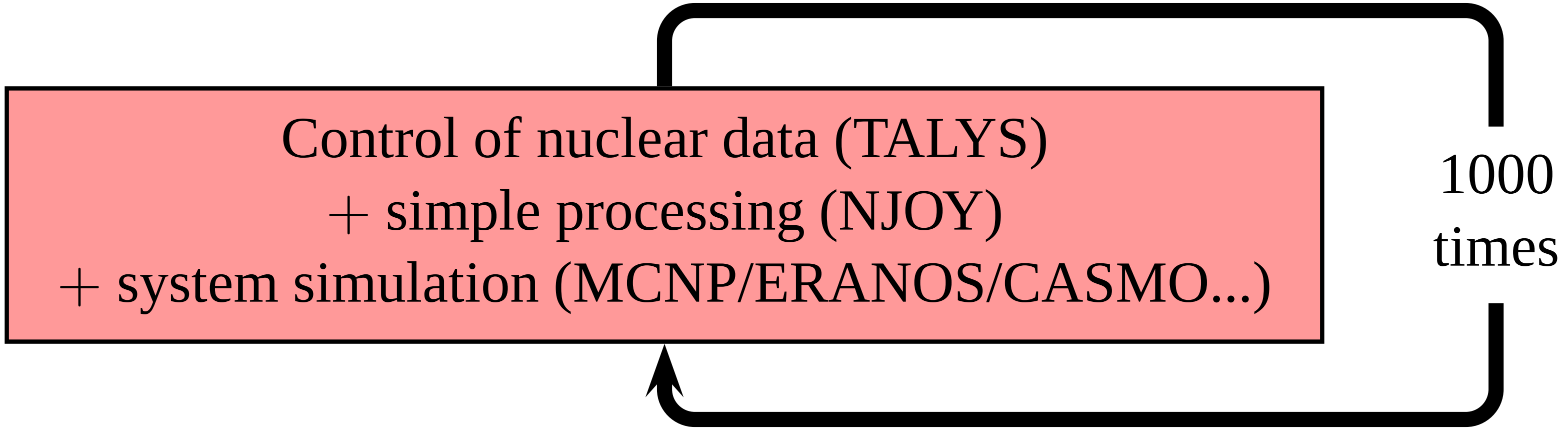


TMC versus Perturbation method

① Obtain uncertainties on a large-scale models due to nuclear data uncertainties

② Systematic approach, reliable and reproducable

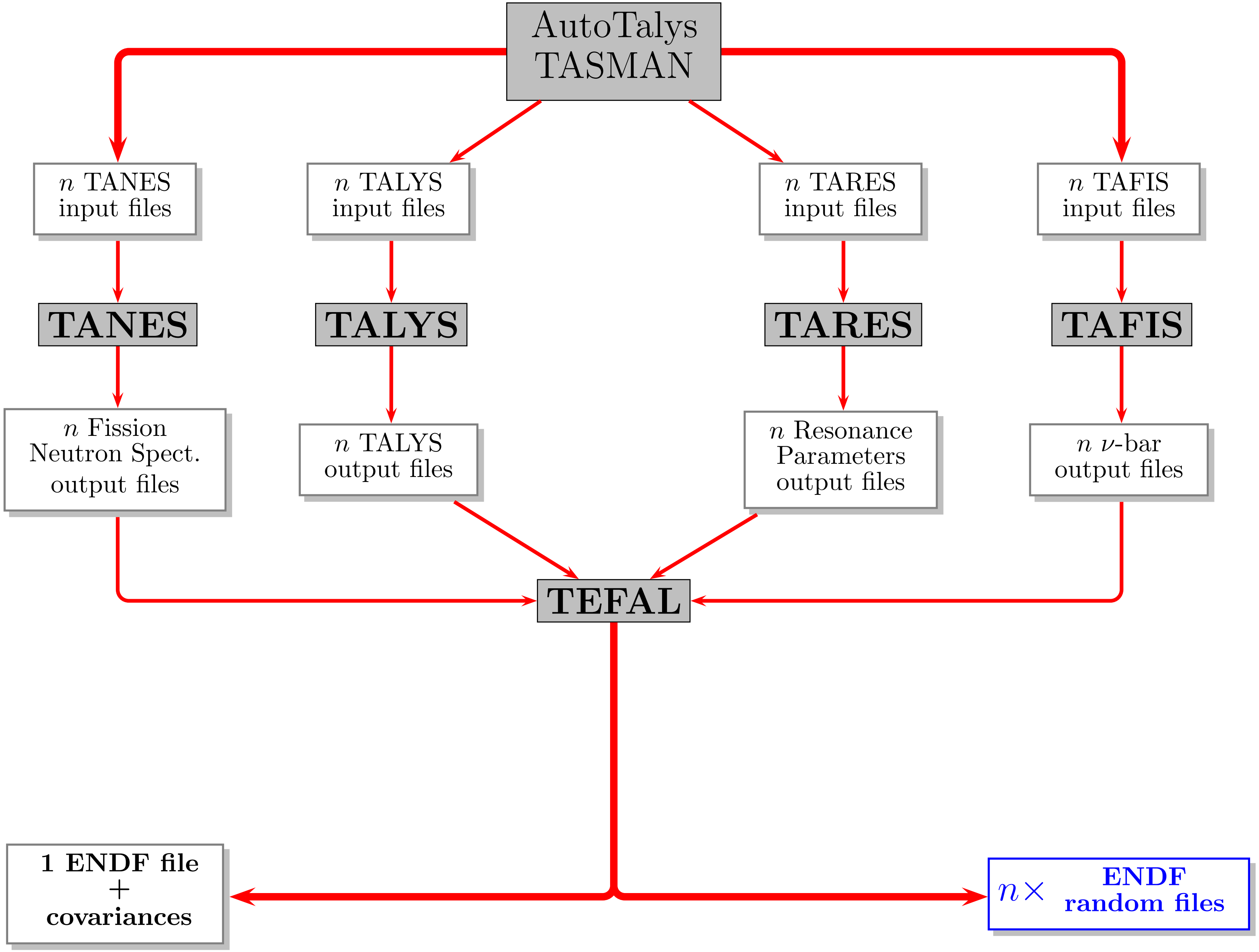
Solution (1): Total Monte Carlo



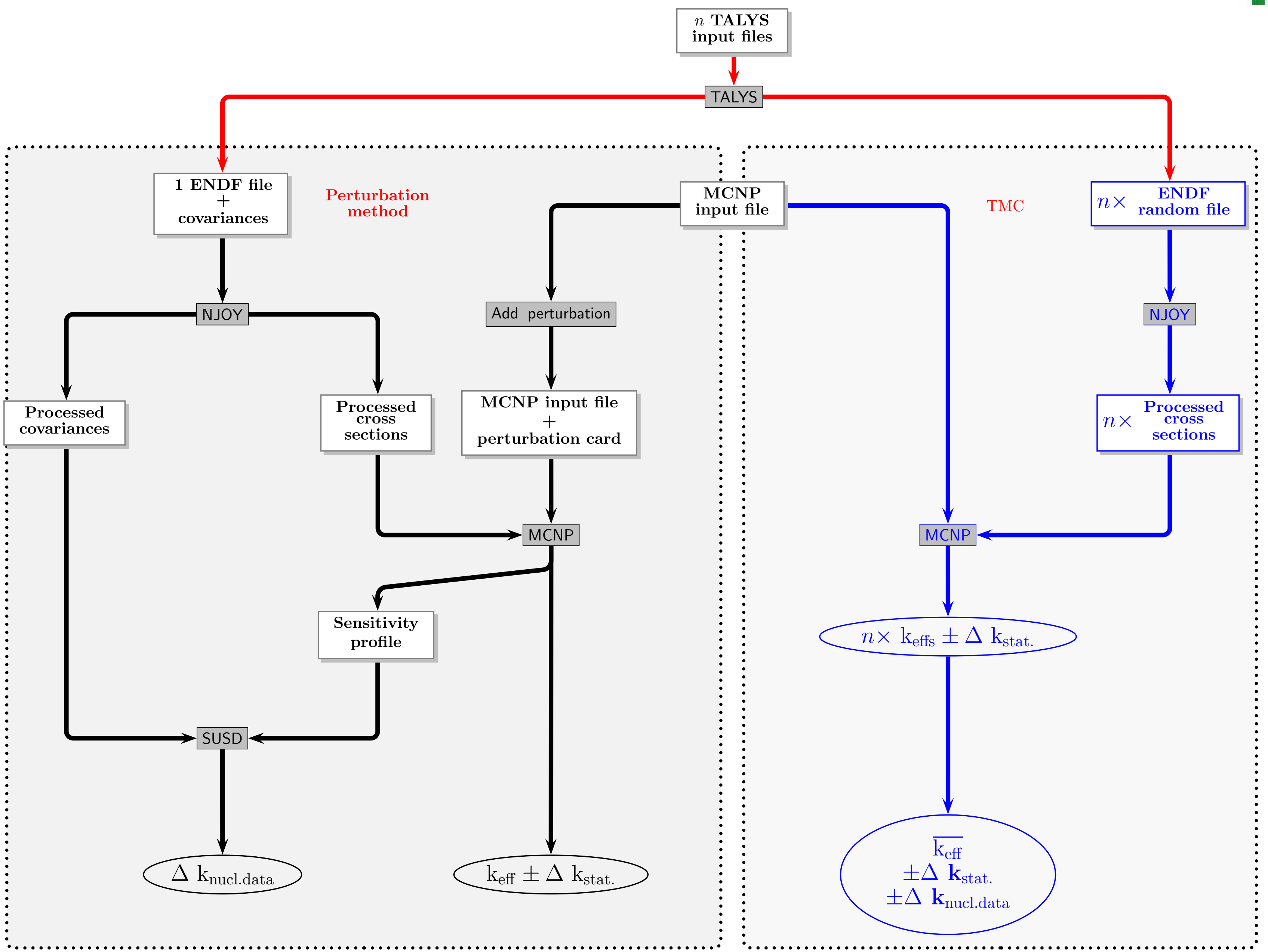
Solution (2): Perturbation method

$\Longrightarrow$MCNP+ Perturbation cards+covariance files

TMC and Perturbation method: File production



TMC and Perturbation method



Necessary software

Common to TMC and Perturbation methods:

☞ TALYS
☞ NJOY (ACE)
☞ MCNP

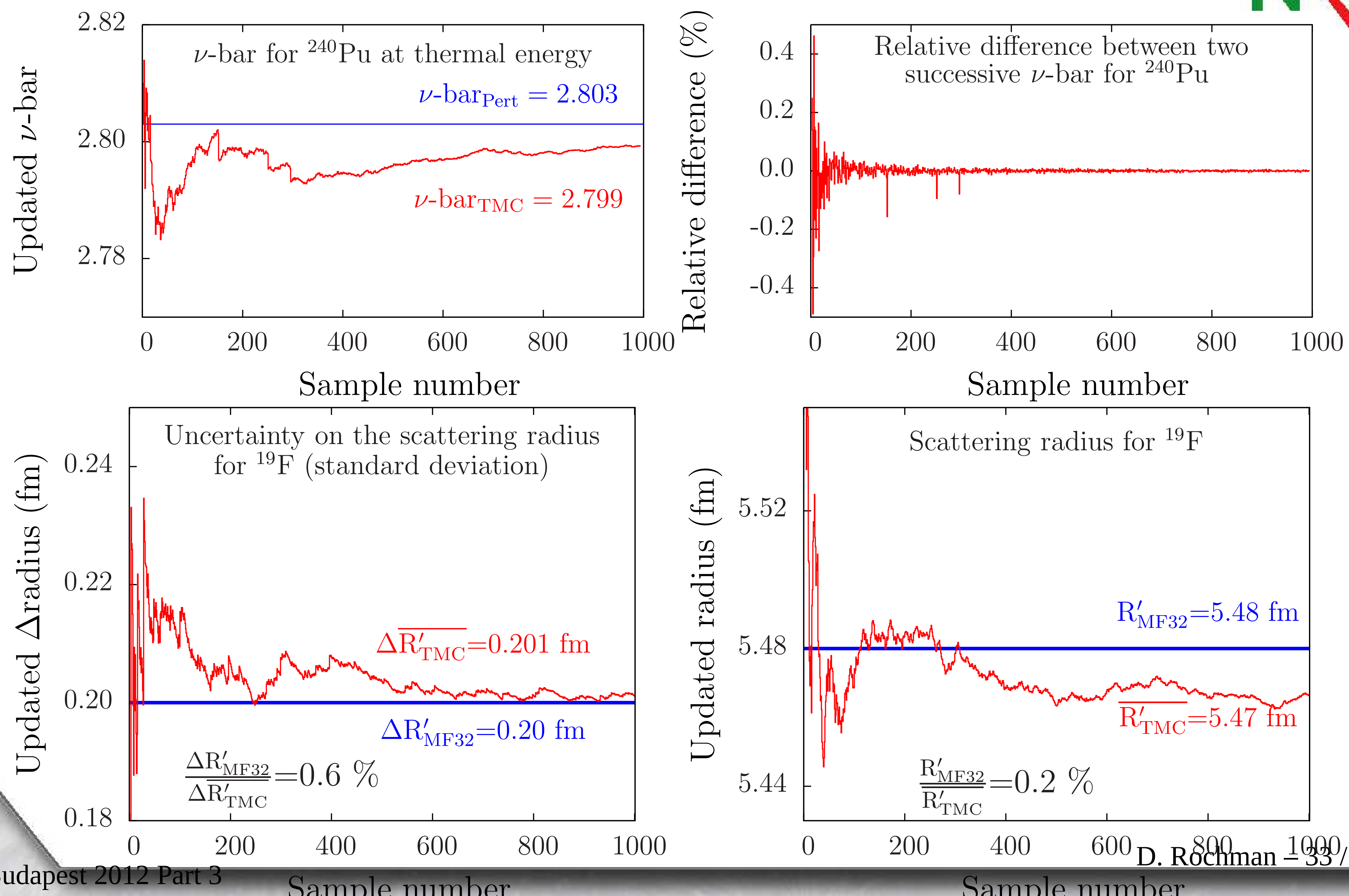
For the TMC method only:

∅

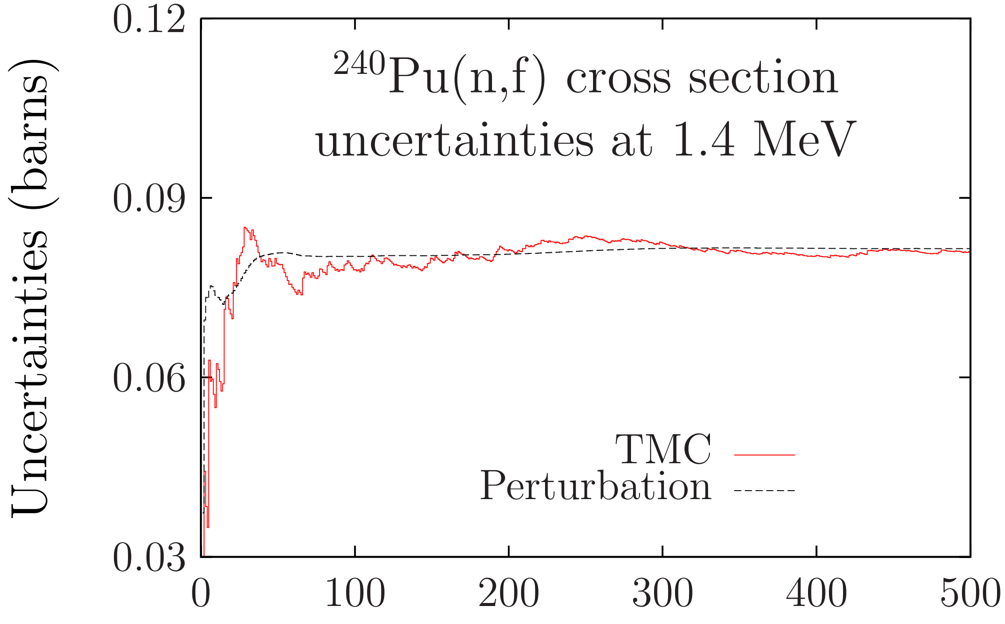
For the Perturbation method only:

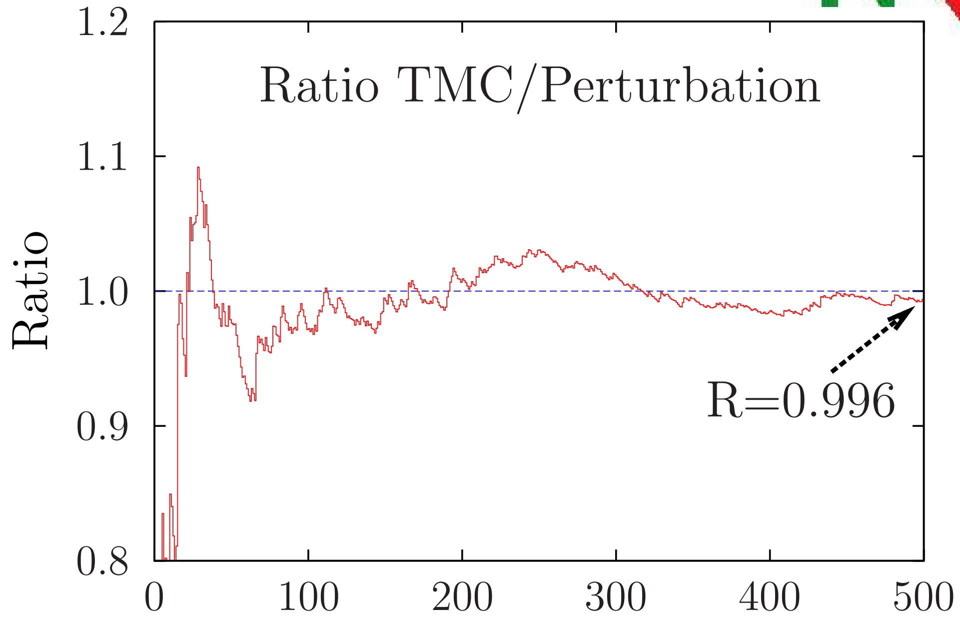
☞ NJOY (ERRORR)/PUFF
☞ Add perturbation
☞ SUSD

Convergence and consistency of ν-bar and resonance parameters



Convergence TMC/Perturbation





TMC versus Perturbation: Results

Comparison TMC-Perturbation methods for a few k_{eff} benchmarks. The ratio in the last column is ”TMC over Perturbation”.

		Total Monte Carlo	Perturbation	Ratio
Benchmark	Isotopes	Uncertainty due to nuclear data (pcm)	Uncertainty due to nuclear data (pcm)	
hst39-6	^{19}F	330	290	1.16
hmf7-34	^{19}F	350	290	1.21
ict3-132	^{90}Zr	190	150	1.29
hmf57-1	^{208}Pb	500	410	1.22
pmf2	^{239}Pu	840	720	1.16
pmf2	^{240}Pu	790	650	1.21

Results: Details of the TMC-Perturbation methods for ^{19}F k_{eff} benchmarks

	hst39-6 ^{19}F			hmf7-34 ^{19}F	
	Δk_{eff} (pcm)			Δk_{eff} (pcm)	
	TMC	Perturbation		TMC	Perturbation
Total	330	290		350	290
MF2	280	240		310	280
MF3	170	160		75	105
MF4	100	-		80	-
MF6	30	-		35	-

Results: Details of the TMC-Perturbation methods for ^{239,240}Pu k_{eff} benchmarks

	pmf2 ^{239}Pu			pmf2 ^{240}Pu	
	Δk_{eff} (pcm)			Δk_{eff} (pcm)	
	TMC	Perturbation		TMC	Perturbation
Total	840	720		790	650
MF1	400	-		370	-
(n,inl)	170	140		70	50
(n,el)	250	240		30	40
(n,γ)	100	100		30	30
(n,f)	720	660		730	640
MF4	20	-		20	-
MF5	50	-		30	-
MF6	50	-		30	-

TMC vs. perturbation: pro and cons

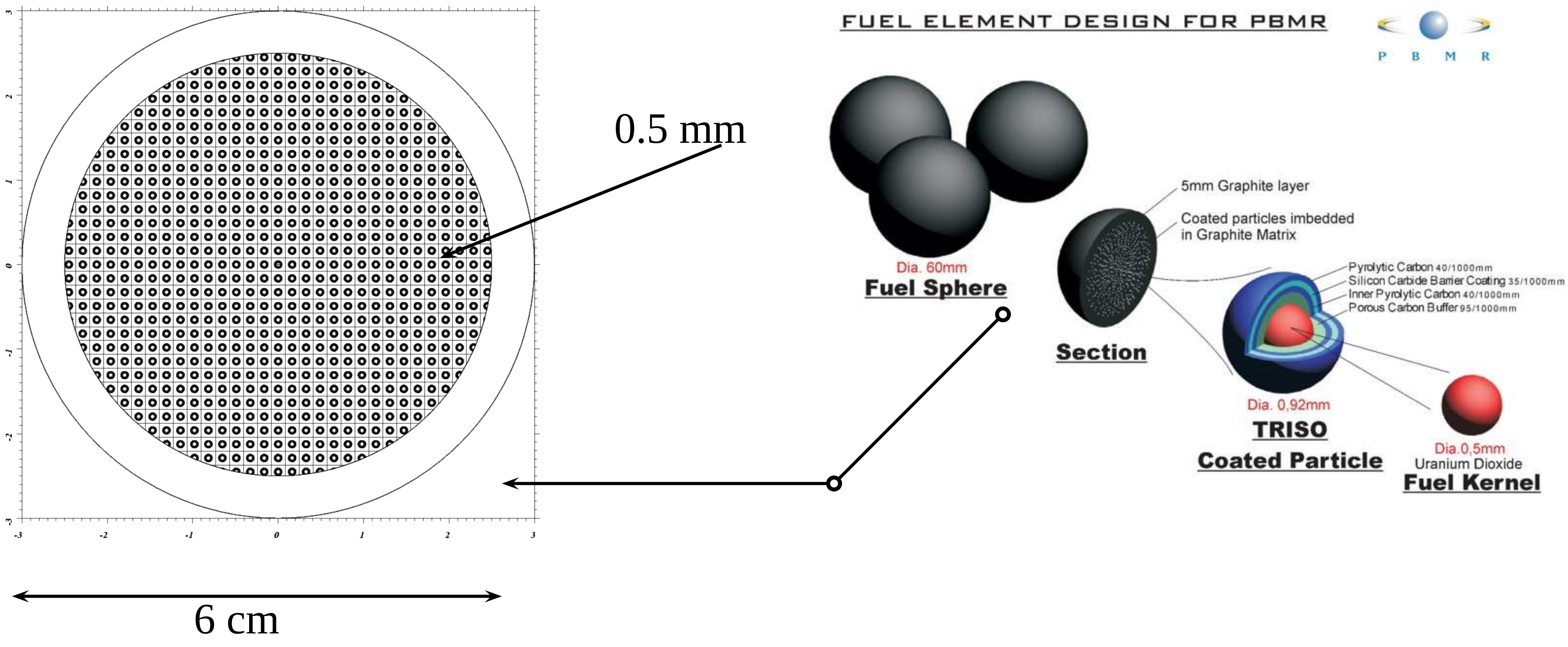
- ☺ First attempt to compare two uncertainty propagation method
- ☺ TMC: more general and exact answer, does not require special codes, more exhaustive
- ☹ but slower

- ☹ Perturbation: approximate, require special processing and codes, limited
- ☺ but faster
- ⌚ TMC uncertainties 15 to 30 % larger than from perturbation

Perturbation approach still dominates the *market*,
but for how long ?

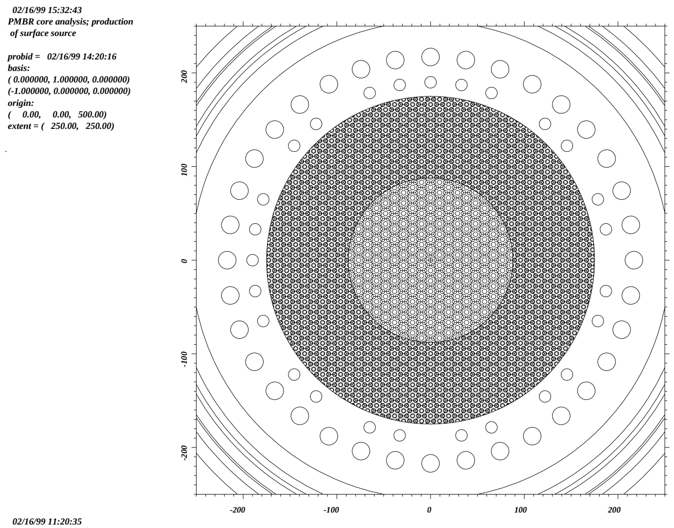
Application to a Pebble Bed Modular Reactor (PBMR)

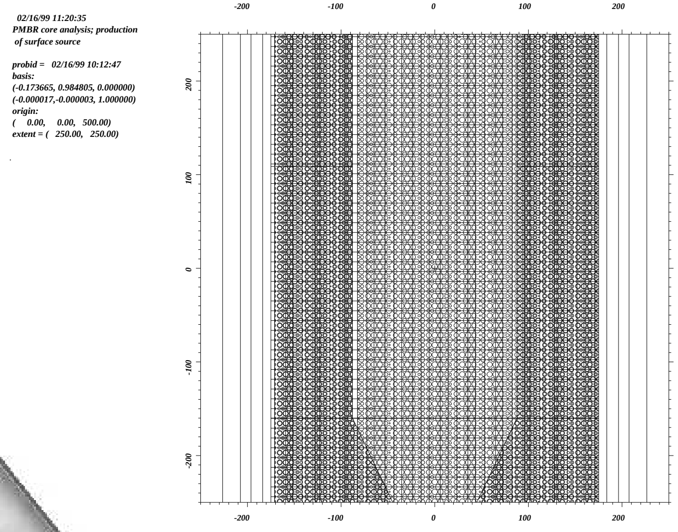
- Model of a fuel pebble
- Fuel particles, surrounded by coating layers, explicitly modeled
- Regular rectangular lattice of fuel particles

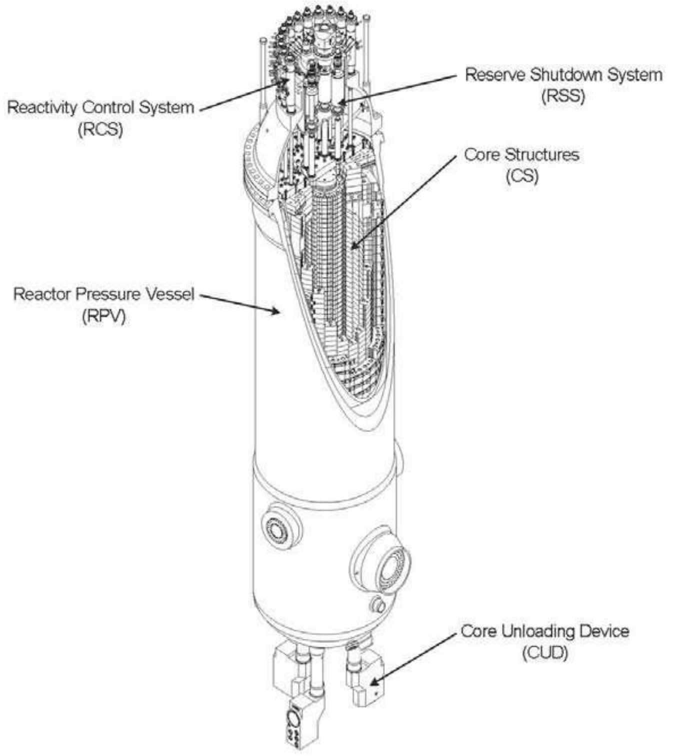


Application to a Pebble Bed Modular Reactor (PBMR)

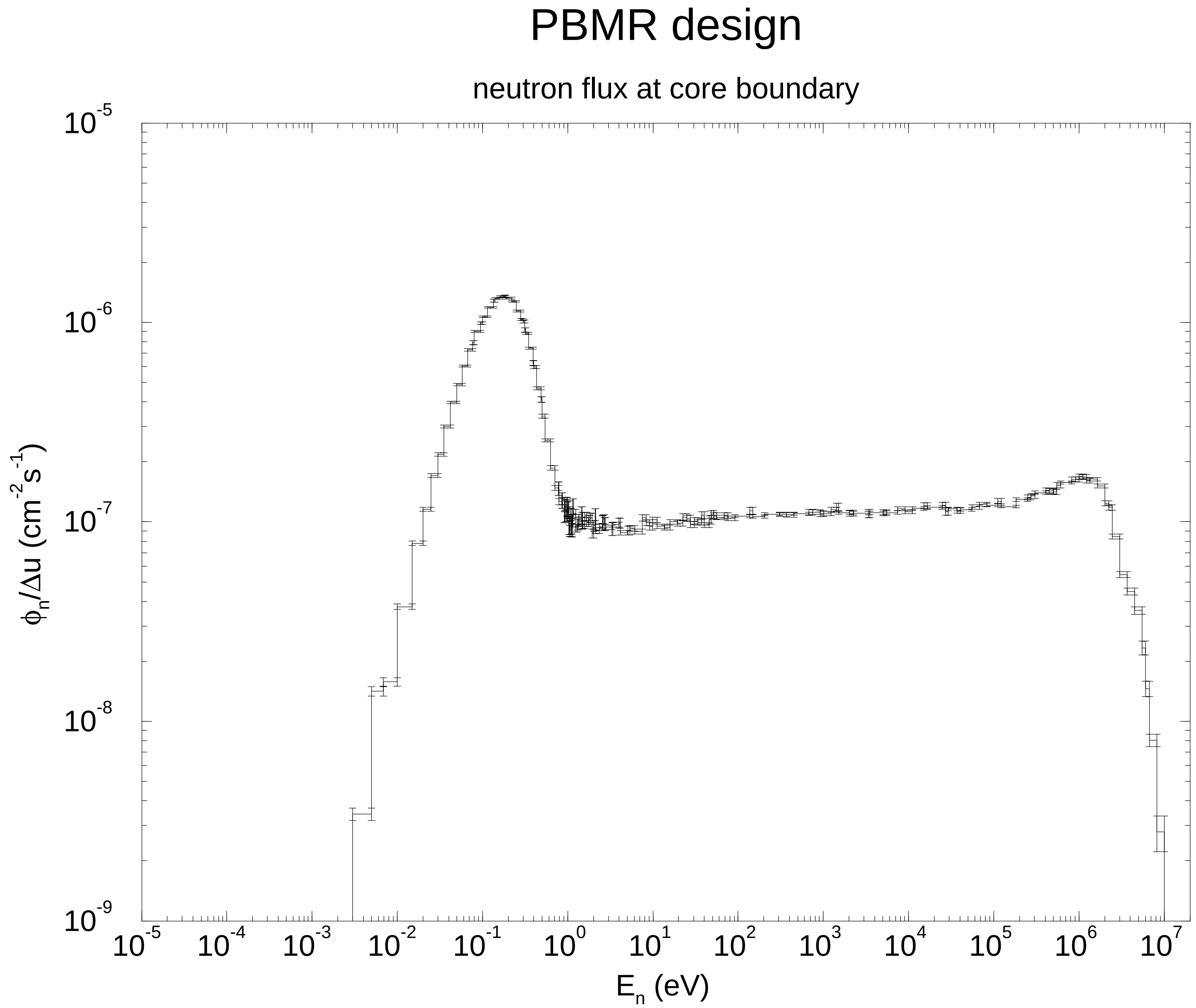
- Hexagonal close packed lattice
- Moderator region consists of homogeneous moderator pebbles as the fuel, reflectors and shields as defined by ESKOM







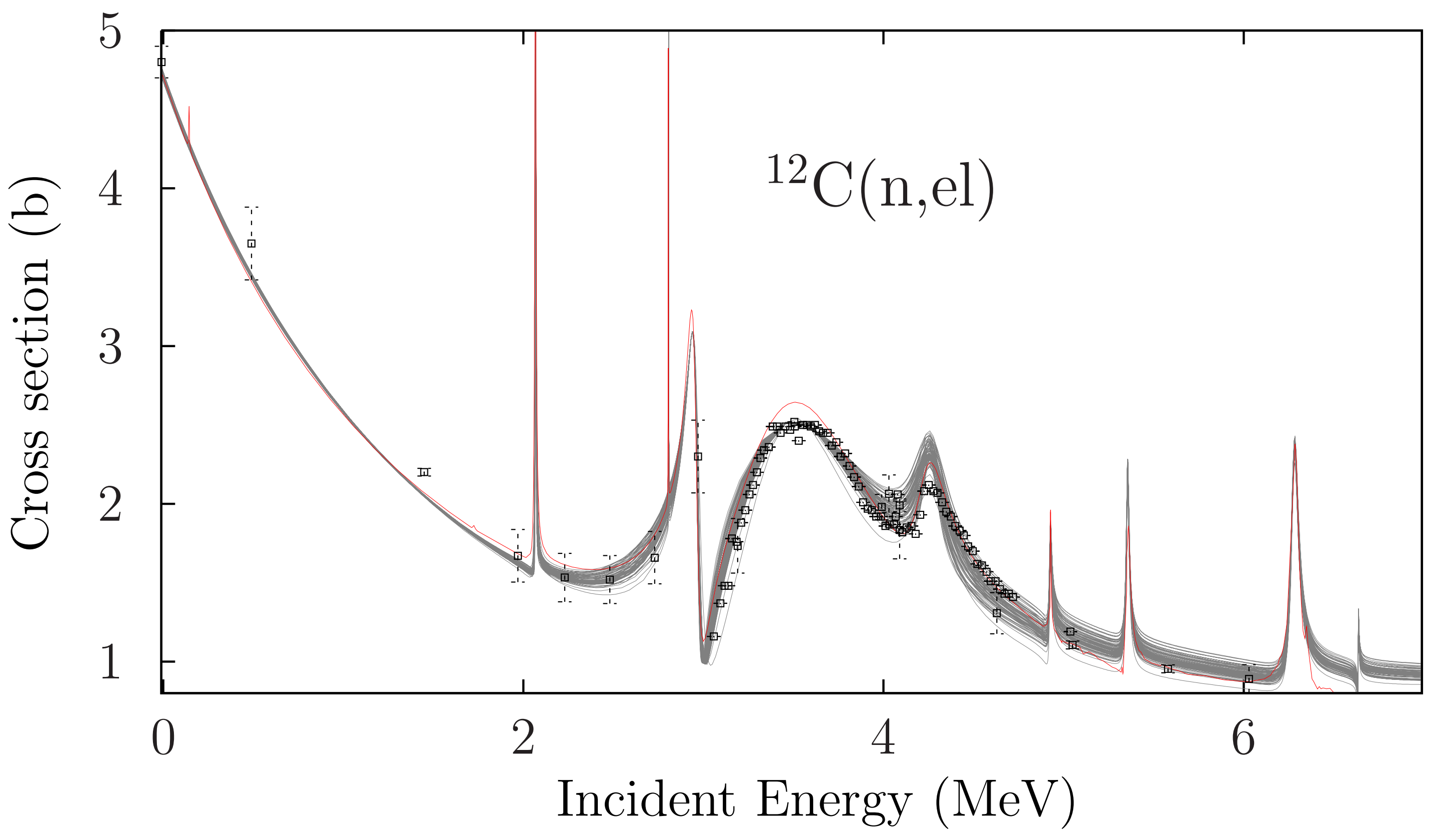
PBMR: Neutron Flux spectrum



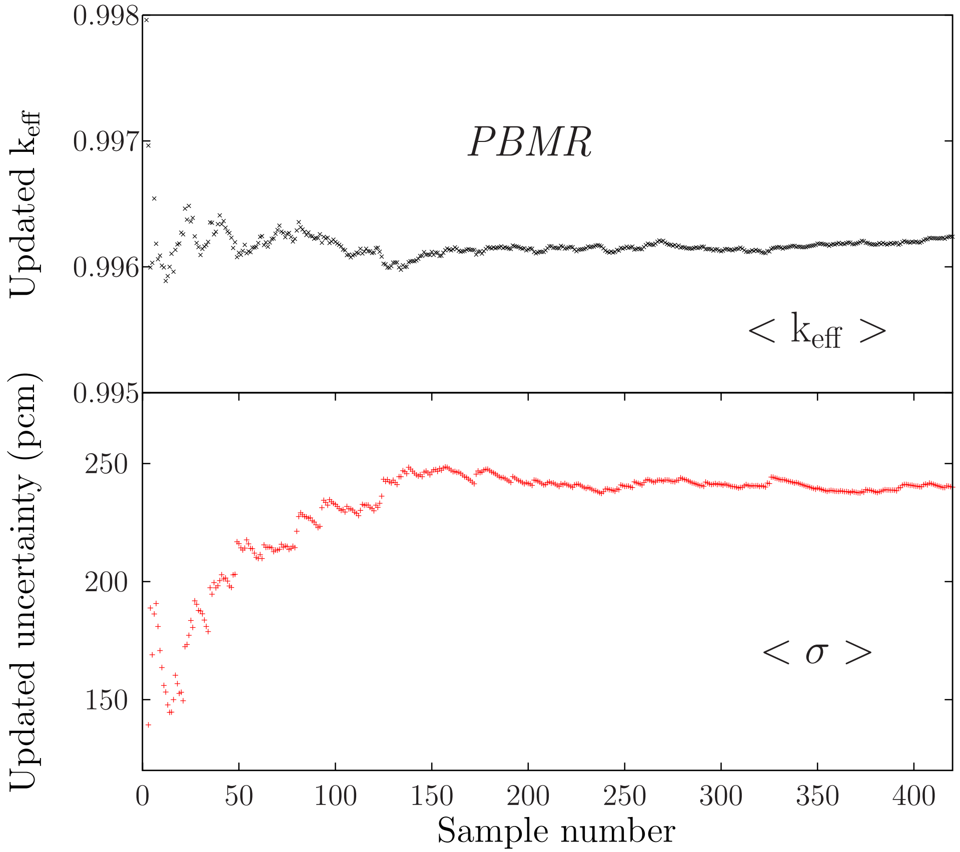
Neutron flux spectrum at the radial core boundary (Almost no neutron above few MeV)

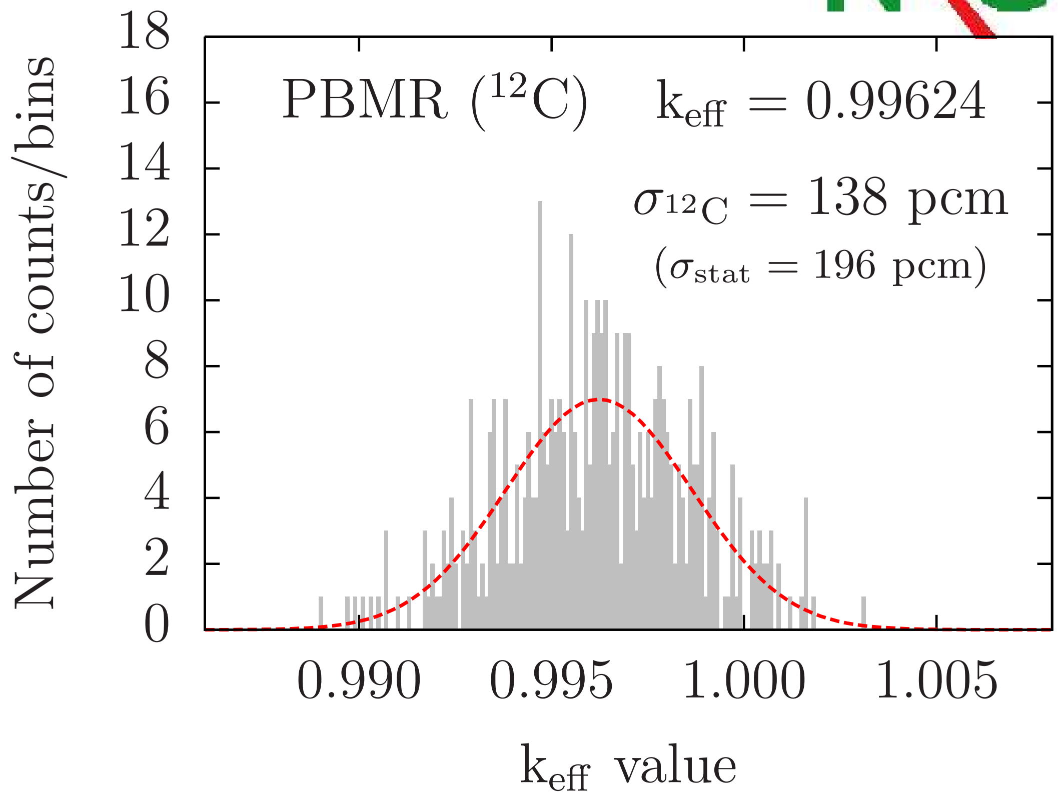
PBMR: ^{12}C nuclear data

- For neutron energy lower than few MeV: only elastic and capture cross sections
- JEFF-3.1: σ_{th} (n,el)$= 4.746 \pm 0.002$b and σ_{th} (n,γ)$= 3.53 \pm 0.07$mb
- All (n,el), (n,γ), angular distribution and emission spectra randomly varied



PBMR: Results





- Convergence achieved after $\simeq$ 350 runs (10 days of 15 CPU)
- More runs would be suitable
- Effect of other isotopes (^{13}C, Si, fission products and of course actinides)

Thermal scattering data (H in H_2O)

TMC is the only method to propagate uncertainties due to thermal scattering data (no covariances exist)

In the case of H in H_2O, the incoherent inelastic scattering is the major component and the coherent and incoherent elastic scattering can be neglected. The inelastic scattering is described by the scattering law S(α,β) at different temperatures.

$$\frac{\partial^2\sigma(E \to E', \mu)}{\partial E' \partial \mu} = \frac{\sigma_b}{2kT}\sqrt{\frac{E'}{E}}S(\alpha,\beta) \quad (2)$$

with E and E' the incident and outgoing neutron energies in the laboratory system, μ is the cosine of the scattering angle in the laboratory system, σ_b is the characteristic bound scattering cross section for the material (water in this case) and kT is the thermal energy in eV. S(α,β) is the asymmetric form of the scattering law, which depends on two variables: the momentum transfer α and the energy transfer β:

$$\alpha = \frac{E + E' - 2\sqrt{EE'\mu}}{AkT} \quad (3)$$

$$\beta = \frac{E' - E}{kT} \quad (4)$$

Thermal scattering data (H in H_2O)

1. create input parameters for the LEAPR module of NJOY,
2. run LEAPR to generate thermal scattering data in ENDF format "MF 7, MT 4" (incoherent inelastic data in terms of $S(\alpha,\beta)$ tables for different temperatures),
3. use the ENDF file with the THERMR module of NJOY to generate pointwise thermal scattering cross sections,
4. use the ENDF file and the output of THERMR with the ACER module of NJOY to generate thermal scattering data for the MCNP code in the ACE format,
5. and finally repeat *n* times the previous steps with random input parameters for LEAPR.

The central (or nominal) values for all model parameters to be used in LEAPR are the values used for the JEFF-3.1.1 evaluation.

Thermal scattering data (H in H_2O)

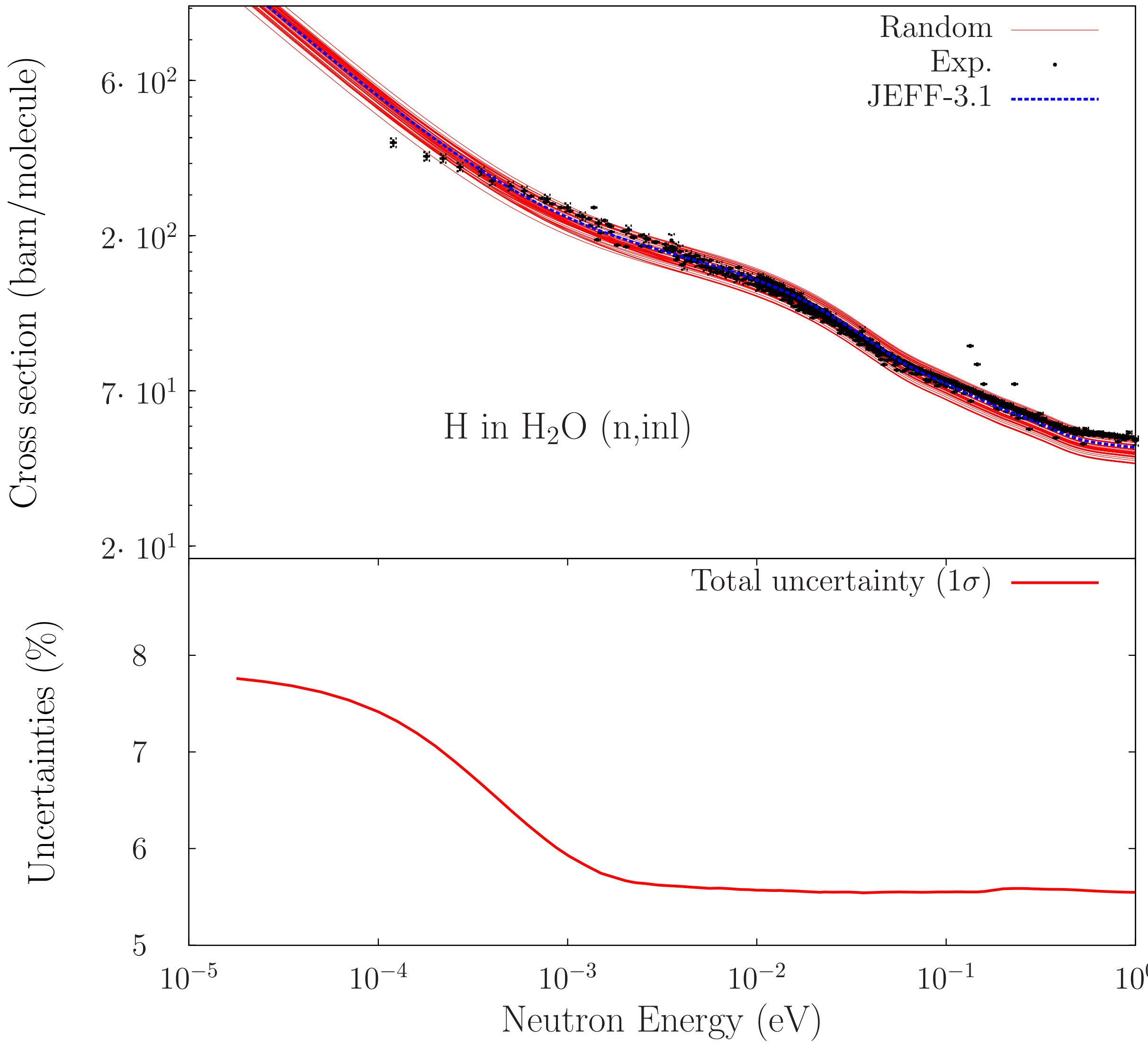


Figure 1: Top: incoherent random inelastic scattering cross section of H in H_2O compared to experimental data and the inelastic cross section from the JEFF-3.1.1 library. Bottom: uncertainties on the inelastic cross section calculated from 1330 random in-

Thermal scattering data (H in H_2O)

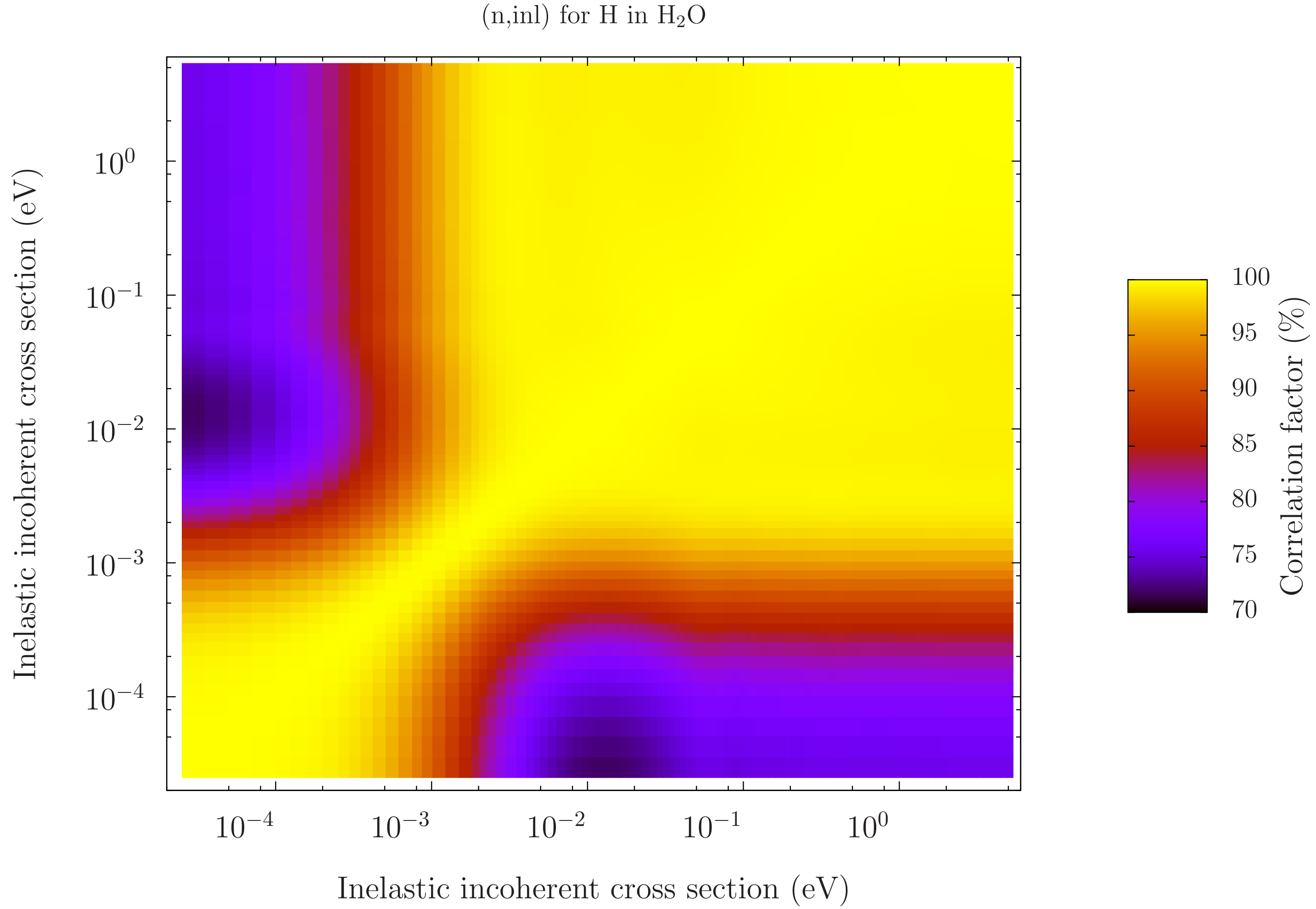


Figure 2: Energy-energy correlation matrix for the incoherent inelastic scattering of H in H_2O. Note that the correlation values are always larger than 0.7.

Thermal scattering data (H in H_2O)

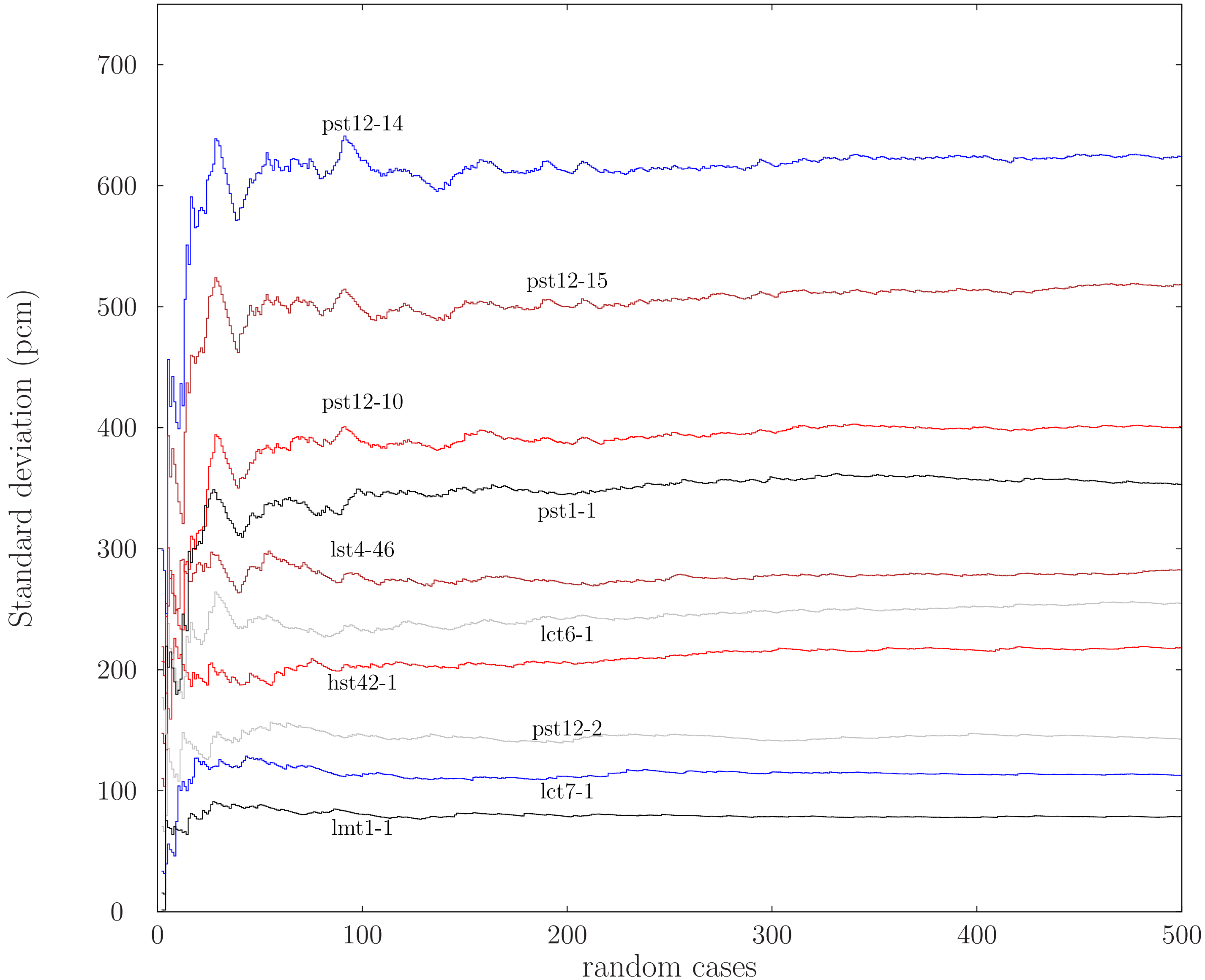
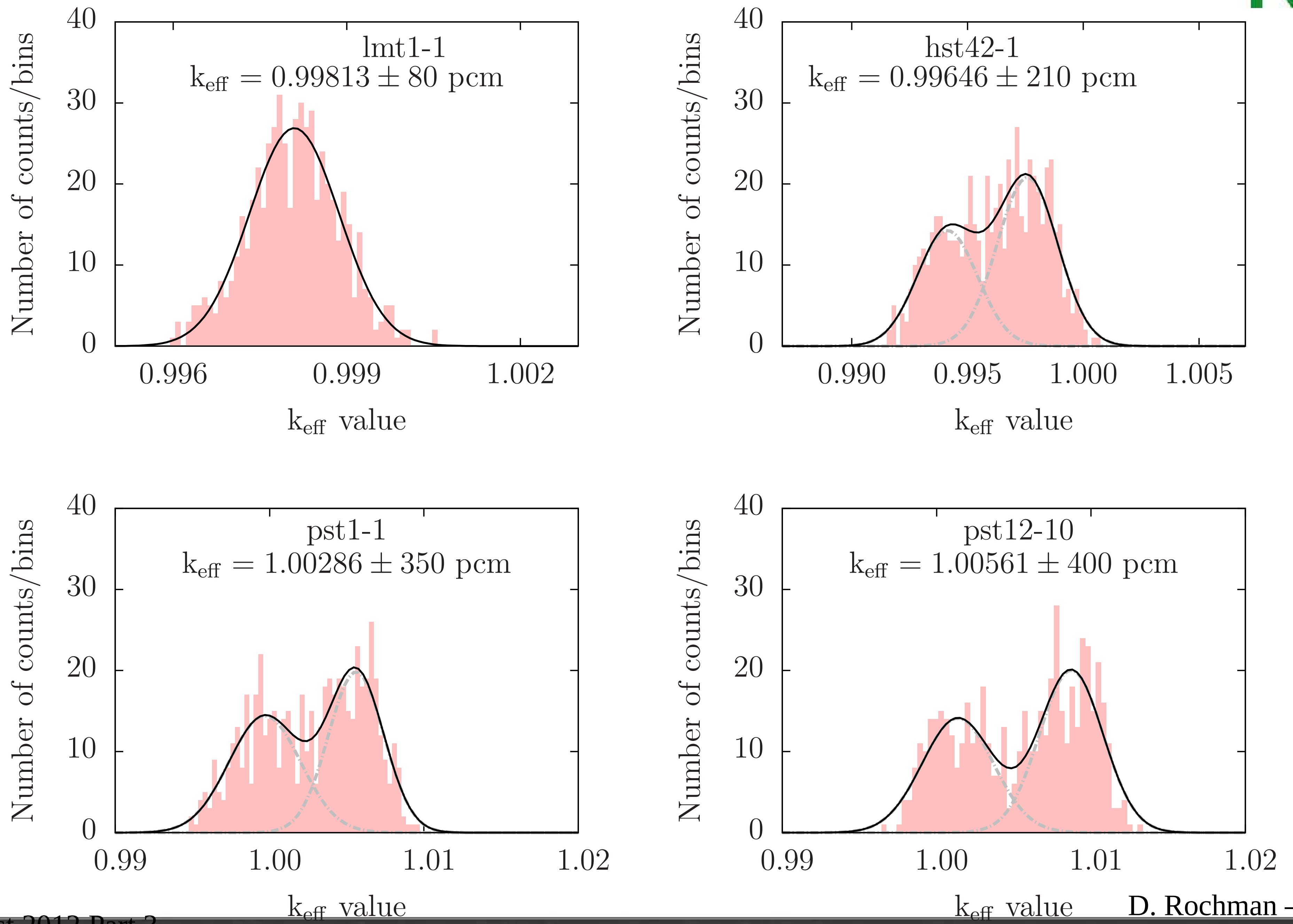


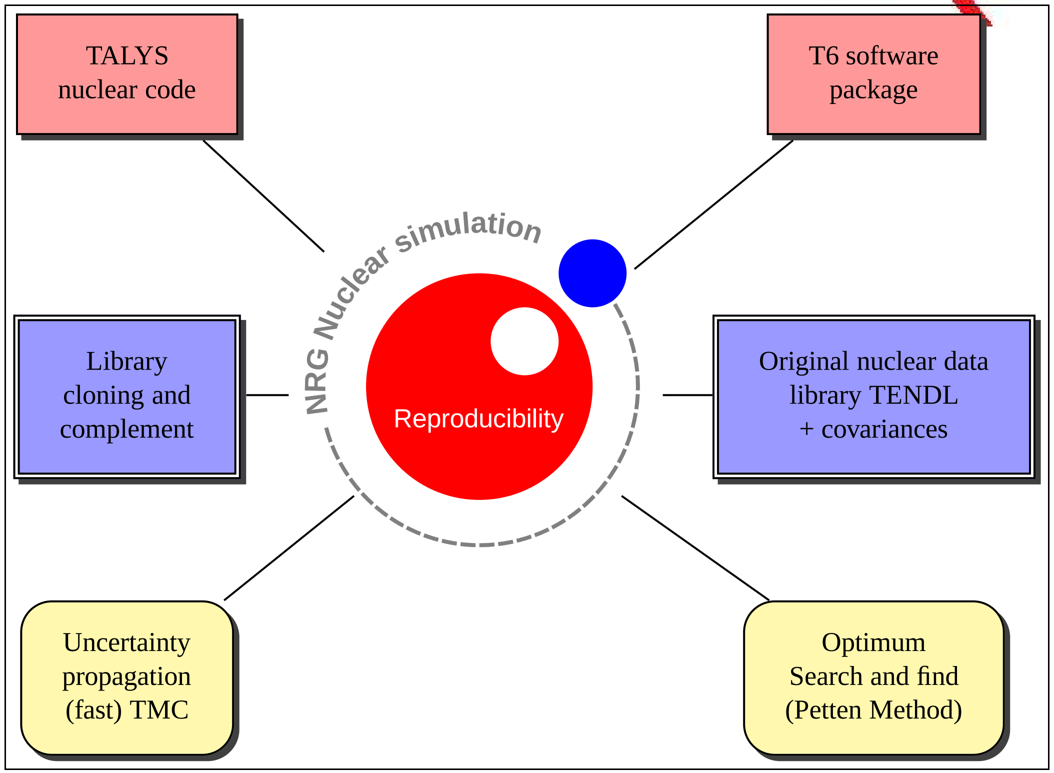
Figure 3: Standard deviations for some benchmarks as a function of the number of random thermal scattering files.

Thermal scattering data (H in H_2O)



Burn-up calculation: 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

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

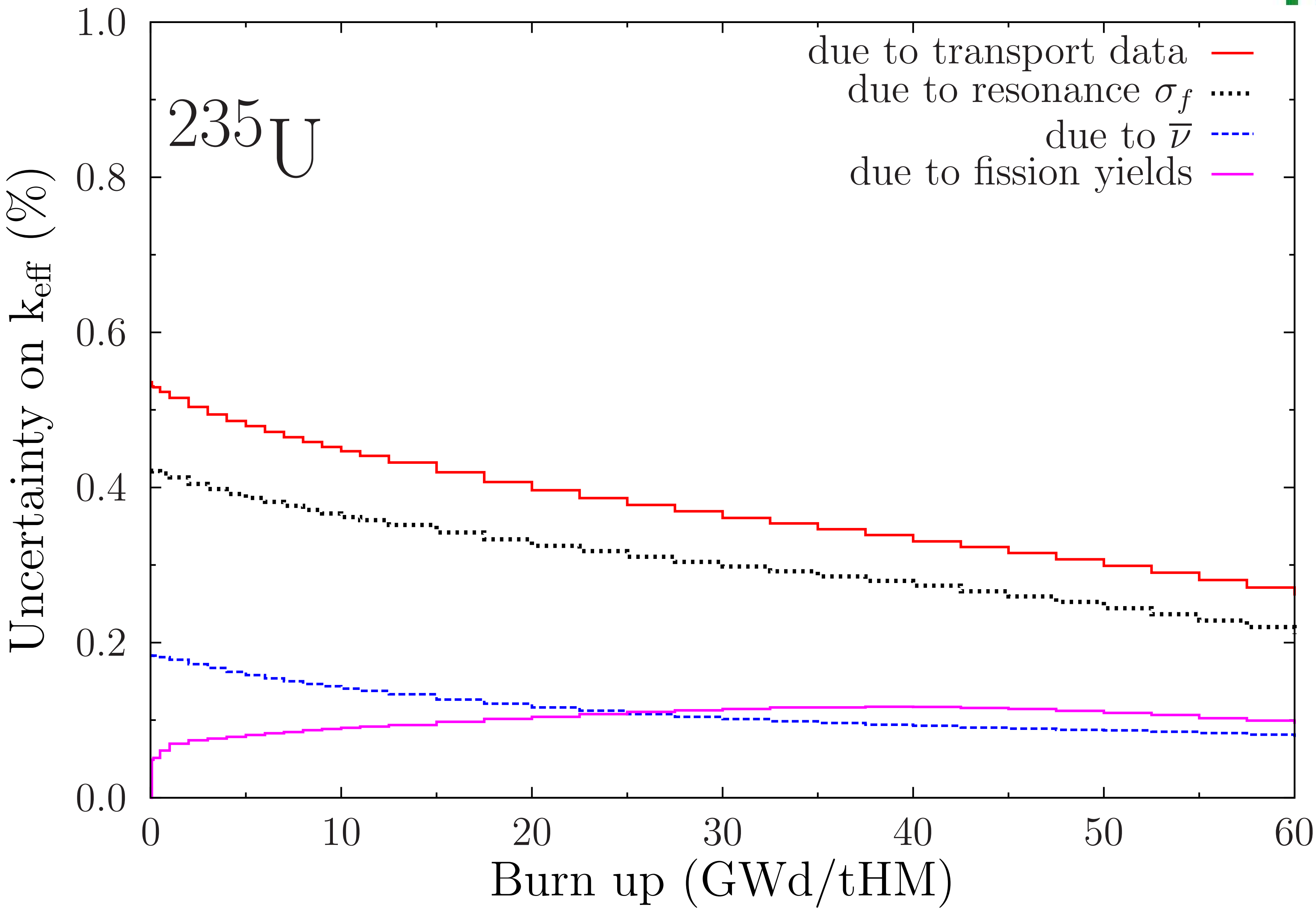
Considered data in TMC

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

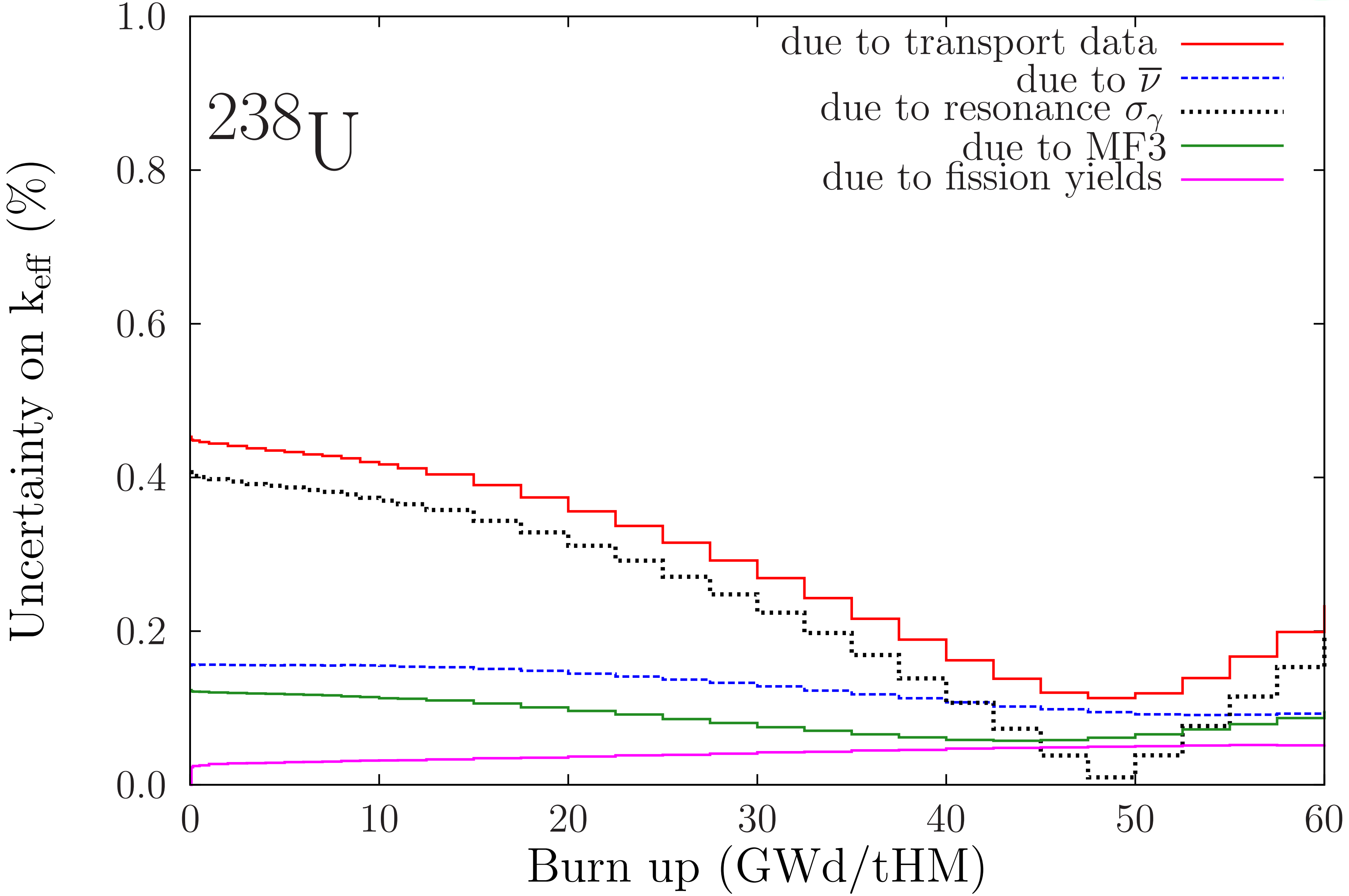
Nuclear data	ENDF-6 name	^{235}U	^{238}U	^{237}Np	^{239}Pu	^{241}Pu	Lumped fiss. prod.
complete ENDF file	MF1-6,10,12,14	X	X		X		X
fission yields	MF8	X	X	X	X	X	
ν-bar	MF1	X	X		X		
Resonance range	MF2	X	X		X		
Fast range	MF3	X	X		X		
Angular distr.	MF4	X	X		X		
Fission neut. spec.	MF5	X	X		X		
(n,γ)	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

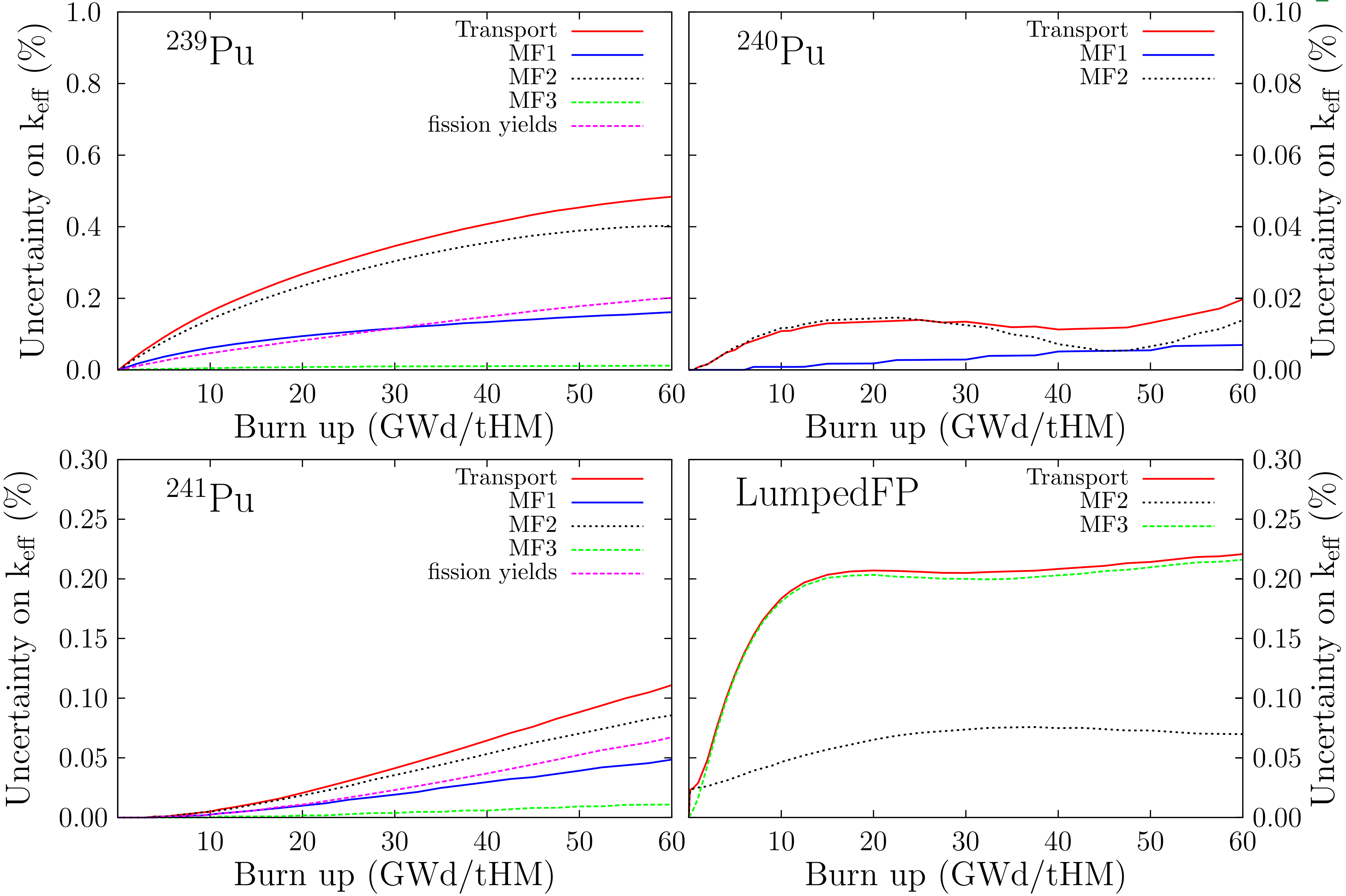
Results on k_∞



Results on k_∞



Results on k_∞



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

	0	20	30	40	50	60	0	300
^{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

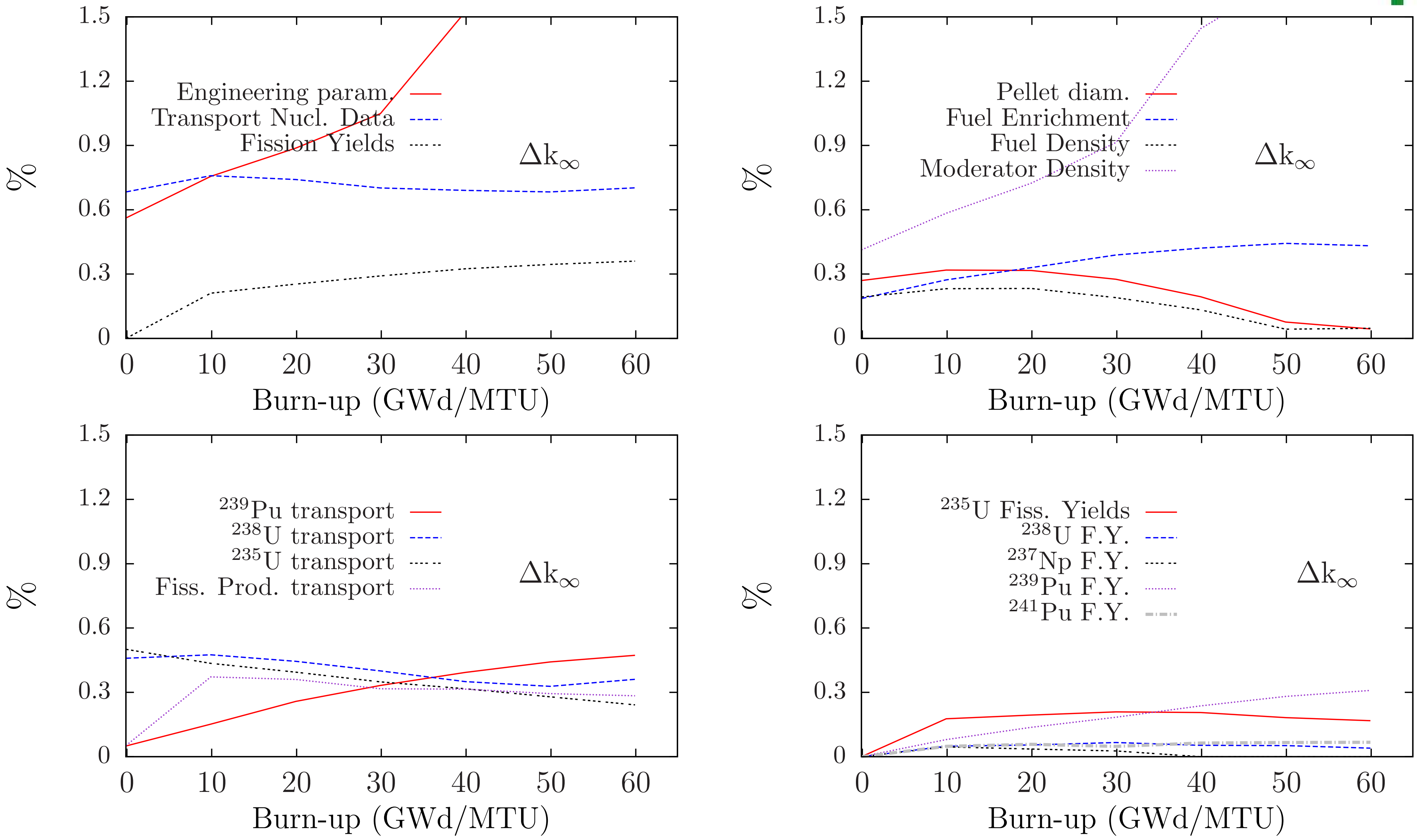
Results on number densities for fission products

	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.

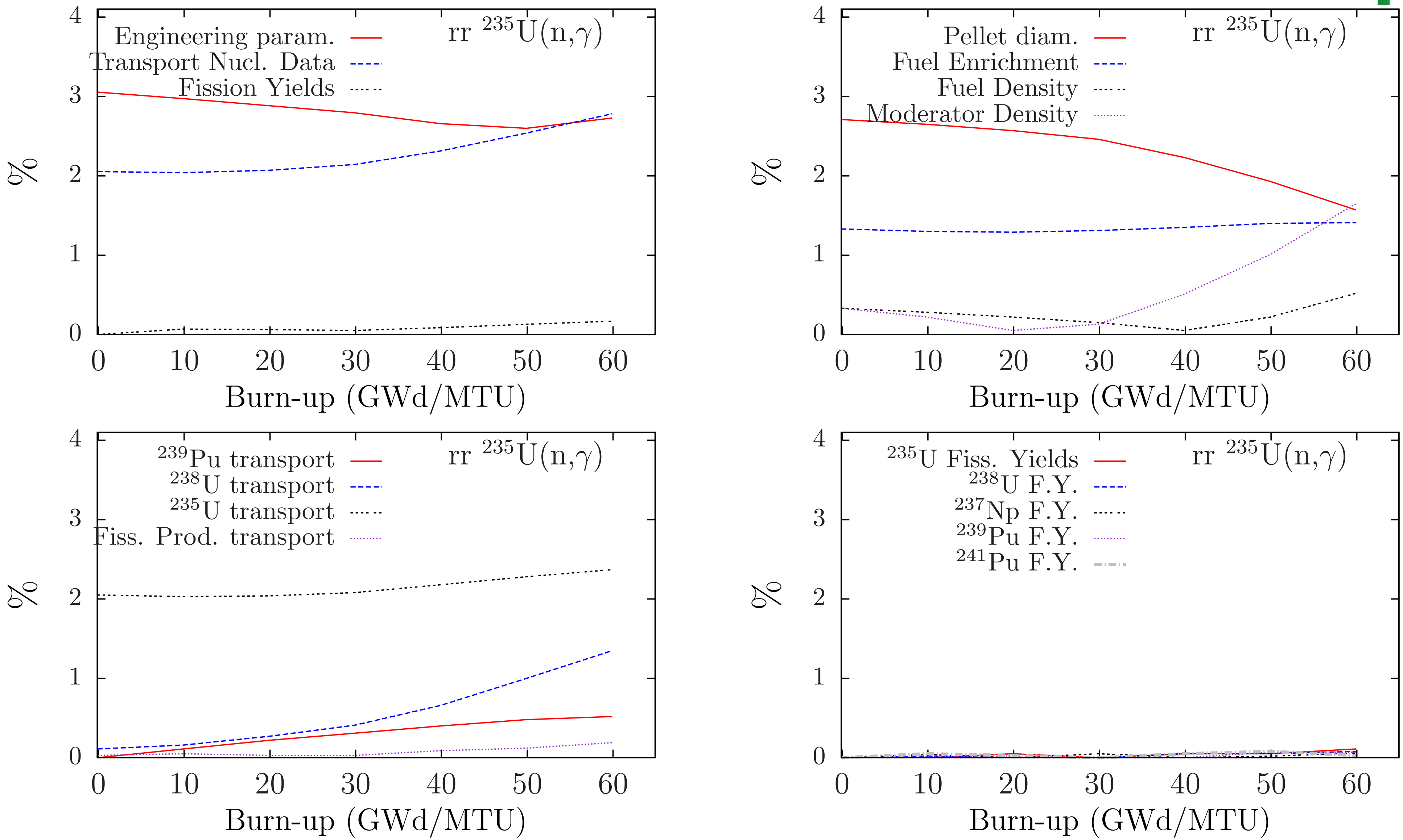
Total uncertainties (due to transport data and fission yields, in %) for

	0	10	30	40	50	60	0	300
^{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

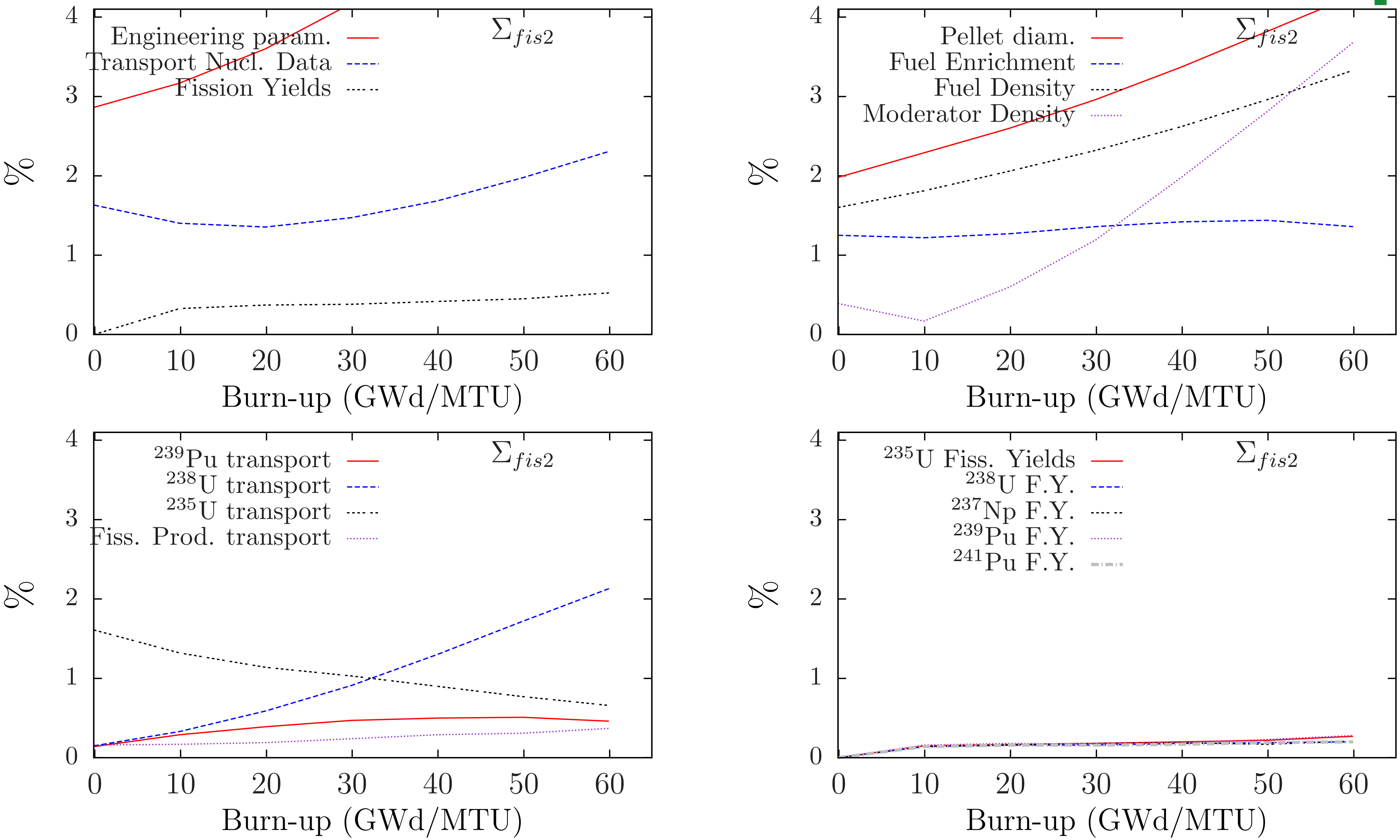
Example for k_∞



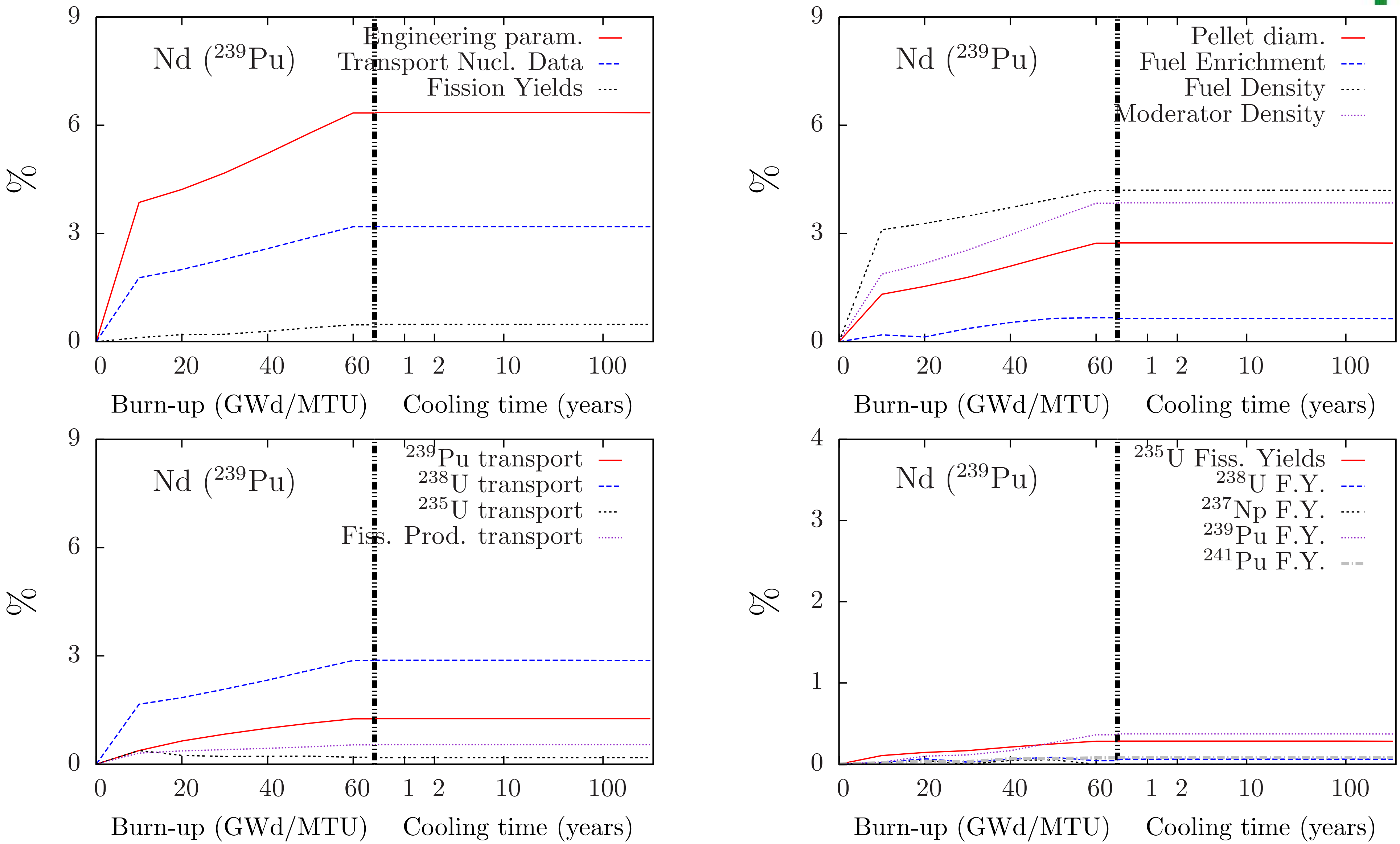
Example for reaction rates



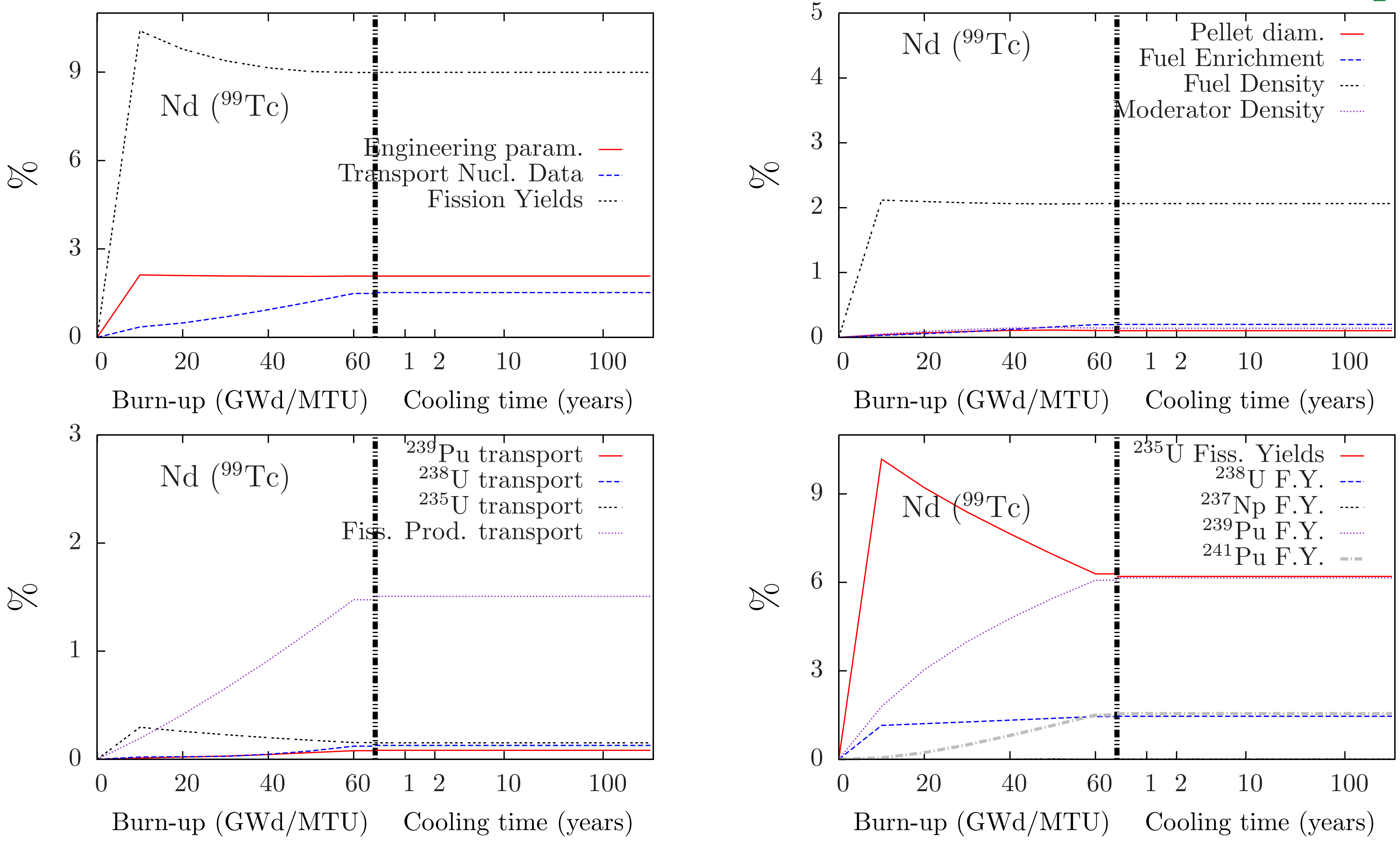
Example for macroscopic cross sections



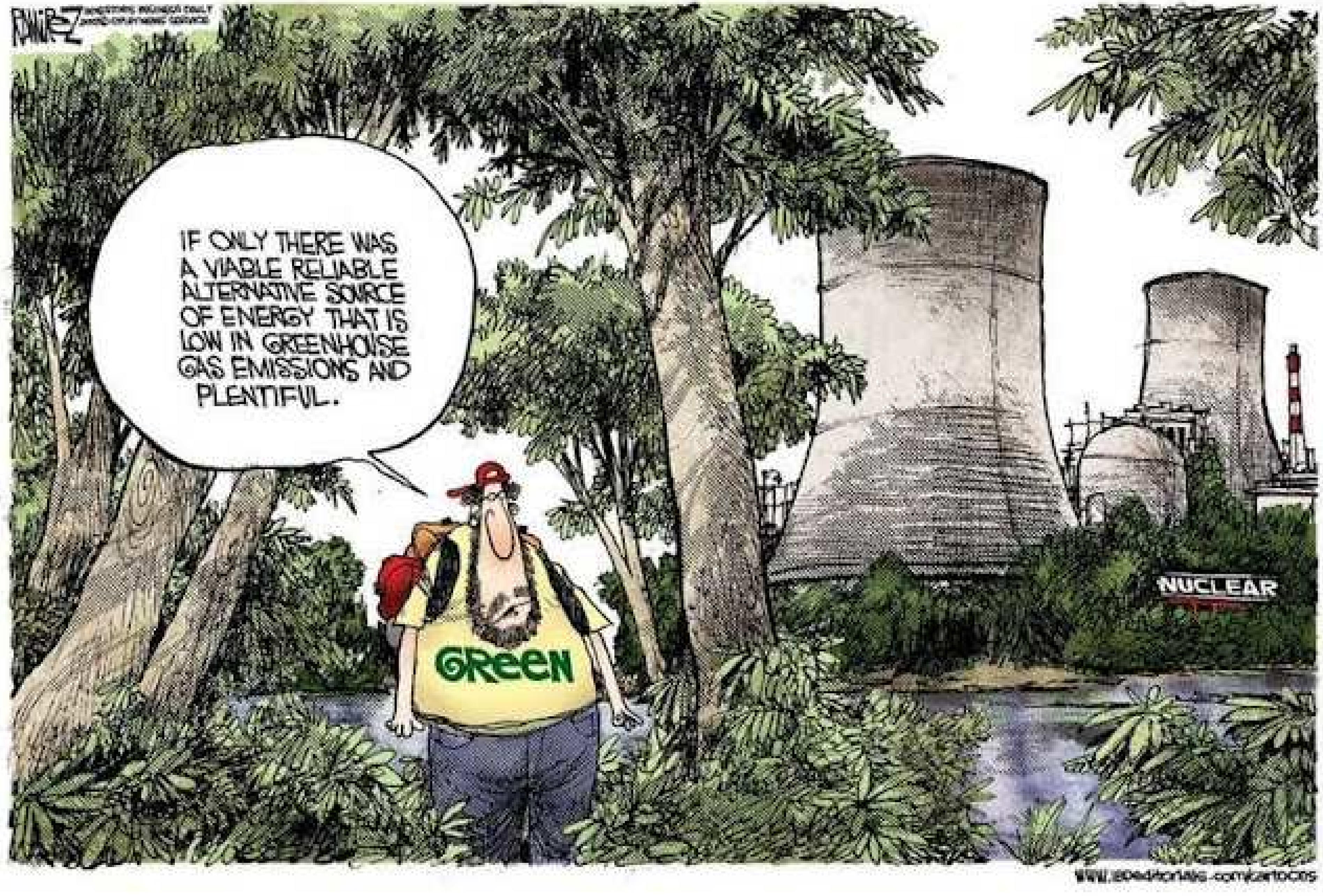
Example for number densities



Example for number densities

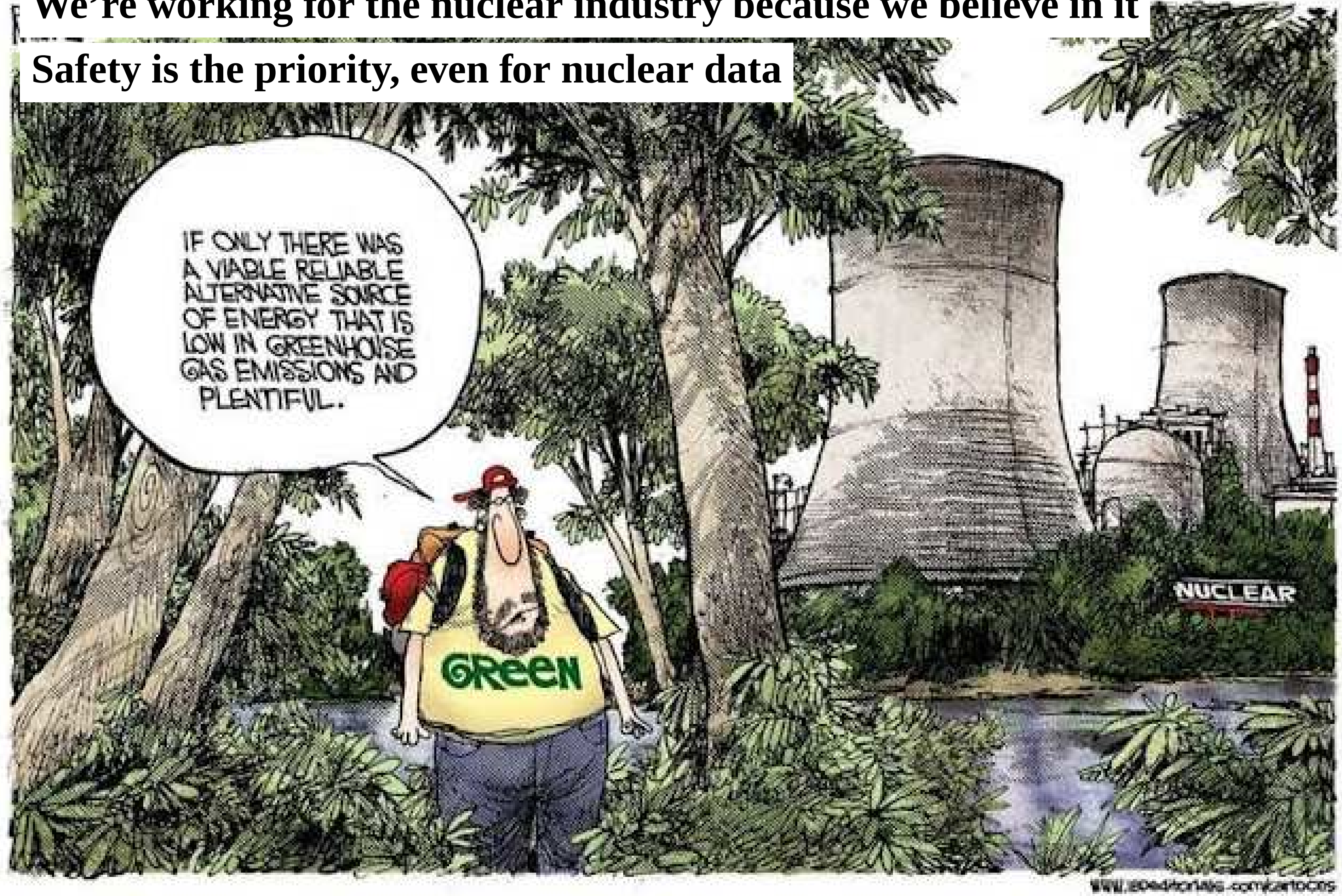


Discussion/conclusion



Discussion/conclusion

- **We're working for the nuclear industry because we believe in it**
- **Safety is the priority, even for nuclear data**



Discussion/conclusion

- **We're working for the nuclear industry because we believe in it**
- **Safety is the priority, even for nuclear data**
- **The general level of QA for nuclear data used by simulation codes is poor**
- **The way to look at nuclear data needs to be upgraded (no more ”tapes”...)**
- **We believe that our approach is improving the situation**
- **with finally, a modern use of computers**