Emerging activities on nuclear data for ATF at CIEMAT

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Nuclear Innovation Unit
CIEMAT

About the Nuclear Innovation Unit - CIEMAT

Solid expertise:

1. Reactor calculations
2. Fuel cycle calculations

With our well validated / verified codes: EVOLCODE, SUMMON, COUNTHER, TREVOL.

3. Nuclear data evaluation and processing (participation in JEFF4)
4. Integral experiments in Cadarache, Minsk, Mol.
5. Nuclear data measurements: cross sections and decay data. ^{237}Np, $^{240,242}Pu$, $^{241,243}Am$, $^{235,238}U$, $^{244,246}Cm$, ^{239}Pu, decay properties of fission fragments.

CIEMAT has coordinated the last three EC nuclear data projects: **ANDES**, **CHANDA** and **SANDA**

European Research Council
Established by the European Commission

Why ATFs?

Accident tolerant fuels (ATF) should:

•**endure the loss of active cooling** in a reactor core for much longer than the current fuel, widen the existing safety margin for nuclear plants,

•**improve nuclear plant performance with fuel that lasts longer**, reduce operational and maintenance costs to pass savings on to electricity consumers.

Nuclear data needs and priorities are driven by two major aspects:

- **Insufficient accuracy of the nuclear data** available for the isotopes present in the new fuel and cladding materials.
- **Higher ^{235}U enrichment / burnup** which will affect the isotopic inventory at the end of the irradiation.

Nuclear data needs for different ATF concepts

New fuel cladding materials / coating

- **Coated and improved Zr alloys**. Ceramic candidates are Ti_3AlC_2, Ti_2AlC and Nb_2AlC. **Improvement of the Ti, Nb (capture) cross sections.**
- **Advanced steels (FeCrAl)**. Up to **70 GWd/t rod average burn-ups** achieved in BR-2 experiments. Need of compensating the neutron absorption in steel with increased U enrichment in about 1%-1.5%. **Improvement of the Fe (capture) isotopes cross sections.**
- **Mo-alloy cladding** (coated with Zr or Al alloy – FeCrAl –). Requires a **higher U enrichment in 1% - 3%** (beyond the licencing limit) maintaining a constant cycle length, depending on the Mo thickness. ^{95}Mo is responsible for most of the capture in Mo. **Improvement of the Mo (capture) isotopes cross sections.**
- **SiC / SiC composite cladding**. ^{28}S has a capture XS uncertainty of ~ 20% in epithermal region (> 0.5 eV) in JEFF-3.3. **Improvement of the Si (capture) isotopes cross sections.**

New fuel materials

- **Improved/doped UO_2**. Cr and Al oxides.
- **(UO_2 – SiC) composite fuel**. **Si cross sections. SiC thermal scattering.**
- **High density fuel – Nitride, UN**. Large (n,p) cross section in ^{14}N would require larger enrichments. Enrichment in ^{15}N would increase costs and avoid larger enrichment. **Improve the $^{14}N(n,p)$ and perhaps other ^{15}N cross sections.**
- **High density fuel - Silicide, U_3Si_2**. **Improve the status of the ^{28}Si capture cross section**.
- **High density fuels** - Carbides, Metallic.
- **TRISO**. **^{235}U enrichment up to 19.9%.**The TRISO particles are embedded in a SiC matrix inside a cylindrical cladding. **Improve the status of the Si capture cross section**. **Thermal scattering xs.**
- **U-Mo fuel**. Capture in ^{95}Mo would require slightly **larger ^{235}U enrichments**. **Improvement of the Mo (capture) isotopes cross sections.**

Candidates to be improved: Ti, Nb, Si, Mo, N, Fe, Cr, Al, SiC…

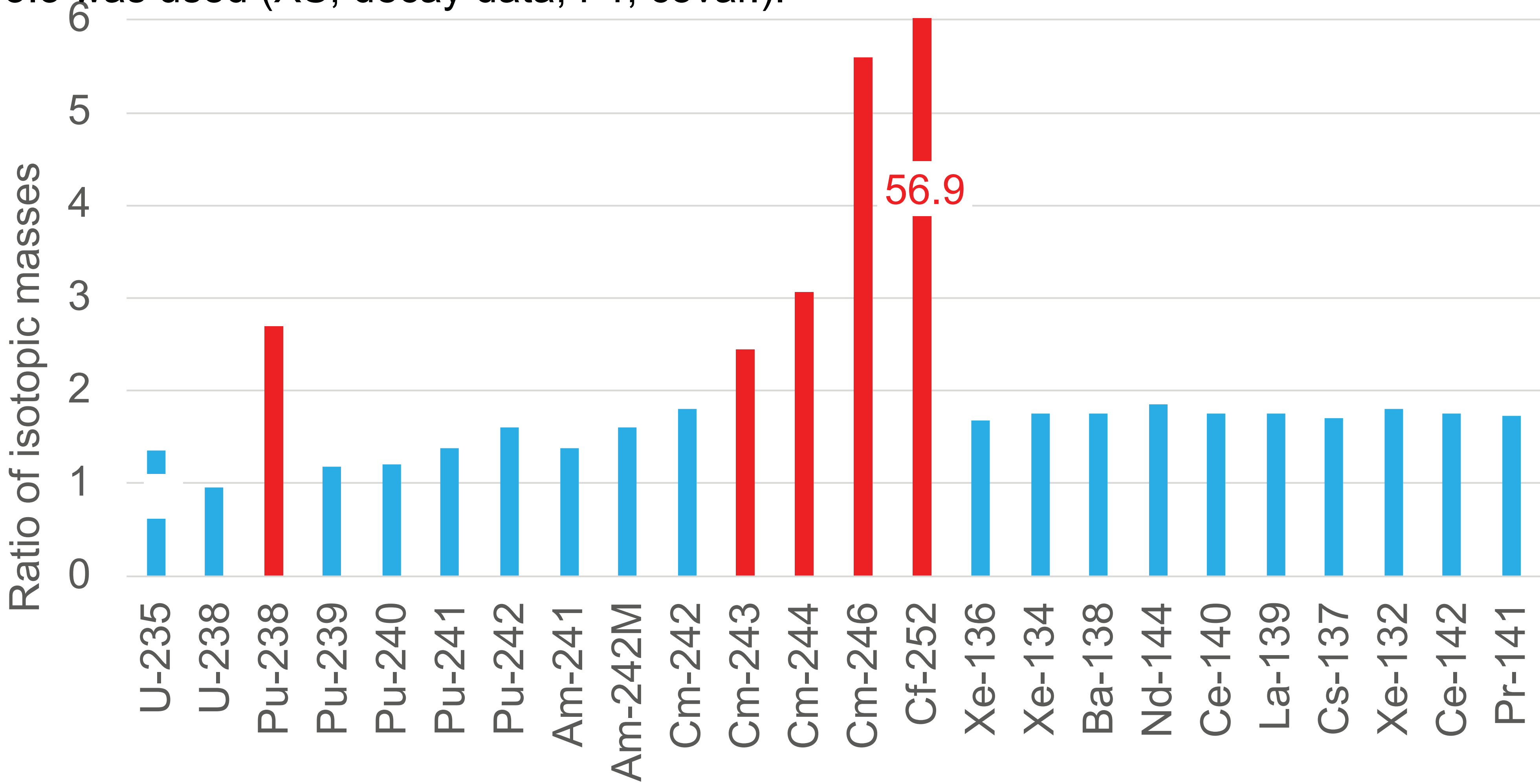
Nuclear data needs for higher burnup scenarios (with ATFs)

Detailed calculations and sensitivity/uncertainty analyses (S/U) addressing different scenarios:

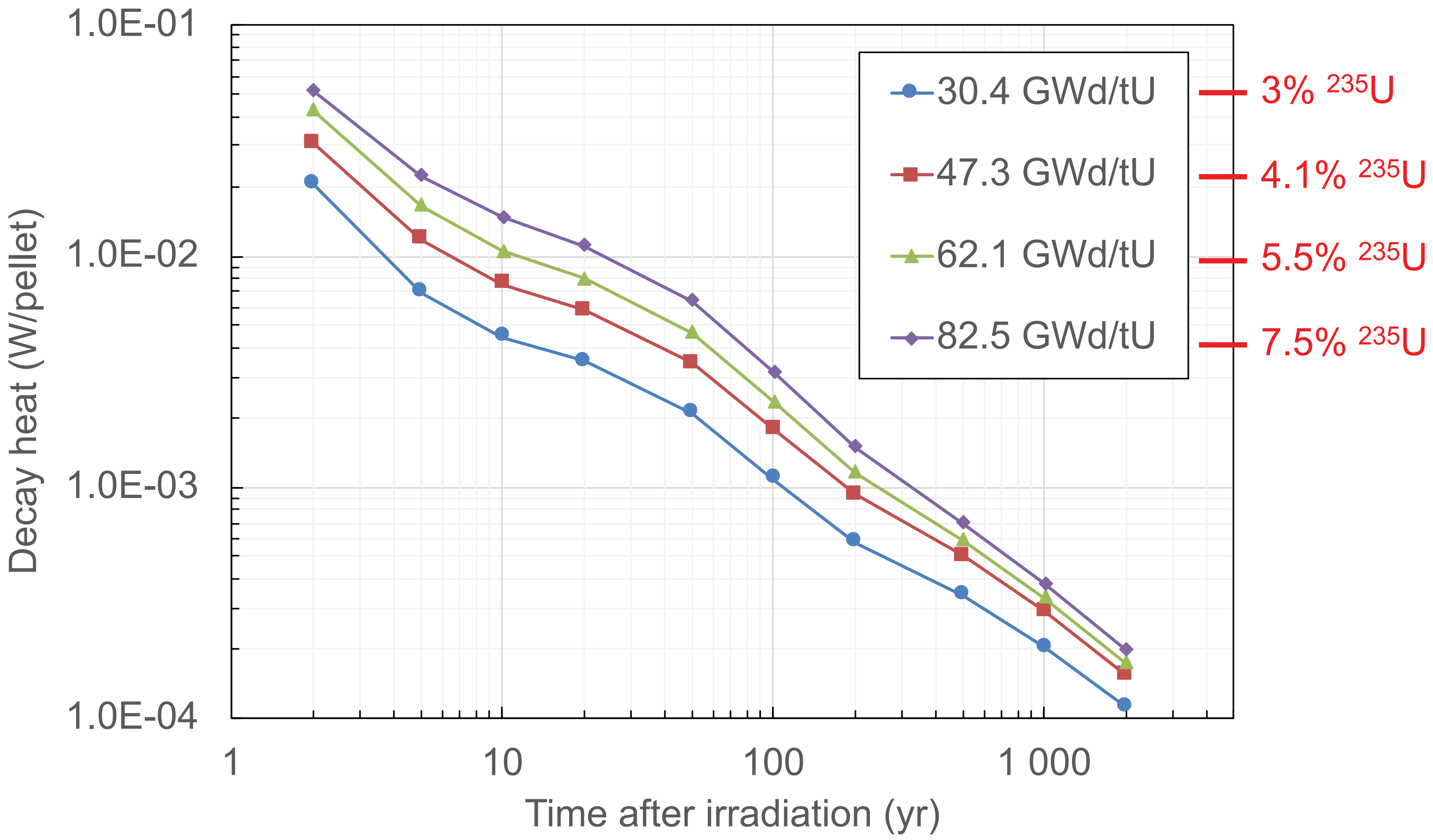
- Reactor operation: neutronics & safety.
- Cooling at the pools: re-criticality, heat.
- Predisposal: heat, neutron emission, thermo-mechanical issues, corrosion…
- (Re-processing: alpha, beta, gamma and neutron emission rates, heat).
- Final disposal: heat, re-criticality, radiotoxicity.

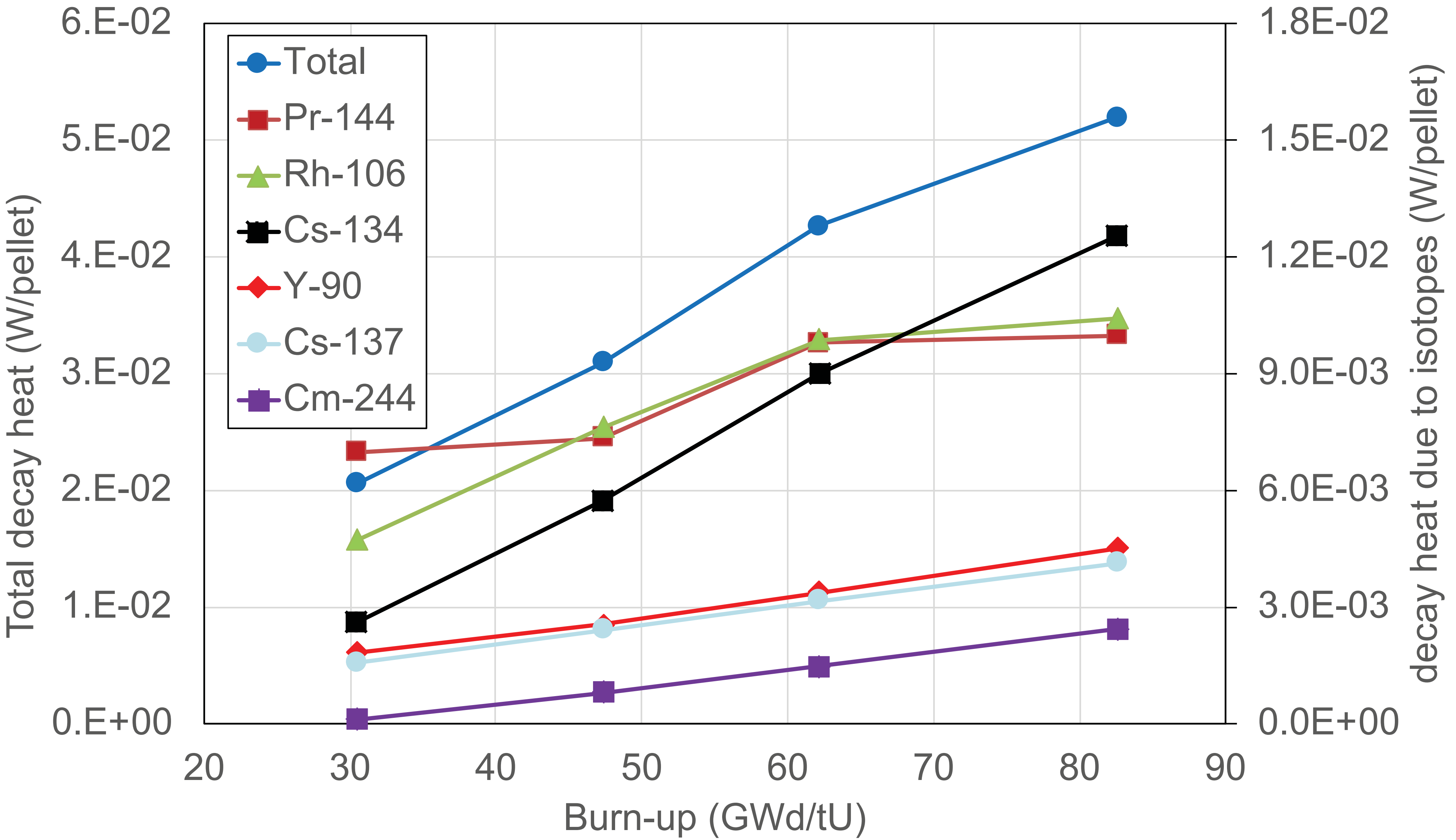
82.5/47.3 (GW/d/tU) burn-up isotope mass ratio

We have performed **very simple** burnup calculations of a fuel pin with different ^{235}U enrichments (3% up to 7.5%) and evaluated the Impact on the isotopic inventory, decay heat and neutron emission after the irradiation and 2 years of cooling. JEFF 3.3 was used (XS, decay data, FY, covar.).

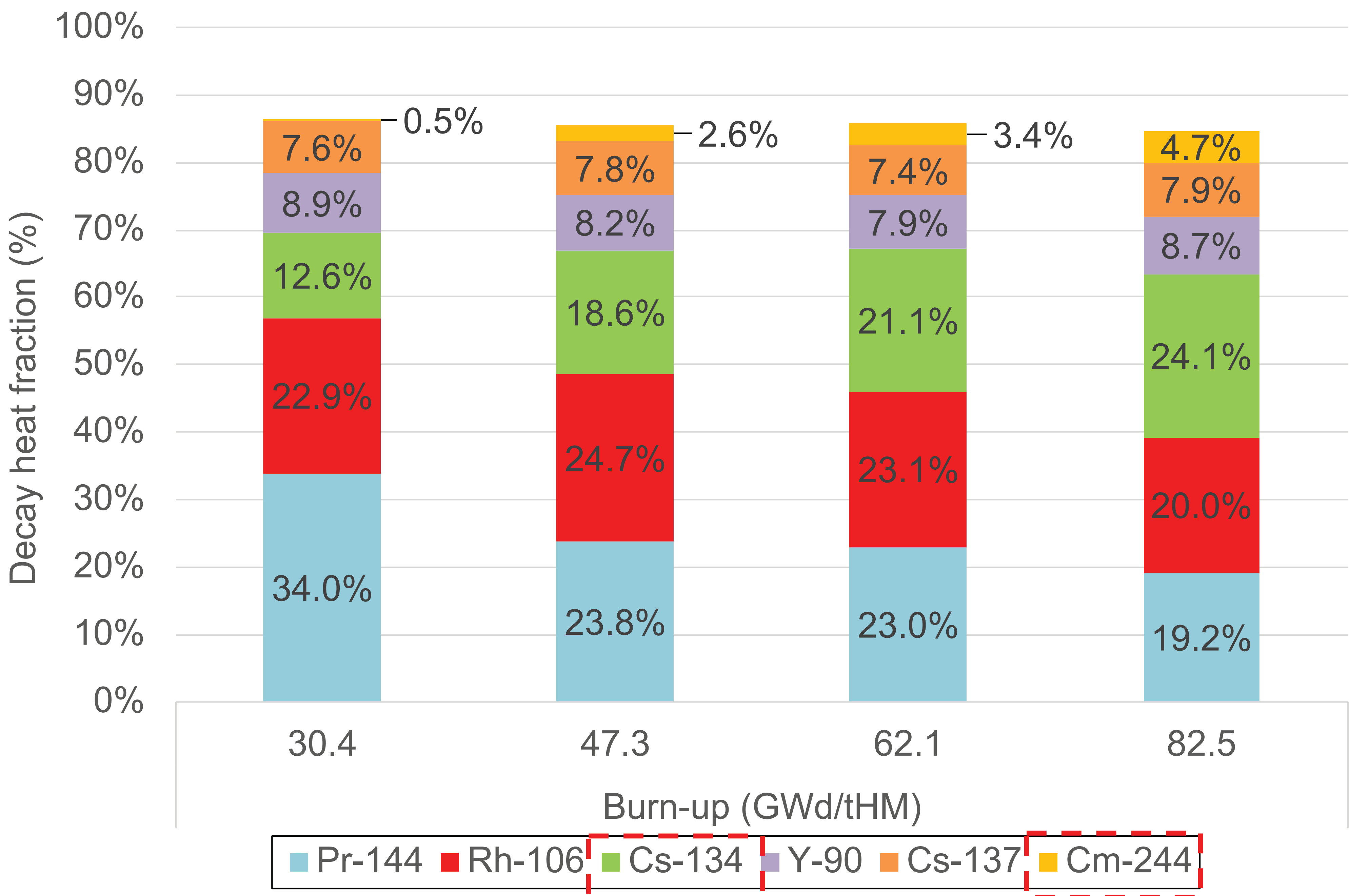


Decay heat with EVOLCODE

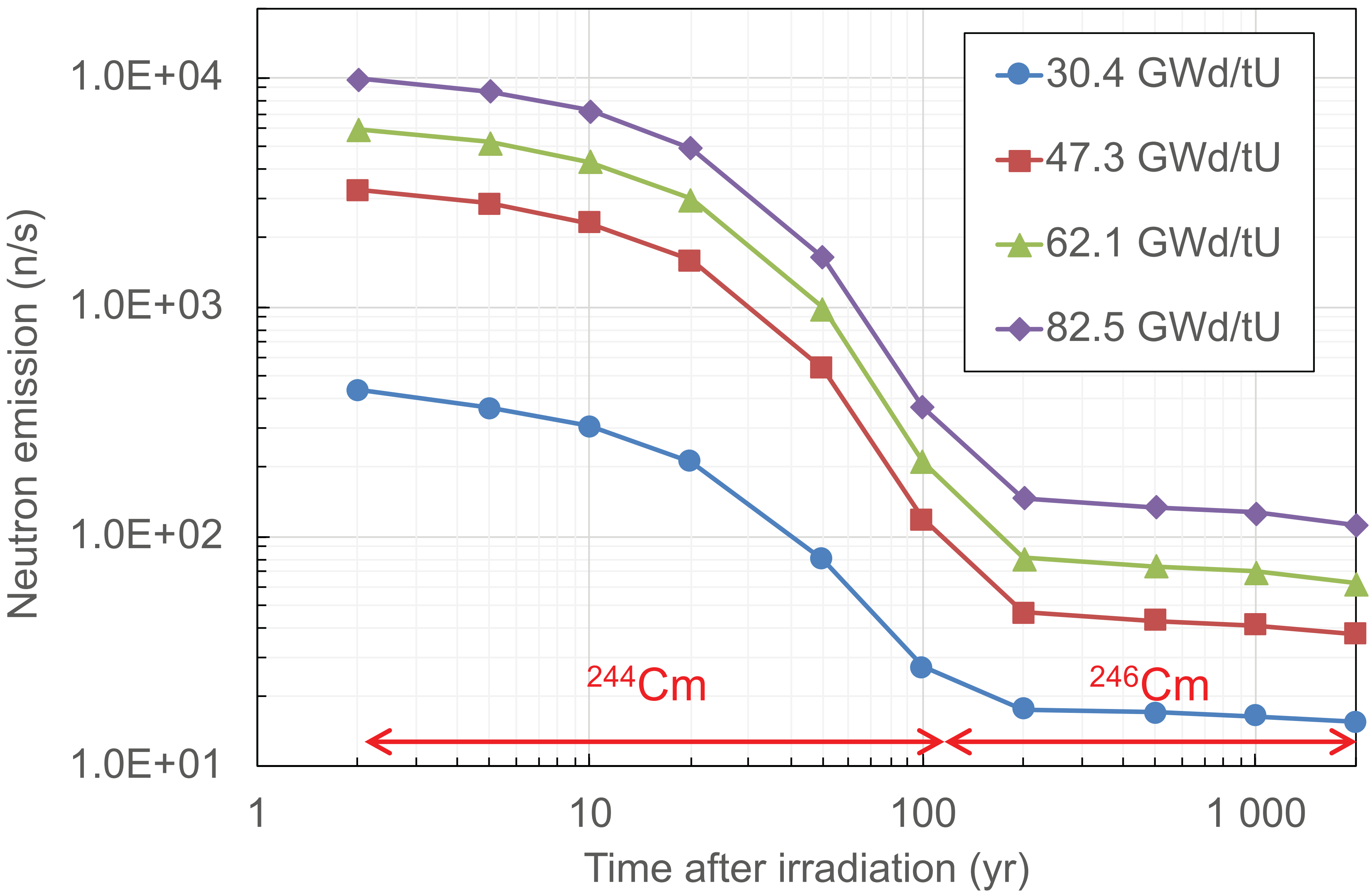


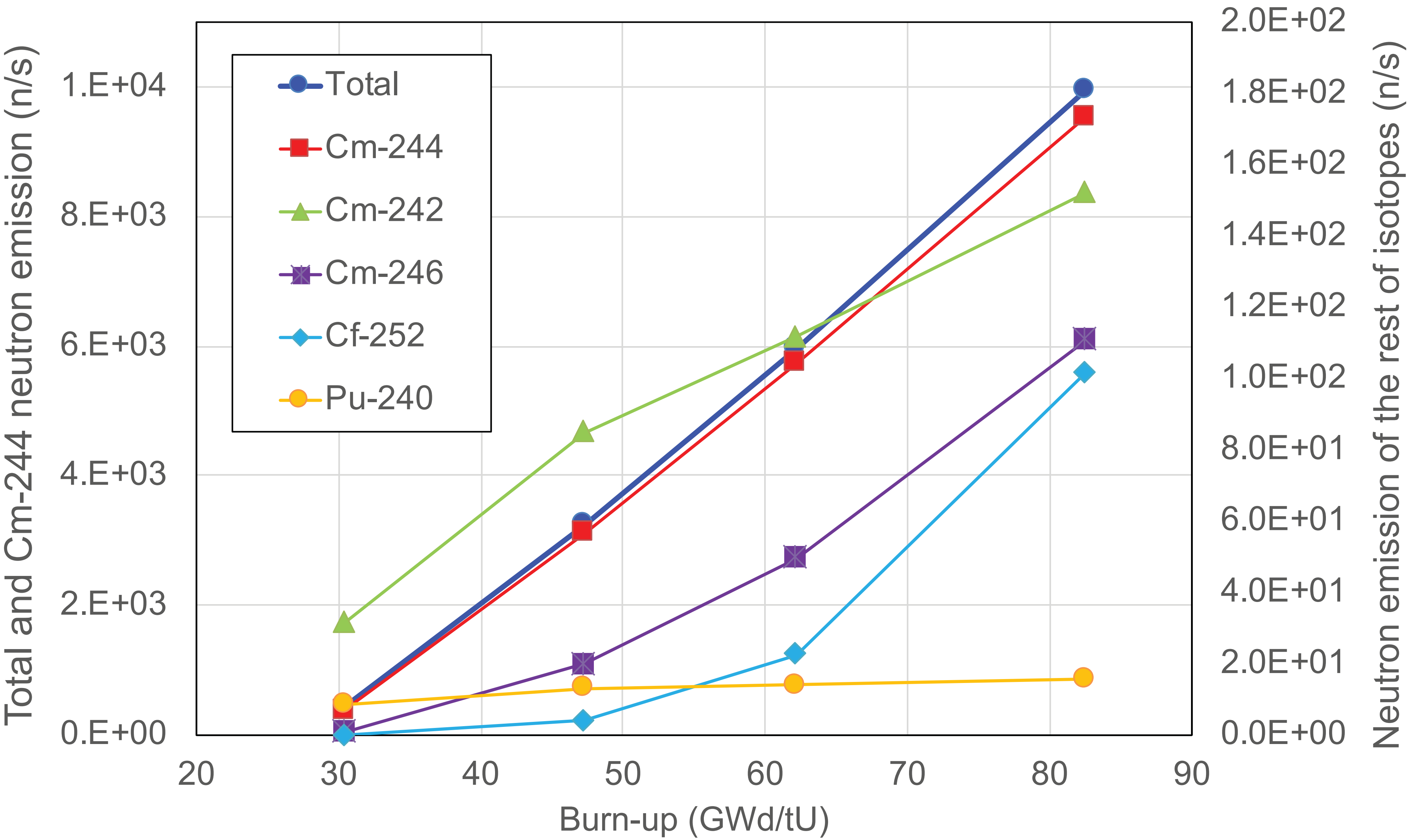


Total
Pr-144
Rh-106
Cs-134
Y-90
Cs-137
Cm-244
6.E-02
5.E-02
4.E-02
3.E-02
2.E-02
1.E-02
0.E+00
1.8E-02
1.5E-02
1.2E-02
9.0E-03
6.0E-03
3.0E-03
0.0E+00
20
30
40
50
60
70
80
90
Total decay heat (W/pellet)
decay heat due to isotopes (W/pellet)
Burn-up (GWd/tU)

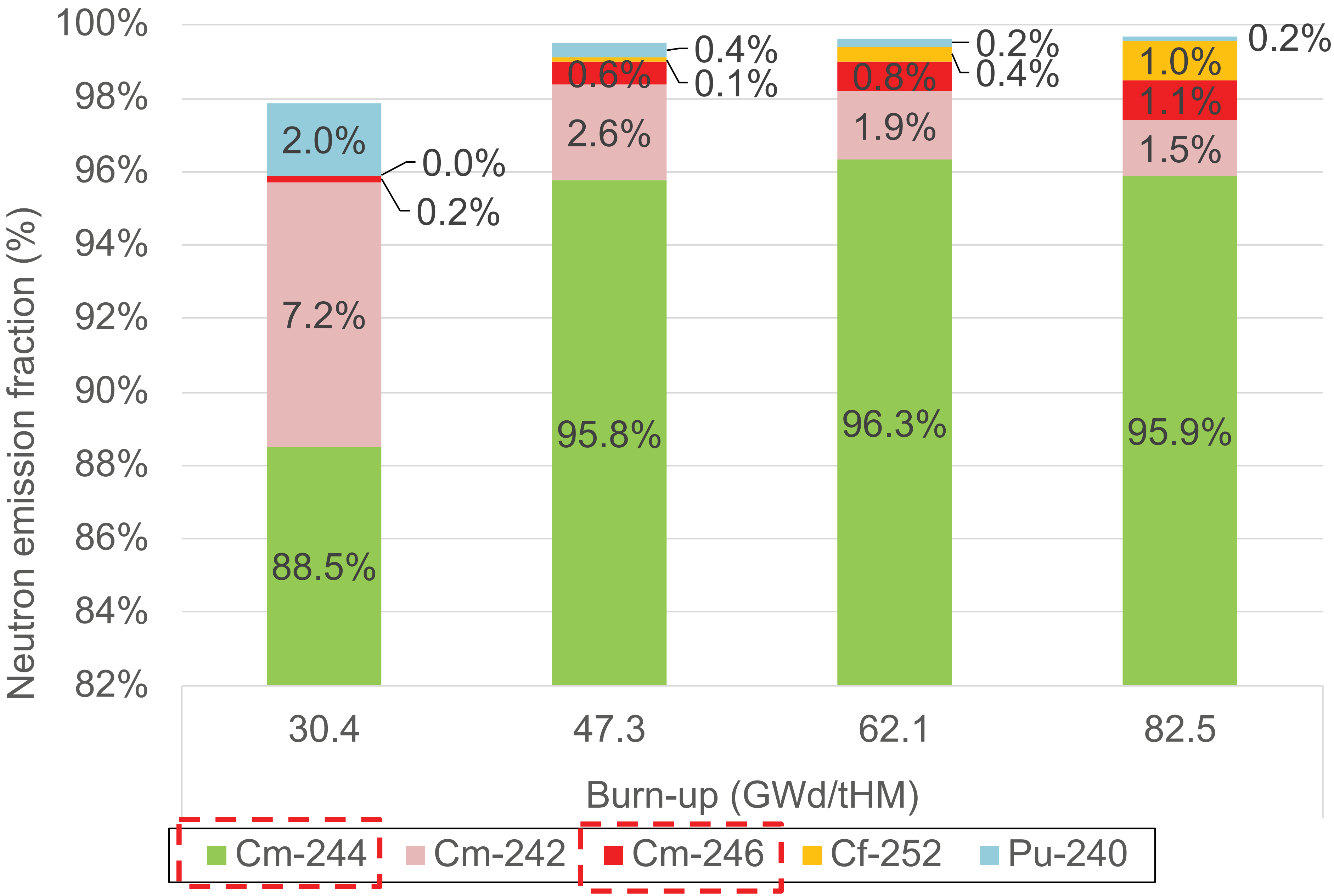

100%
90%
80%
70%
60%
50%
40%
30%
20%
10%
0%
Decay heat fraction (%)
0.5%
7.6%
8.9%
12.6%
22.9%
34.0%
2.6%
7.8%
8.2%
18.6%
24.7%
23.8%
3.4%
7.4%
7.9%
21.1%
23.1%
23.0%
4.7%
7.9%
8.7%
24.1%
20.0%
19.2%
30.4
47.3
62.1
82.5
Burn-up (GWd/tHM)
Pr-144
Rh-106
Cs-134
Y-90
Cs-137
Cm-244

Neutron emission (after a 2 yr cooling time)





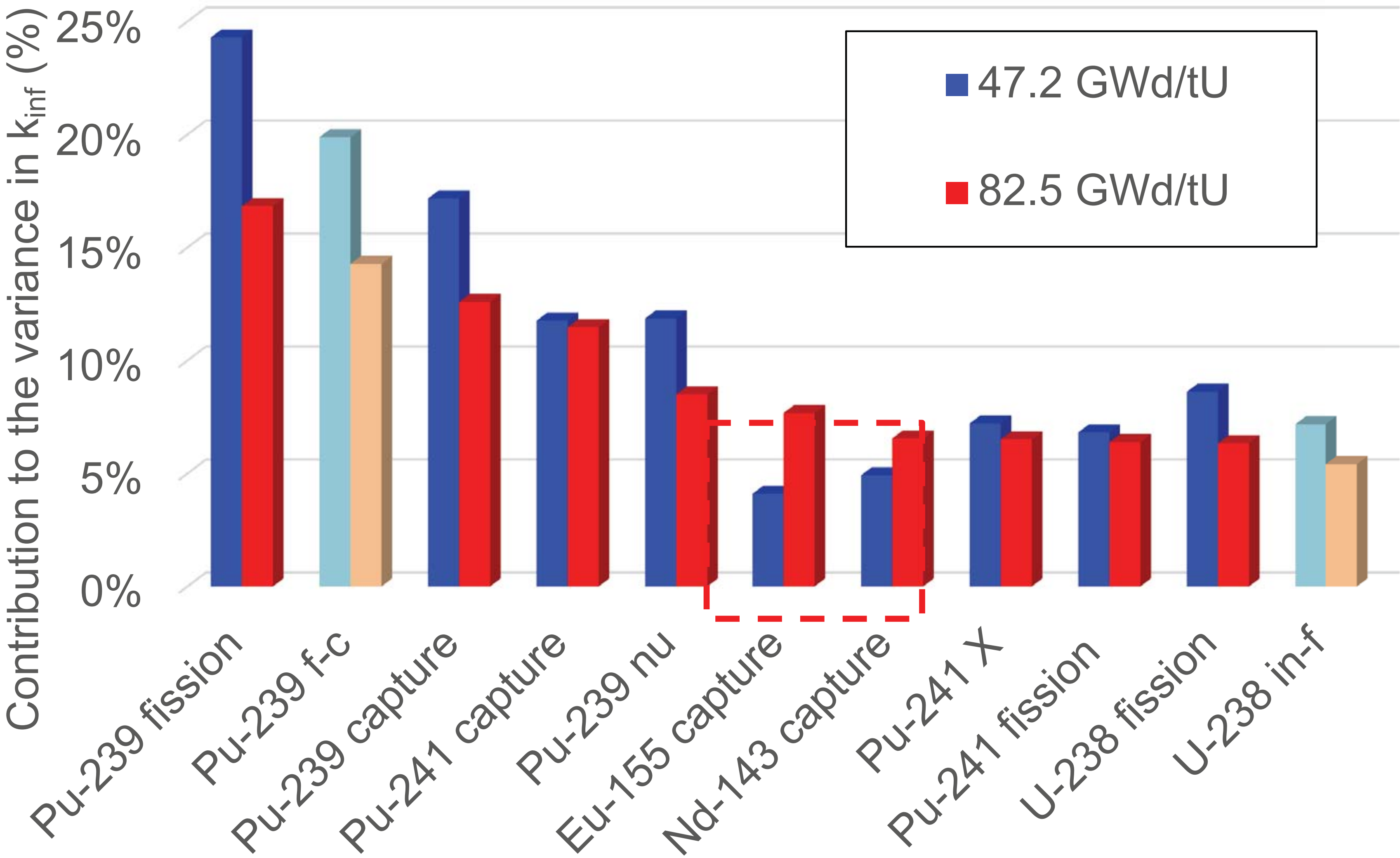
Total
Cm-244
Cm-242
Cm-246
Cf-252
Pu-240
Total and Cm-244 neutron emission (n/s)
1.E+04
8.E+03
6.E+03
4.E+03
2.E+03
0.E+00
Neutron emission of the rest of isotopes (n/s)
2.0E+02
1.8E+02
1.6E+02
1.4E+02
1.2E+02
1.0E+02
8.0E+01
6.0E+01
4.0E+01
2.0E+01
0.0E+00
20
30
40
50
60
70
80
90
Burn-up (GWd/tU)



100%
98%
96%
94%
92%
90%
88%
86%
84%
82%
Neutron emission fraction (%)
2.0%
0.0%
0.2%
7.2%
88.5%
0.4%
0.1%
0.6%
2.6%
95.8%
0.2%
0.4%
0.8%
1.9%
96.3%
0.2%
1.0%
1.1%
1.5%
95.9%
30.4
47.3
62.1
82.5
Burn-up (GWd/tHM)
Cm-244
Cm-242
Cm-246
Cf-252
Pu-240

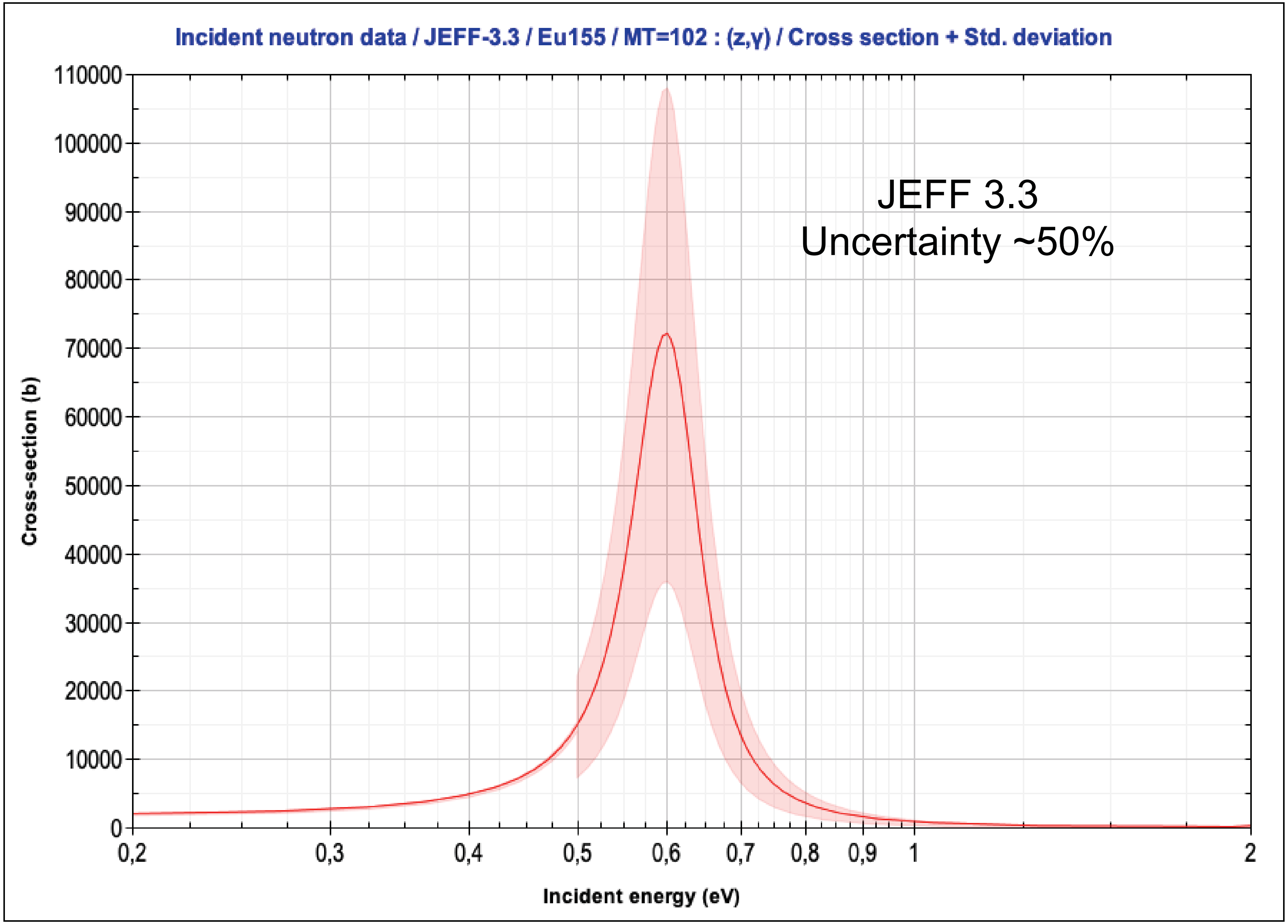
S/U analysis with SUMMON: uncertainty in k_{inf} due to XS

Normal burn-up of 47.2 GWd/tU: Uncertainty in k_{inf} of 0.65% due to cross sections
Long burn-up of 82.5 GWd/tU: Uncertainty in k_{inf} of 0.72% due to cross sections

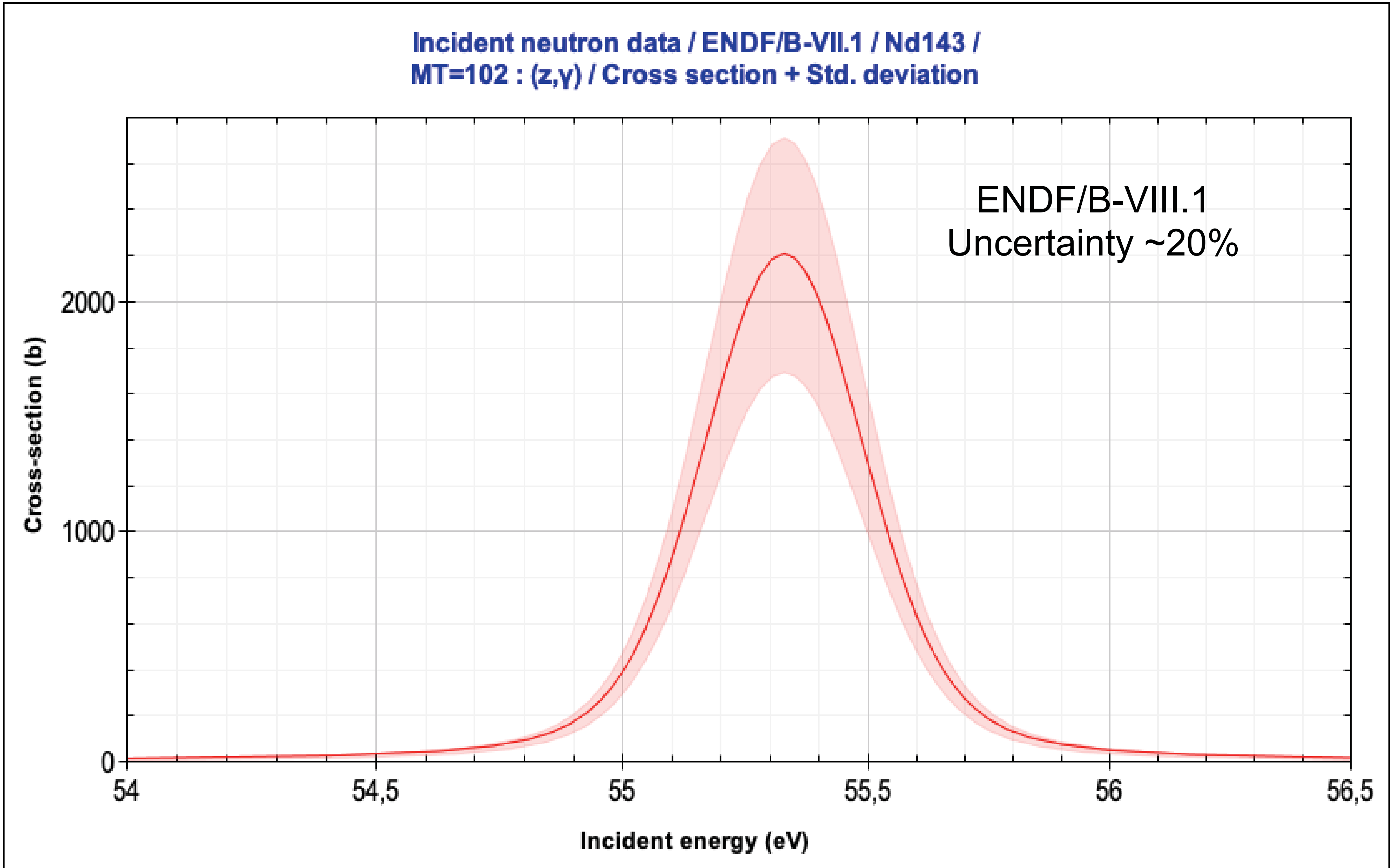


In cyan and orange: anticorrelations

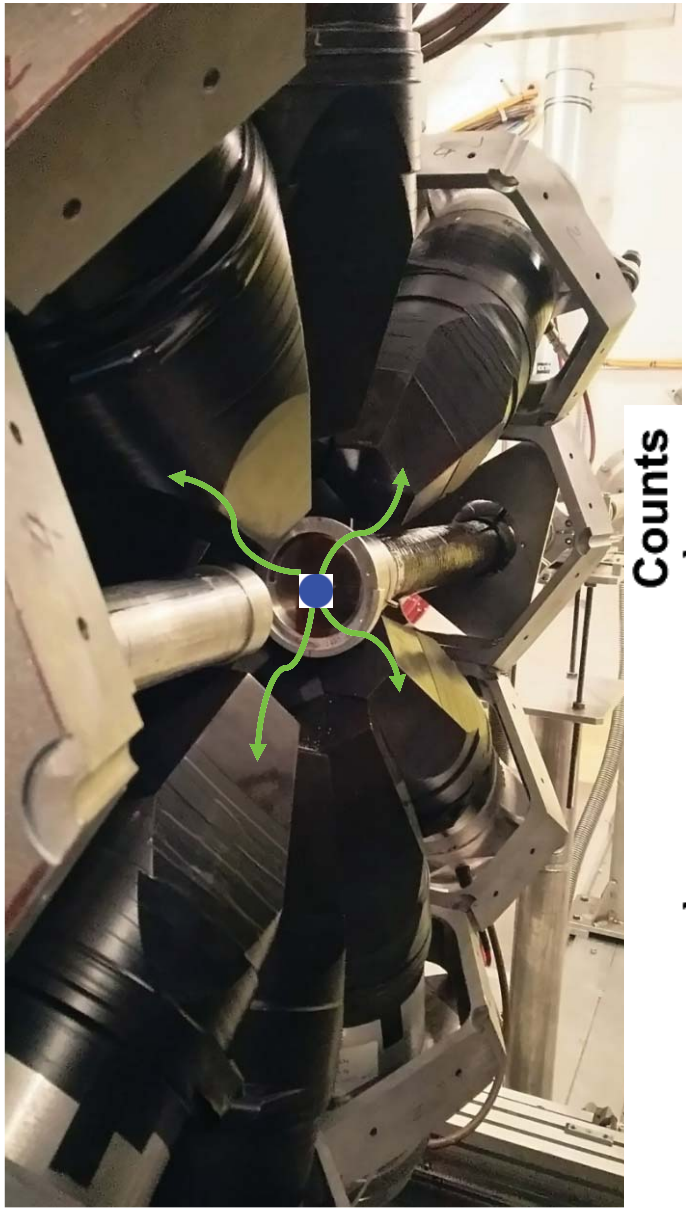
^{155}Eu(n,γ) cross section ($T_{1/2}$ = 4.753 yr)



^{143}Nd(n,γ) cross section

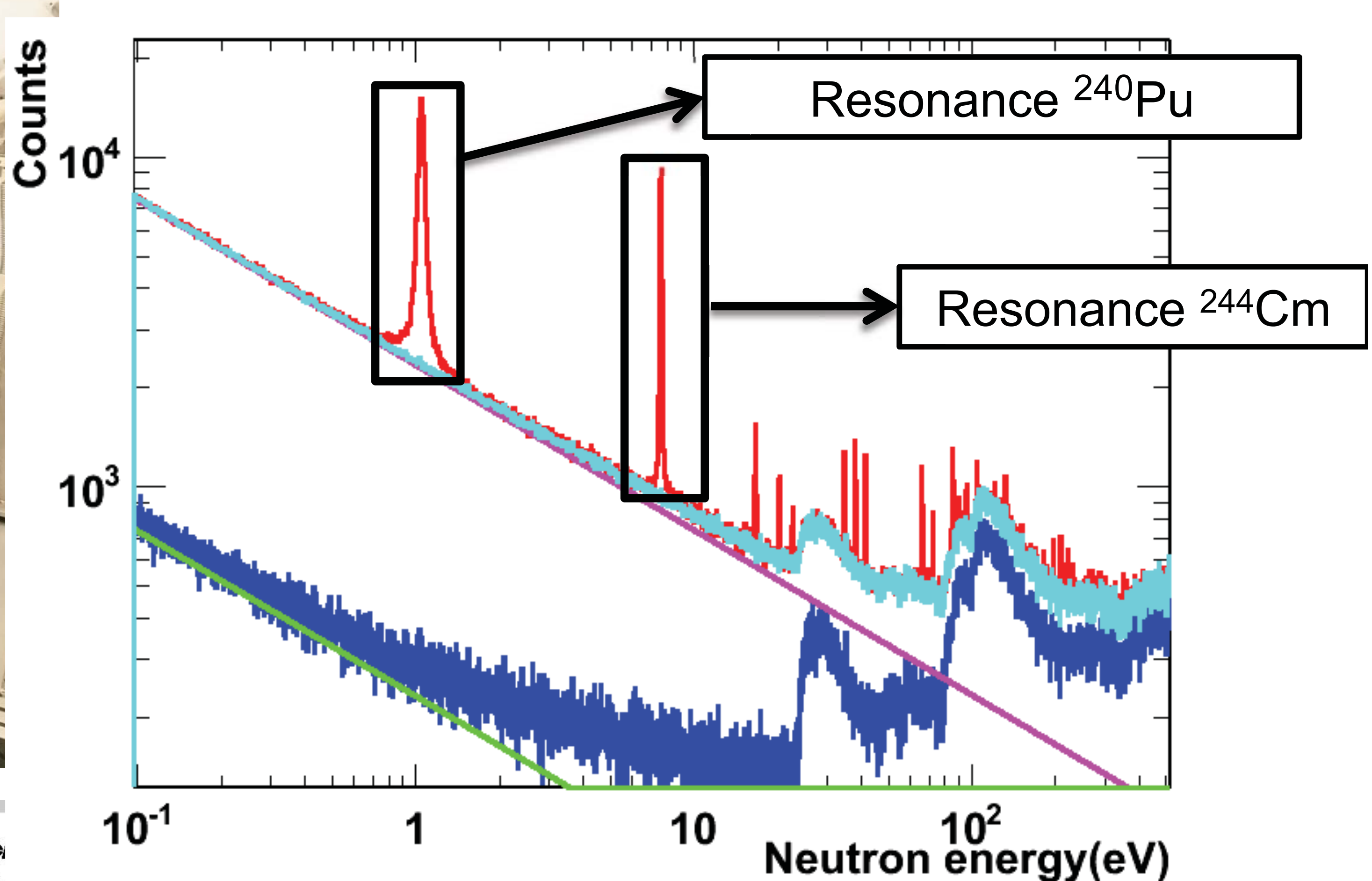


$^{244}Cm(n,\gamma)$ cross section measurement at CERN n_TOF

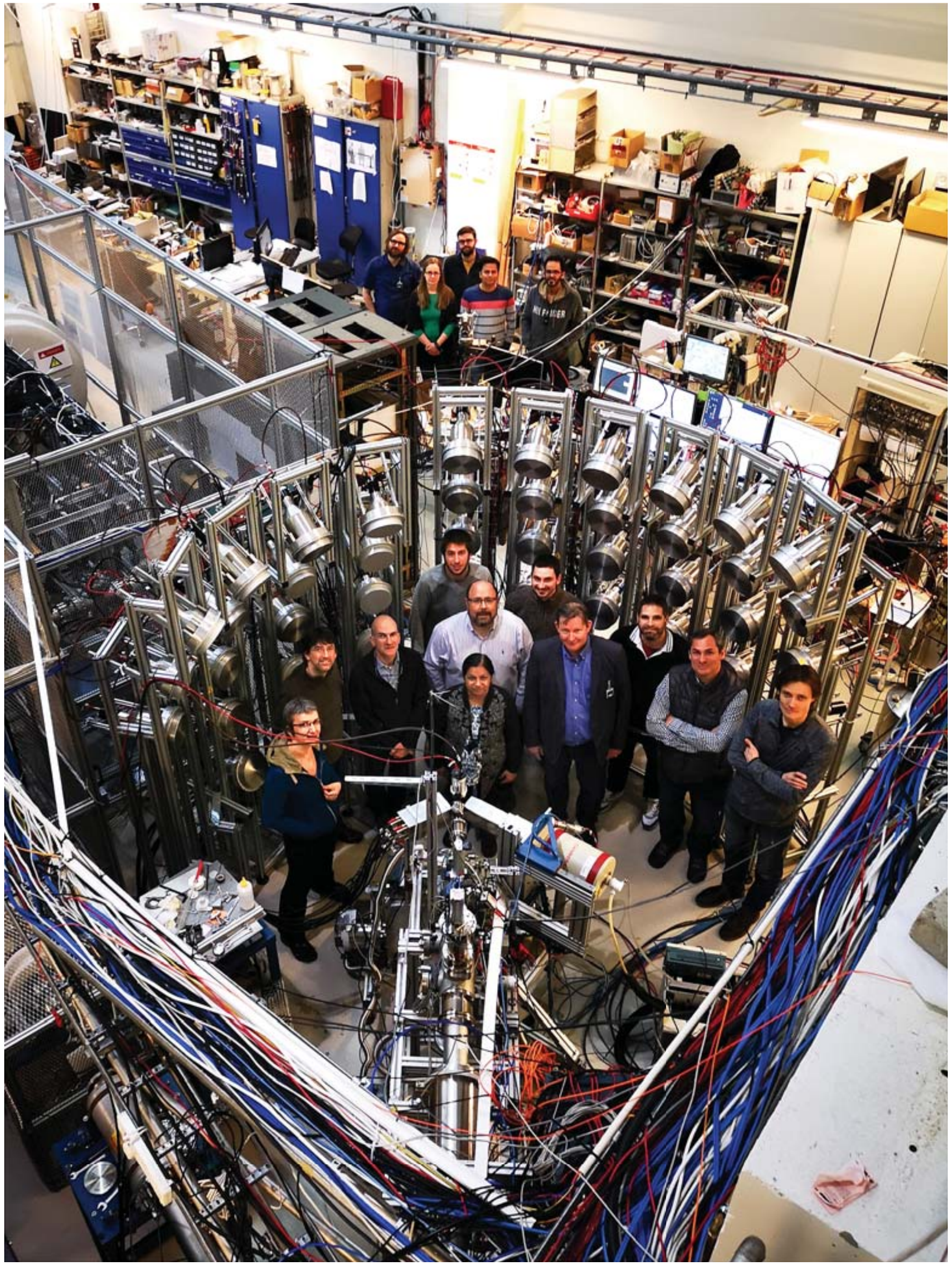


Measurement with a ~1 mg sample.

PhD thesis of Víctor Alcayne



“the Accident Tolerant Fuels for LWRs”



MONSTER

MOdular Neutron SpectromeTER

CIEMAT, VECC (India), JYFL (Finland), IFIC (Valencia), UPC (Barcelona)

^{85}As βn-decay at the University of Jyväskylä

PhD thesis of A. Pérez de Rada (CIEMAT)

CIEMAT's future activities on nuclear data for ATFs

I. Simulation of reactor cores loaded with ATFs with the propagation of nuclear data uncertainties **(EVOLCODE + SUMMON):**

- Irradiation / depletion calculations for obtaining a detailed isotopic inventory and all possible derived quantities: decay heat, neutron emission rates... (EVOLCODE)
- Impact of nuclear data on reactor safety parameters: k_{eff}, β_{eff}...
- Sensitivity and uncertainty (S/U) analyses for the determination of nuclear data priorities (SUMMON code).

II. Assessment of the impact of nuclear data on **fuel cycle calculations including ATF (TREVOL code)**

- Complex fuel cycle simulations with uncertainty propagation.
- S/U analyses for determining the most relevant nuclear data for cooling, predisposal and final disposal.

III. Measurement of prioritary nuclear data for ATFs (cross sections, decay properties). Started with the $^{244,\,246}Cm$ cross section measurements.