SCK•CEN

STUDIECENTRUM VOOR KERNENERGIE
CENTRE D'ETUDE DE L'ENERGIE NUCLEAIRE

Sensitivity analysis of the nuclear data for MYRRHA reactor modelling

A.Stankovskiy, G.Van den Eynde
SCK•CEN, Mol, Belgium

C.J.Diez, O.Cabellos
UPM, Madrid, Spain

P.Schillebeeckx, J.Heyse
EC-JRC-IRMM, Geel, Belgium

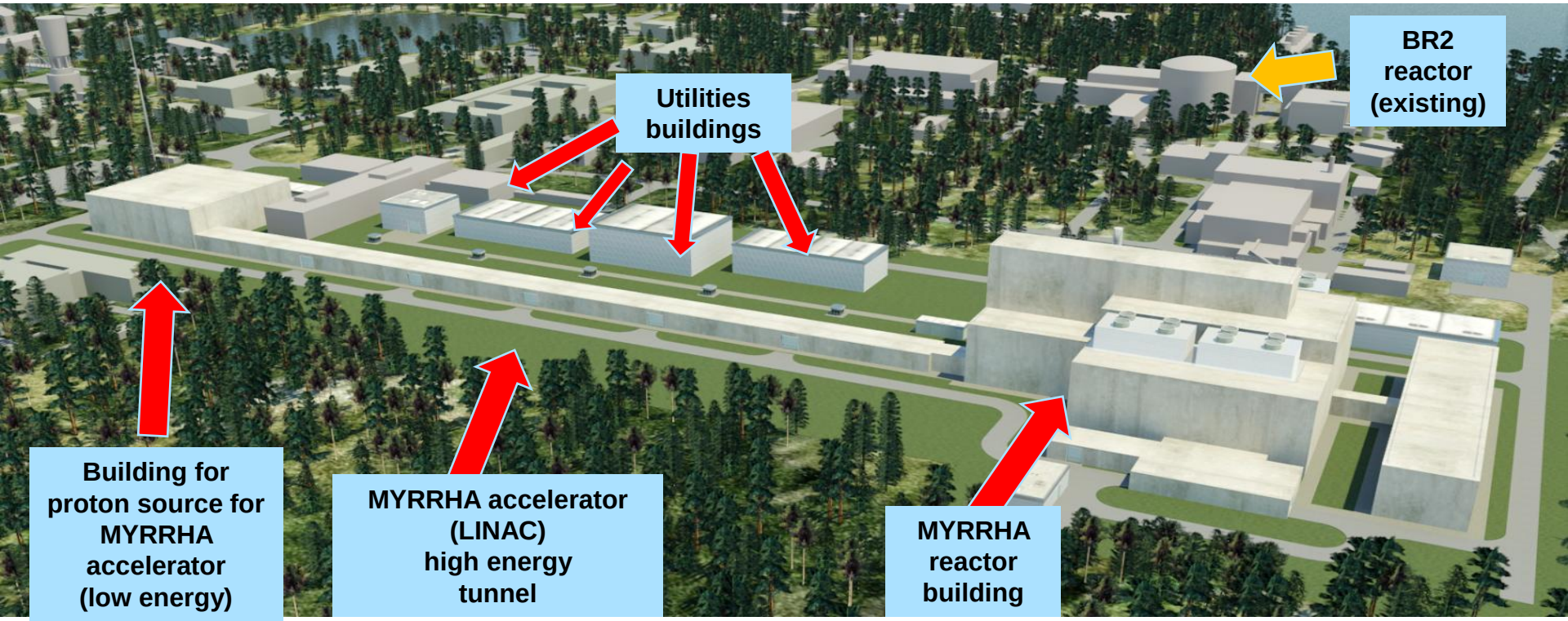
alexey.stankovskiy@sckcen.be

Overview

- MYRRHA project
- Neutronic characteristics
- Nuclear data sensitivities and priority list
- K_{eff} sensitivity analysis to ^{56}Fe data
- K_{eff} sensitivity analysis to ^{238}Pu data

MYRRHA at SCK•CEN

Multipurpose h**Y**brid **R**esearch **R**eactor for **H**igh-tech **A**pplications

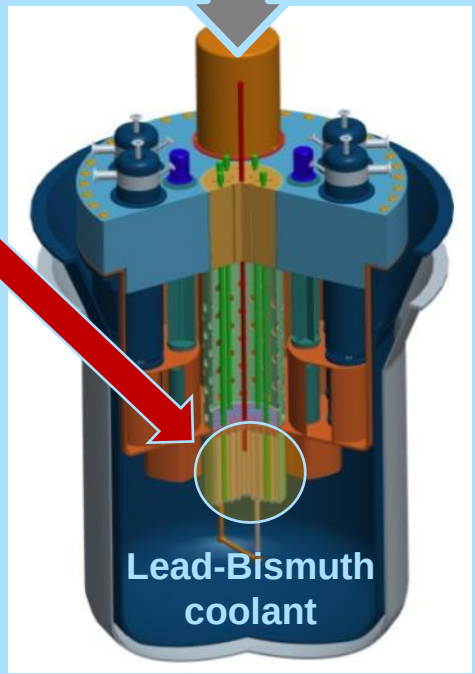


to JRC-IRMM

MYRRHA – Accelerator Driven System

Accelerator
(600 MeV - 4 mA proton)

Reactor
- Subcritical or Critical modes
- 65 to 100 MWth



Innovative & Unique

Spallation Source

Fast Neutron Source

Multipurpose Flexible Irradiation Facility

Multipurpose facility

Fuel research
in entire reactor

Material research
in large volumes

Fission GEN IV

Highly representative conditions
below target

Improved conditions
in dedicated positions

Fusion

Transmutation
in entire reactor

Waste

Multipurpose
h**Y**brid
Research
Reactor for
High-tech
Applications

High energy LINAC
600 MeV – 1 GeV
Long irradiation time

Fundamental research

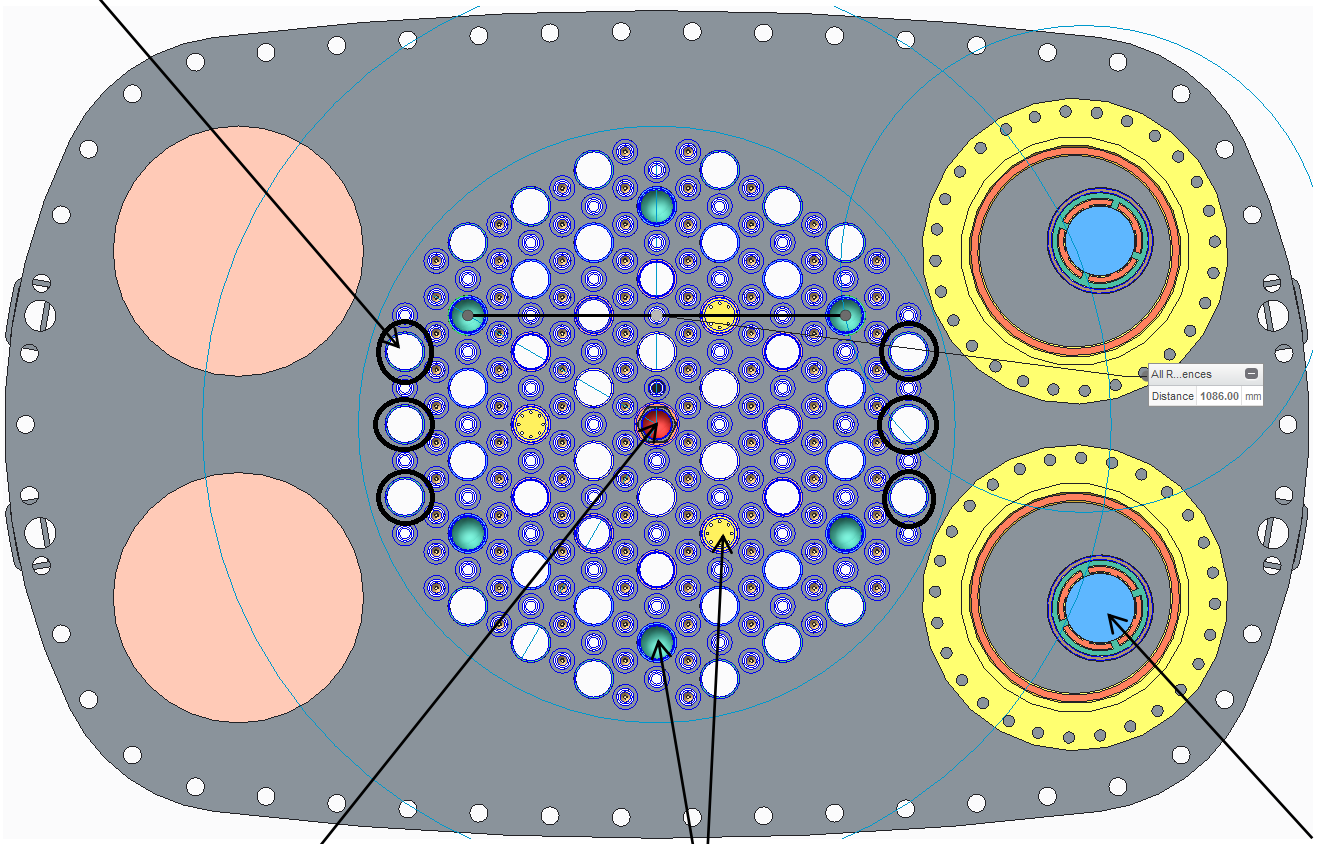
Radio-isotopes

Tc-99m
in dedicated positions

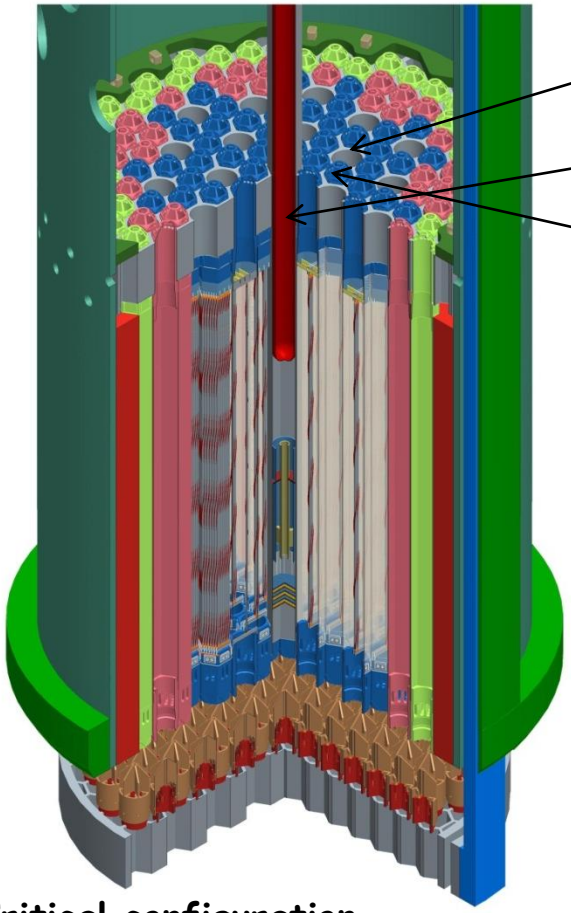
Silicon doping

Large ingots
outside of reactor core

CORE LAYOUT



Critical and sub-critical

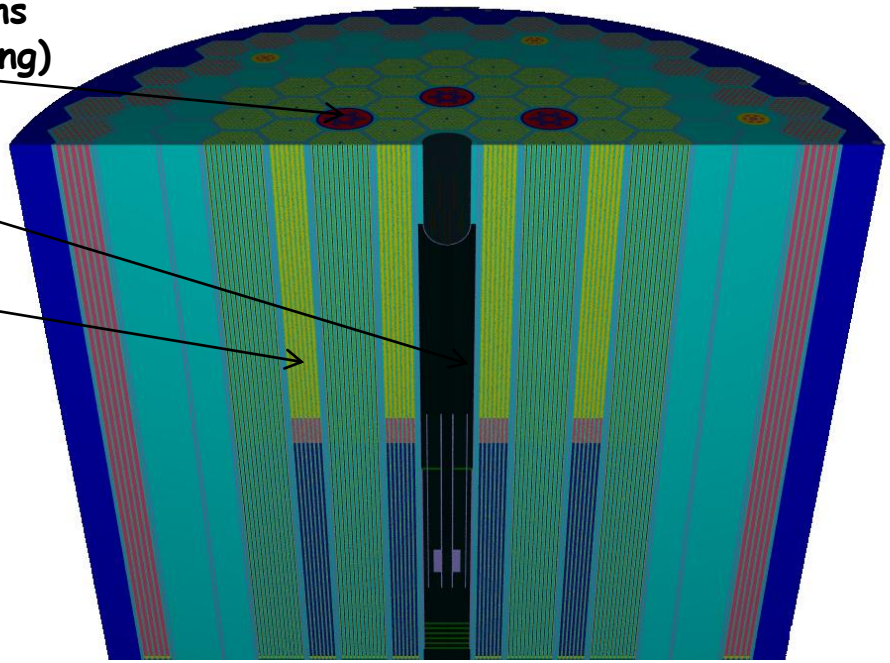


Critical configuration
~100 MWth

In-Pile Sections
(material testing)

Spallation
target

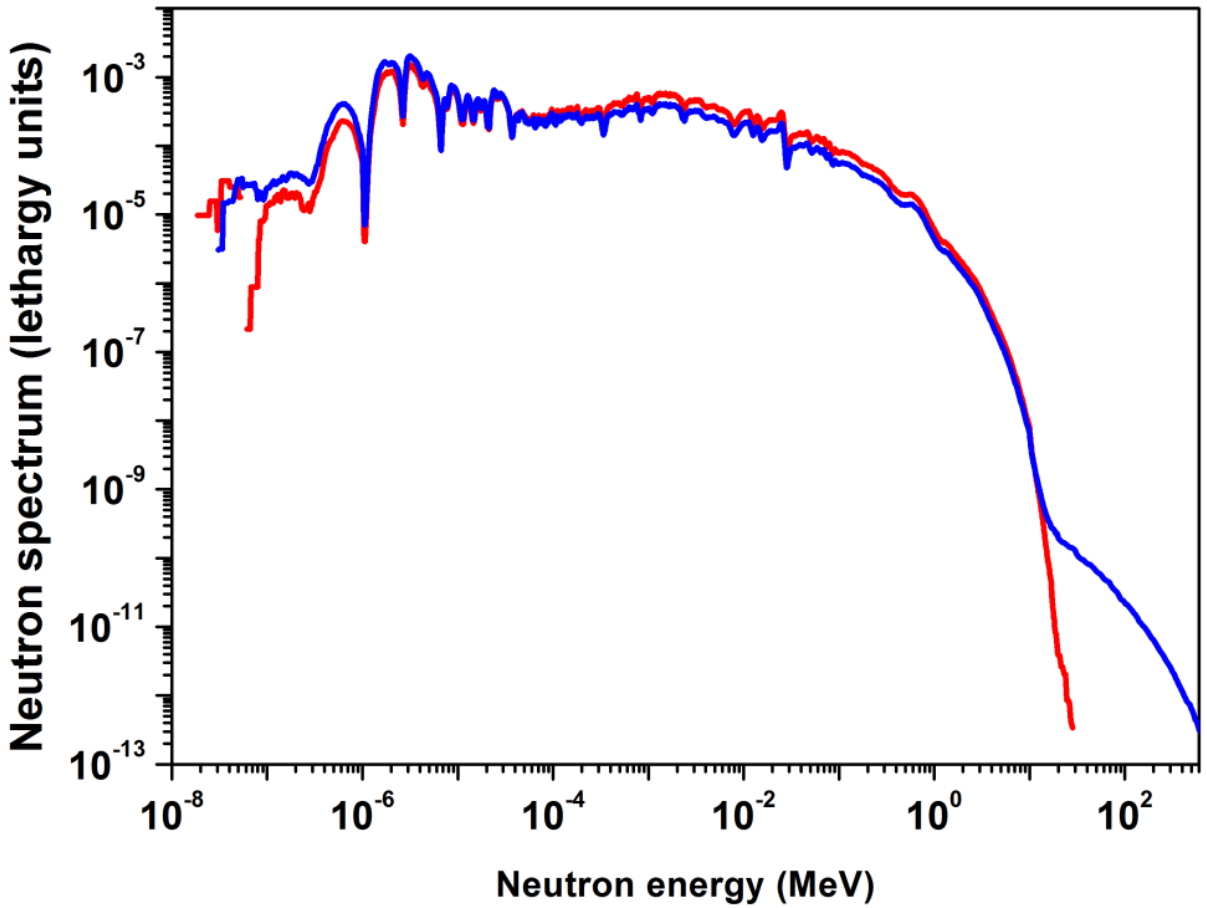
FA
(MOX)



Sub-critical configuration:

- 600 MeV, < 4mA proton beam
- 65 - 75 MWth
- K_{eff} ~0.95-0.96
- LBE spallation target (with beam window)
- LBE pool

Neutron spectra: critical vs. subcritical

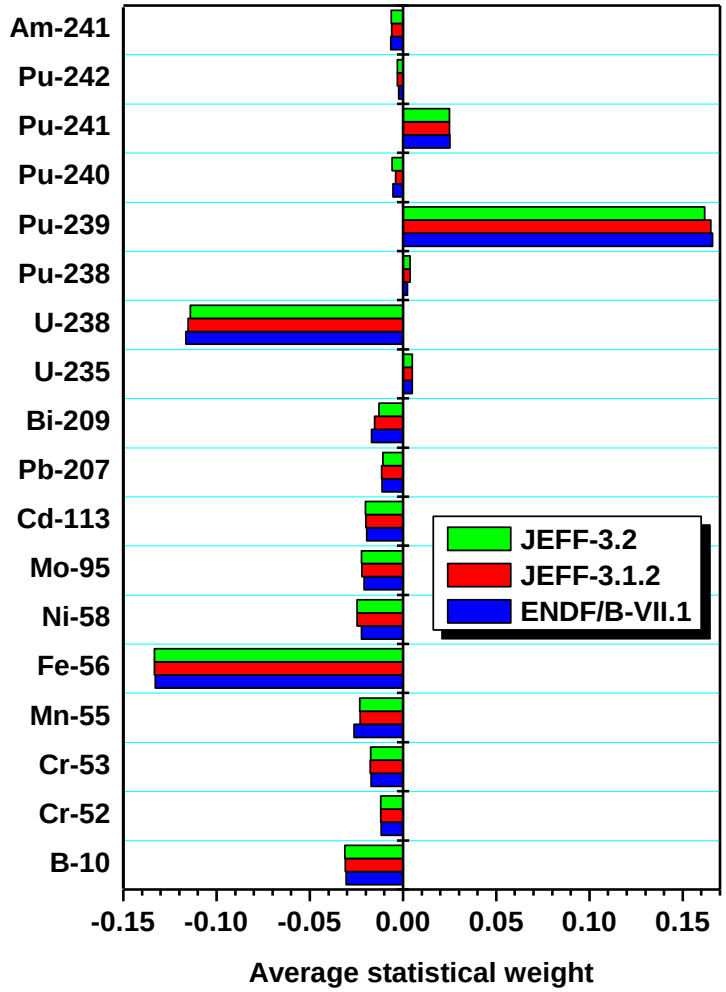


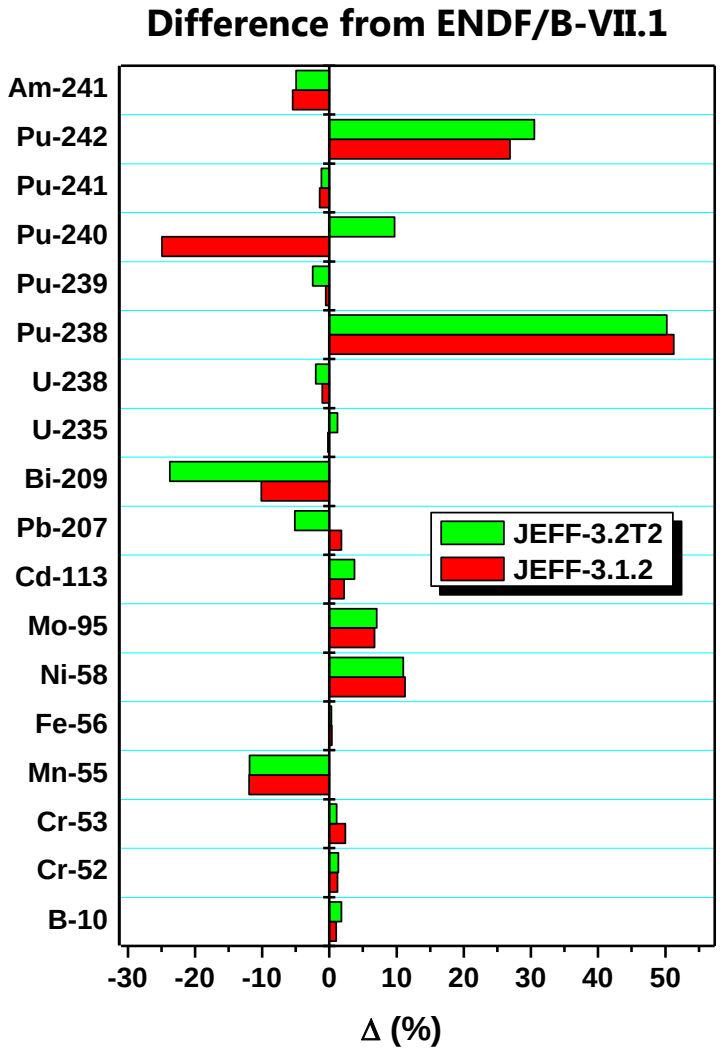
Sensitivity analysis of general purpose neutron data

Requirement of licensing authorities:

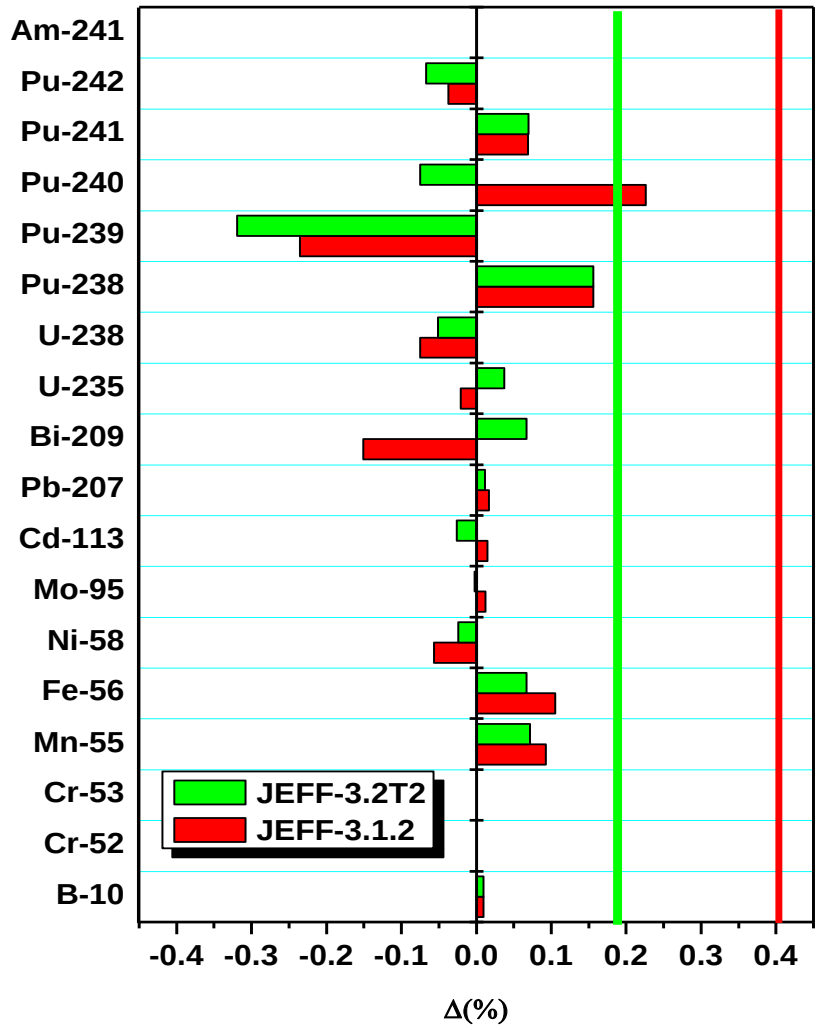
The codes and data libraries must be selected and validated. This validated set of codes & data must be used for further safety-related neutronics calculations of MYRRHA

Net neutron production (MCNPX 2.7.0)





Sensitivity of k_{eff} to nuclear data library change



Vertical lines reflect global library changes:

ENDF/B-VII.1 to JEFF-3.1.2 → +390 pcm

ENDF/B-VII.1 to JEFF-3.2T2 → +180 pcm

Priority list to improve uncertainty data

- ✓ **^{239}Pu neutron capture and fission neutron yields,**
- ✓ **^{238}Pu fission, capture and (n,2n) cross-sections,**
- ✓ **^{240}Pu fission neutron yield,**
- ✓ **^{241}Pu fission and elastic scattering cross-sections,**
- ✓ **^{56}Fe neutron capture, elastic and inelastic scattering cross-sections,**
- ✓ **^{55}Mn neutron capture cross section,**
- ✓ **^{209}Bi neutron capture and (n,2n) cross-sections**

Calculation of sensitivity profiles: model and tools

SCALE-6.1

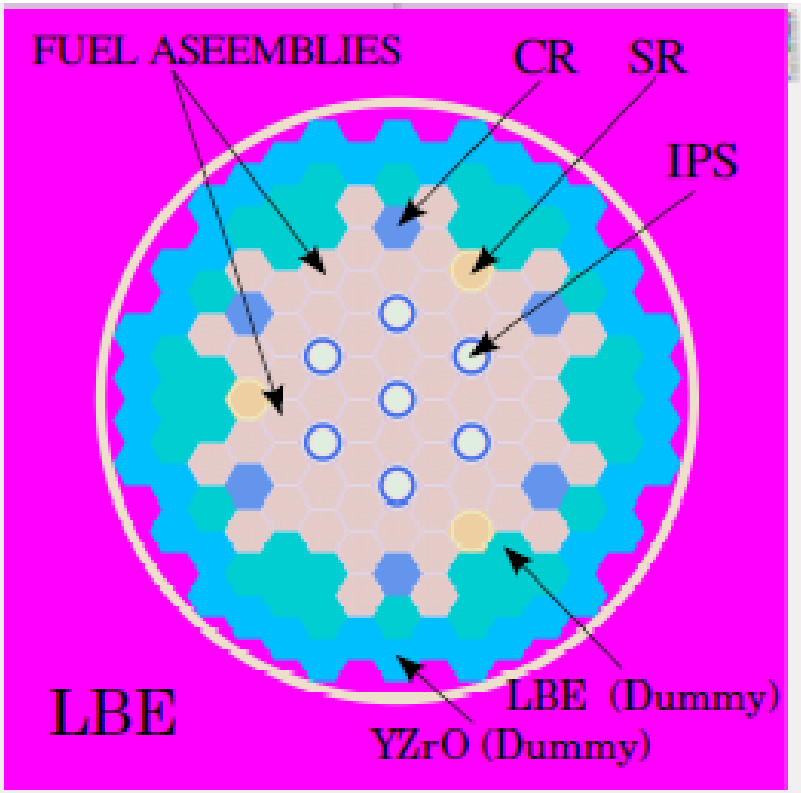
TSUNAMI-3D

KENO-VI: Forward + adjoint transport
SAMS: sensitivity coefficient generation

$$\sigma + \bar{\nu} + \chi$$

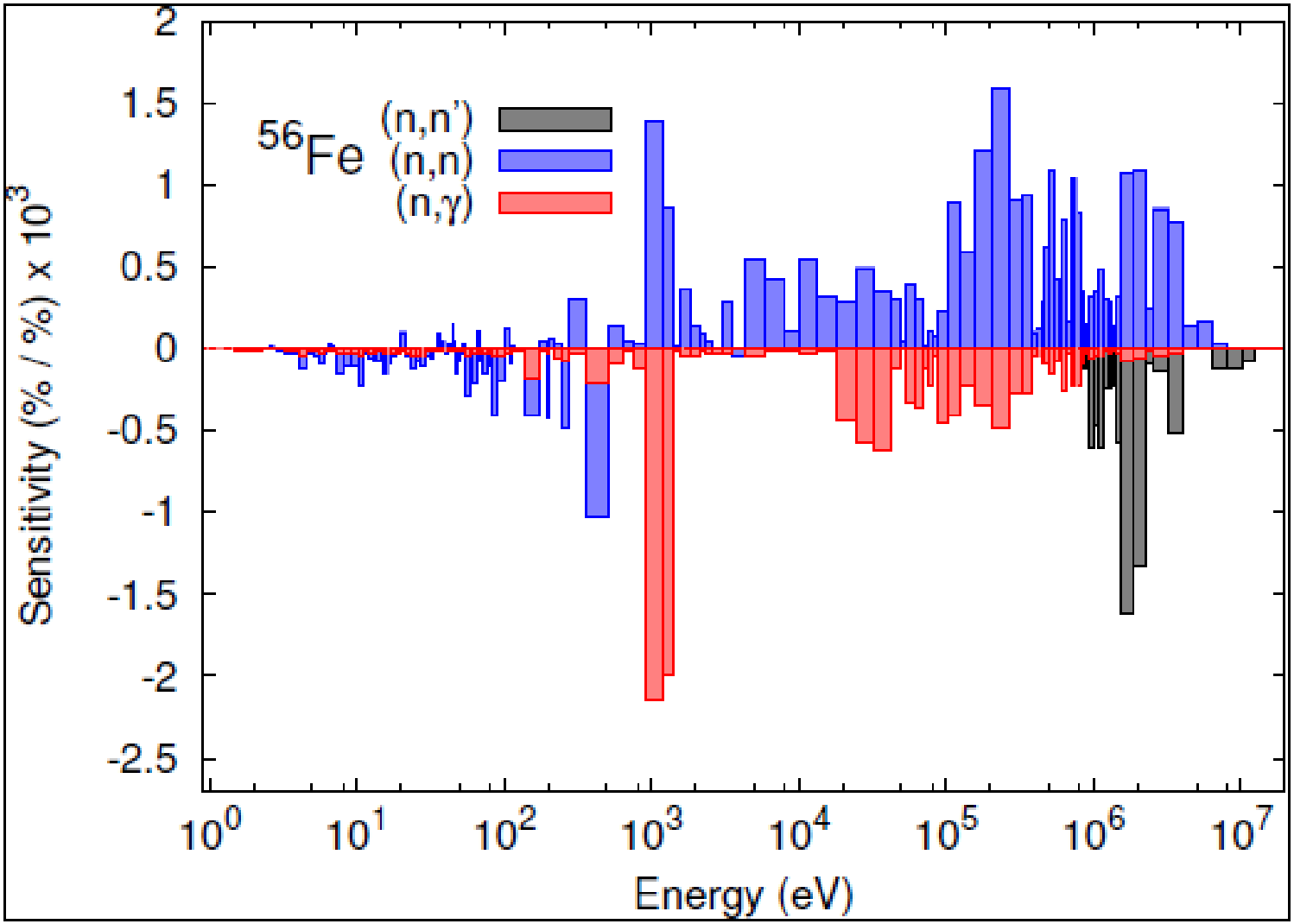
Energy dependence of sensitivity coefficients $\frac{\partial k_{eff}}{k_{eff}} \Big/ \frac{\partial \sigma_i}{\sigma_i}$

Integrated sensitivity coefficients

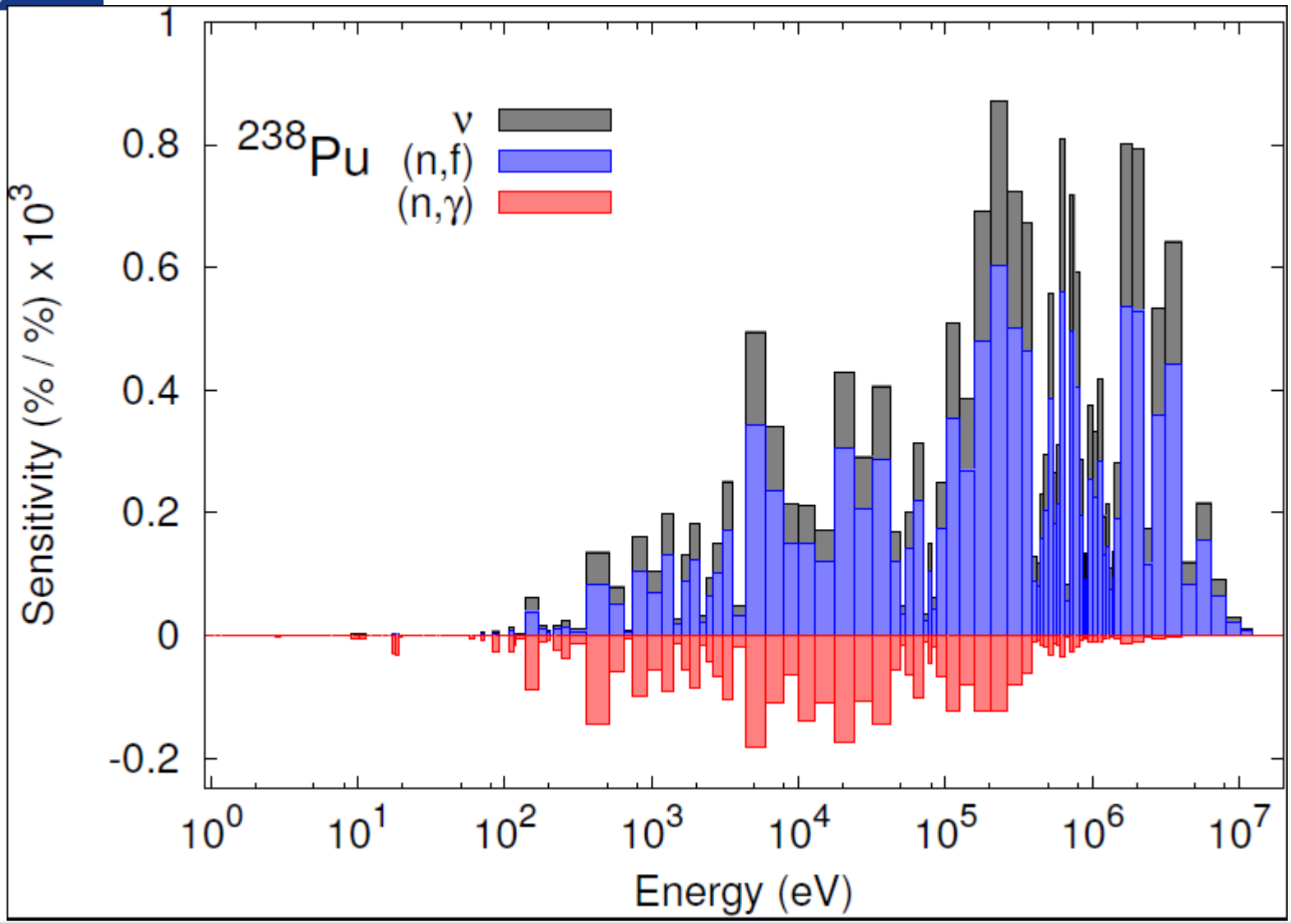


Critical BOL core (FASTEF- CDT FP7 Project)
57 FA detailed modelling,
rest SA homogenized (LBE dummy and reflector assemblies)

Sensitivity profiles: ^{56}Fe



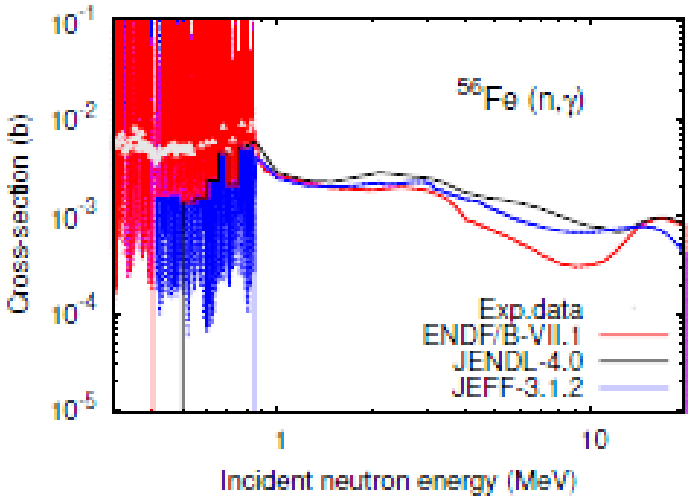
Sensitivity profiles: ^{238}Pu

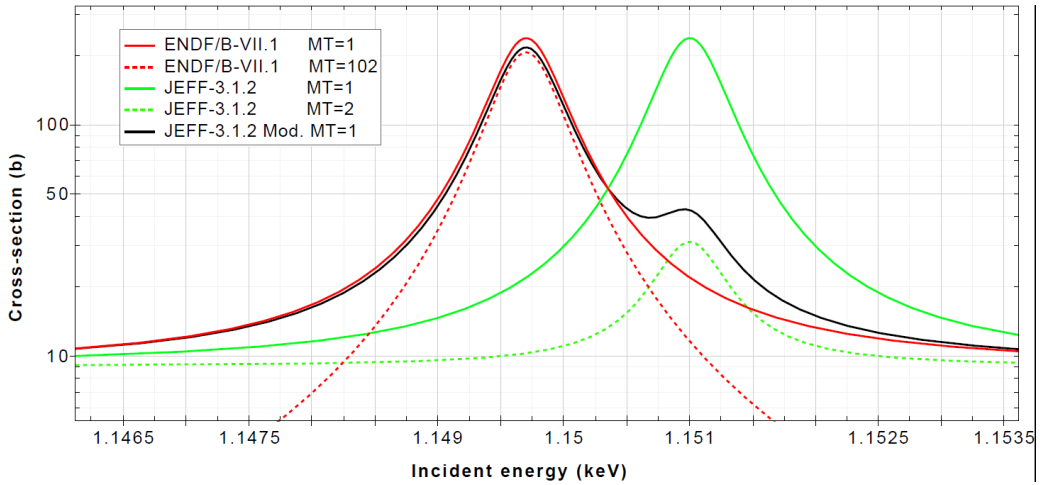


Integrated Sensitivity Coefficients

Isotope	Reaction	Int. Sens. Coef.	Std. dev (1σ), %
^{239}Pu	$\bar{\nu}$	0.6969	0.01
^{239}Pu	(n,f)	0.4779	0.01
^{241}Pu	$\bar{\nu}$	0.1035	0.01
^{238}U	(n,γ)	-0.1023	0.02
^{240}Pu	$\bar{\nu}$	0.08256	0.01
^{241}Pu	(n,f)	0.07122	0.01
^{238}U	$\bar{\nu}$	0.06128	0.02
^{239}Pu	(n,γ)	-0.05764	0.03
^{240}Pu	(n,f)	0.05577	0.02
^{209}Pu	(n,n)	0.05039	0.30
^{238}U	(n,f)	0.03767	0.04
^{240}Pu	(n,γ)	-0.02564	0.03
^{56}Fe	**(n,n)**	**0.02215**	**1.28**
^{56}Fe	**(n,γ)**	**-0.01409**	**0.02**
^{56}Fe	**(n,n')**	**-0.00733**	**0.96**
^{238}Pu	**$\bar{\nu}$**	**0.01961**	**0.01**

Comparison of ^{56}Fe data





Main differences:

- *Number of resonances* JEFF-3.1.2 > JENDL-4.0 > ENDF/B-VII.1
- *Different resonance parameters*
- *Background (n,γ) cross section* ENDF/B-VII.1 and JENDL-4.0

History of evaluations:

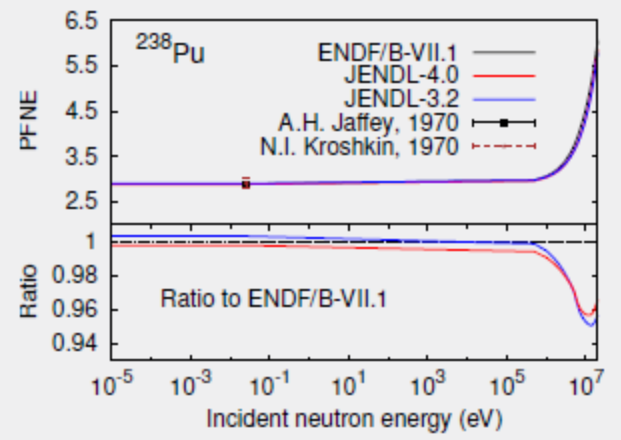
- ENDF/B-VII.1 ← ENDF/B-VI.1 (1991)
- JENDL-4.0 ← JENDL-3.3 (re-evaluated JENDL-3.2 in 2000)
- JEFF-3.1.2 ← EFF 3.1 (2001)

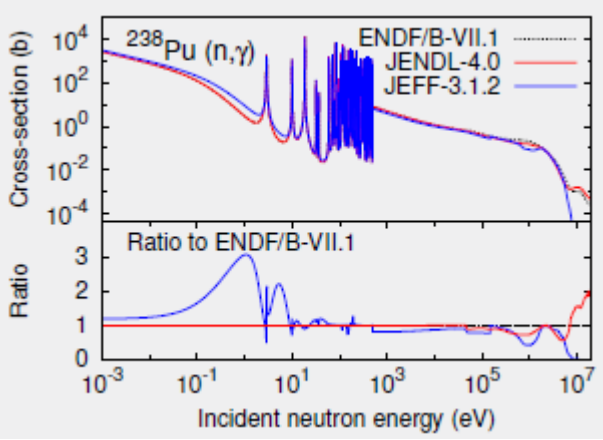
Comparison of ^{56}Fe data

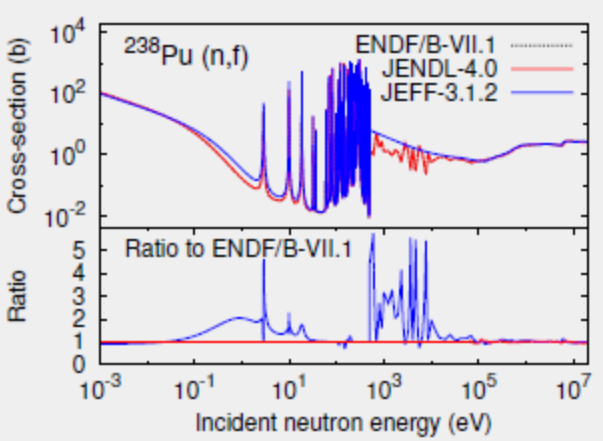
MCNPX 2.7.0 calculations
Base library: JEFF-3.1.2

^{56}Fe file	Modified part	k_{eff}	Δk_{eff} (pcm)
JEFF-3.1.2	-	1.05374±0.00008	-
ENDF/B-VII.1	-	1.05224±0.00008	-150
JEFF-3.1.2	(n,n')	1.05343±0.00007	-31
JEFF-3.1.2	(n;γ)	1.05275±0.00008	-99

Comparison of ^{238}Pu data







Main differences:

- *Different thermal cross section values to adjust RRR*
- *URR parameters for cross section & self-shielding (JEFF-3.1.2)*
- *URR parameters for self-shielding only (JENDL-4.0 & ENDF/B-VII.1)*

History of evaluations:

- ENDF/B-VII.1 ← JENDL-4.0 (< 60 keV)+ new evaluation (> 60 keV, 2011)
- JENDL-4.0 ← new evaluation (2010)
- JEFF-3.1.2 ← JENDL-3.2 (1989) + BROND-2.2 (URR 0.5-150 keV)

Comparison of ^{238}Pu data

MCNPX 2.7.0 calculations
Base library: JEFF-3.1.2

^{238}Pufile	Modified part	k_{eff}	Δk_{eff} (pcm)
JEFF-3.1.2	-	1.05363±0.00008	-
ENDF/B-VII.1	-	1.05178±0.00008	-185
JEFF-3.1.2	$\bar{\nu}$	1.05378±0.00008	+15
JEFF-3.1.2	(n;f)	1.05201±0.00008	-162
JEFF-3.1.2	(n,γ)	1.05319±0.00008	-44

Summary

- Priority list to improve uncertainty data has been created
- Sensitivity analysis of keff to ^{56}Fe and ^{238}Pu data has been done
- ^{56}Fe: review of background cross section and resonance parameters → new evaluation from CIELO ?
- ^{238}Pu: ENDF/B-VII.1 seems to be more reliable while JEFF-3.1.2 underestimates neutron capture and overestimates fission due to URR parameters
- S/U analysis to be continued in the framework of CHANDA FP7 project sponsored by EC (kick-off 1 December 2013):

WP10 "Development of nuclear data for Myrrha reactor safety analyses"

Nuclear data required for the development, safety assessment and licensing of MYRHHA will be studied and recommendations for improvements will be given. This work package will provide support to evaluation projects, such as JEFF, by identifying issues in current nuclear data files for MYRRHA-relevant elements and isotopes

K_{eff} S/U analysis to be complemented by S/U analysis for other safety parameters (reactivity coefficients, delayed neutrons…)

MYRRHA: EXPERIMENTAL ACCELERATOR DRIVEN SYSTEM
A pan-European, innovative and unique facility at Mol (BE)