SCALE/AMPX Sensitivity and Uncertainty Analysis Capabilities and Data

Presented to:

OECD NEA Working Party on Evaluation Cooperation (WPEC) Subgroup 44 on Investigation of Covariance Data in General Purpose Nuclear Data Libraries

OECD/NEA Headquarters, Paris, France
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ORNL is managed by UT-Battelle
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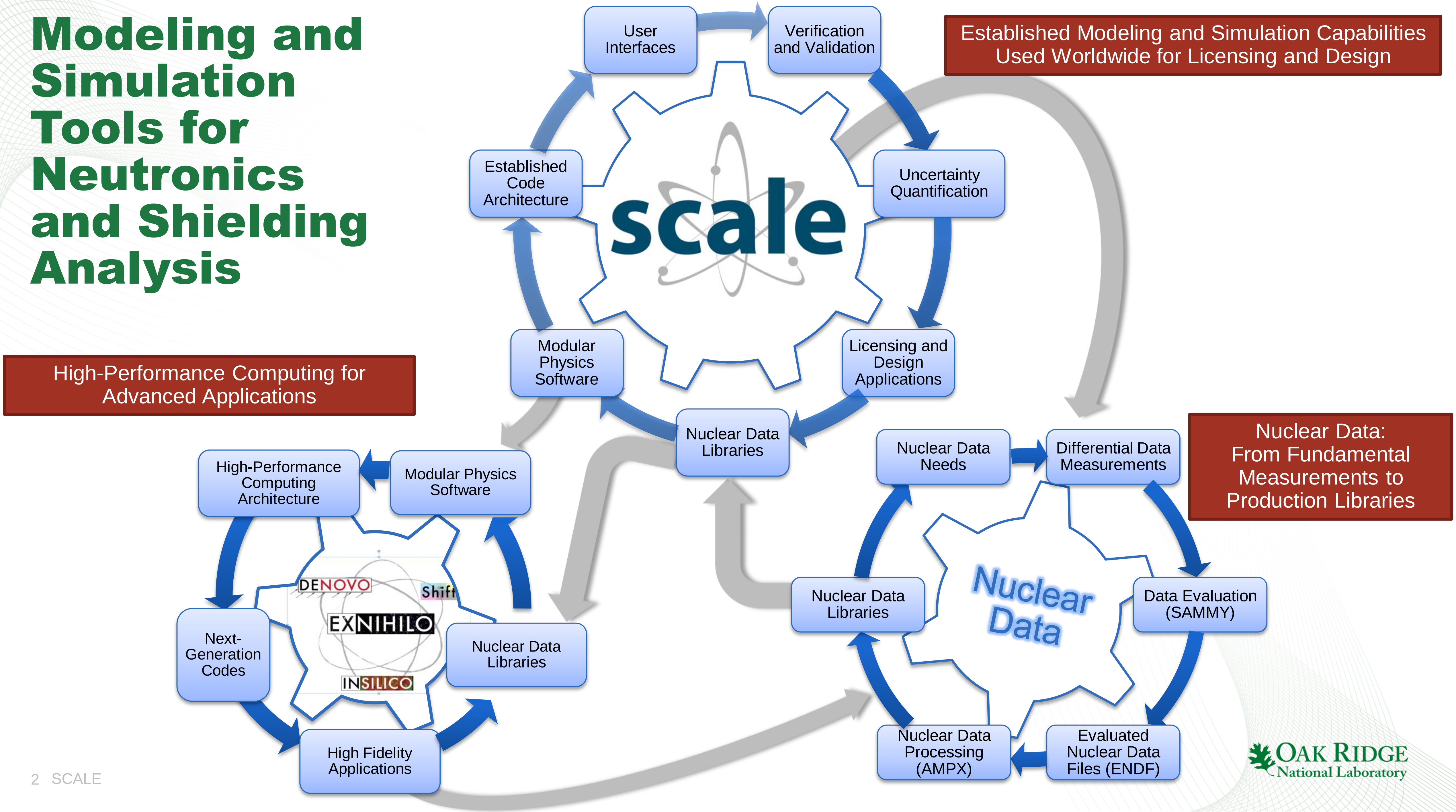
Modeling and Simulation Tools for Neutronics and Shielding Analysis



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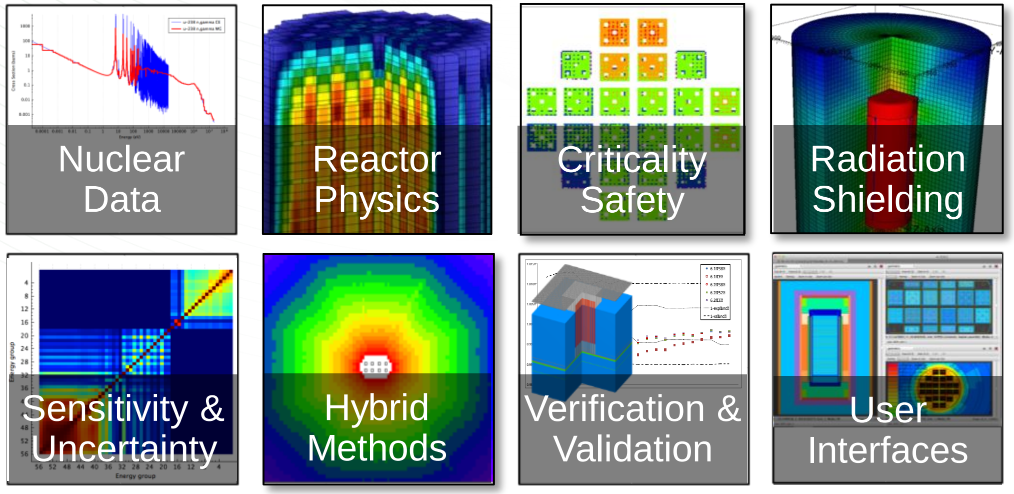
SCALE Code System

Neutronics and Shielding Analysis Enabling Nuclear Technology Advancements

http://scale.ornl.gov

Practical Tools Relied Upon for Operations and Regulation



Global Distribution: 8000 Users in 56 Nations

Professional Training for Practicing Engineers and Regulators

FY16 Statistics:

- **12 week-long courses**
- **1 conference tutorial**
- **150 participants from 15 nations**

Primary Sponsors

Robust Quality Assurance Program Based on Multiple Standards

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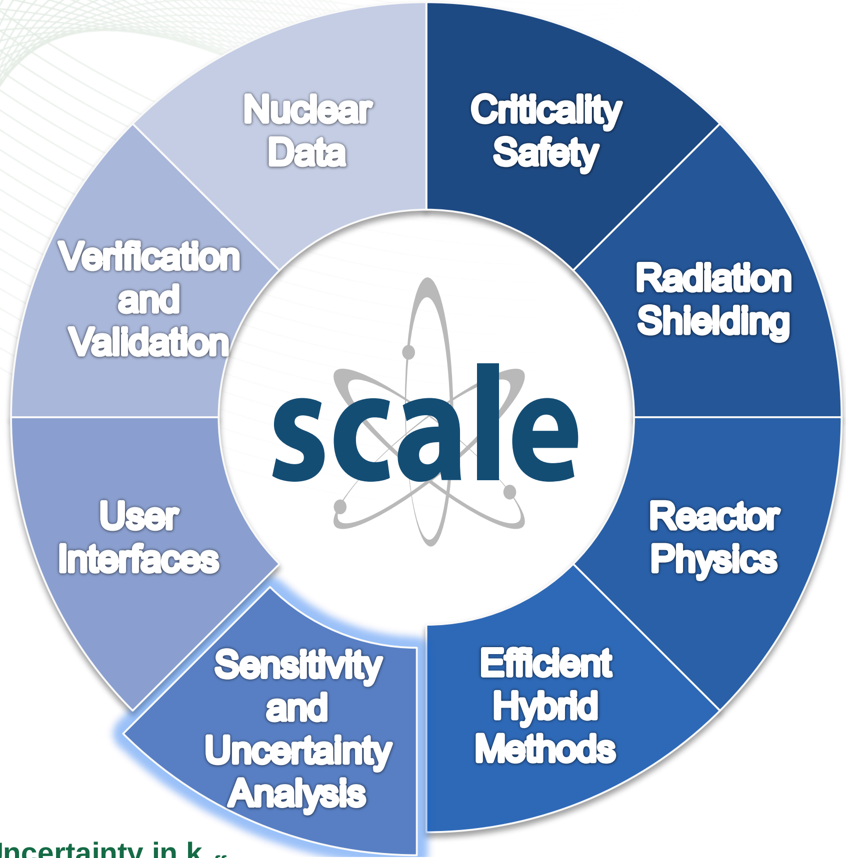
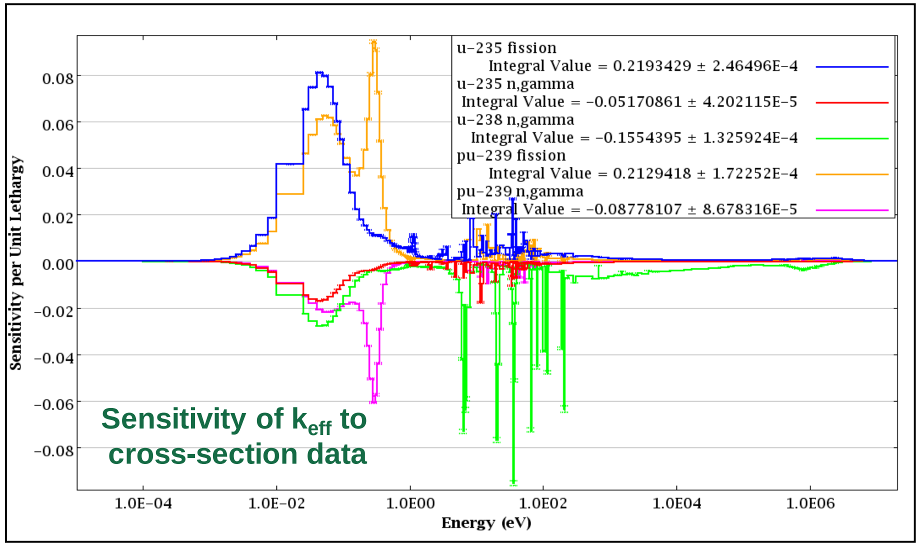
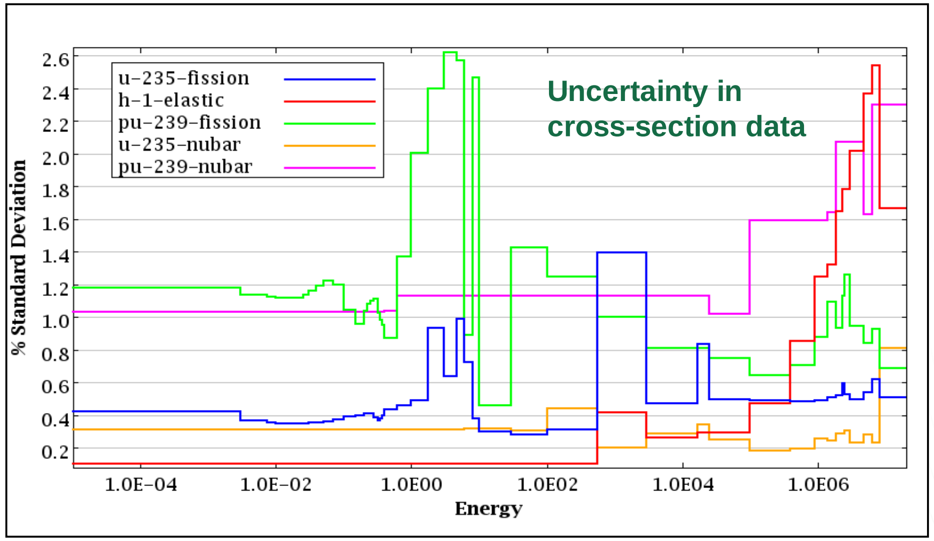
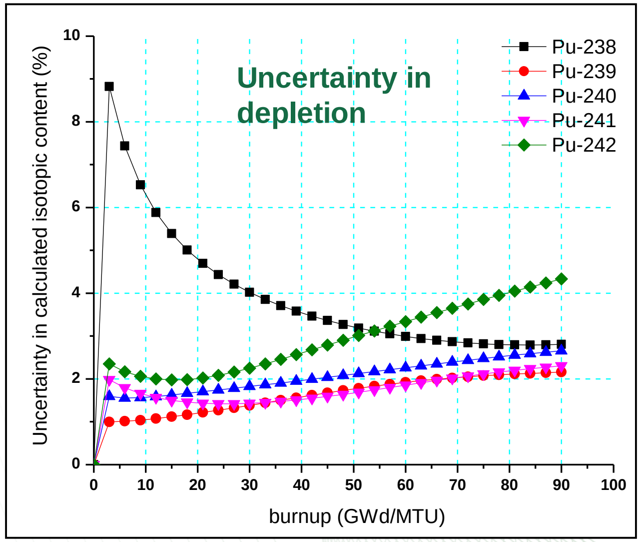
Sensitivity and Uncertainty Analysis









Uncertainty in k_{eff}

Covariance Matrix		Unc. in % dk/k
Nuclide-Reaction	Nuclide-Reaction	Due to this Matrix
^{239}Pu nubar	^{239}Pu nubar	4.0032E-01 ± 2.5625E-06
^{238}U n,gamma	^{238}U n,gamma	1.9457E-01 ± 1.2387E-05
^{239}Pu fission	^{239}Pu fission	1.5501E-01 ± 1.0838E-05
^{235}U nubar	^{235}U nubar	1.3981E-01 ± 5.0038E-07
^{239}Pu fission	^{239}Pu n,gamma	1.2261E-01 ± 4.3564E-06

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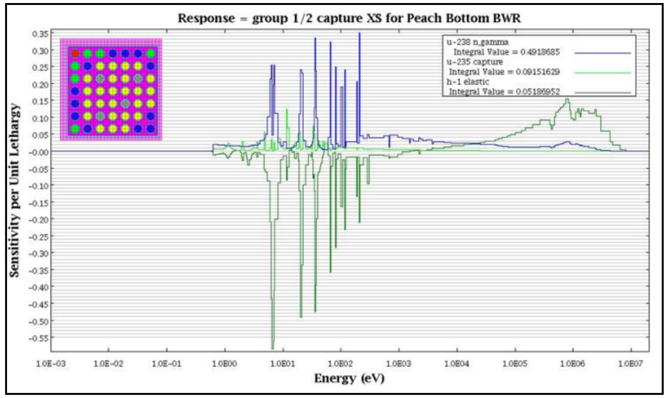
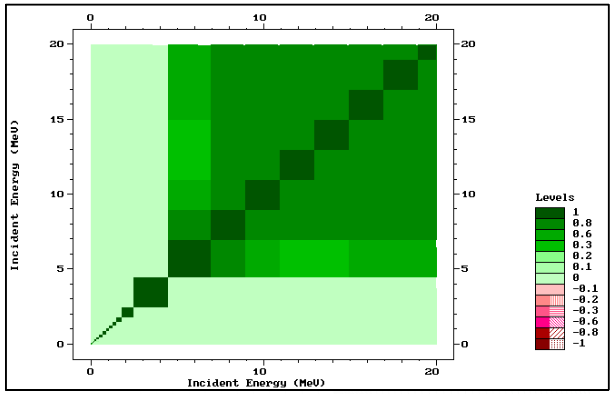
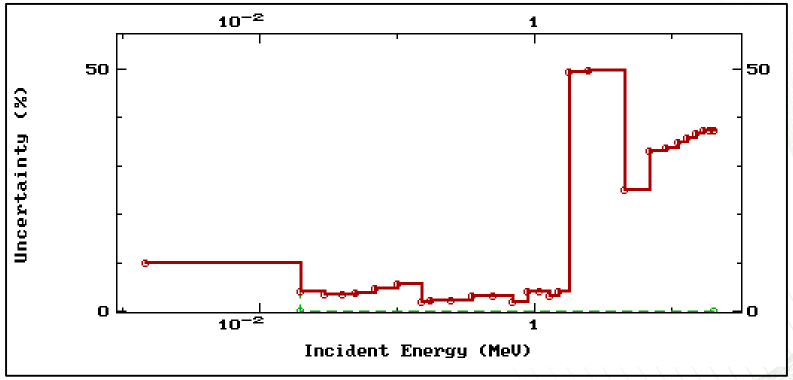
Sensitivity and Uncertainty Enhancements

- Sensitivity computations with continuous-energy Monte Carlo for eigenvalue and GPT responses "CE-TSUNAMI"
- New statistical sampling code "Sampler" for general UQ of Scale MG Sequences
- Covariance data
    - Neutron cross section covariance library based on ENDF/B-VII.1 and previous SCALE 6.1 data
    - New fission product yield covariances
    - New decay data covariances
    - New gamma yield covariances
- Calculation of experimental correlation coefficients with Sampler



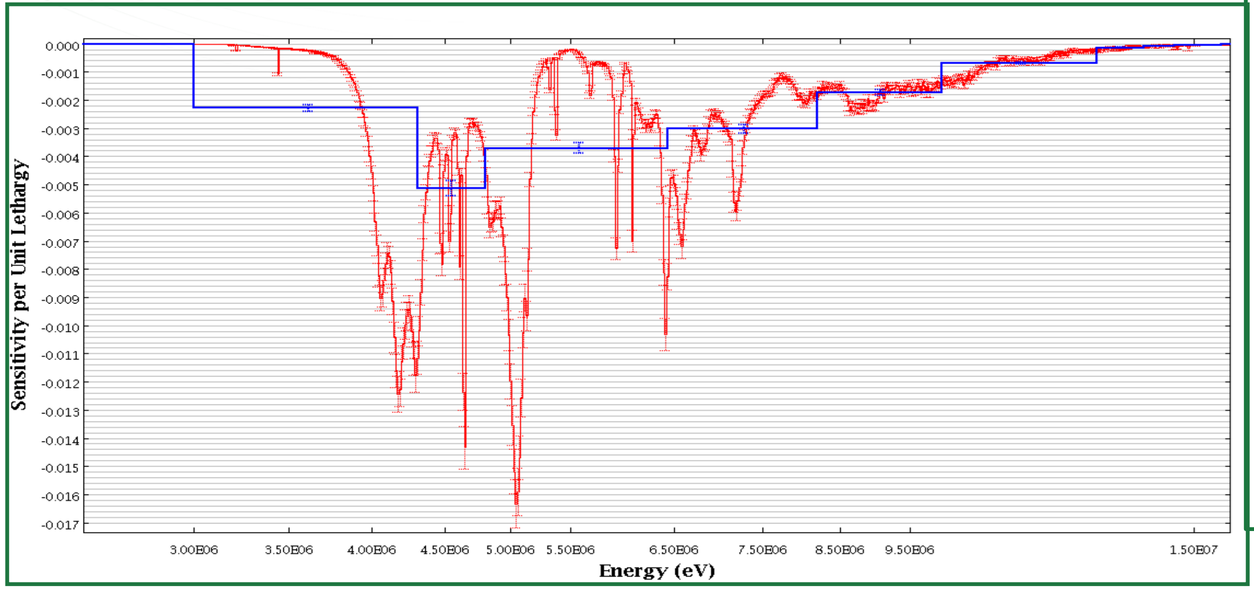




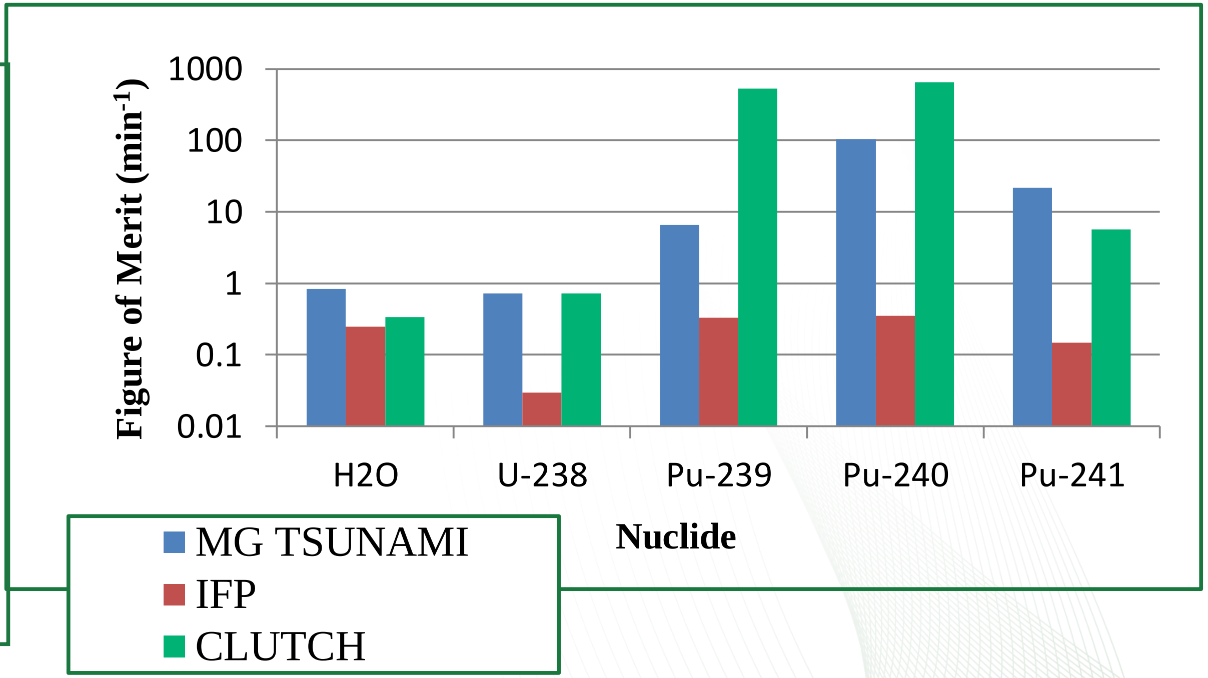
Continuous-energy TSUNAMI-3D

- In SCALE6.2 the multigroup TSUNAMI-3D code has been extended to perform continuous-energy (CE) sensitivity coefficient calculations.
  - ➢ This work involved the development of the CLUTCH sensitivity method, a new and efficient approach for calculating eigenvalue sensitivity coefficients.

O-16 Capture Sensitivity
238-group VS
Microgroup CLUTCH



MIX-COMP-THERM-004-001
FoM Comparison

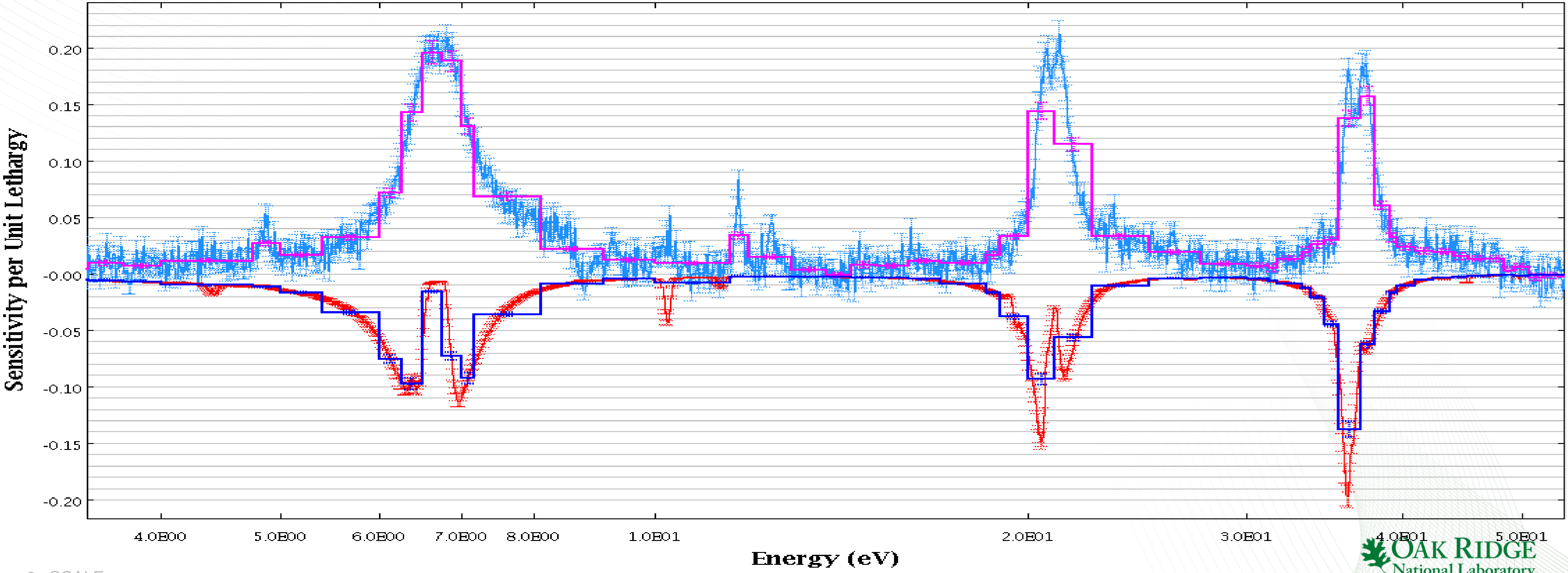


C. M. Perfetti, B. T. Rearden, and W. R. Martin, "SCALE Continuous-Energy Eigenvalue Sensitivity Coefficient Calculations," *Nucl. Sci. Eng.* **182**(3), 332-353 (2016).

CE TSUNAMI-3D Sensitivity Capabilities use High-Fidelity Monte Carlo Methods

H-1 Elastic Scatter Sensitivity
238-group CLUTCH VS
Microgroup CLUTCH

U-238 Capture Sensitivity
238-group CLUTCH VS
Microgroup CLUTCH



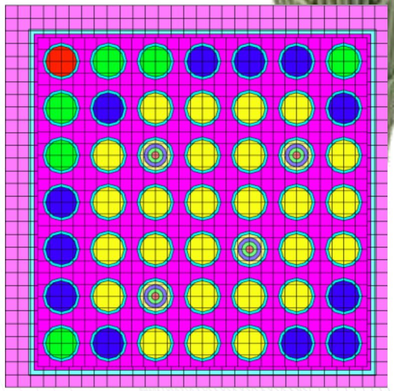
Generalized Perturbation Theory

- Recent developments have enabled the calculation of generalized response sensitivity coefficients using high-fidelity, continuous-energy Monte Carlo methods.

- Generalized Perturbation Theory (GPT) calculates sensitivity coefficients for any system response that can be expressed as the ratio of reaction rates.

$$R = \frac{\langle S_1 f \rangle}{\langle S_2 f \rangle}$$

- Applications for GPT sensitivity/uncertainty analysis include:
  - Relative powers
  - Isotope Conversion Ratios
  - Multigroup Cross Sections
  - Experimental Parameters



NUMBER	EXPERIMENT	Type	Format	Value	Xsec Uncert
1	k_infinity	keff	Relative	1.1083E+0	4.98551E-1 % dk/k
2	fission_grp_1	gpt	Relative	1.9155E-3	6.91925E-1 % dR/R
3	fission_grp_2	gpt	Relative	2.7748E-2	3.23440E-1 % dR/R
4	absorpt_grp_1	gpt	Relative	7.1637E-3	8.36728E-1 % dR/R
5	absorpt_grp_2	gpt	Relative	5.3702E-2	2.38082E-1 % dR/R
6	cornerrod_fpf	gpt	Relative	1.1458E+0	1.67147E-1 % dR/R

OECD UAM GPT Benchmark Phase 1-2 Results

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GPT Calculations in CE TSUNAMI-3D

- The generalized importance function for a response can be expressed as the sum of two terms: the intra-generation effect term and the inter-generational effect term.
  - The **intra-generation** effect describes how much importance a neutron generates after an event occurs.
  - The **inter-generational** effect describes the importance that is generated by the daughter fission neutrons of the original particle.

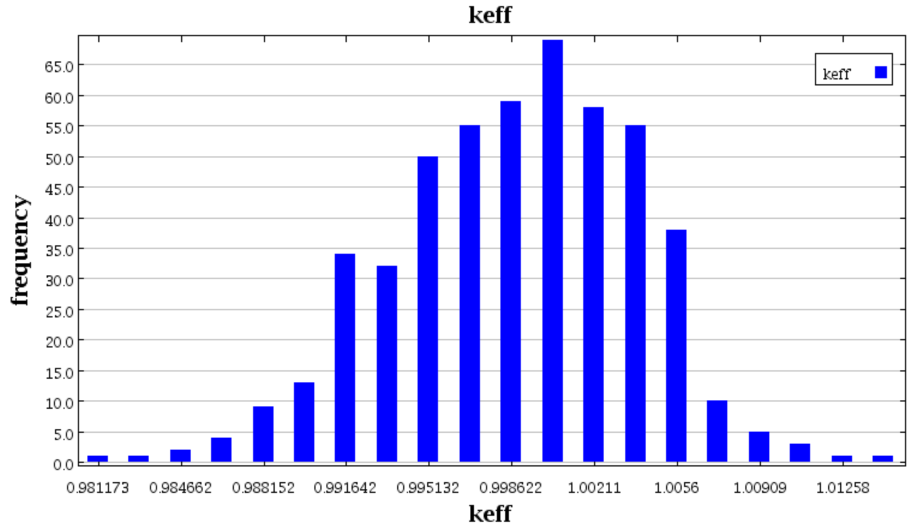
$$\Gamma^{\dagger}(\tau_s) = \boxed{\frac{1}{Q_s}\langle \frac{1}{R}\frac{\partial R}{\partial \phi}(r)\phi(\tau_s \rightarrow r)\rangle} + \boxed{\frac{\lambda}{Q_s}\langle \Gamma^{\dagger}(r)P(r)\phi(\tau_s \rightarrow r)\rangle}$$

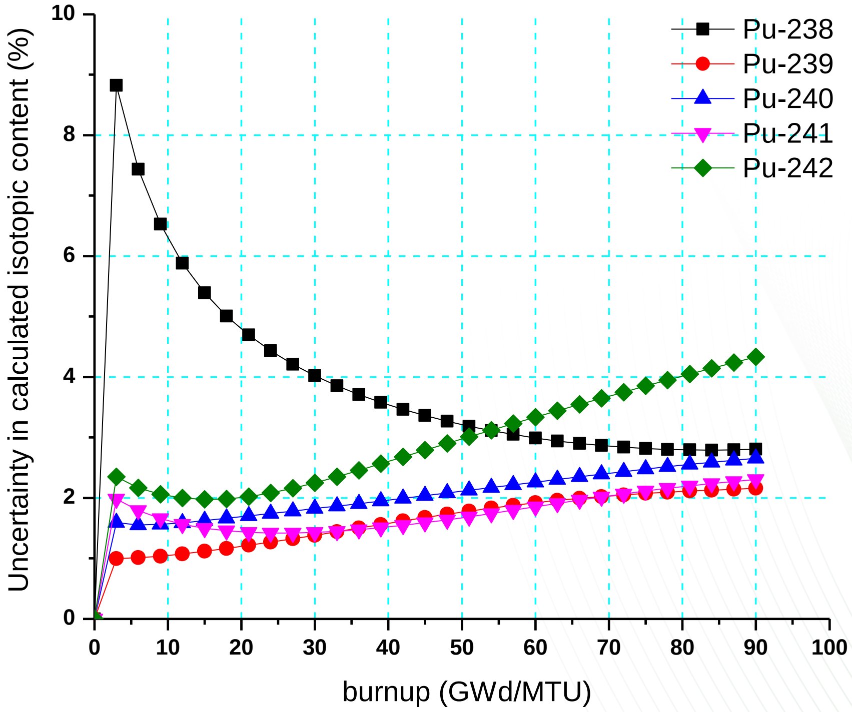
- CE TSUNAMI-3D uses the **CLUTCH** sensitivity method to calculate the intra-generation term, and an **Iterated Fission Probability**-based approach to calculate the inter-generational term.

C. M. Perfetti and B. T. Rearden, "Development of a Generalized Perturbation Theory Method for Uncertainty and Sensitivity Analysis Using Continuous-Energy Monte Carlo Methods," *Nucl. Sci. Eng.* **182**(3), 354-368 (2016).

Sampler: A Module for Statistical Uncertainty Analysis with SCALE Sequences

- Sampler provides uncertainty in any computed result from any SCALE sequence due to uncertainties in:
  - neutron cross sections
  - fission yield and decay data
  - geometry and composition
- Sampler propagates uncertainties through complex analysis sequences such depletion calculations
- Correlations between systems are also computed





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SCALE 6.0–6.1 Covariance Library (401 materials)

Data Source	Materials
ENDF/B-VII.0	$^{152,154-158,160}$Gd,^{191,193}Ir,^{7}Li, ^{99}Tc, ^{232}Th
ENDF/B-VII-p	^{197}Au, ^{209}Bi, ^{59}Co, ^{23}Na, ^{93}Nb, ^{58}Ni, ^{239}Pu, ^{48}Ti, ^{233,235,238}U*, V
ENDF/B-VI	^{27}Al, ^{241}Am, C, C-graphite, $^{50,52-54}$Cr, ^{65}Cu, ^{156}Dy, $^{54,56-58}$Fe, In, ^{55}Mn, $^{60-62,64}$Ni, $^{206-208}$Pb, ^{242}Pu, ^{185,187}Re, ^{45}Sc, Si, $^{28-30}$Si, ^{89}Y
JENDL 3.3	^{11}B, ^{240,241}Pu
JENDL 3.3+BLO	^{16}O
SG-26	^{234,236}U, ^{242,242m}Am, $^{242-245}$Cm, ^{237}Np, ^{238}Pu
BLO LANL evaluation +JENDL 3.3	^{10}B, ^{1}H, H-ZrH, H-poly, Hfreegas
BLO LANL evaluation	^{6}Li
BLO Approximate Data	$^{225-227}$Ac, ^{107,109,110m,111}Ag, ^{243,244,244m}Am, ^{36,38,40}Ar,$^{74-75}$As, $^{130,132,133,135-138,140}$Ba, ^{7,9}Be, Bebound, ^{249,250}Bk, ^{79,81}Br, Ca, $^{40,42-44,46,48}$Ca, Cd, $^{106,108,110-114,115m,116}$Cd, $^{136,138,139-144}$Ce, $^{249-254}$Cf, Cl, ^{35,37}Cl, $^{241,246-250}$Cm, ^{58,58m}Co, $^{133-137}$Cs, ^{63}Cu, $^{158,160-164}$Dy, $^{162,64,166-168,170}$Er, $^{253-255}$Es, $^{151-157}$Eu, ^{19}F, ^{255}Fm, Ga, ^{69,71}Ga, ^{153}Gd, $^{70,72-74,76}$Ge, ^{2,3}H, Dfreegas,^{3,4}He, Hf, $^{174,176-180}$Hf, $^{196,198-202,204}$Hg, ^{165}Ho, $^{127,129-131,135}$I, ^{113,115}In, K, $^{39-41}$K, $^{78,80,82-86}$Kr, $^{138-140}$La, ^{175,176}Lu, Mg, $^{24-26}$Mg, Mo, $^{92,97-100}$Mo, ^{14,15}N, ^{94,95}Nb, $^{142-148,150}$Nd, ^{59}Ni, ^{235,236,238,239}Np, ^{17}O, ^{31}P, $^{231-233}$Pa, ^{204}Pb, $^{102,104-108,110}$Pd, ^{147,148,148m,149,151}Pm, $^{141-143}$Pr, ^{236,237,243,244,246}Pu, $^{85-87}$Rb, ^{103,105}Rh, $^{96,98-106}$Ru, S, $^{32-34,36}$S, $^{121,123-126}$Sb, $^{74,76-80,82}$Se, $^{144,147-154}$Sm, $^{112-120,122-125}$Sn, $^{84,86-90}$Sr, ^{181,182}Ta, ^{159,160}Tb, $^{120,122-126,127m,128,129m,130}$Te, $^{227-230,233,234}$Th, Ti, ^{46,47,49,50}Ti, $^{232,237,239-241}$U, W, $^{182-184,186}$W, $^{123,124,126,128-136}$Xe, ^{90,91}Y, Zr, $^{90-96}$Zr

ENDF/B-VII.0: evaluated covariance data released with ENDF/B-VII.0	**JENDL-3.3**: evaluated covariance data in JENDL-3.3
ENDF/B-VII-p: recently evaluated data proposed for future release of ENDF/B-VII.1	**BLO approximate data**: lo-fi covariances from BLO project (**B**rookhaven, **L**os Alamos, **O**RNL)
ENDF/B-VI: evaluated covariance data released with ENDF/B-VI	**BLO LANL evaluation**: LANL R-matrix evaluation from BLO project
* ^{235}U thermal nubar data from JENDL 3.3	**SG-26**: approximate covariances from WPEC Subgroup-26

ENDF/B-VII.1 Covariance Nuclides

h1
h2
he4
li6
li7
be9
b10
b11
c
n15
o16
f19
na23
mg24
mg25
mg26
al27
si28
si29
si30
cl35
cl37
k39
k41
ti46
ti47
ti58
ti49
ti50
cr50
cr52
cr53
cr54
mn55
fe54
fe56
fe57
co59
ni58
ni60
y89
zr90
zr91
zr92
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nb95
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i127
i129
xe131
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es255
fm255
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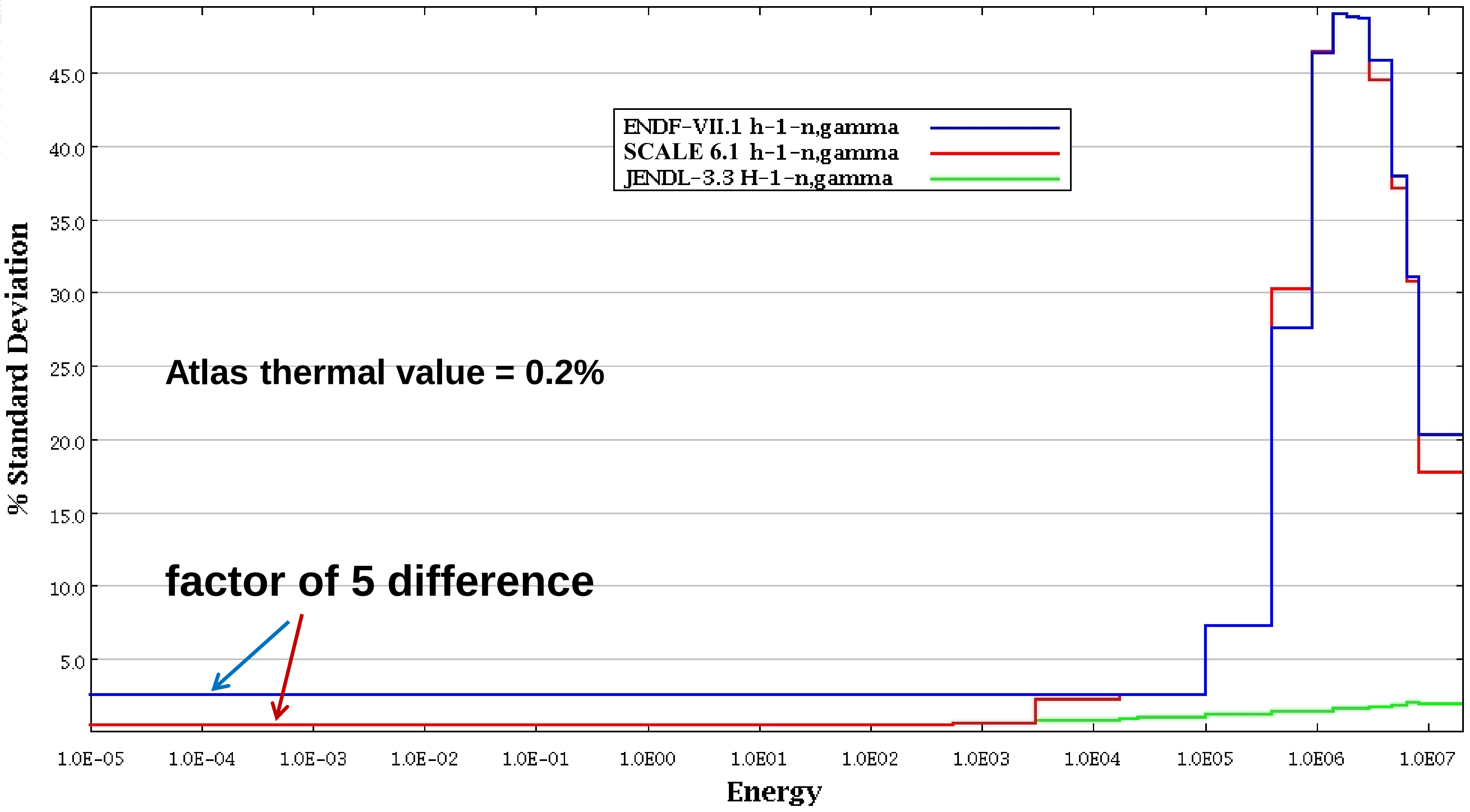
fe_thermal
bebound
be-beo
h-h2o
d-d20
h-poly
h-zr2
o-beo
o-u2o
si28_si2o
si29_si2o
si30_si2o
u235_u2o
zr90_zr_zrh
zr91_zr_zrh
zr92_zr_zrh
zr93_zr_zrh
zr94_zr_zrh
zr95_zr_zrh
zr96_zr_zrh
h-benzene
benzene
graphite
h-liquid_ch4
d-cryo_ortho
h-cryo_ortho
d-cryo_para
h-cryo_para
h-solid_ch4

H Capture Uncertainty



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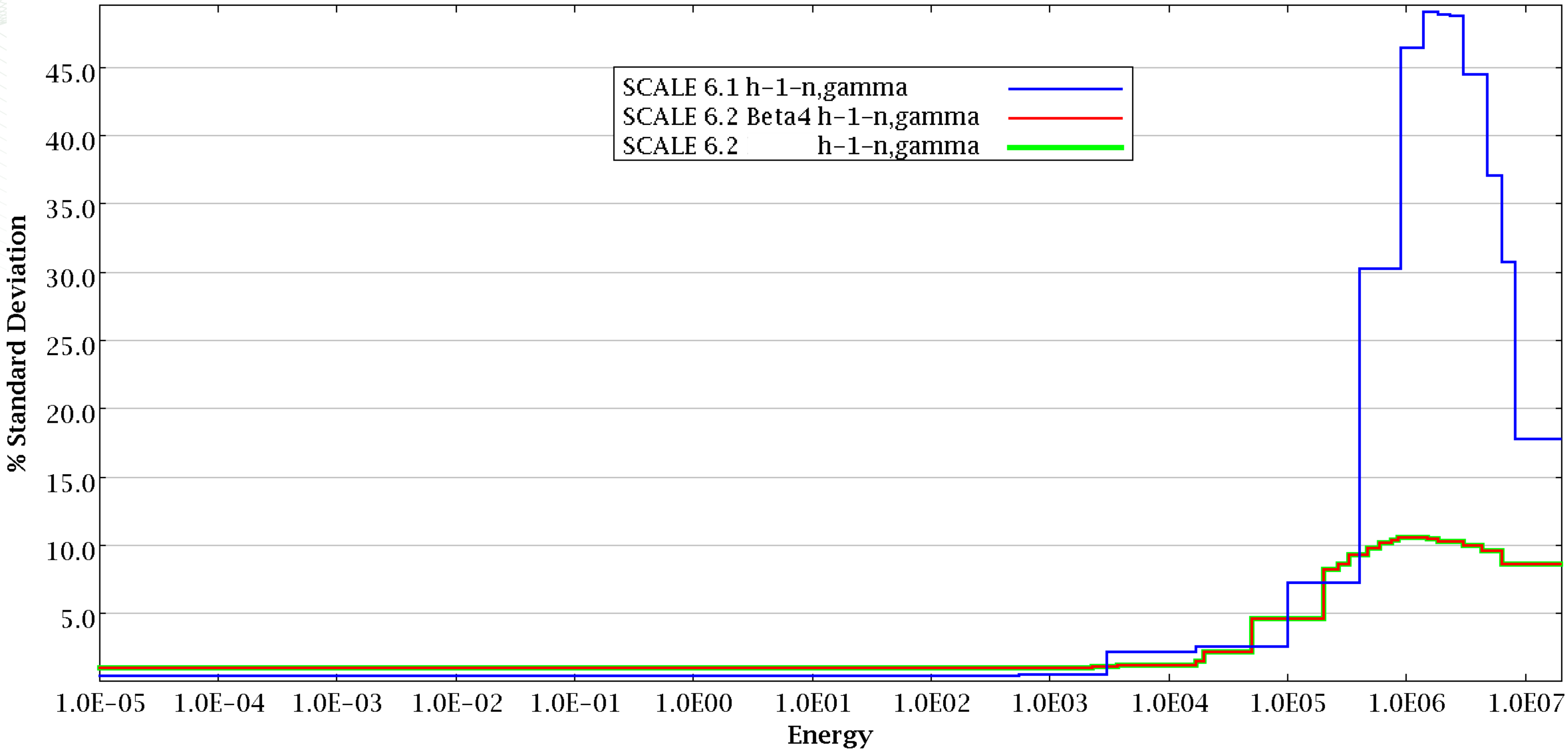
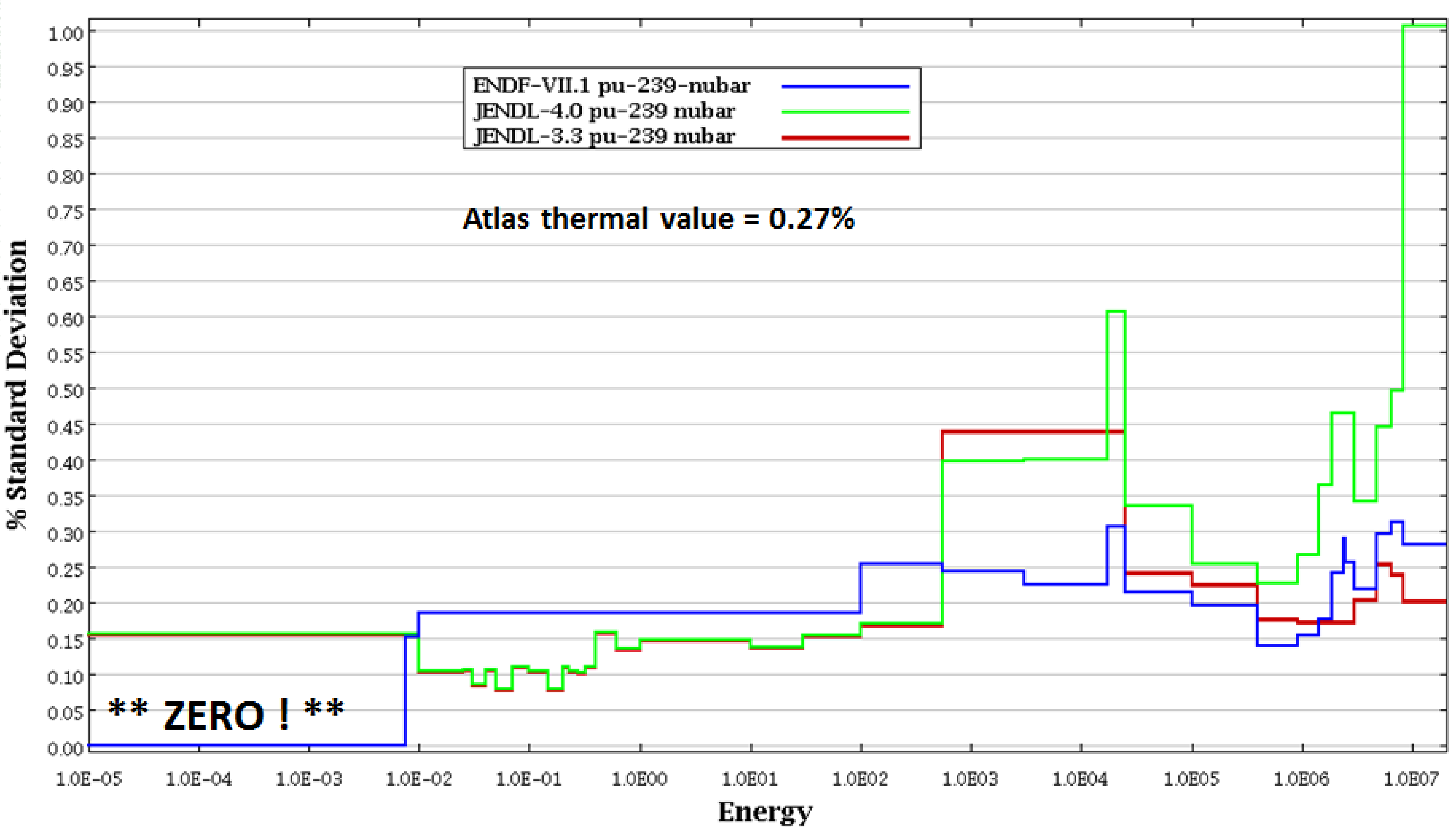
H Capture Uncertainty



Pu-239 nubar uncertainty



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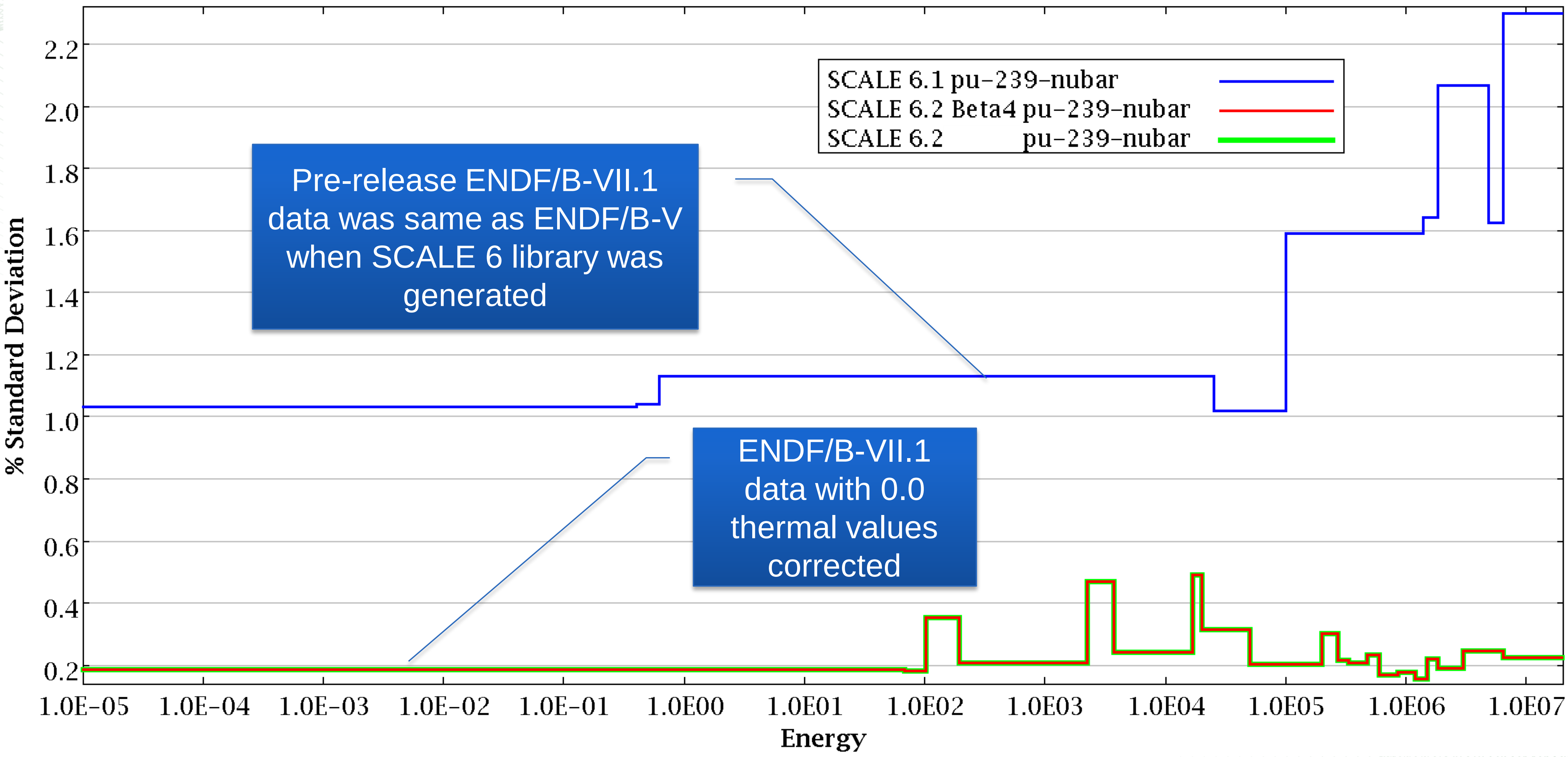
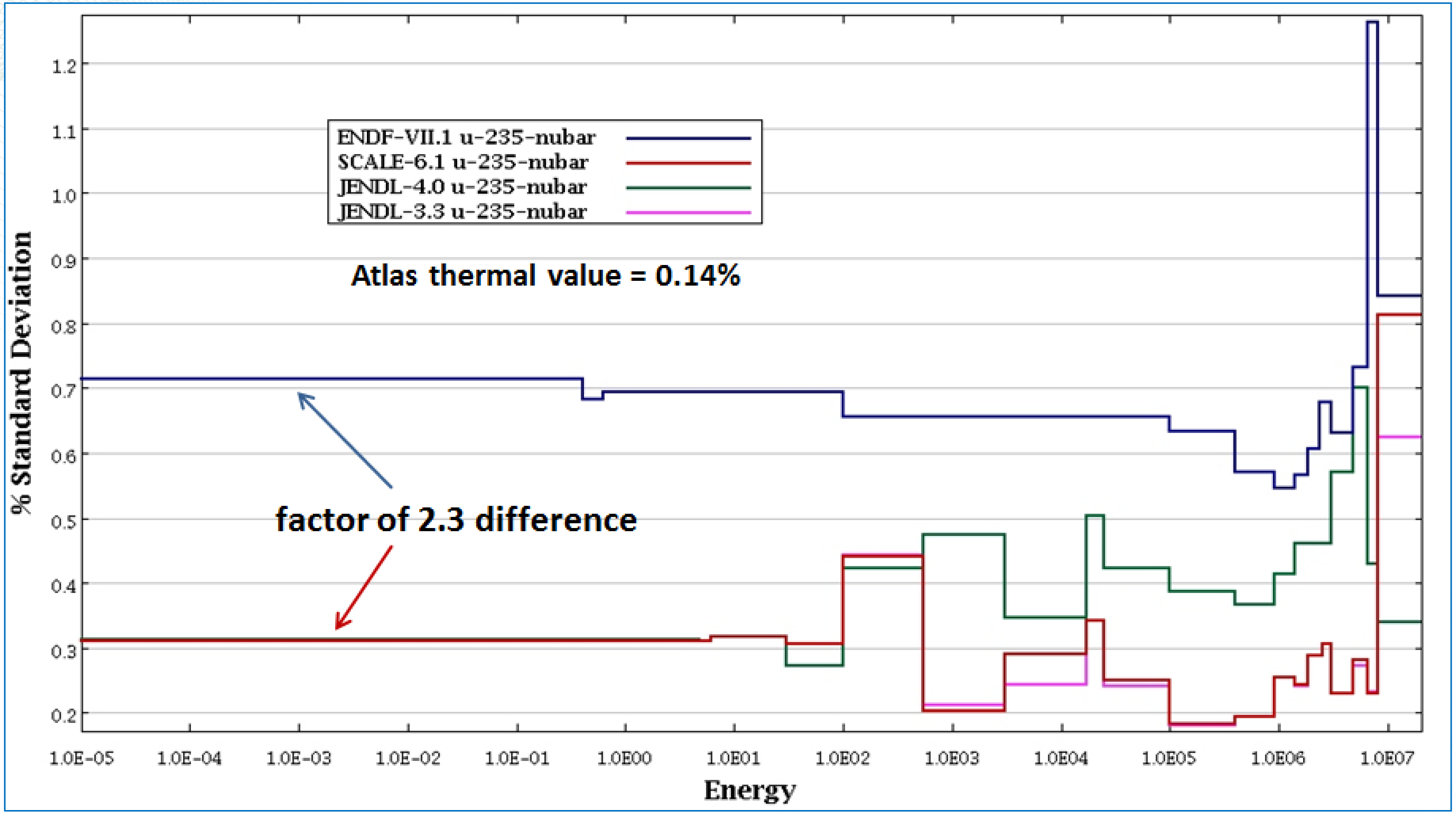
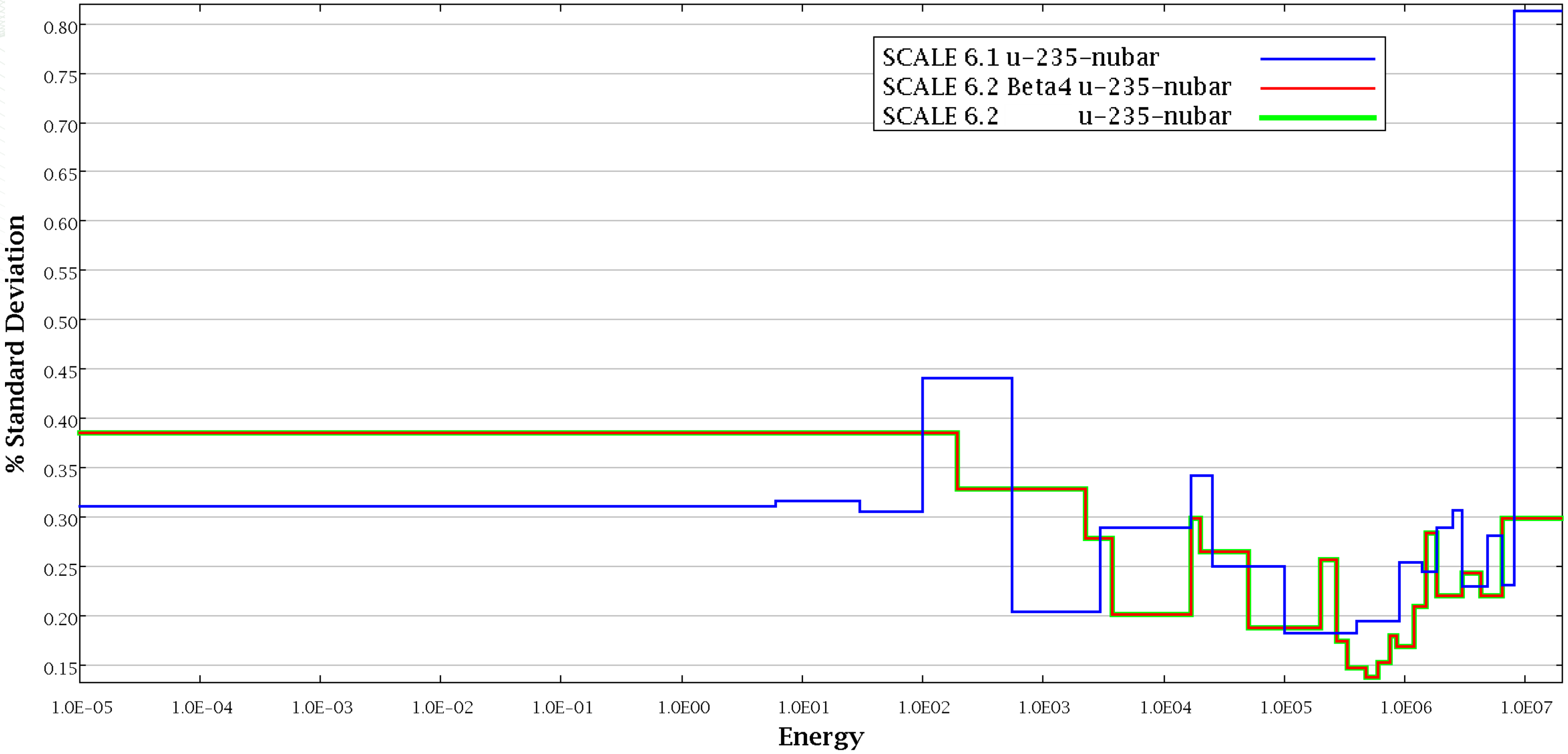
Pu-239 nubar uncertainty



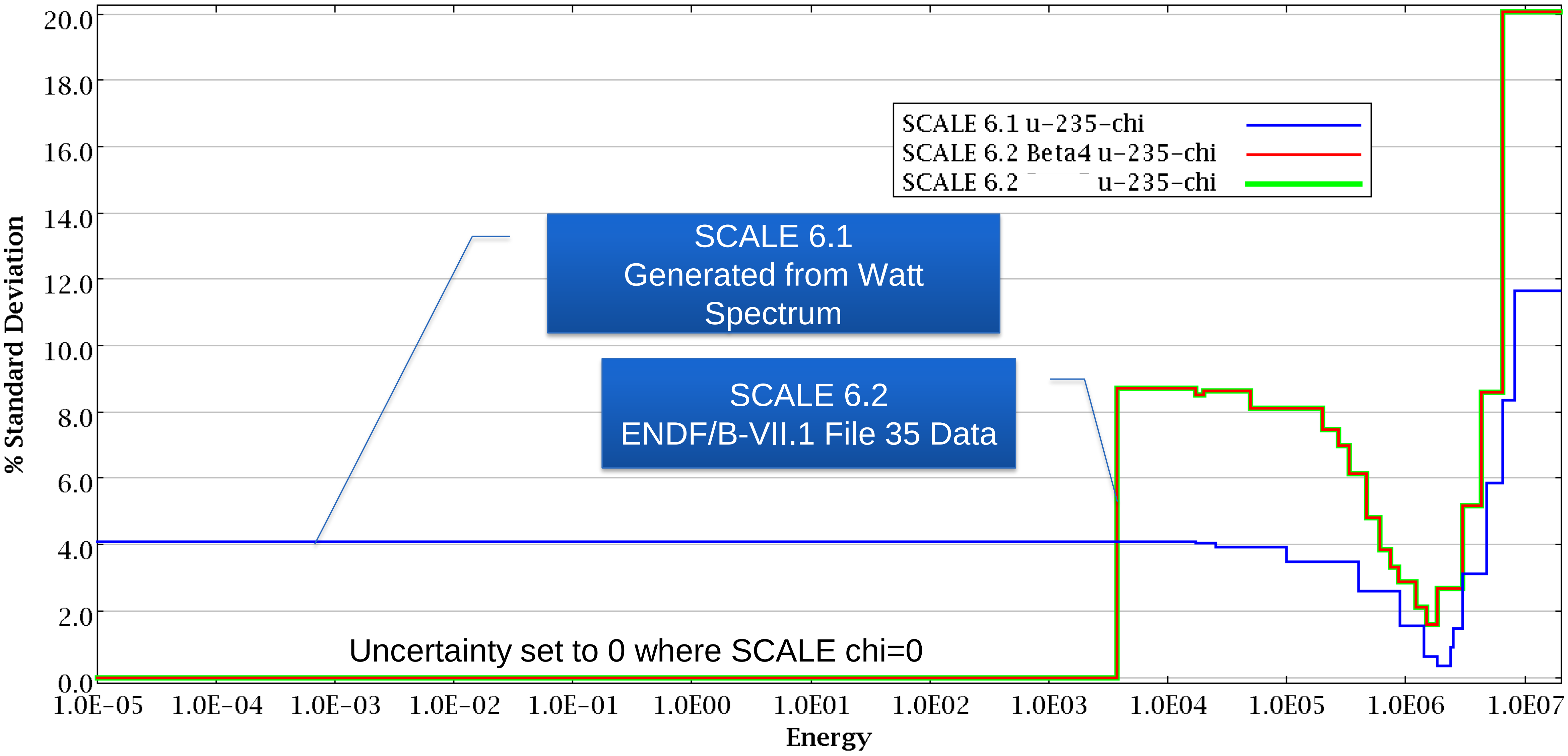
U-235 nubar uncertainty



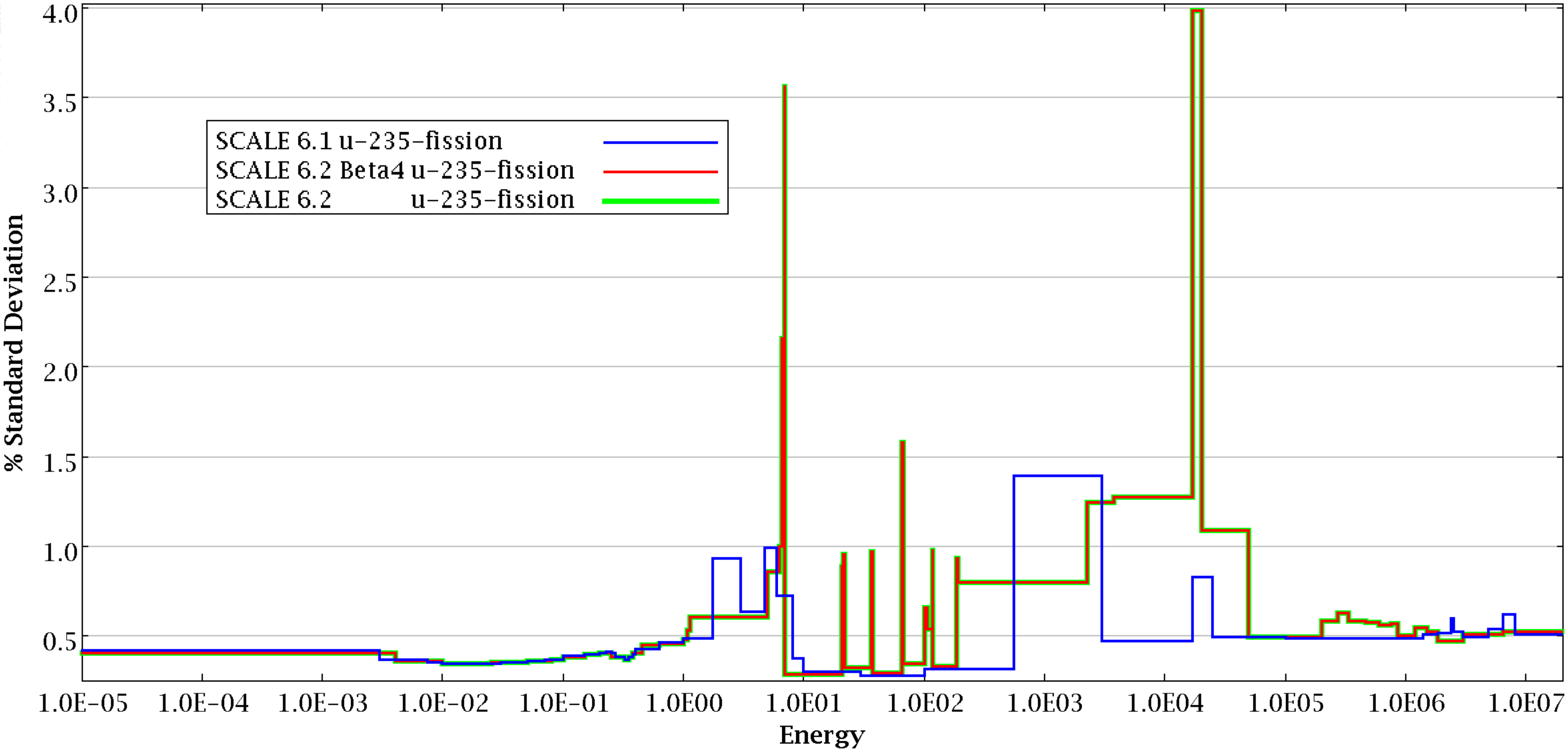
U-235 nubar uncertainty



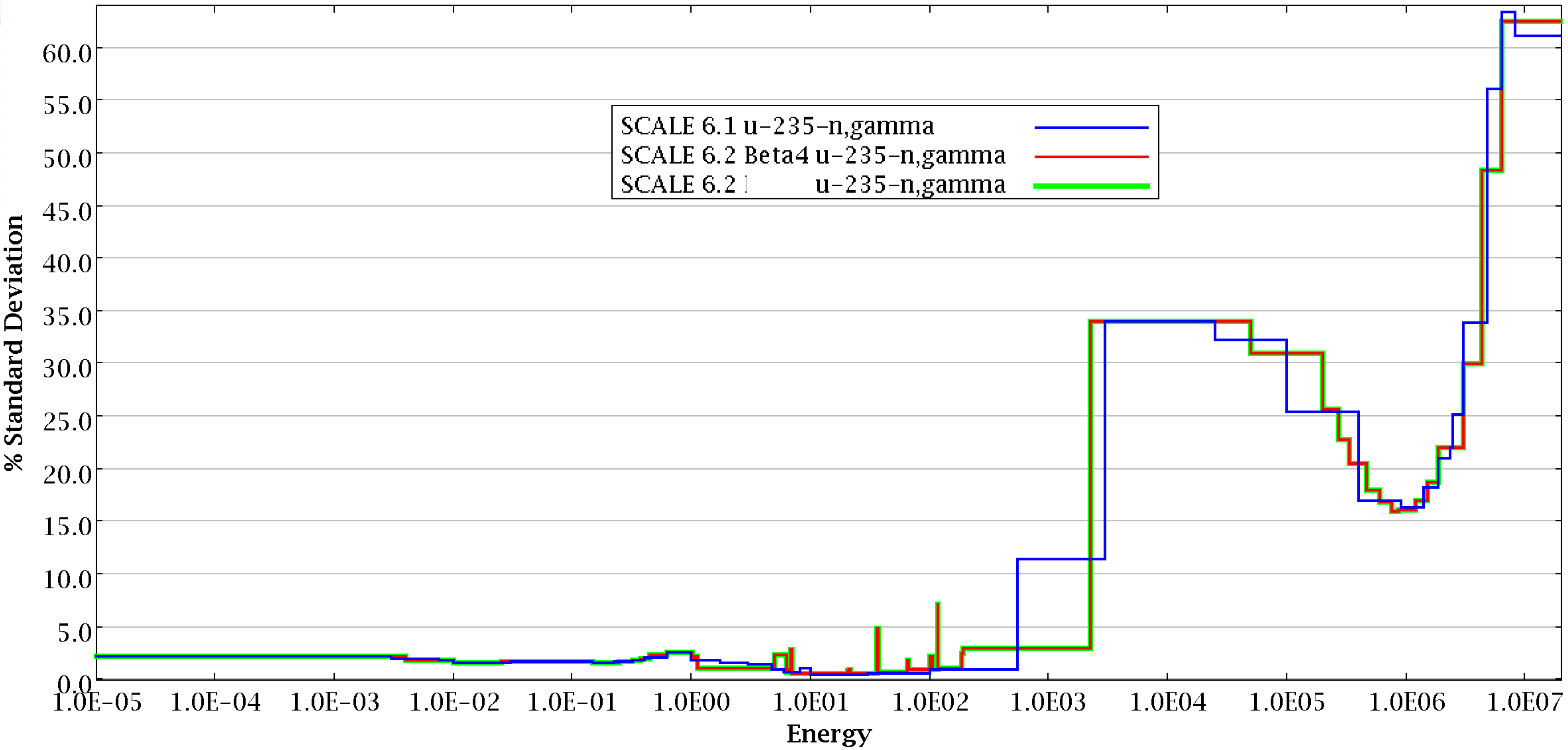
U-235 chi uncertainty



U-235 fission uncertainty

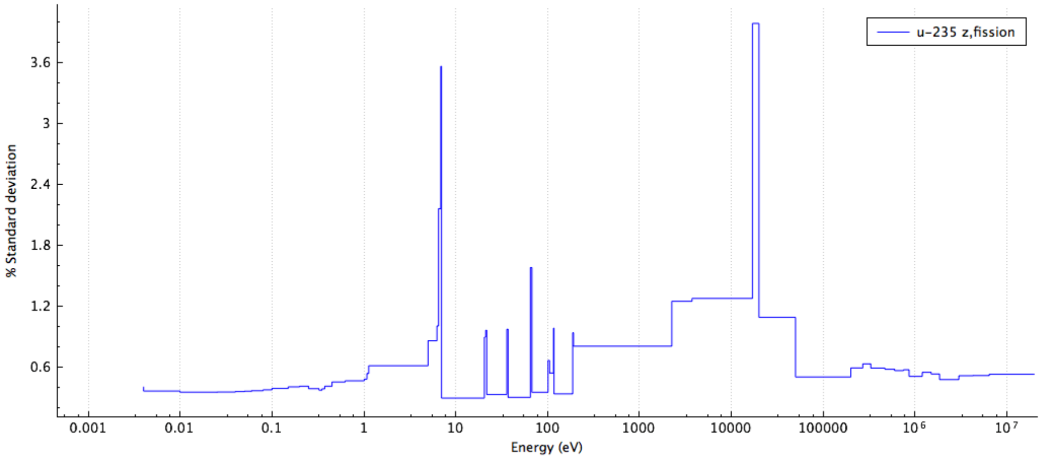


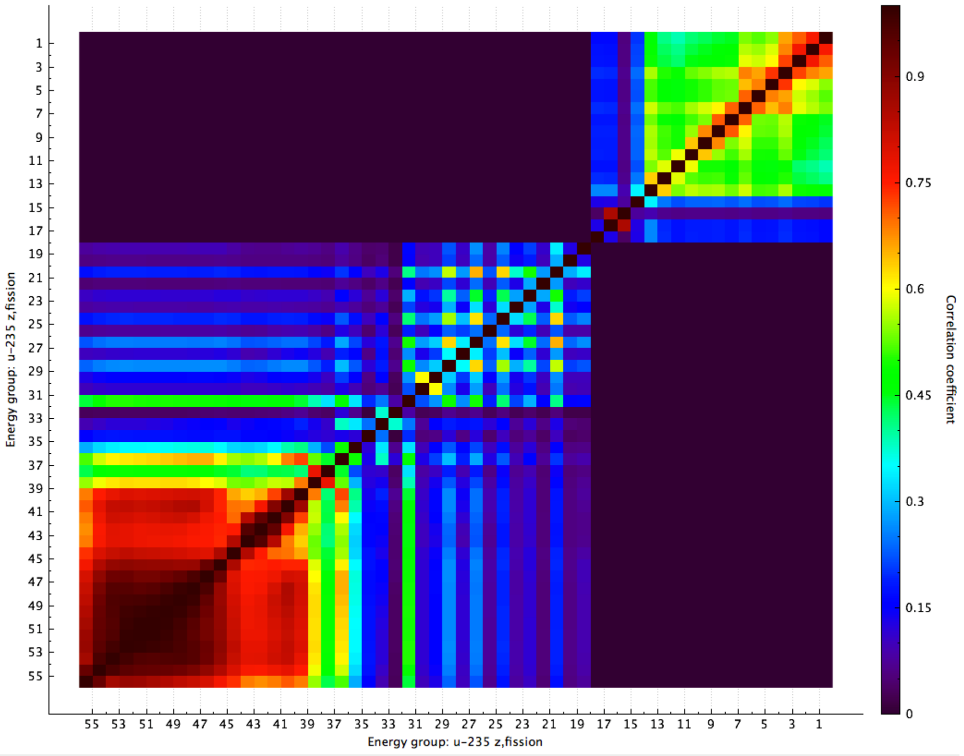
U-235 n,gamma uncertainty



SCALE 6.2 Covariance Library

- ENDF/B-VII.1 for 187 isotopes
- SCALE 6.1 data retained for ~215 missing nuclides
- Modified ENDF/B-VII.1 ^{239}Pu nubar, ^{235}U nubar, H capture, and several fission product uncertainties, with data contributed back to ENDF/A repository
- Fission spectrum (chi) uncertainties processed from ENDF/B-VII.1 and from JENDL 4.0 (minor actinides)
  - Previous SCALE chi uncertainties were generated from Watt spectrum data and data were missing for minor actinides
- 56- and 252-group energy structures





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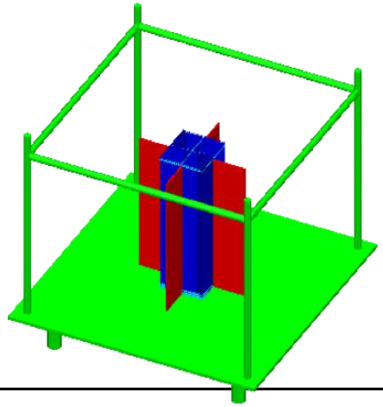
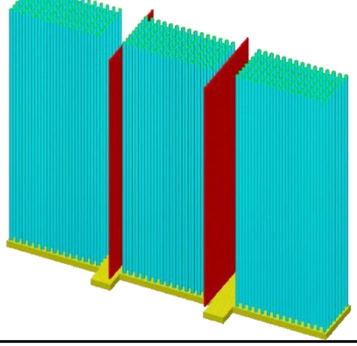
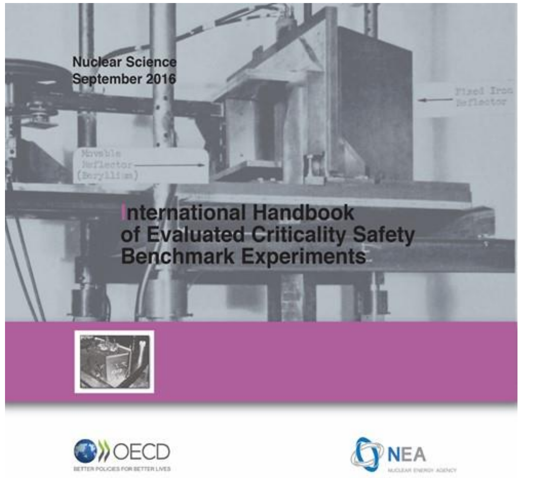
Validation with critical benchmarks for many types of systems

- 411 configurations from International Criticality Safety Benchmark Evaluation Project (ICSBEP)







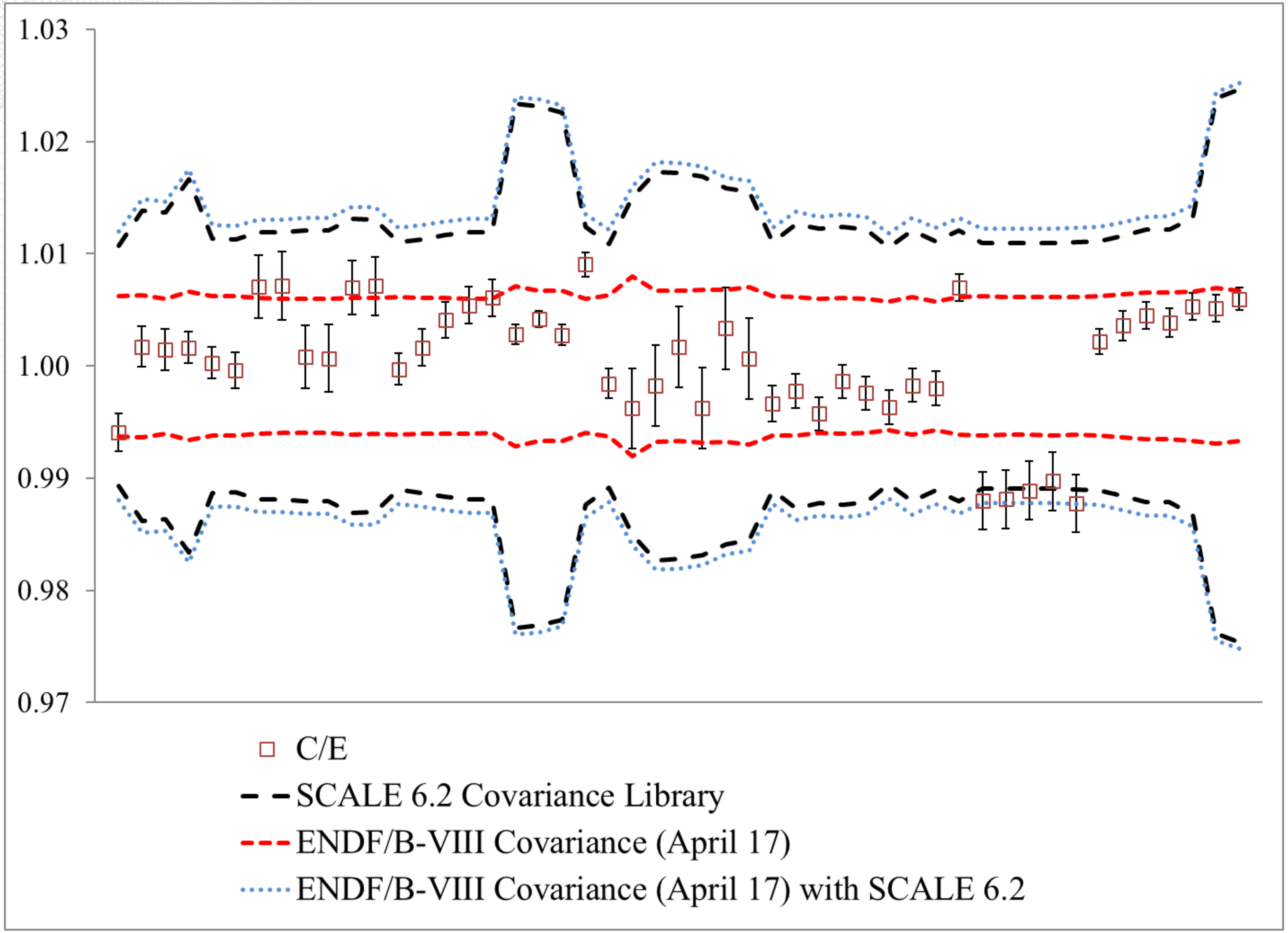
Sequence / Geometry	Experiment class	ICSBEP case numbers	Number of configurations
CSAS5 / KENO V.a	HEU-MET-FAST	15, 16, 17, 18, 19, 20, 21, 25, 30, 38, 40, 65	18
	HEU-SOL-THERM	1, 13, 14, 16, 28, 29, 30	52
	IEU-MET-FAST	2, 3, 4, 5, 6, 7, 8, 9	8
	LEU-COMP-THERM	1, 2, 8, 10, 17, 42, 50, 78, 80	140
	LEU-SOL-THERM	2, 3, 4	19
	MIX-MET-FAST	5, 6	2
	MIX-COMP-THERM	1, 2, 4	21
	MIX-SOL-THERM	2	3
	PU-MET-FAST	1, 2, 5, 6, 8, 10, 18, 22, 23, 24	10
	PU-SOL-THERM	1, 2, 3, 4, 5, 6, 7, 11, 20	81
CSAS6 / KENO-VI	HEU-MET-FAST	5, 8, 9, 10, 11, 13, 24, 80, 86, 92, 93	27
	IEU-MET-FAST	19	2
	MIX-COMP-THERM	8	28

- **Fissile materials**
  - High-enriched uranium (HEU),
  - Intermediate-enriched uranium (IEU)
  - Low-enriched uranium (LEU)
  - Plutonium (Pu)
  - Mixed uranium/plutonium oxides (MOX)
- **Fuel form**
  - Metal (MET),
  - Fissile solution (SOL)
  - Multi-material composition (e.g. fuel pins – COMP)
- **Neutron spectra**
  - Fast
  - Thermal

Cross section uncertainty for critical experiments

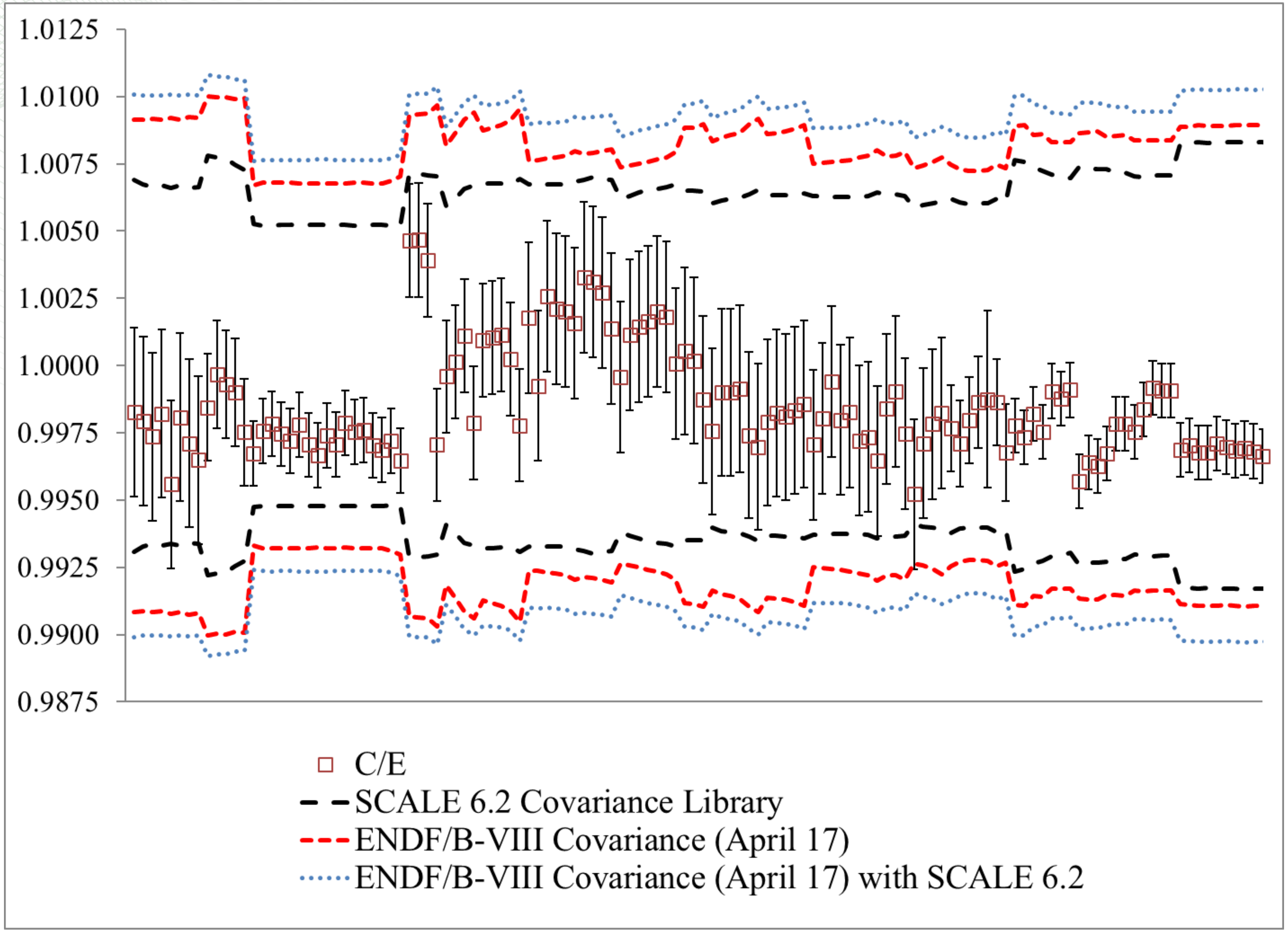
- Purpose:
  - Generate k_{eff} uncertainty due to covariances for critical experiments in VALID library maintained at ORNL
  - Compare variability of predictions with resulting uncertainty band
  - Plots frequently presented by Mark Williams, Brad Rearden, and others
- Methodology:
  - TSUNAMI-IP will calculate k_{eff} uncertainty resulting from covariance data
    - Covariance patching turned off for data testing
    - “uncert” and “values” keywords in parameters block
  - Covariances propagated with sensitivities to determine uncertainty in k_{eff}
  - Detailed uncertainty edit can also be generated for each element in the covariance matrix

Results (1): HEU-MET-FAST



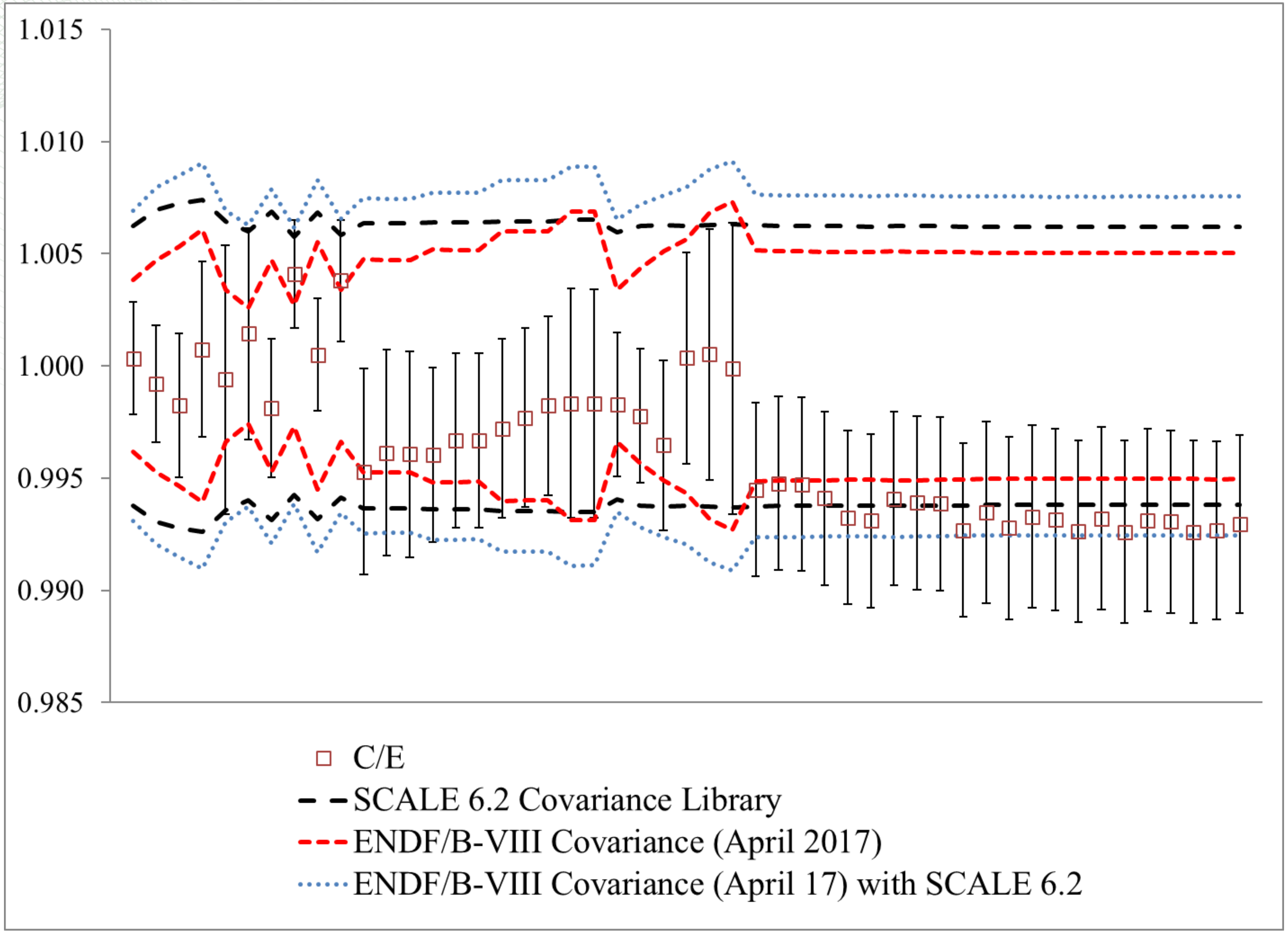
- 49 critical configurations
- C/E plotted for 252-group ENDF/B-VII.1 library
- Error bar is 1σ experimental uncertainty
- X variable is case number: no attempt to find trends
- HMF cases have generally not shown much impact in either ENDF/B-VII.1 or ENDF/B-VIII

Results (2): LEU-COMP-THERM



- 140 critical configurations
- C/E and error bar same as previous plot
- Noticeably larger uncertainty band with ENDF/B-VIII than SCALE 6.2.1
- Traced to ^{235}U nubar and ^{1}H capture and scatter
  - Reintroduction of errors identified in ENDF/B-VII.1 data during preparation and testing of SCALE 6.2 covariance library

Results (3): MIX-COMP-THERM



- 49 critical configurations
- New zigzag in the middle of the data is caused by ^{1}H capture
- Generally fewer points inside the uncertainty band for Pu-fueled systems in SCALE 6.2 and beyond
  - MCT-008 appears biased low, especially Cases 7-28
- Uncertainty may be creeping back – or is it just the ^{1}H problem?

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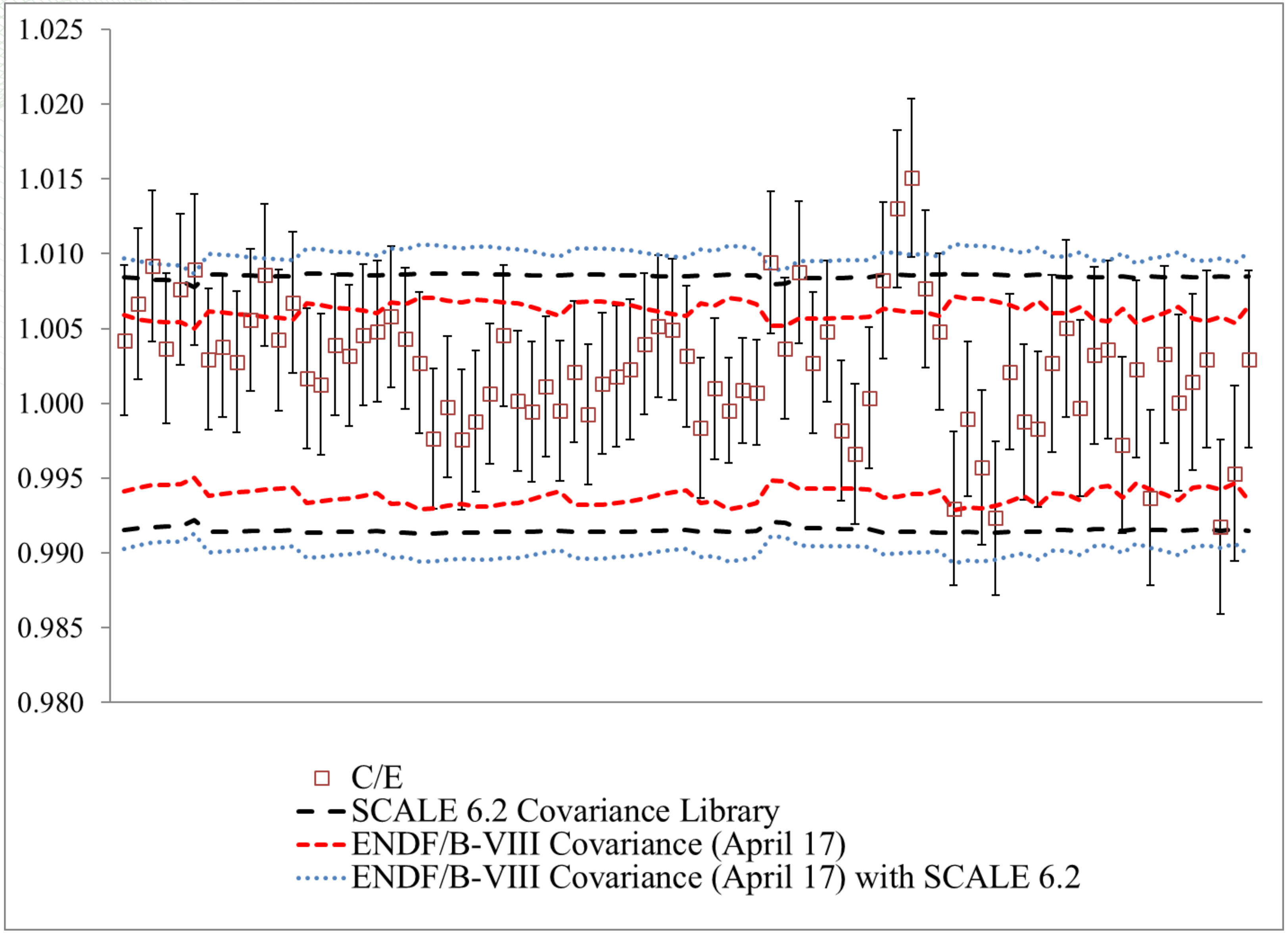
Results (4): PU-SOL-THERM



- 81 critical configurations
- ^{239}Pu covariance data missing, so not tested
  - Obviously would have large impact here
- Again, results looked better with ^{239}Pu nubar from ENDF/B-VII.1
  - Is that unrealistically low?

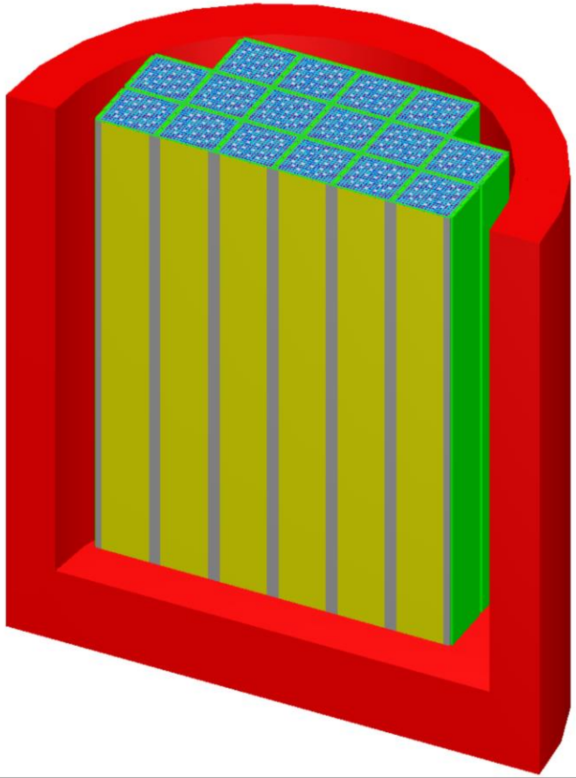
Results (5): Summary Table

Category	Number of Cases	Avg C/E (CE_V7.1)	Avg Exp Unc	St. Dev. Of C/Es	Avg 1σ XS Unc		% of Cases Within	
					SCALE 6.2	E8+SCALE	Exp Unc Band	E8+SCALE XS Band
HMF	49	1.00014	194	477	1366	1474	24.5	100
HST	52	0.99802	494	588	1050	1288	48.1	96.2
IMF	13	1.00329	269	367	1528	1591	23.1	100
LCT	140	0.99956	195	167	677	934	45.7	100
LST	19	0.99866	318	266	716	1180	63.2	100
MCT	49	0.99649	400	337	633	768	46.9	100
PMF	10	1.00020	204	128	586	584	80.0	100
PST	81	1.00302	497	420	850	995	72.8	95.1

The cross section uncertainty band is TOO WIDE! The experimental uncertainty band is in generally good agreement with the variation seen in the critical experiments.

Criticality Analysis for Used Nuclear Fuel (UNF)

- UNF contains many actinides and fission products
- A limiting condition on UNF cask storage is sub-criticality margin
- NRC allows "burnup credit" for burned fuel and certain FP nuclides
- Uncertainties are important consideration
  - SCALE 6.1 used in S/U for U.S. NRC



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NRC Interim Staff Guidance 8, Revision 3

Division of Spent Fuel Storage and Transportation
Interim Staff Guidance - 8
Revision 3

Issue: Burnup Credit in the Criticality Safety Analyses of PWR Spent Fuel in Transportation and Storage Casks

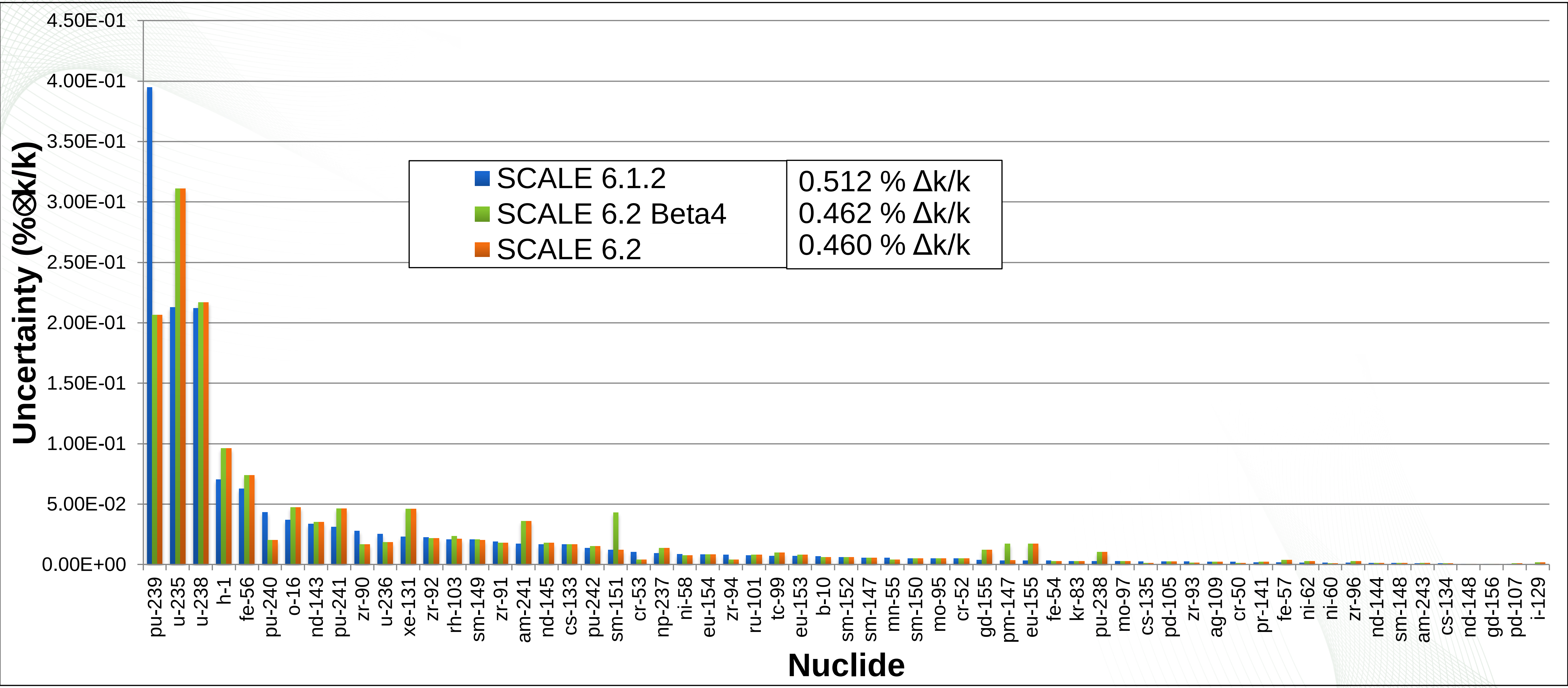
Introduction:

Title 10 of the Code of Federal Regulations (10 CFR) Part 71, *Packaging and Transportation of Radioactive Material*,[1] and 10 CFR Part 72, *Licensing Requirements for the Independent Storage of Spent Nuclear Fuel, High-Level Radioactive Waste, and Reactor-Related Greater Than Class C Waste*,[2] require that spent nuclear fuel (SNF) remain subcritical in transportation and storage, respectively. Unirradiated reactor fuel has a well-specified nuclide composition that provides a straightforward and bounding approach to the criticality safety analysis of transportation and storage systems. As the fuel is irradiated in the reactor, the nuclide composition changes and, ignoring the presence of burnable poisons, this composition change will cause the reactivity of the fuel to decrease. Allowance in the criticality safety analysis for the decrease in fuel reactivity resulting from irradiation is termed burnup credit. Extensive investigations have been performed both within the United States and by other countries in an effort to understand and document the technical issues related to the use of burnup credit.

This Interim Staff Guidance (ISG) provides recommendations to the staff for accepting, on a design-specific basis, a burnup credit approach in the criticality safety analysis of pressurized water reactor (PWR) SNF storage and transportation systems. This revision to ISG-8 incorporates the results of burnup credit-related research that has been conducted since Revision 2 (Rev. 2) was published in September 2002. Based on the detailed results of this research and the technical judgment of the U.S. Nuclear Regulatory Commission (NRC) staff, ISG-8, Rev. 3, includes two major changes in the recommendations to staff reviewing burnup credit applications for SNF transportation and storage systems: (1) optional credit for fission product and minor actinide neutron absorbing isotopes in the SNF composition, and (2) misload analyses and additional administrative procedures in lieu of a burnup measurement at the time of loading. This ISG revision also includes an increase in the maximum assembly average burnup recommended for burnup credit.

"In order to achieve an **appropriate estimate of the *keff* bias and bias uncertainty due to fission products**, ORNL developed a methodology based on the SCALE *Tools for Sensitivity and Uncertainty Methodology Implementation* (TSUNAMI) code, developed as part of the SCALE code system. **This methodology uses the nuclear data uncertainty estimated for each fission product cross section known as the cross section covariance data. These data are provided with the ENDF/B-VII cross section library.** The TSUNAMI code is used to propagate the cross section uncertainties represented by the covariance data into *keff* uncertainties for each fission product isotope used in a particular application. The theoretical basis of this validation technique is that computational biases are primarily caused by errors in the cross section data, which are quantified and bounded, with a 1σ confidence, by the cross section covariance data. The validity of this theoretical basis is discussed in greater detail in NUREG/CR-7109."

Top Contributors to Uncertainty for GBC-32



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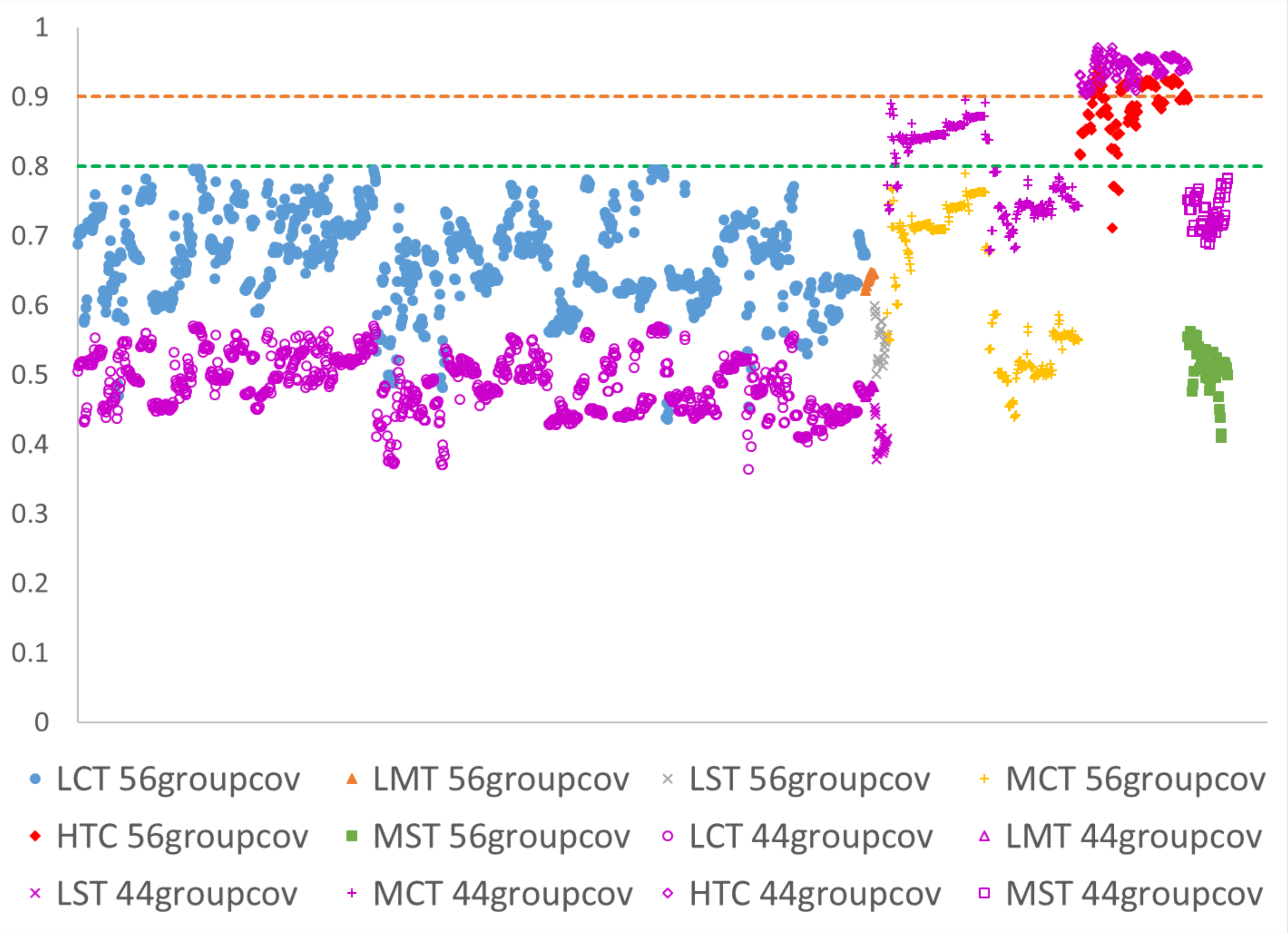
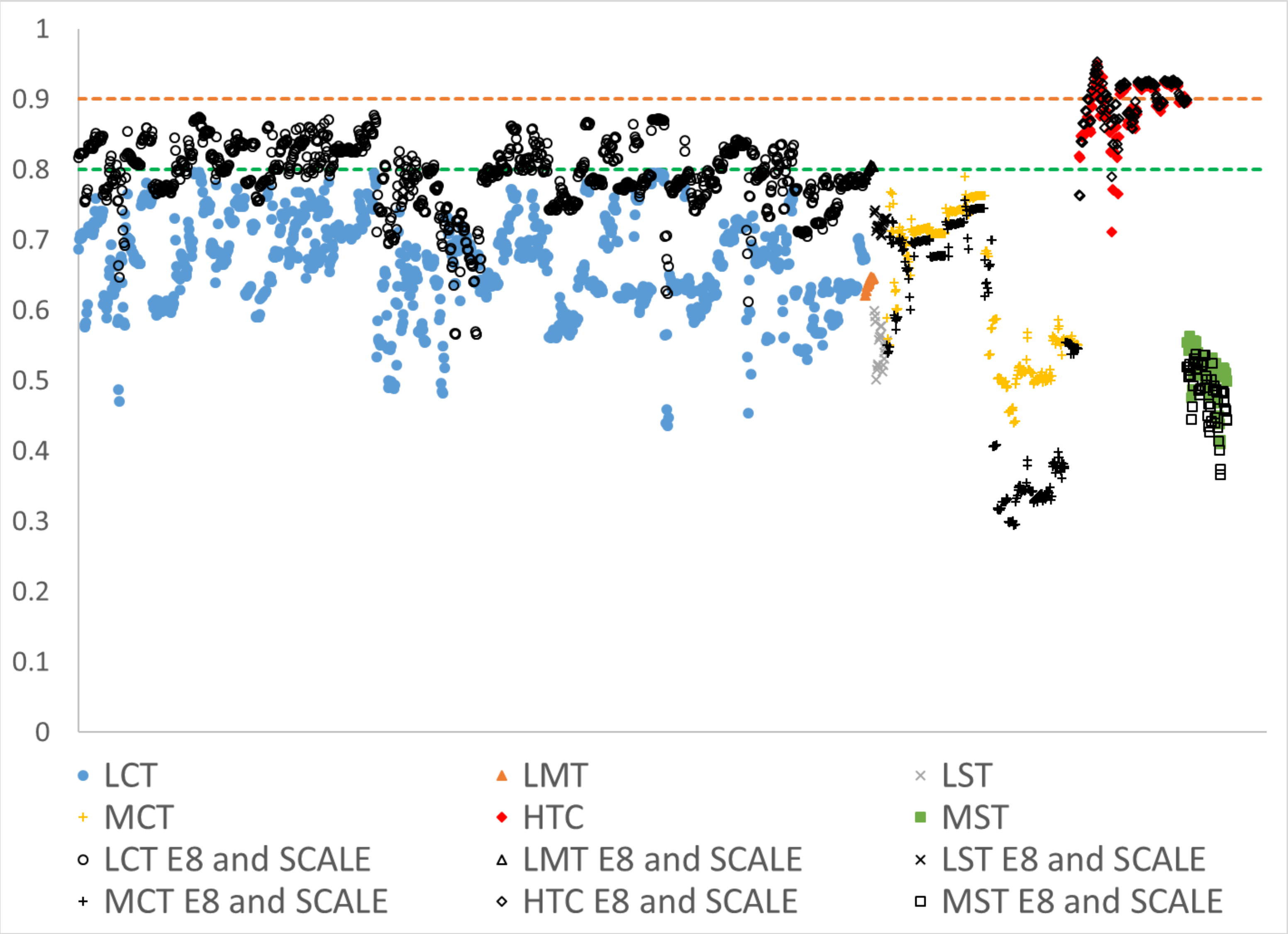
c_k results –SCALE 6.1 to SCALE 6.2



- 1643 unique critical experiments
- PWR SNF cask with fuel at representative discharge burnup
- SCALE 6.1 (purple)
- SCALE 6.2 (various)
- This change caused significant turmoil for use of c_k to select similar experiments for validation.

c_k results – SCALE 6.2 & ENDF/B-VIII



- Same critical experiments and PWR SNF
- SCALE 6.2 (various)
- ENDF/B-VIII plus SCALE data (black)
- This change further reduces MCT systems and increases LCT systems. The result doesn't make sense – LEU can't be representative of SNF.

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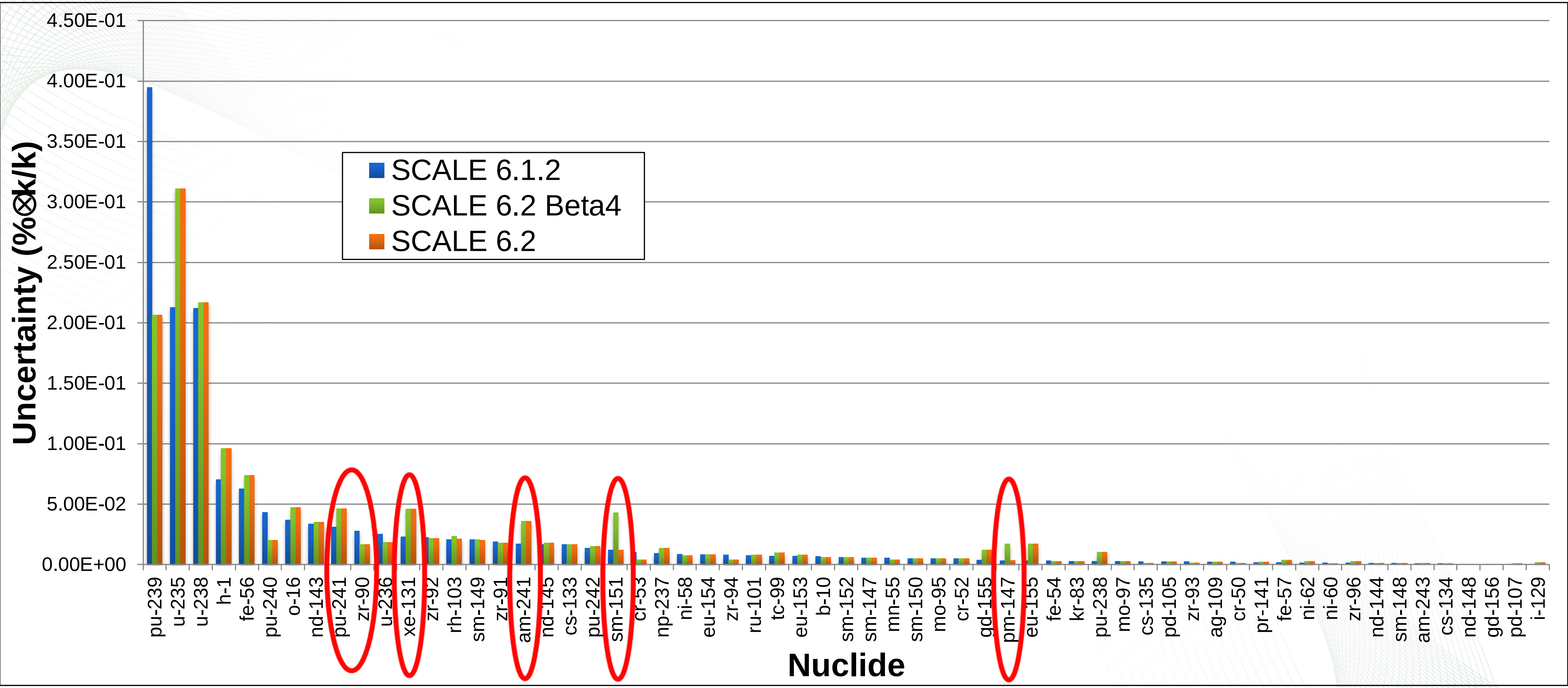
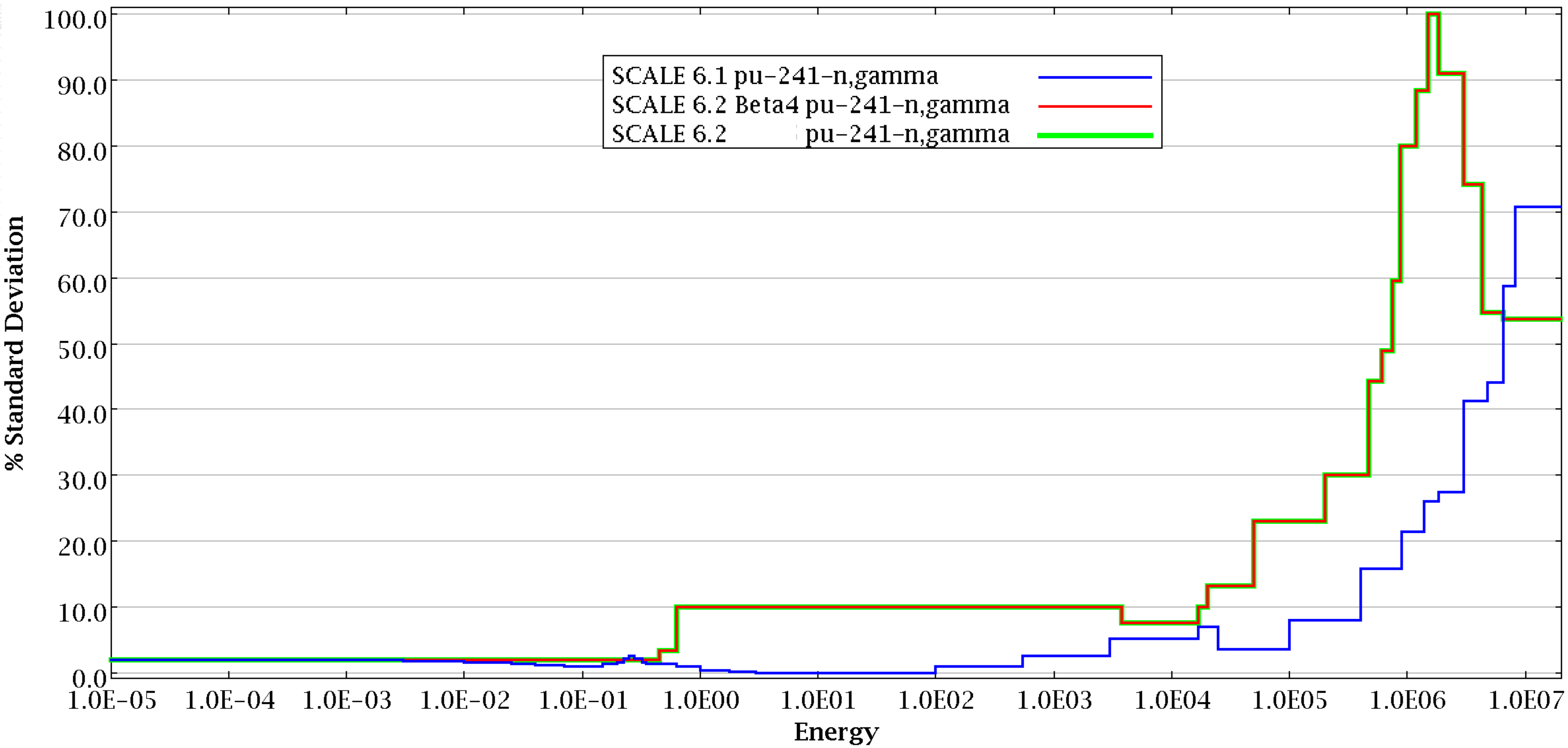
c_k summary

- Increased ^{235}U nubar uncertainty results in higher similarity for LCT systems with SNF storage/transportation models
  - This exacerbates a change resulting from ENDF/B-VII.1 covariance changes to ^{235}U (bigger) and ^{239}Pu (smaller)
- Without reliable covariance data, S/U methods cannot be used to select appropriate experiments for validation
  - Reliability of covariance data is also a significant problem for data adjustment methods (e.g., TSURFER)

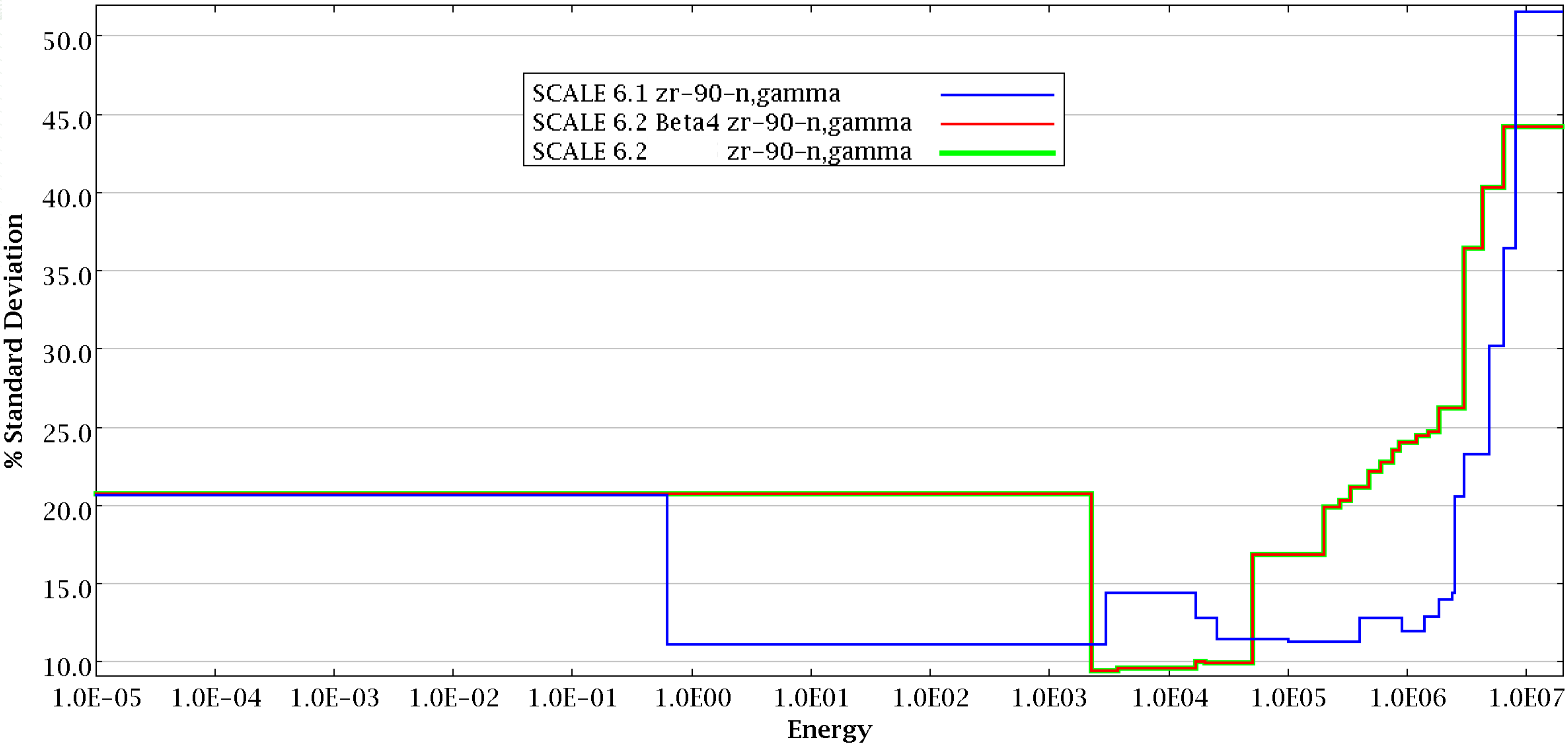
More Contributors to Uncertainty for GBC-32



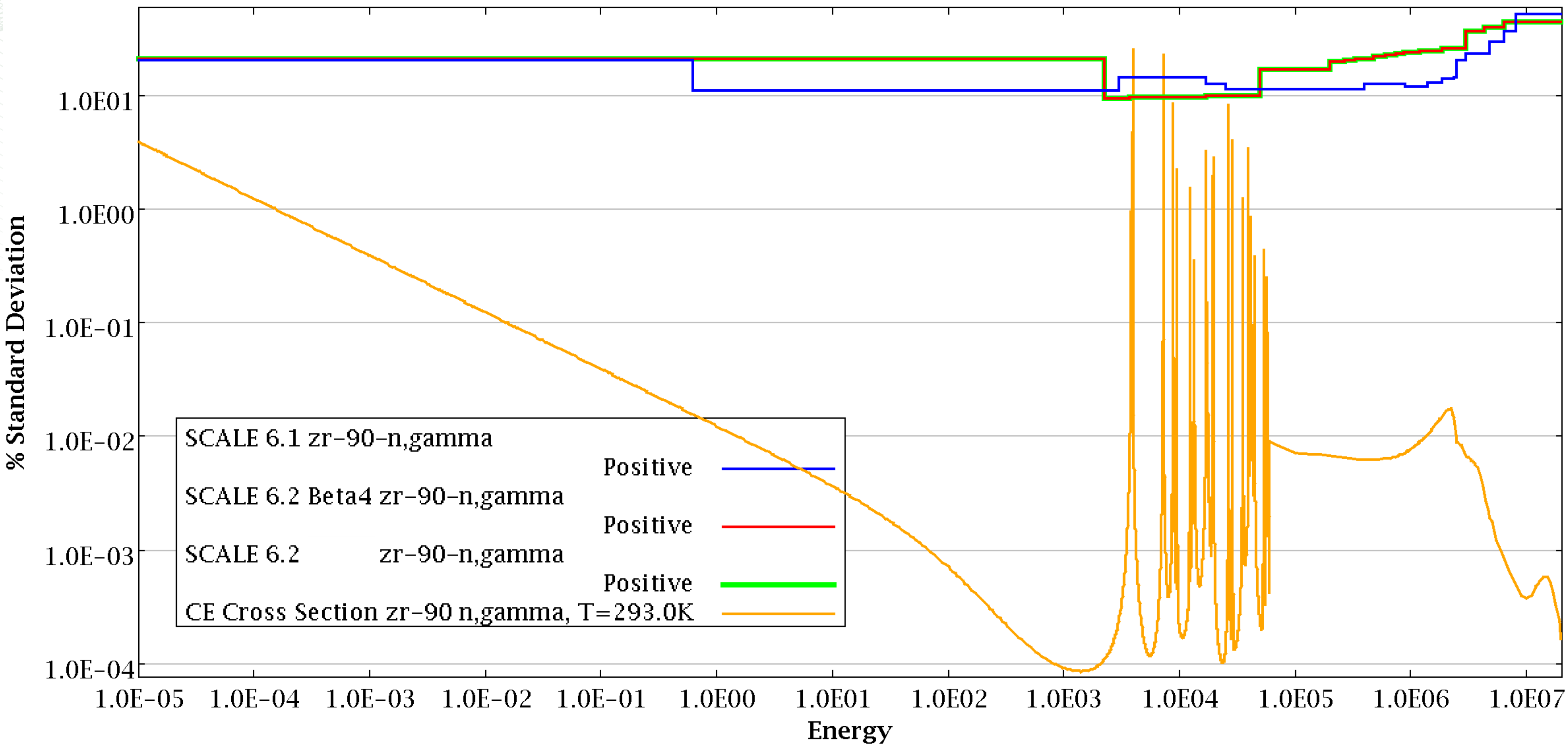
Pu-241 n,gamma uncertainty



Zr-90 n,gamma uncertainty

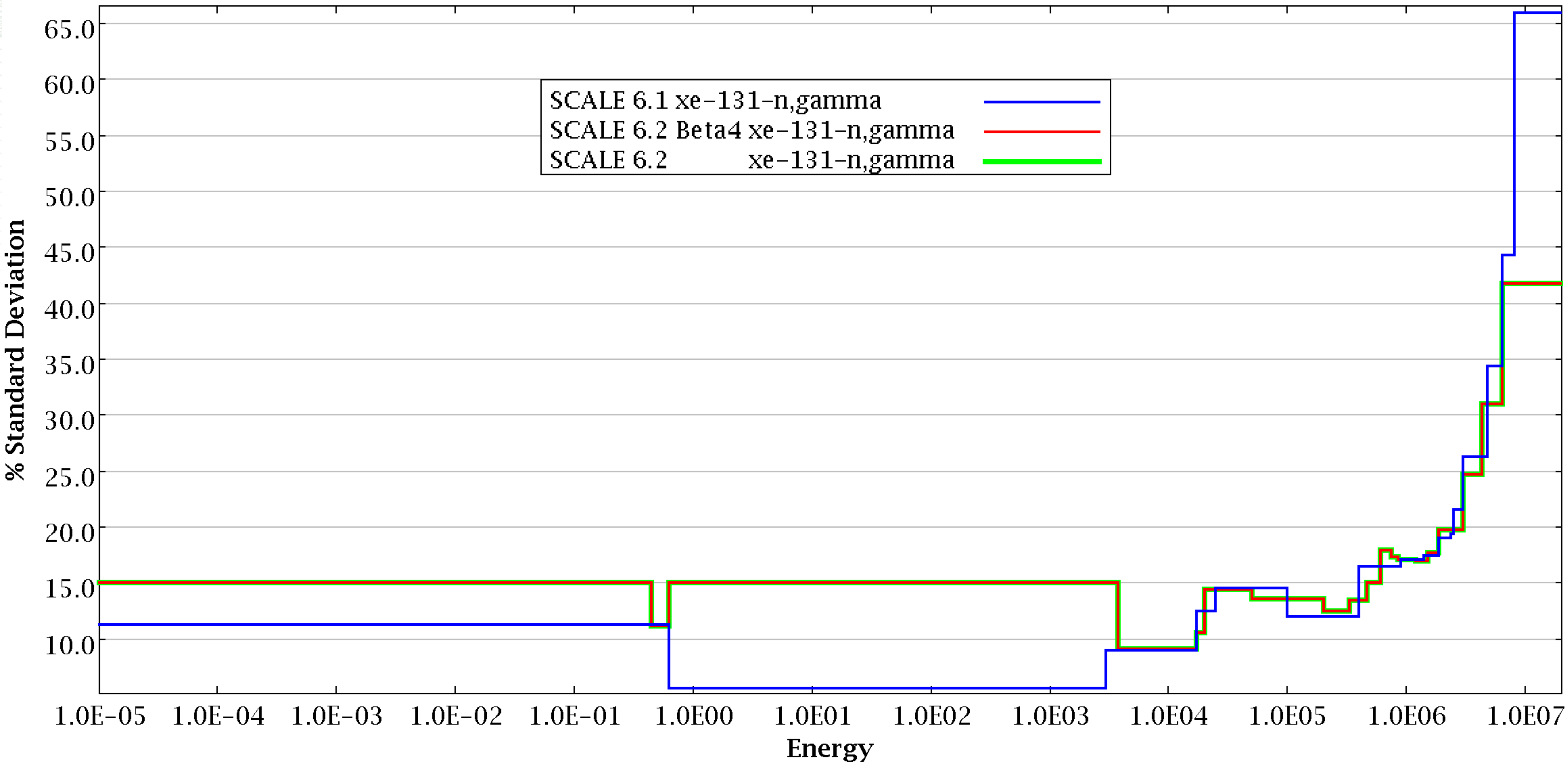


Zr-90

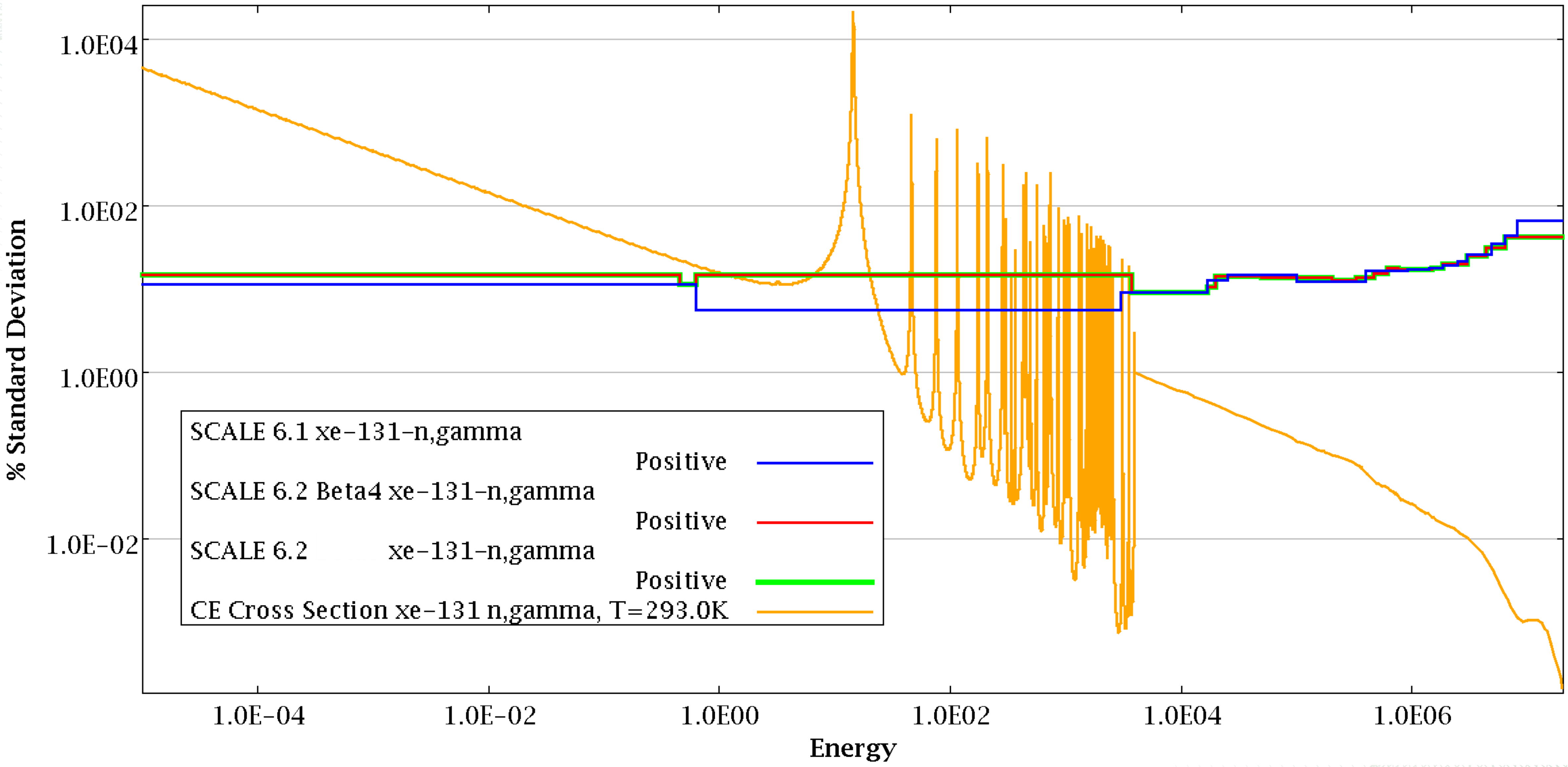


% Standard Deviation
1.0E01
1.0E00
1.0E-01
1.0E-02
1.0E-03
1.0E-04
1.0E-05
1.0E-04
1.0E-03
1.0E-02
1.0E-01
1.0E00
1.0E01
1.0E02
1.0E03
1.0E04
1.0E05
1.0E06
1.0E07
Energy
SCALE 6.1 zr-90-n,gamma
Positive
SCALE 6.2 Beta4 zr-90-n,gamma
Positive
SCALE 6.2 zr-90-n,gamma
Positive
CE Cross Section zr-90 n,gamma, T=293.0K

Xe-131 n,gamma uncertainty



Xe-131



% Standard Deviation
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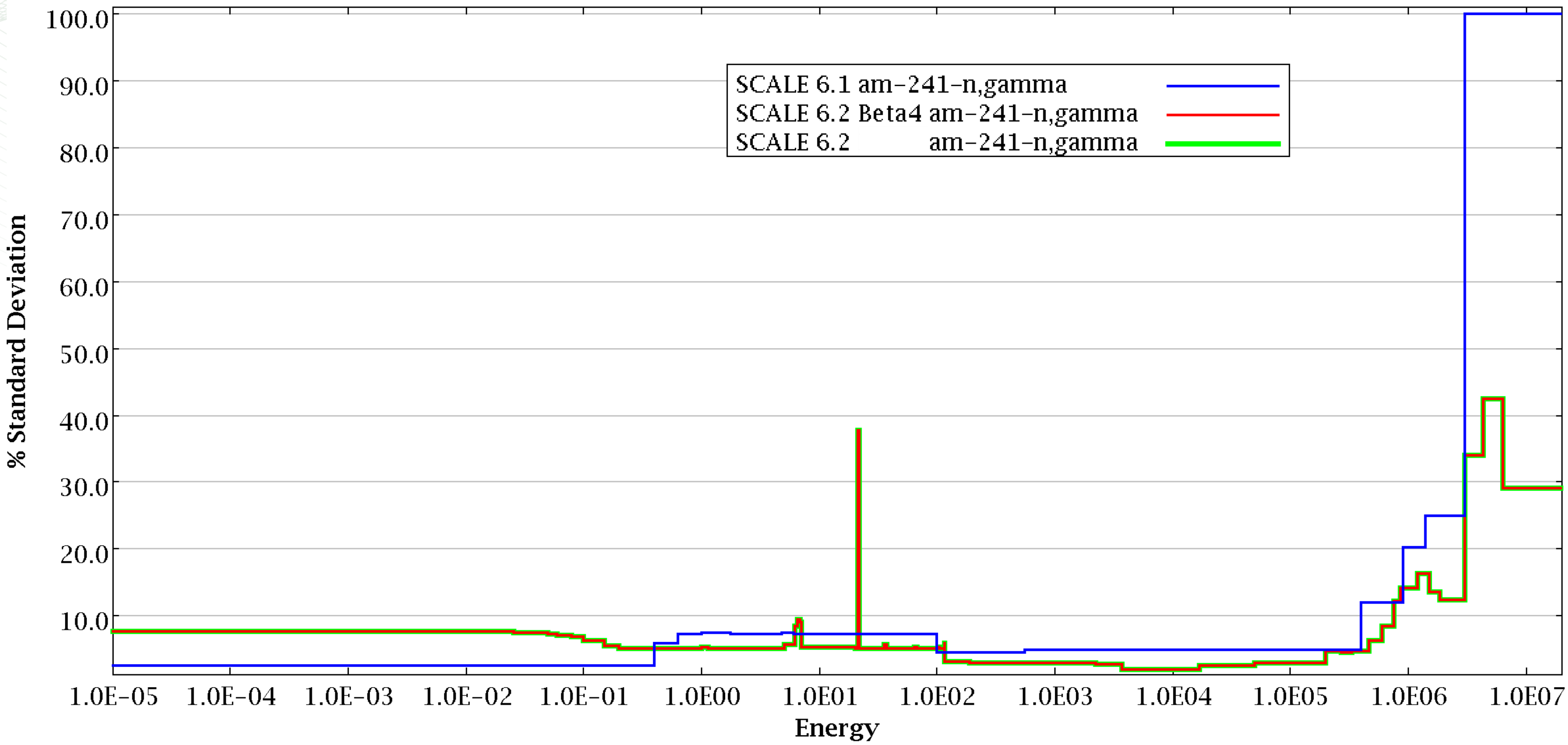
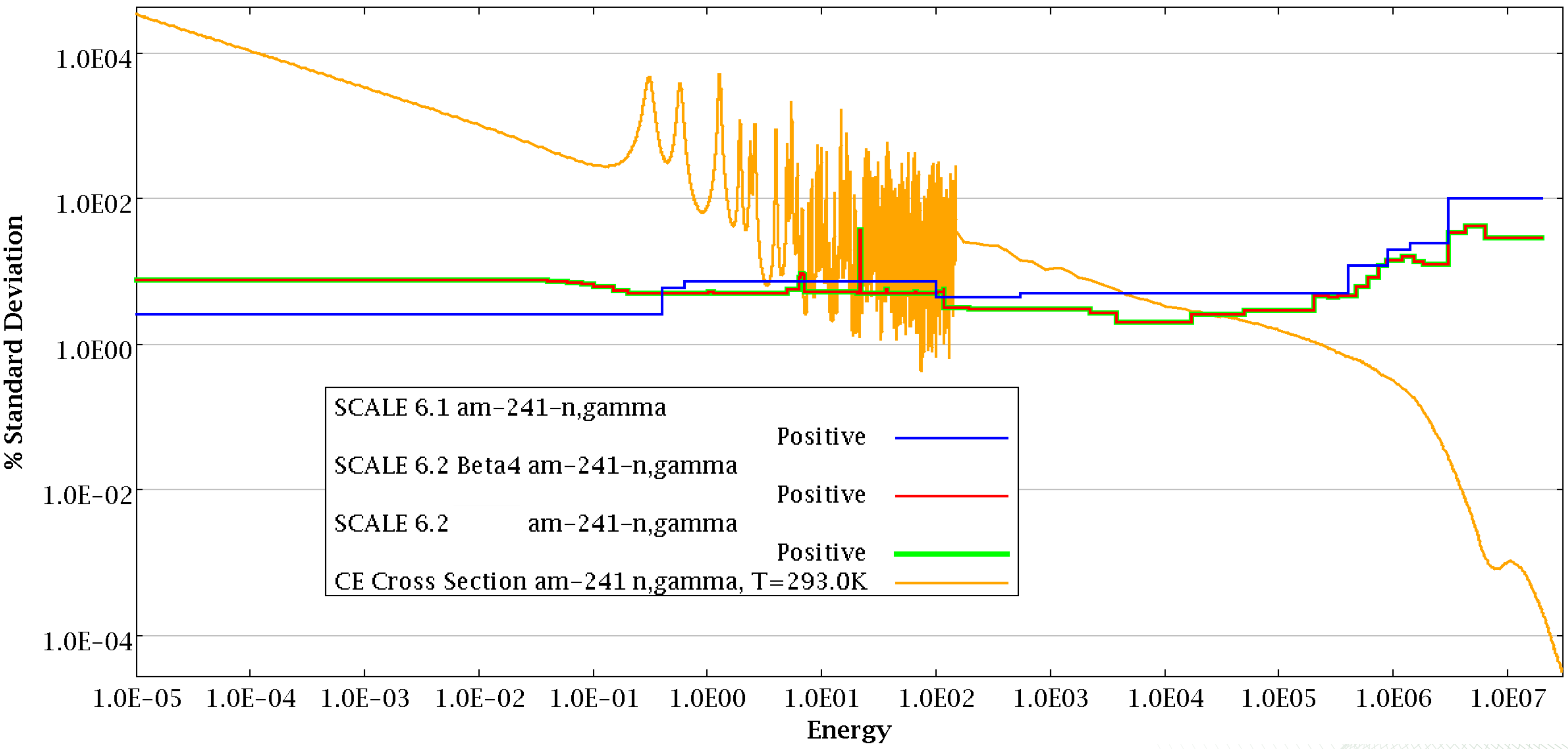
1.0E-02
1.0E-04
1.0E-02
1.0E00
1.0E02
1.0E04
1.0E06
Energy
SCALE 6.1 xe-131-n,gamma
Positive
SCALE 6.2 Beta4 xe-131-n,gamma
Positive
SCALE 6.2 xe-131-n,gamma
Positive
CE Cross Section xe-131 n,gamma, T=293.0K

Am-241 n,gamma uncertainty



Am-241



% Standard Deviation
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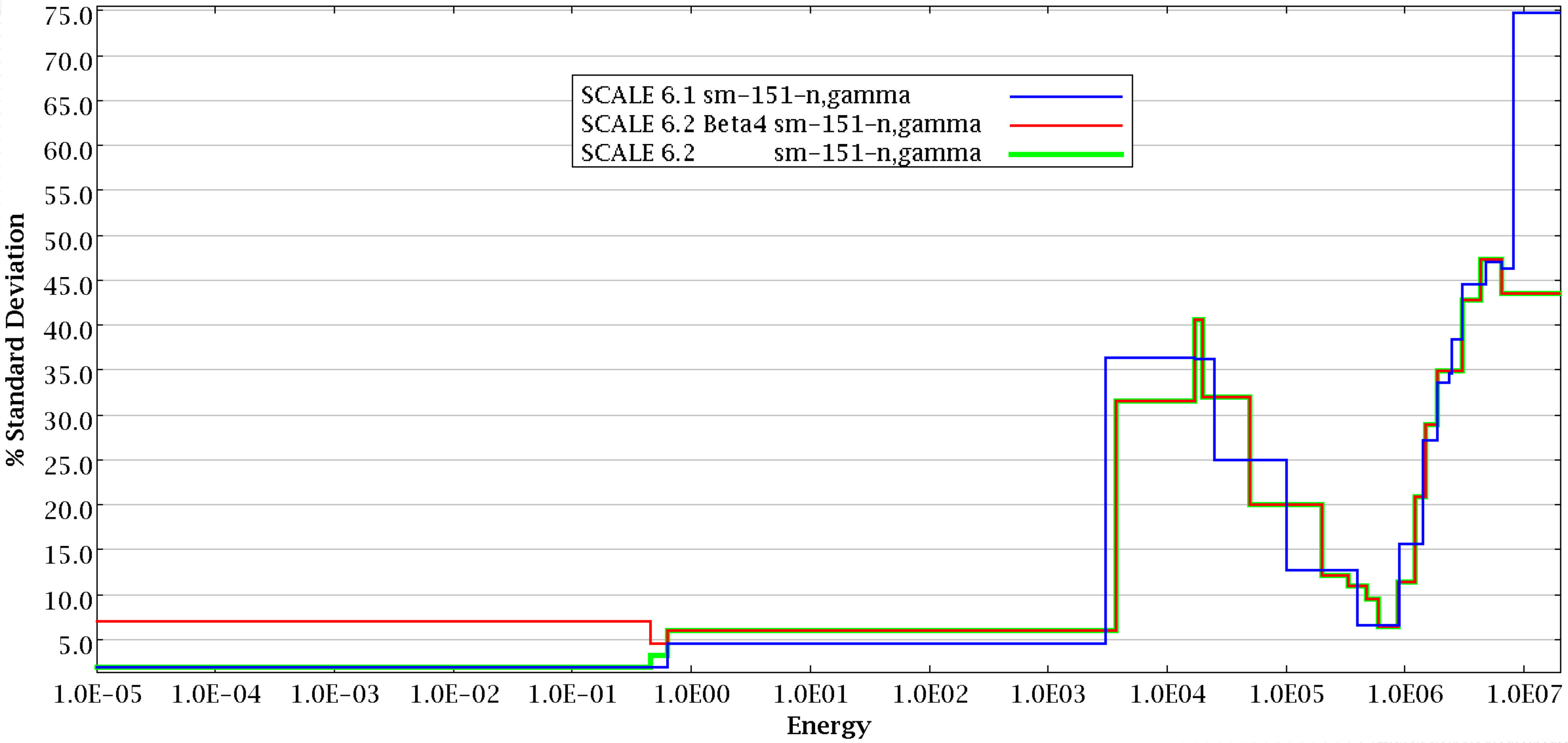
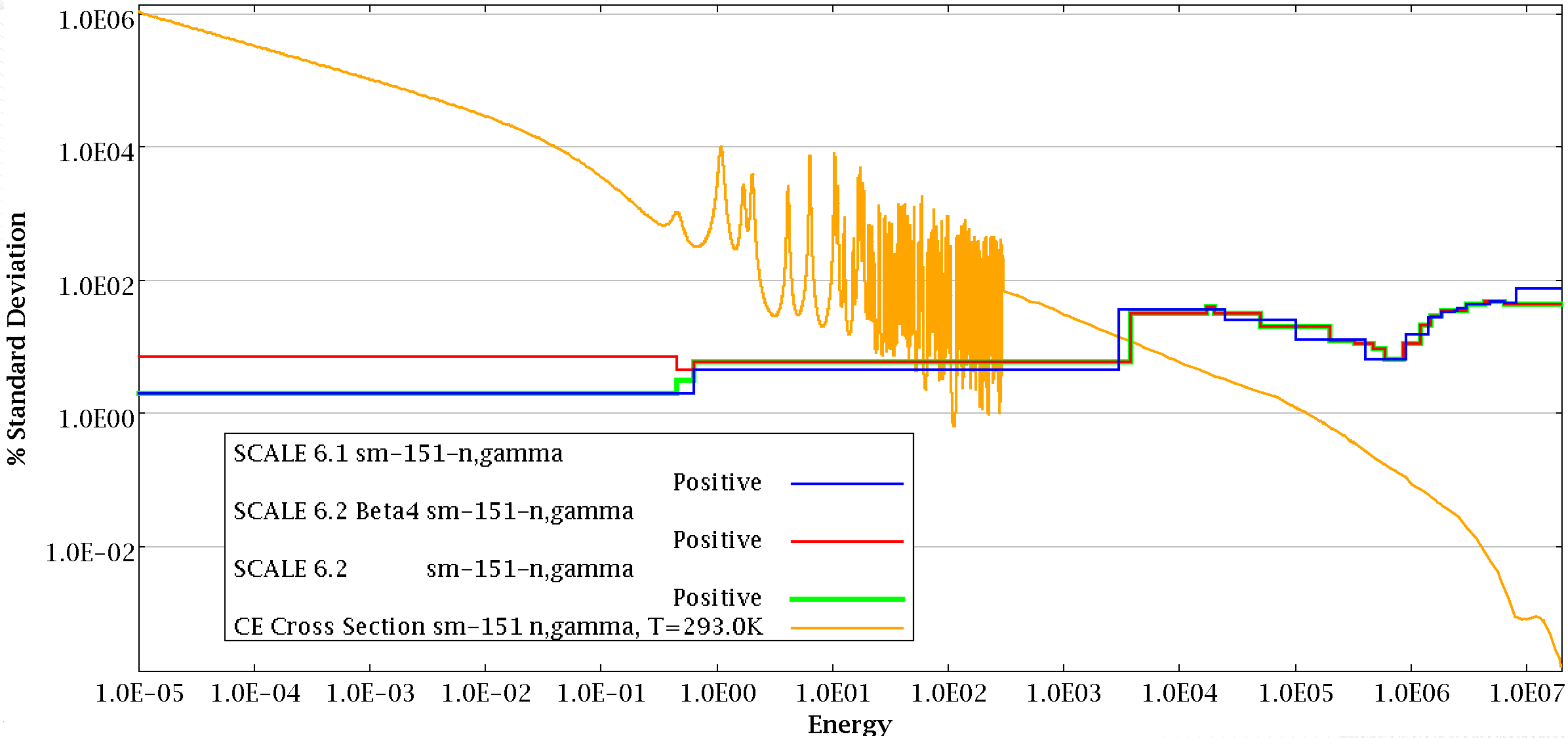
1.0E00
1.0E-02
1.0E-04
1.0E-05
1.0E-04
1.0E-03
1.0E-02
1.0E-01
1.0E00
1.0E01
1.0E02
1.0E03
1.0E04
1.0E05
1.0E06
1.0E07
Energy
SCALE 6.1 am-241-n,gamma
Positive
SCALE 6.2 Beta4 am-241-n,gamma
Positive
SCALE 6.2 am-241-n,gamma
Positive
CE Cross Section am-241 n,gamma, T=293.0K

Sm-151 n,gamma uncertainty

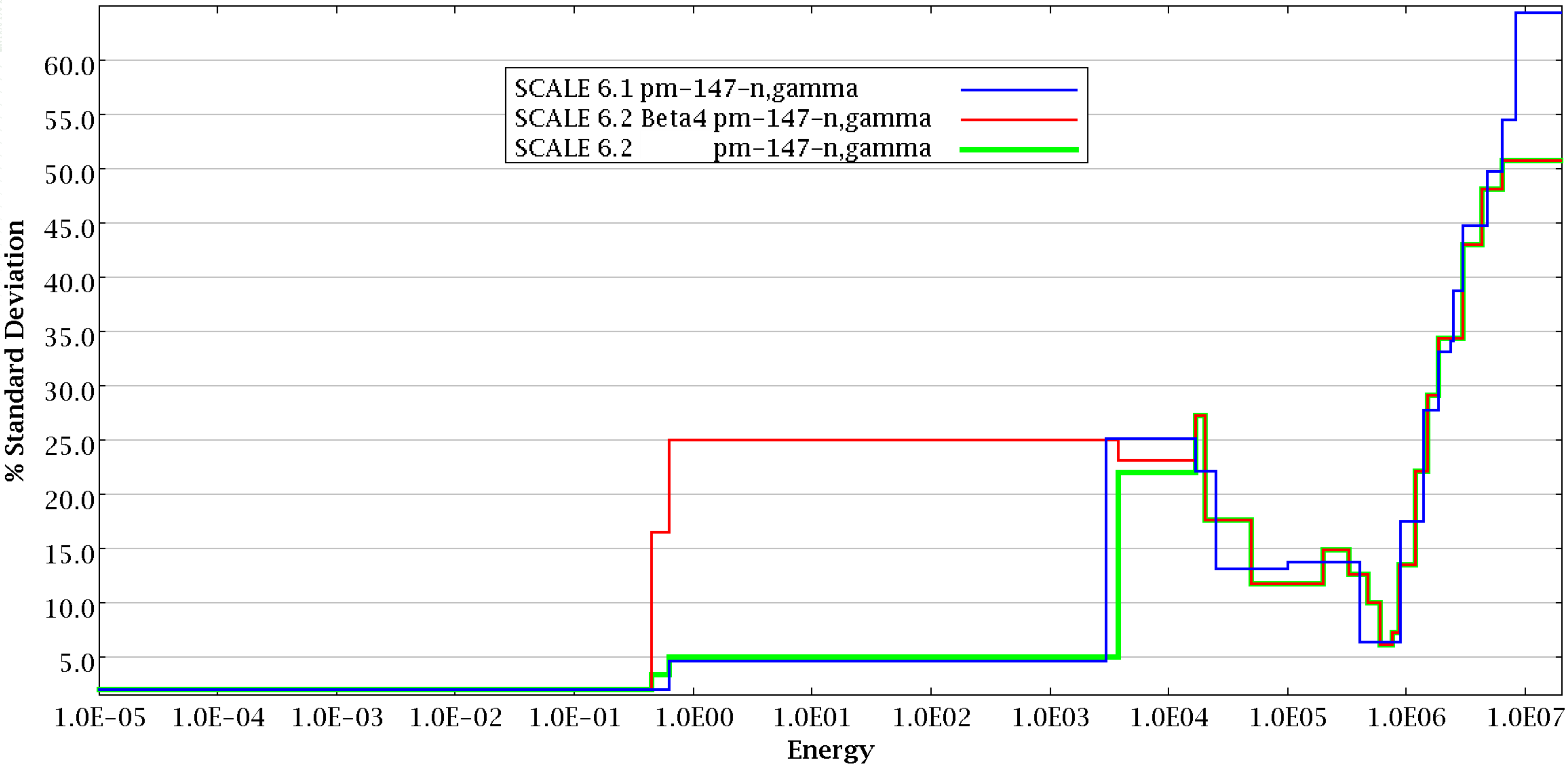


Sm-151

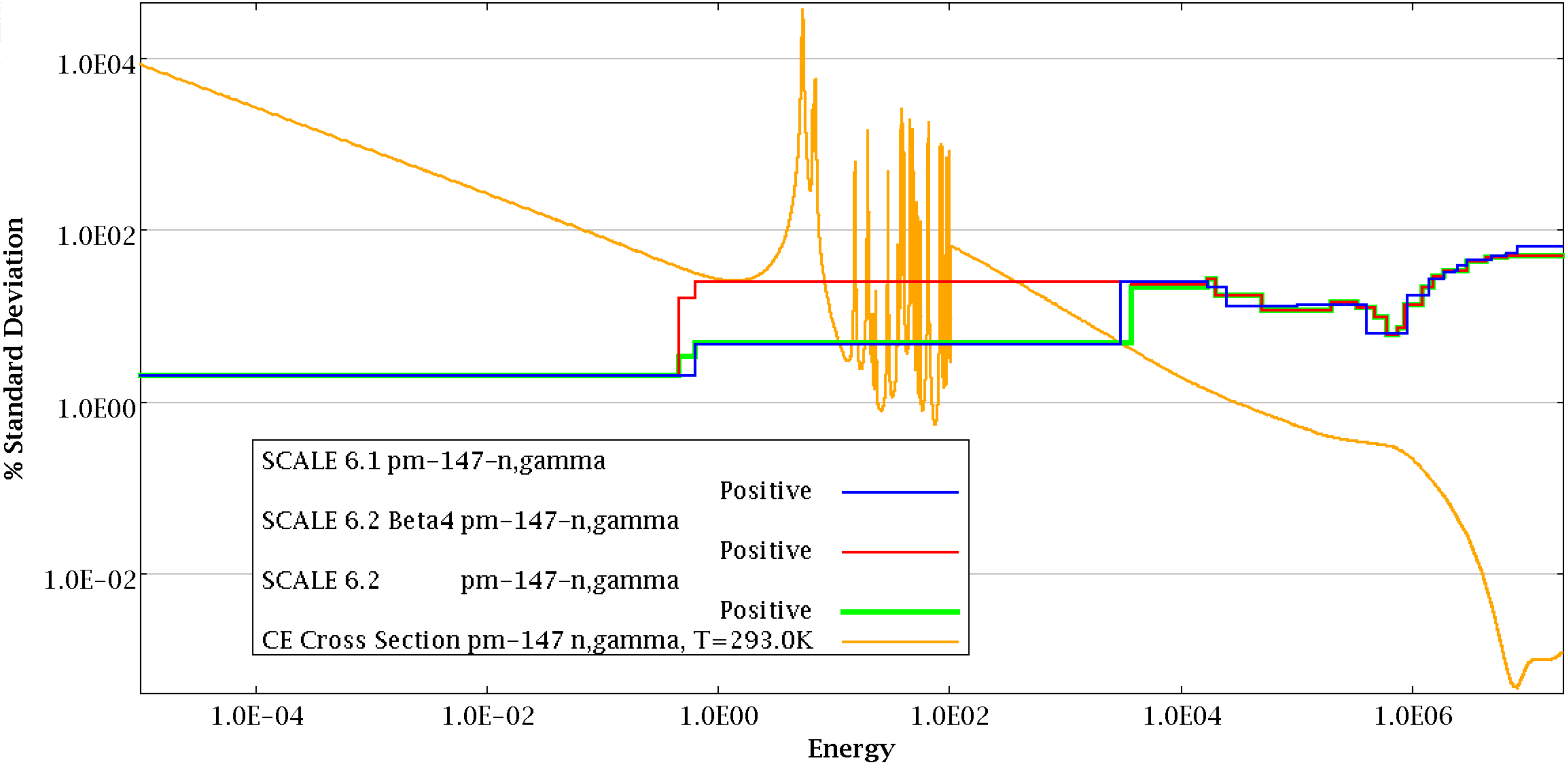


1.0E06
1.0E04
1.0E02
1.0E00
1.0E−02
% Standard Deviation
1.0E−05
1.0E−04
1.0E−03
1.0E−02
1.0E−01
1.0E00
1.0E01
1.0E02
1.0E03
1.0E04
1.0E05
1.0E06
1.0E07
Energy
SCALE 6.1 sm−151−n,gamma
Positive
SCALE 6.2 Beta4 sm−151−n,gamma
Positive
SCALE 6.2 sm−151−n,gamma
Positive
CE Cross Section sm−151 n,gamma, T=293.0K

Pm-147 n,gamma uncertainty



Pm-147

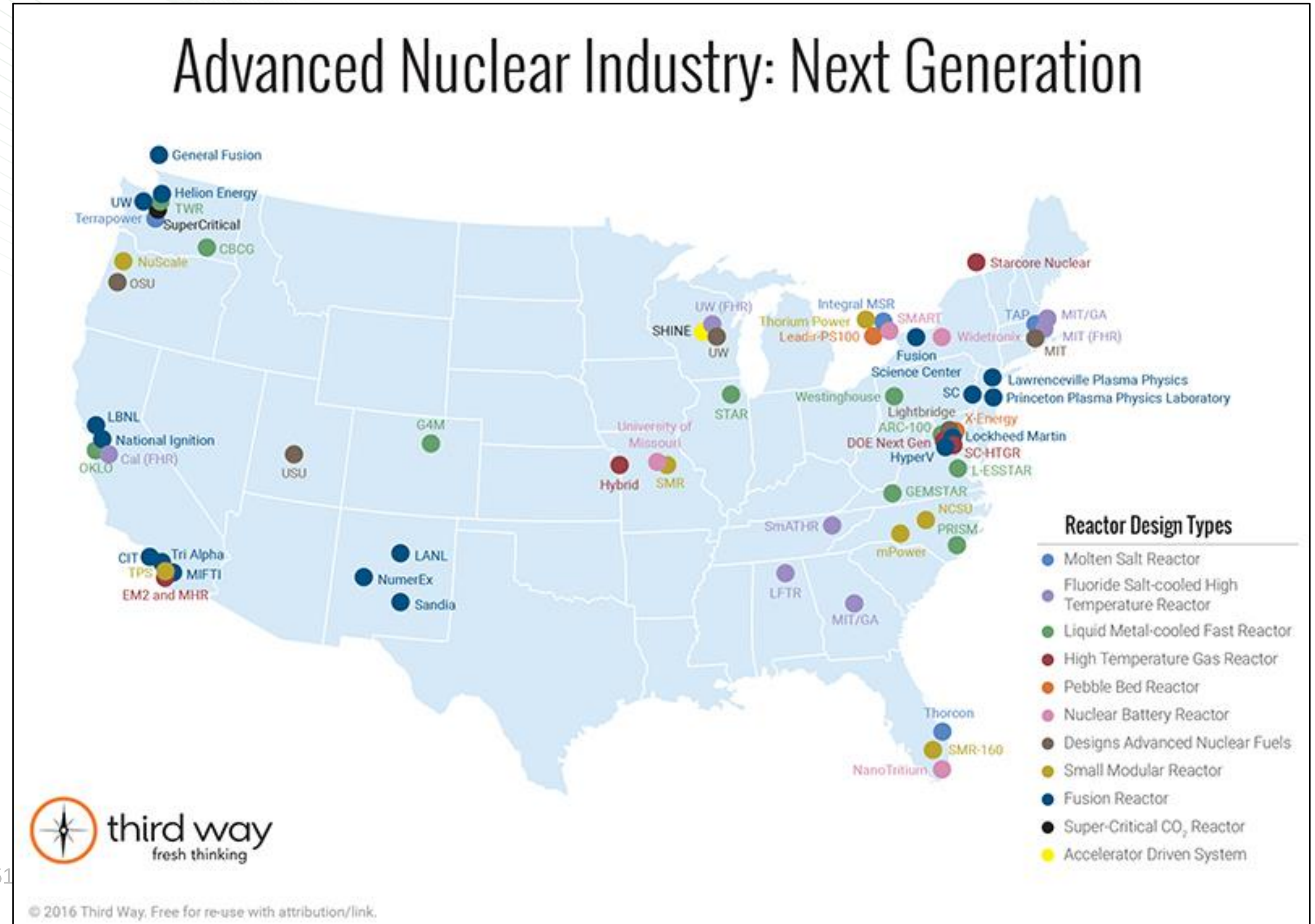


% Standard Deviation
1.0E04
1.0E02
1.0E00
1.0E-02
1.0E-04
1.0E-02
1.0E00
1.0E02
1.0E04
1.0E06
Energy
SCALE 6.1 pm-147-n,gamma
Positive
SCALE 6.2 Beta4 pm-147-n,gamma
Positive
SCALE 6.2 pm-147-n,gamma
Positive
CE Cross Section pm-147 n,gamma, T=293.0K

Uncertainties for Spent Fuel Systems (Same cases & models as NUREG/CR-7109 Table 6.10)

	SFP				GBC32				BWR	
	10 GWd/MTU		40 GWd/MTU		10 GWd/MTU		40 GWd/MTU		11 GWd/MTU	
Category	SCALE 6.1	SCALE 6.2	SCALE 6.1	SCALE 6.2	SCALE 6.1	SCALE 6.2	SCALE 6.1	SCALE 6.2	SCALE 6.1	SCALE 6.2
Major Actinides	0.00463	0.00469	0.00476	0.00416	0.00455	0.00449	0.00466	0.00406	0.00393	0.00447
Minor Actinides	0.00007	0.00007	0.00027	0.00022	0.00007	0.00007	0.00025	0.00022	0.00013	0.00011
Major Fission Products	0.00022	0.00022	0.00052	0.00054	0.00024	0.00024	0.00052	0.00055	0.00023	0.00024
Other Actinides	0.00003	0.00006	0.00003	0.00006	0.00000	0.00000	0.00001	0.00001	0.00000	0.00000
Other Fission Products	0.00015	0.00021	0.00034	0.00062	0.00008	0.00015	0.00024	0.00047	0.00014	0.00025
Other (structural)	0.00081	0.00123	0.00073	0.00102	0.00106	0.00145	0.00104	0.00127	0.00080	0.00099
ALL	0.00471	0.00486	0.00486	0.00437	0.00468	0.00472	0.00481	0.00432	0.00402	0.00459

A number of private U.S. companies are pursuing conceptual and technological development of advanced reactors



AREVA

GE

HITACHI

Westinghouse

X-energy

TerraPower

TERRESTRIAL ENERGY

Kairos Power

TRANSATOMIC

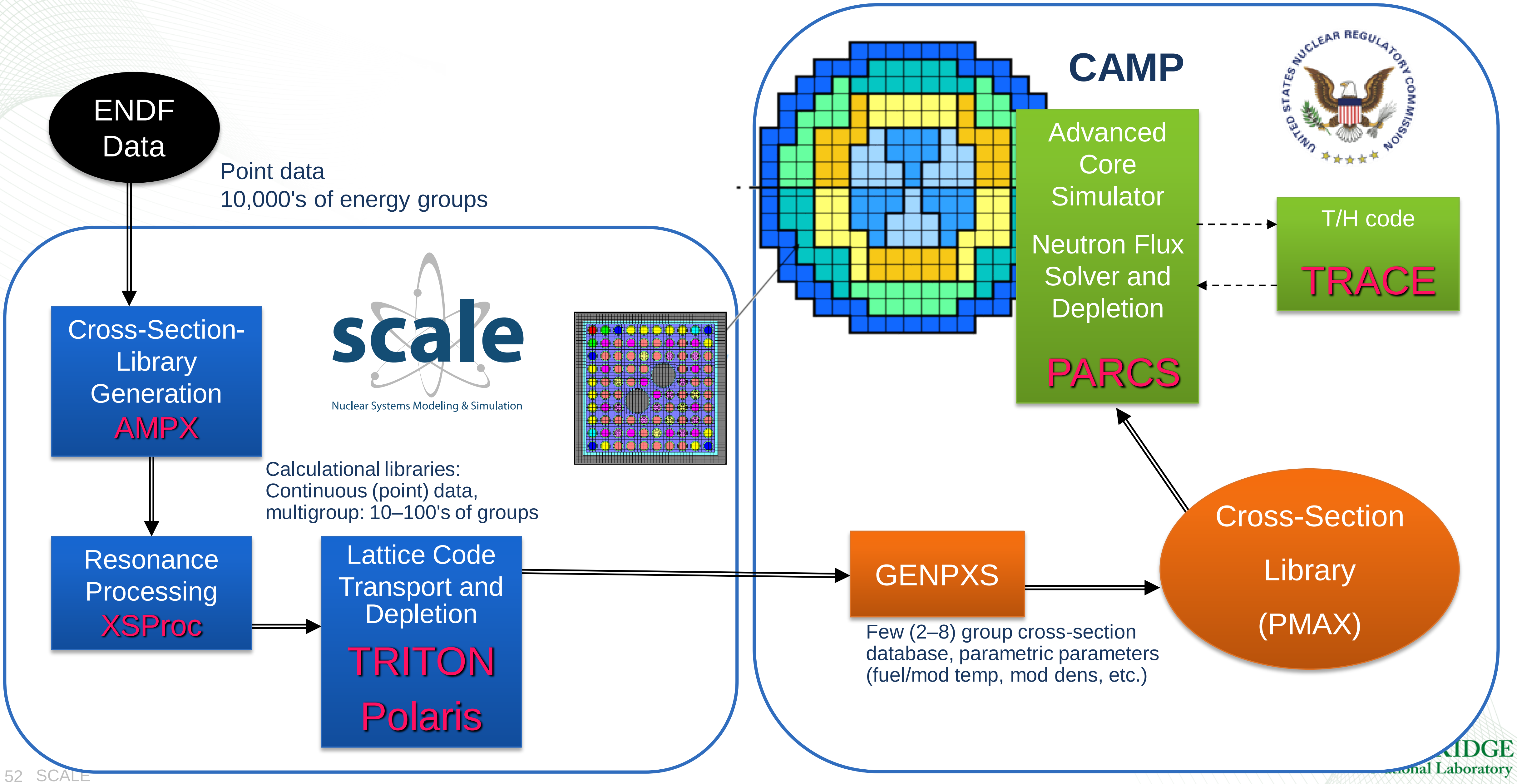
ARC
ADVANCED REACTOR CONCEPTS, LLC

GENERAL ATOMICS
AND AFFILIATED COMPANIES

SOUTHERN NUCLEAR
A SOUTHERN COMPANY

OAK RIDGE
National Laboratory

SCALE is a part of NRC's reactor licensing path

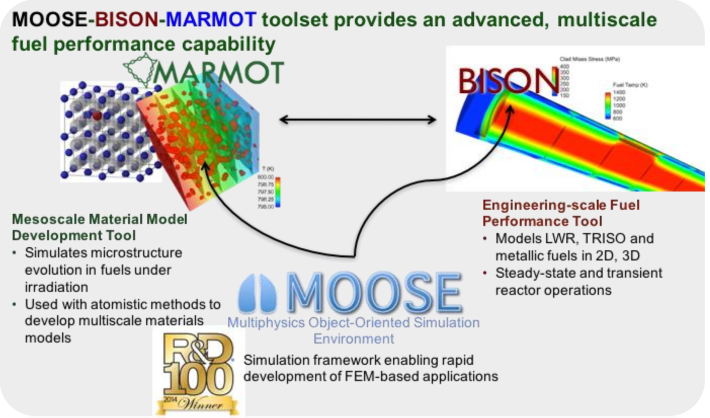


NEAMS (Nuclear Energy Advanced Modeling and Simulation) Program

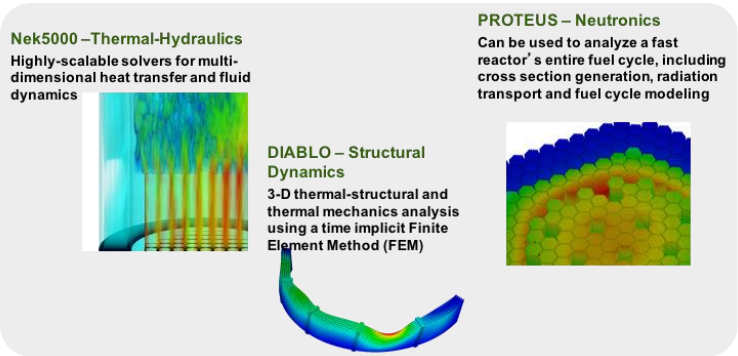
U.S. DEPARTMENT OF ENERGY
Nuclear Energy

Aim: Develop, apply, deploy, and support a predictive modeling and simulation toolkit for the design and analysis of current and future nuclear energy systems using computing architectures from laptops to leadership class facilities.

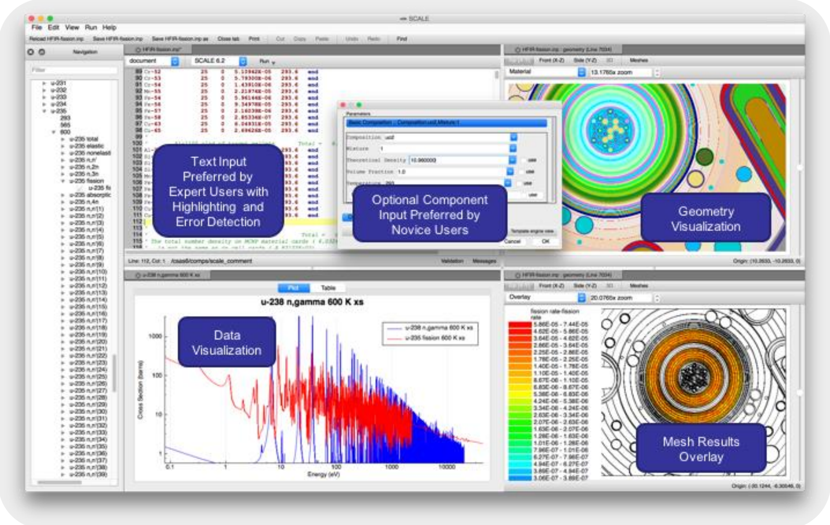
Fuels Product Line



Reactor Product Line



Integration Product Line



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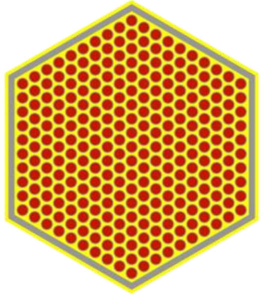
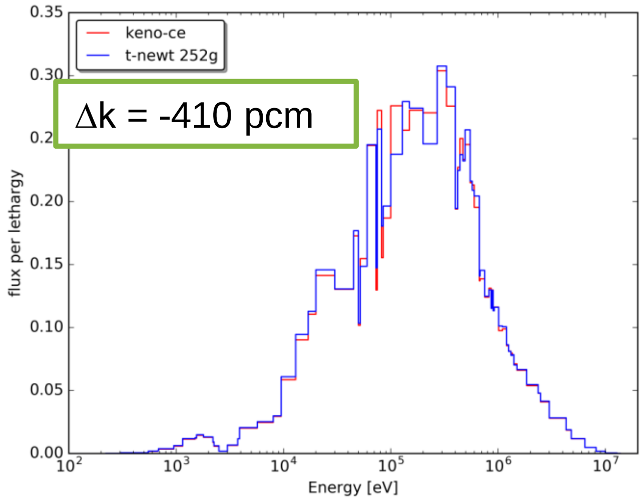
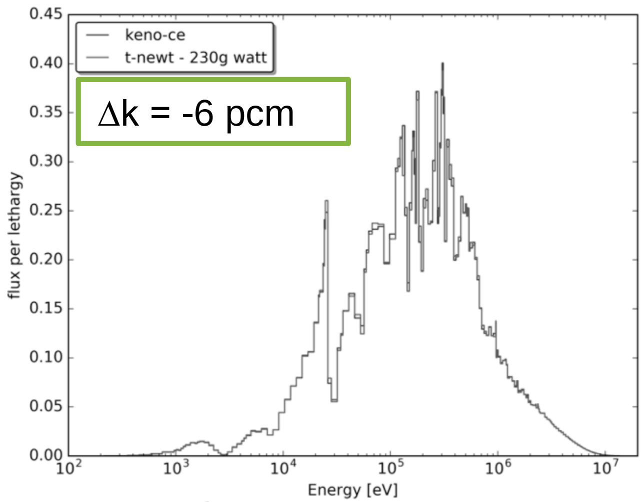
Generation of multigroup libraries and covariance data for advanced reactors – in collaboration with GRS



- Continuous-energy data serve as reference solution to confirm multigroup approximations
- SCALE 6.2 includes multigroup neutronics libraries that are optimized for LWRs
- Multigroup cross sections can be generated for any type of system
  - LWR, HTGR, MSR, FHR, SFR, etc. with appropriate energy group structure and weighting spectrum
- Uncertainties in cross sections (covariance data) quantify confidence in deployed data libraries
- Example for SFR:



Incorrect group structure/weighting



Correct group structure/weighting

Uncertainty in k_{eff} Due to Nuclear Data Uncertainties: 1,435 pcm!

covariance matrix nuclide-reaction		with	nuclide-reaction		% Δk/k due to this matrix
u-238	n,n'		u-238	n,n'	1.2053(9)
na-23	elastic		na-23	elastic	0.3242(2)
fe-56	elastic		fe-56	elastic	0.2590(3)
u-238	n,gamma		u-238	n,gamma	0.2435(1)
fe-56	n,n'		fe-56	n,n'	0.2388(1)

Summary and conclusions

- Spent fuel, advanced reactors, and other challenging application areas require advanced methods for design optimization, validation, and licensing
- High quality and consistent covariance data is needed for setting safety limits and optimizing design
- Overall uncertainties in ENDF appear to be too large
- Drastic changes and/or errors in covariance data erode confidence in S/U methods
- The community needs SG-44 to stimulate coordination and provide the highest quality covariance data