DE LA RECHERCHE À L'INDUSTRIE

UTILISATION DES EXPÉRIENCES INTÉGRALES POUR LES INCERTITUDES SUR LES DONNÉES NUCLÉAIRES ÉVALUÉES

Workshop NACRE « incertitudes »

CEA Cadarache, 11-12 Juin 2018

Gilles Noguere, Pascal Archier, David Bernard, Pierre Leconte,
Cyrille De Saint Jean, Jean Tommasi

DEN/DER/SPRC/LEPh

Context

Two main schools of thought

- some users would like to have adjusted covariances that will give small uncertainties in the calculated integral observables

- others want unadjusted libraries that they can adjust for their particular class of applications

⇒ **How far we can go with the use of integral parameters in the evaluation process so that the evaluated library is still a "general purpose library" (see IAEA report INDC(NDS)-0746)**

Outlines

☐ Long standing issue

☐ « Clean integral experiment »

☐ Generic mathematical framework in CONRAD

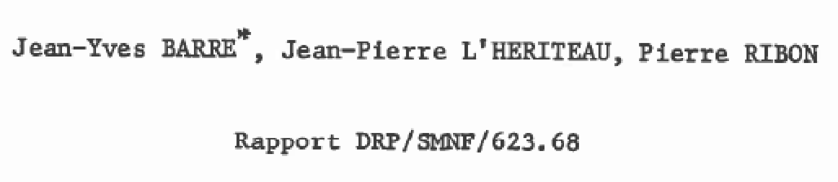
☐ New Pu239 evaluation
⇒ **Integral Data Assimilation of the CERES data for MOX fuel calculations**

☐ Evaluation of the cumulative fission yields as a function of the neutron energy
⇒ **Integral Data Assimilation of the PROFIL trends**

Long standing issue

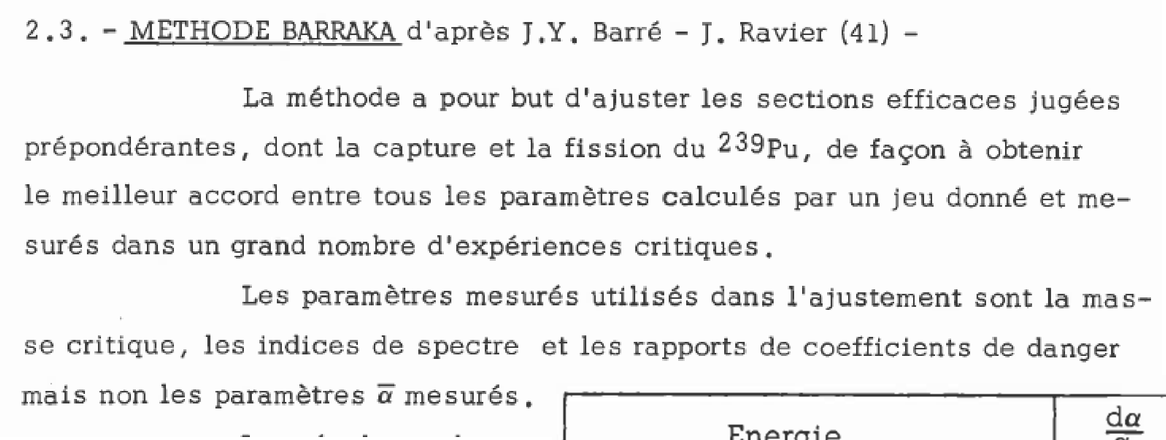
Integral Data Assimilation (IDA) exists since the early years of the nuclear data evaluation activities

1968 ⇒ IDA method,called BARRAKA, applied to determine the capture-to-fission ratio of Pu239 (CEA Cadarache)



Jean-Yves BARRE*, Jean-Pierre L'HERITEAU, Pierre RIBON

Rapport DRP/SMNF/623.68

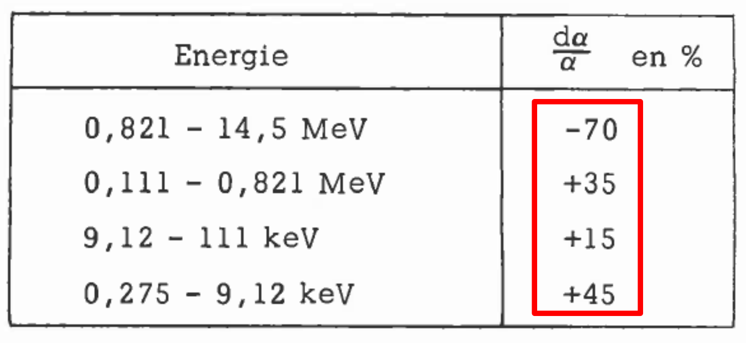


2.3. - METHODE BARRAKA d'après J.Y. Barré - J. Ravier (41) -

La méthode a pour but d'ajuster les sections efficaces jugées prépondérantes, dont la capture et la fission du ^{239}Pu, de façon à obtenir le meilleur accord entre tous les paramètres calculés par un jeu donné et mesurés dans un grand nombre d'expériences critiques.

Les paramètres mesurés utilisés dans l'ajustement sont la masse critique, les indices de spectre et les rapports de coefficients de danger mais non les paramètres $\bar{\alpha}$ mesurés.

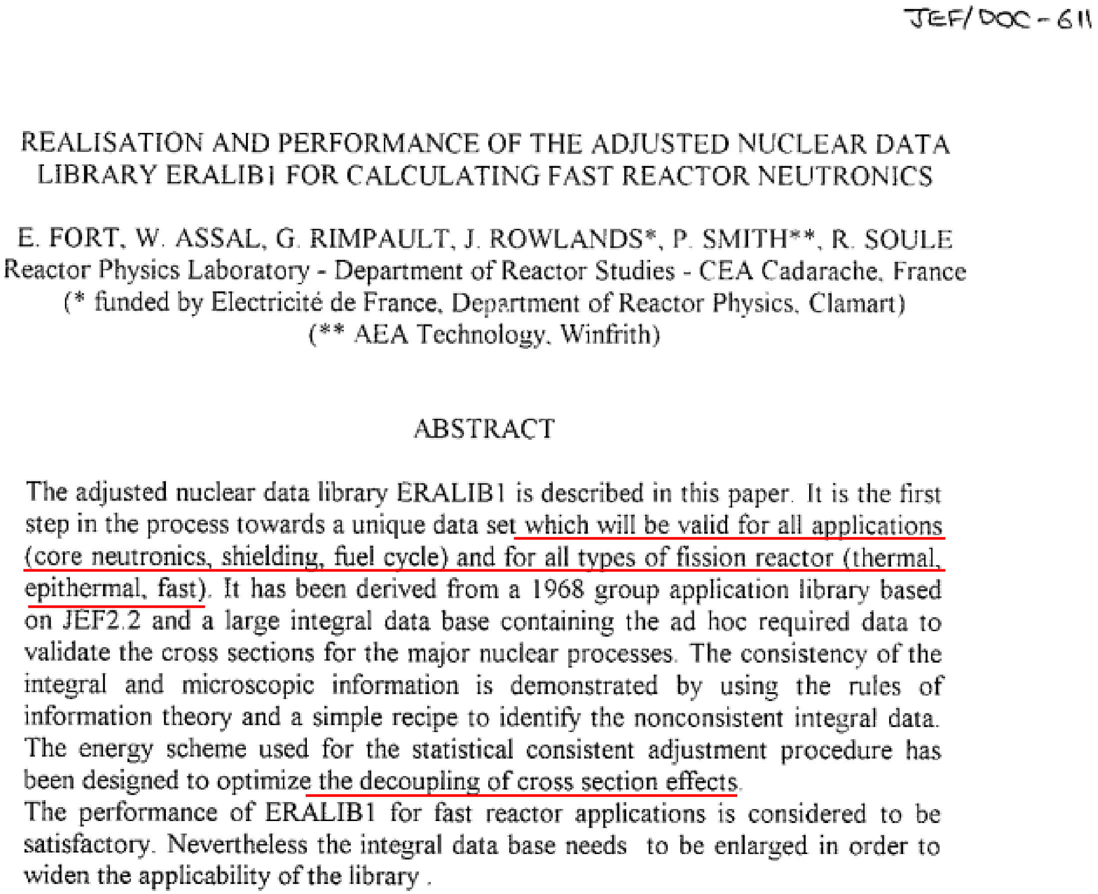
IDA is usefull to identify major problems in existing nuclear data



Energie	$\frac{d\alpha}{\alpha}$ en %
0,821 - 14,5 MeV	-70
0,111 - 0,821 MeV	+35
9,12 - 111 keV	+15
0,275 - 9,12 keV	+45

Long standing issue

1996 ⇒ Production of the ERALIB-1 application library for Fast Reactors by using a standard least-squares method (CEA Cadarache)



JEF/DOC-611

REALISATION AND PERFORMANCE OF THE ADJUSTED NUCLEAR DATA LIBRARY ERALIB1 FOR CALCULATING FAST REACTOR NEUTRONICS

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Reactor Physics Laboratory - Department of Reactor Studies - CEA Cadarache, France
(* funded by Electricité de France, Department of Reactor Physics, Clamart)
(** AEA Technology, Winfrith)

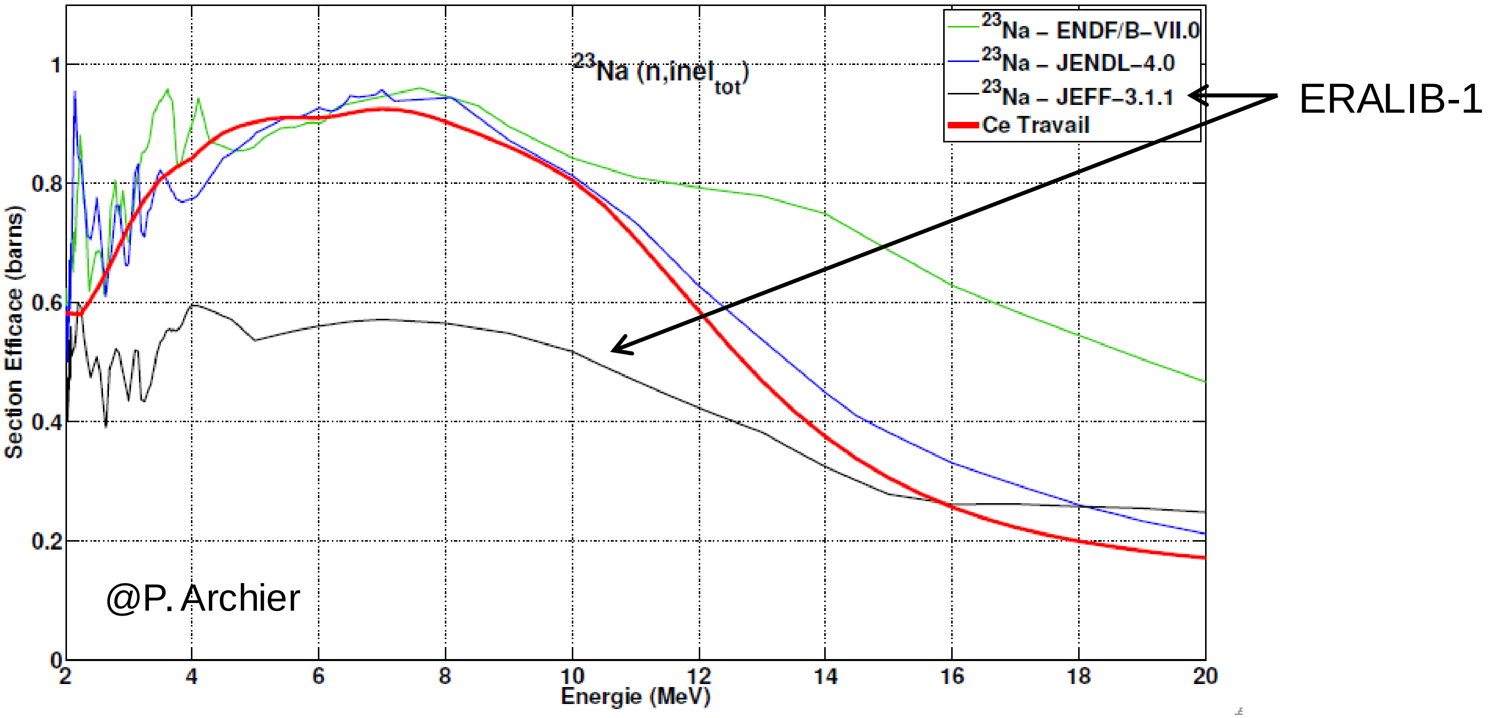
ABSTRACT

The adjusted nuclear data library ERALIB1 is described in this paper. It is the first step in the process towards a unique data set which will be valid for all applications (core neutronics, shielding, fuel cycle) and for all types of fission reactor (thermal, epithermal, fast). It has been derived from a 1968 group application library based on JEF2.2 and a large integral data base containing the ad hoc required data to validate the cross sections for the major nuclear processes. The consistency of the integral and microscopic information is demonstrated by using the rules of information theory and a simple recipe to identify the nonconsistent integral data. The energy scheme used for the statistical consistent adjustment procedure has been designed to optimize the decoupling of cross section effects.
The performance of ERALIB1 for fast reactor applications is considered to be satisfactory. Nevertheless the integral data base needs to be enlarged in order to widen the applicability of the library .

Long standing issue

1996 ⇒ Production of the ERALIB-1 application library for Fast Reactors by using a standard least-squares method (CEA Cadarache)

However, new evaluation of Na23 (only based on nuclear models) confirms the incorrect (n,n') cross section established in the frame of ERALIB-1 (ajustement of integral data)

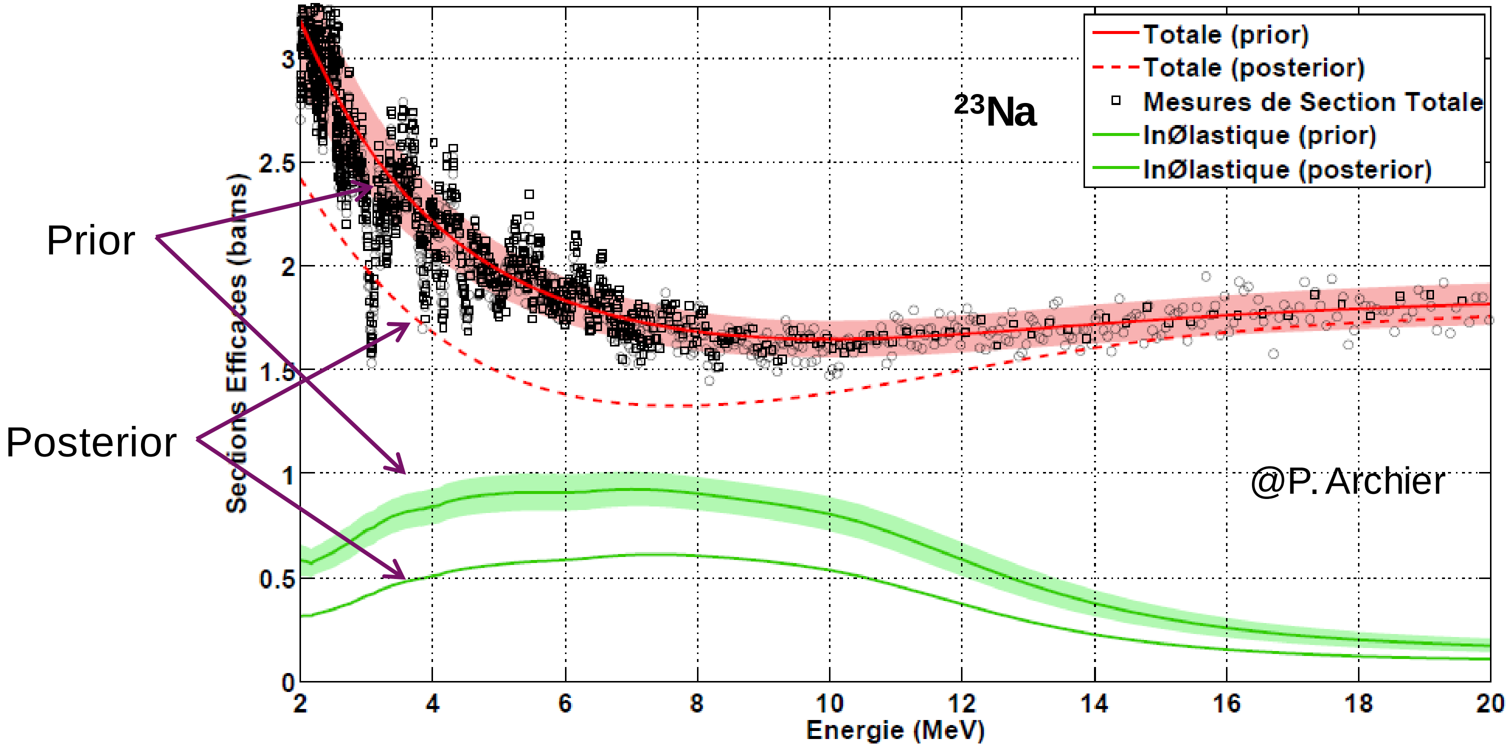


@P. Archier

Long standing issue

1996 ⇒ Production of the ERALIB-1application library for Fast Reactors by using a standard least-squares method (CEA Cadarache)

It was also possible to reproduce the « mistake » observed in ERALIB-1 (by ajusting the same integral data) ⇒ the problem mainly comes from the lack of **constraints** in the fitting procedure



Prior

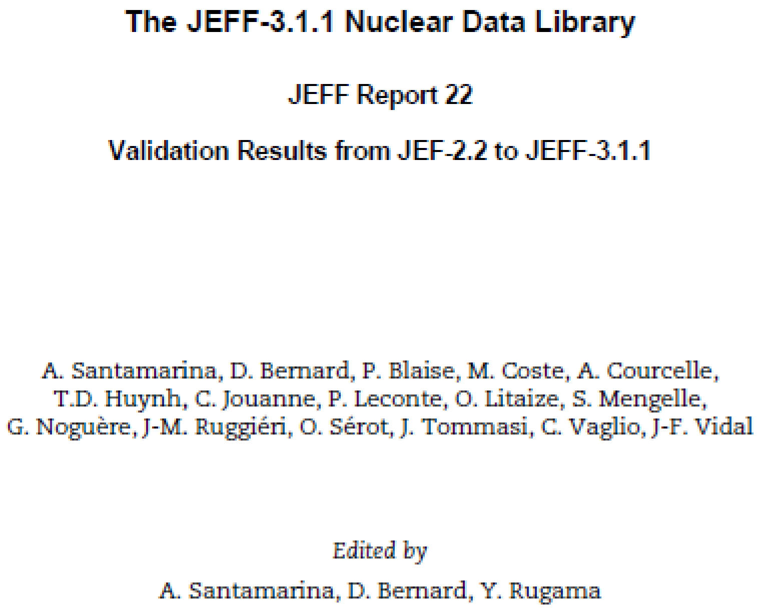
Posterior

@P. Archier

Long standing issue

2005 ⇒ Release of the JEFF-3.1.1 library

Integral trends (coming from mock-up reactors and power reactors) were considered as « <u>decision-making support tool</u> » to get an integral feedback (experimental validation) for improving the nuclear data of interest for reactors applications ⇒ good performances for Light Water Reactors, but **compensation between nuclear data not solved**



The JEFF-3.1.1 Nuclear Data Library

JEFF Report 22

Validation Results from JEF-2.2 to JEFF-3.1.1

A. Santamarina, D. Bernard, P. Blaise, M. Coste, A. Courcelle,
T.D. Huynh, C. Jouanne, P. Leconte, O. Litaize, S. Mengelle,
G. Noguère, J-M. Ruggiéri, O. Sérot, J. Tommasi, C. Vaglio, J-F. Vidal

Edited by

A. Santamarina, D. Bernard, Y. Rugama

Long standing issue

Since 2010: New working groups were proposed in the frame of the Working Party on International Nuclear Data Evaluation Co-operation (WPEC) in order to solve such difficulties

33 Methods and issues for the combined use of integral experiments and covariance data — Volume 33 (2013)
Co-ordinator: G. Palmiotti and M. Salvatores

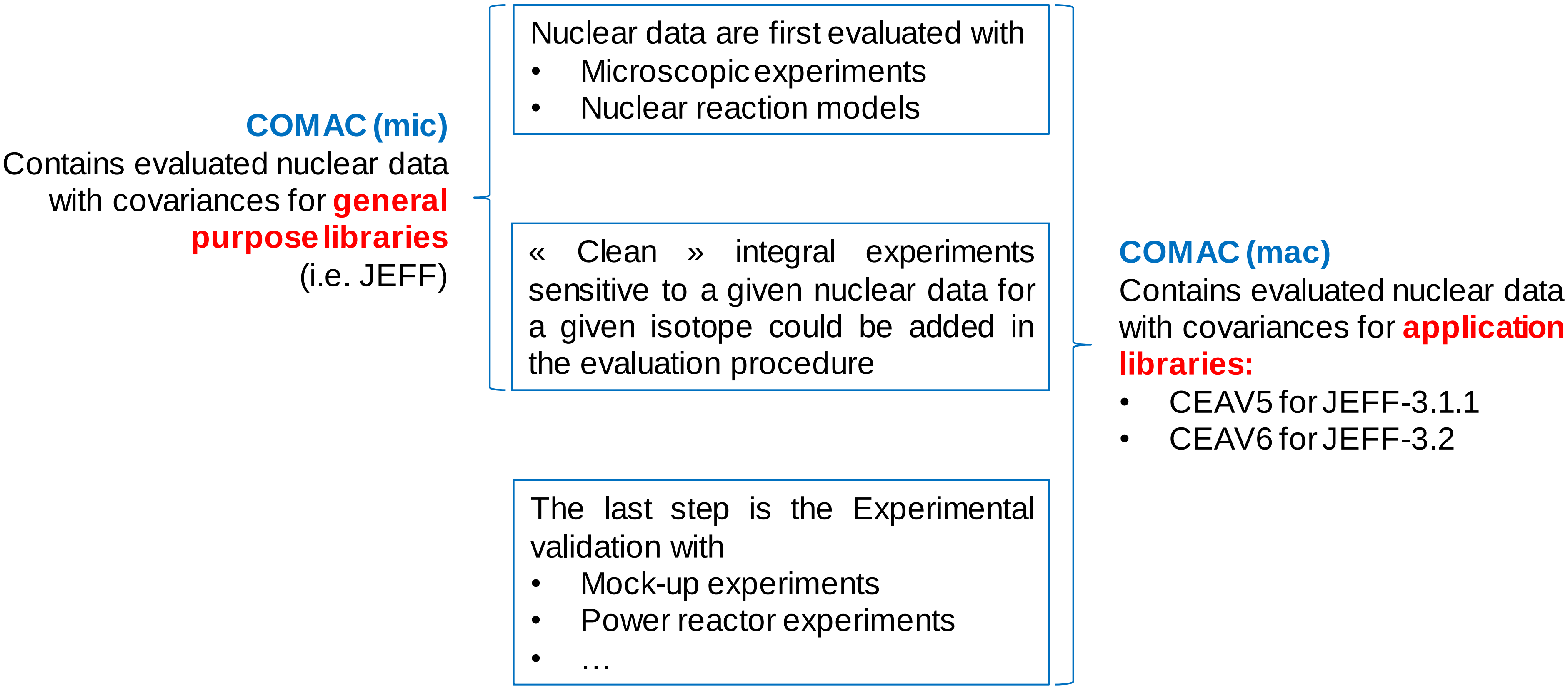
39 Methods and approaches to provide feedback from nuclear and covariance data adjustment for improvement of nuclear data files — Status: Ongoing
Co-ordinator: G. Palmiotti and M. Salvatores

44 Investigation of Covariance Data in General Purpose Nuclear Data Libraries
Co-ordinators: V. Sobes and C.de Saint Jean

46 Efficient and Effective Use of Integral Experiments for Nuclear Data Validation
Co-ordinators: M. Salvatores and G. Palmiotti
Monitors: A. Plompen, M. Herman

Long standing issue

Since 2012: Developpement of the covariance data base **COMAC** (last version COMAC-V2) with the following strategy ⇒ COMAC is structured in two parts



COMAC (mic)
Contains evaluated nuclear data with covariances for **general purpose libraries** (i.e. JEFF)

Nuclear data are first evaluated with
- Microscopic experiments
- Nuclear reaction models

« Clean » integral experiments sensitive to a given nuclear data for a given isotope could be added in the evaluation procedure

The last step is the Experimental validation with
- Mock-up experiments
- Power reactor experiments
- …

COMAC (mac)
Contains evaluated nuclear data with covariances for **application libraries:**
- CEAV5 for JEFF-3.1.1
- CEAV6 for JEFF-3.2

⇒ Not always easy to apply this strategy and to respect such an ideal separation between « general purpose library » and « application library » !

Outlines

- ☐ Long standing issue
- ☐ « Clean integral experiment »
- ☐ Generic mathematical framework in CONRAD
- ☐ New Pu239 evaluation

⇒ **Integral Data Assimilation of the CERES data for MOX fuel calculations**

- ☐ Evaluation of the cumulative fission yields as a function of the neutron energy

⇒ **Integral Data Assimilation of the PROFIL trends**

« Clean » integral experiment

How to define a « clean » integral experiment in order to avoid overlap with the « mic » and « mac » steps ?

A large variety of integral benchmarks exists ⇒ **ICSBEP** is widely used in nuclear evaluation

Only few ICSBEP benchmarks can be used **with care** for evaluation, such as the famous GODIVA, JEZEBEL, BIGTEN, FLATTOP, …

$$k_{eff} \equiv \frac{\nu\sigma_f}{\sigma_a} \quad \Rightarrow$$

Highly sensitive to PFNS, (n,n'), H($H_2$0)…

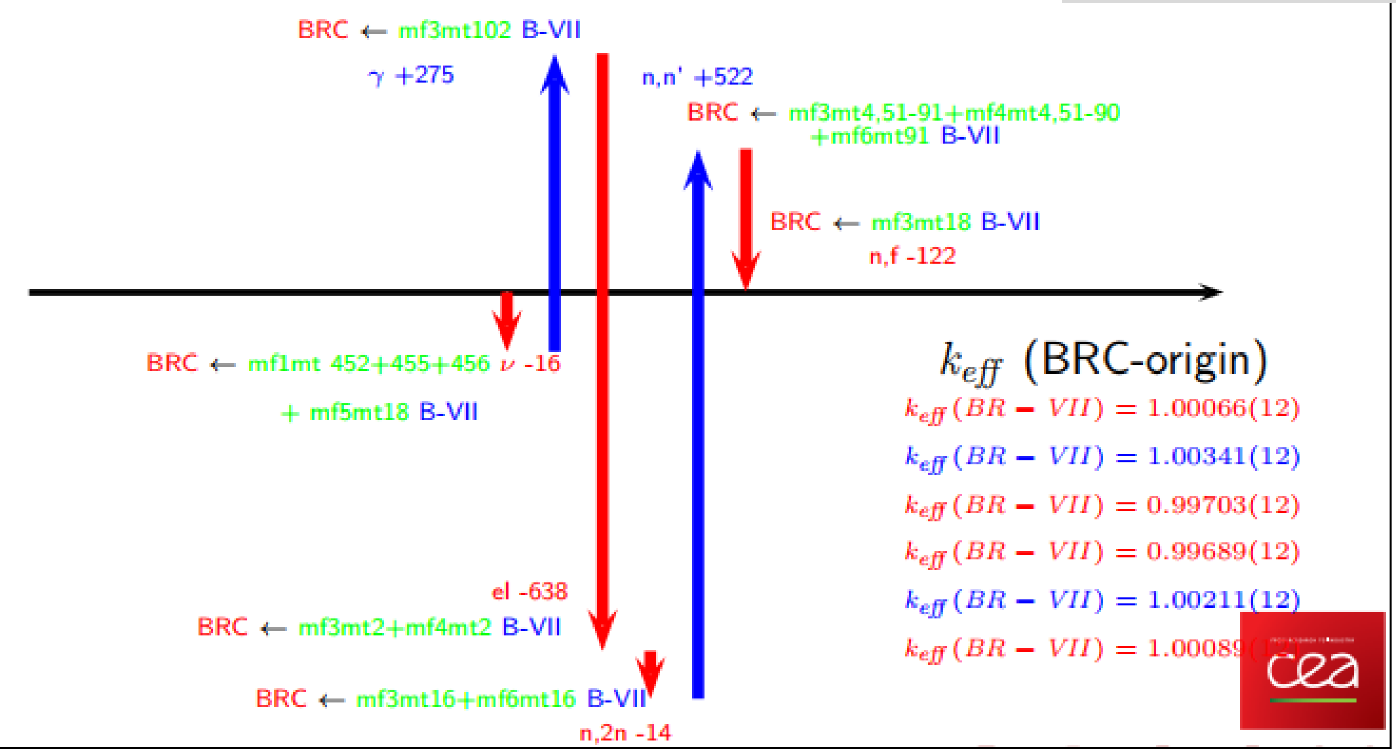
ICSBEP are sensitive to various nuclear data ⇒ neutron multiplicities are often used as adjustable parameters

« Clean » integral experiment

Exemple of compensation between nuclear data

MCNP study of the JEZEBEL critical benchmark

$k_{eff}(BRC) = 1.00082(11)$ $k_{eff}(B\text{-}VII) = 1.00060(12)$

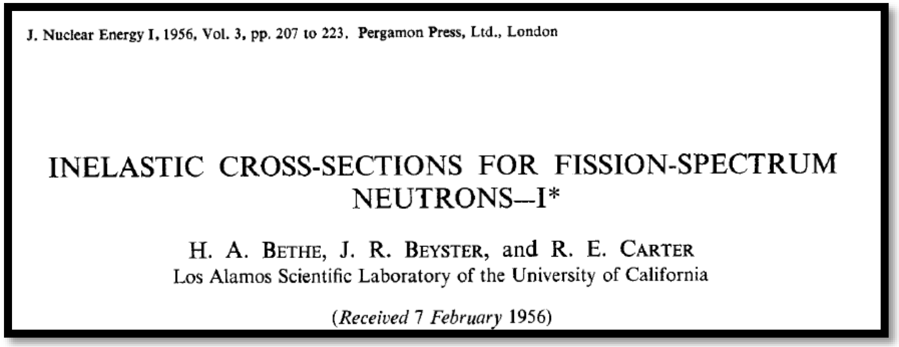


Strong compensation between (n,n) and (n,n') reactions
⇒ Dedicated experiments and needed to quantify each contribution

H. Duarte, B. Morillon, P. Romain, JEFDOC-1485

« Clean » integral experiment

The contribution of the inelastic cross section is a well known problematic

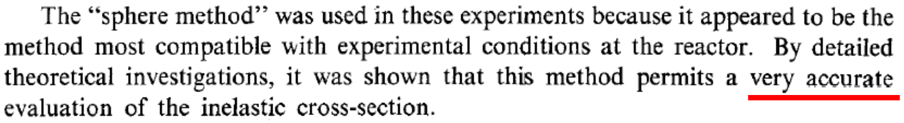


J. Nuclear Energy I, 1956, Vol. 3, pp. 207 to 223. Pergamon Press, Ltd., London

INELASTIC CROSS-SECTIONS FOR FISSION-SPECTRUM NEUTRONS—I*

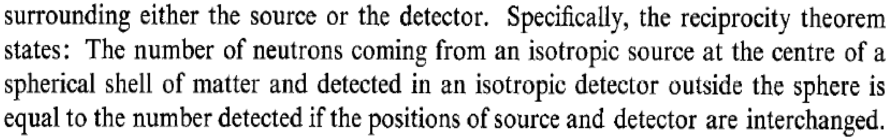
H. A. Bethe, J. R. Beyster, and R. E. Carter
Los Alamos Scientific Laboratory of the University of California

(*Received 7 February* 1956)



The "sphere method" was used in these experiments because it appeared to be the method most compatible with experimental conditions at the reactor. By detailed theoretical investigations, it was shown that this method permits a very accurate evaluation of the inelastic cross-section.

Optimistic statement !



surrounding either the source or the detector. Specifically, the reciprocity theorem states: The number of neutrons coming from an isotropic source at the centre of a spherical shell of matter and detected in an isotropic detector outside the sphere is equal to the number detected if the positions of source and detector are interchanged.

Complementary integral trends are needed to avoid « **compensation » between nuclear data**, otherwize the obtained evaluated file is not compatible with a « general purpose library »

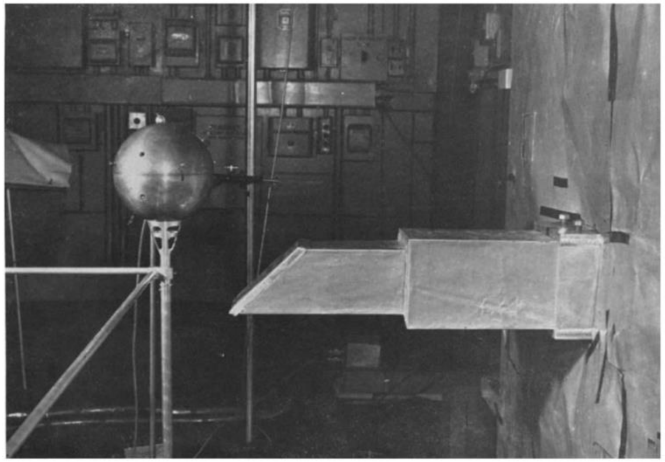


Fig. 9.2.—External collimator with source and a typical 8-inch (outside diameter) sphere on its supporting cone.

« Clean » integral experiment

Several experimental programs were performed in the CEA and European facilities (MINERVE, EOLE, MELUSINE, RAPSODIE, MASURCA, CALIBAN, GELINA) and French Power Reactors (PHENIX, PWR) for improving **reactor parameter calculations** and **nuclear data**

⇒ Integral benchmarks of interest are those which provide information <u>on a given isotope and nuclear reaction</u>

« Clean » integral experiment

Exemples of CEA programs :

- **Oscillation experiments** in MINERVE reactor (CEA Cadarache) such as BUC (fission product), OSMOSE (actinides), OCEAN (absorbant), MAESTRO (structural materials), CERES-Pu and CNA (Mox fuel)

  ⇒ Provide Thermal Neutron Constants measured in well caracterised thermal neutron flux

- **Post-Irradiation Experiments of separate isotopes** such as PROFIL programs (fast reactor PHENIX of CEA Marcoule), TACO experiments (fast reactor RAPSODIE of CEA Cadarache) and ICARE-R, ICARE-R and SHERWOOD (thermal reactor MELUSINE of CEA Grenoble)

  ⇒ Provide effective cross sections and fission yields

- **Post-Irradiated fuel Experiments in PWR reactor** can be used with care for evaluation

⇒ **How to use these experimental results in the evaluation procedure ?**

Outlines

- ❑ Long standing issue
- ❑ « Clean integral experiment »
- ❑ Generic mathematical framework in CONRAD
- ❑ New Pu239 evaluation

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Generic mathematical framework in CONRAD

« Variance Penalty » introduced by Muir:

D.W . Muir, « The contribution of individual correlated parameters to the uncertainty of integral quantities», Nucl. Inst. Meth. A 644, 55 (2011)

The model parameters and the full covariance matrix has to be partitioned as:

$$\mu = \begin{pmatrix} x \\ \theta \end{pmatrix} \qquad \Sigma = \begin{pmatrix} \Sigma_{11} & \Sigma_{12} \\ \Sigma_{21} & \Sigma_{22} \end{pmatrix}$$

By definition, the observable parameters x are the « **passive** » parameters (nuclear data ajusted on microscopic and/or integral data) and the latent or nuissance parameters θ are the « **active** » parameters (experimental corrections, model parameters with known uncertainties)

Generic mathematical framework in CONRAD

The « zero variance penalty » condition lead to*

$$\Sigma = \begin{pmatrix} \Sigma_{11} & \Sigma_{12} \\ \Sigma_{21} & \Sigma_{22} \end{pmatrix} \qquad \begin{cases} \Sigma_{11} = M_x + \left(G_x^T G_x\right)^{-1} G_x^T G_\theta M_\theta G_\theta{}^T G_x \left(G_x^T G_x\right)^{-1} \\ \Sigma_{12} = -(G_x^T G_x)^{-1} G_x^T G_\theta M_\theta \\ \Sigma_{22} = M_\theta \end{cases}$$

For a vector z of general dimension k, the derivative matrix $G=(G_x,G_\theta)$ of the quantity z to the parameters x and θ is defined as:

$$G_x = \begin{pmatrix} \frac{\partial z_1}{\partial x_1} & \cdots & \frac{\partial z_1}{\partial x_n} \\ \vdots & \ddots & \vdots \\ \frac{\partial z_k}{\partial x_1} & \cdots & \frac{\partial z_k}{\partial x_n} \end{pmatrix} \qquad G_\theta = \begin{pmatrix} \frac{\partial z_1}{\partial \theta_1} & \cdots & \frac{\partial z_1}{\partial \theta_m} \\ \vdots & \ddots & \vdots \\ \frac{\partial z_k}{\partial \theta_1} & \cdots & \frac{\partial z_k}{\partial \theta_m} \end{pmatrix}$$

---

*« non-zero variance penalty » is used to account for a « defect model »

Generic mathematical framework in CONRAD

M_x covariance matrix provided by the fitting procedure
- retroactive analysis
- iterative least-squares method
- Bayesian Monte-Carlo (BMC)

\+ constraints if needed

Expression when only **nuissance parameter uncertainties** (experimental corrections) are « marginalized »

$$\boxed{\begin{cases} \boxed{\Sigma_{11} = \boxed{M_x} + \left(G_x^T G_x\right)^{-1} G_x^T G_\theta M_\theta G_\theta{}^T G_x \left(G_x^T G_x\right)^{-1}} \\ \Sigma_{12} = -(G_x^T G_x)^{-1} G_x^T G_\theta M_\theta \\ \Sigma_{22} = M_\theta \end{cases}}$$

Generalization of the marginalization procedure to any **model parameters with known uncertainties**

$\Rightarrow$ M_θ is the covariance matrix between the nuissance and lattent parameters

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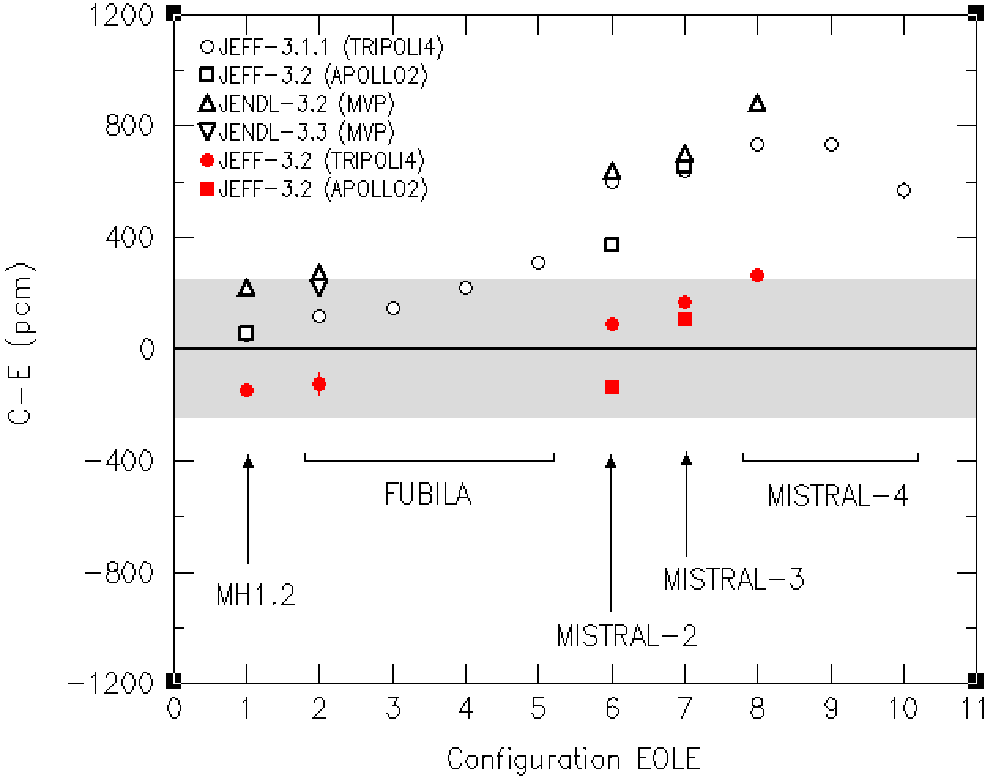
Integral Data Assimilation for MOX fuel calculations

Context ⇒ with JEFF-3.1.1, increasing overestimation of the experimental value with the Pu aging



(CEA Cadarache)

New evaluation of three isotopes

- Am241
- **Pu239**
- Pu240

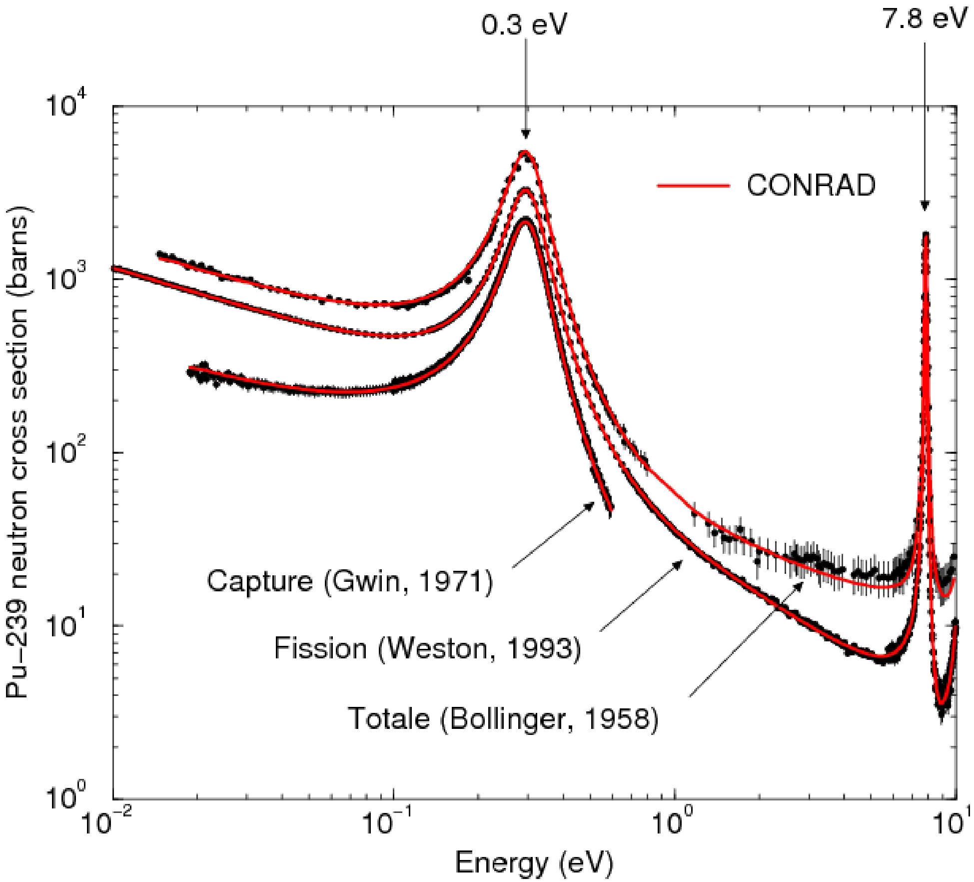
New evaluation of the resonance range of Pu239

Time-of-flight data available in EXFOR (E<2500 eV)

References	Energie (eV)	Installation	Mesures
Bollinger et al. (1956)	**0.01 – 1.0**	**Chopper**	**Section efficace totale**
Gwin et al. (1971)	**0.01 – 0.5**	**ORELA**	**Capture at 25.6 m**
Gwin et al. (1976)	1.0 – 100.0	ORELA	Capture at 40.0 m
Gwin et al. (1984)	0.01 – 20.0	ORELA	Fission at 8 m
Weston et al. (1984)	9.0 – 2500.0	ORELA	Fission at 18.9 m
Weston et al. (1988)	100.0 – 2500.0	ORELA	Fission at 86 m
Weston et al. (1993)	**0.02 – 40.0**	**ORELA**	**Fission at 18.9 m**
Wagemans et al. (1988)	0.002 – 20.0	GELINA	Fission at 8 m
Wagemans et al. (1993)	0.01 – 1000.0	GELINA	Fission at 8 m
Harvey et al. (1985)	0.7 – 30.0	ORELA	Transmission at 18 m
Harvey et al. (1985)	30.0 – 2500.0	ORELA	Transmission at 80 m

New evaluation of the resonance range of Pu239

Reich-Moore analysis with the CONRAD code

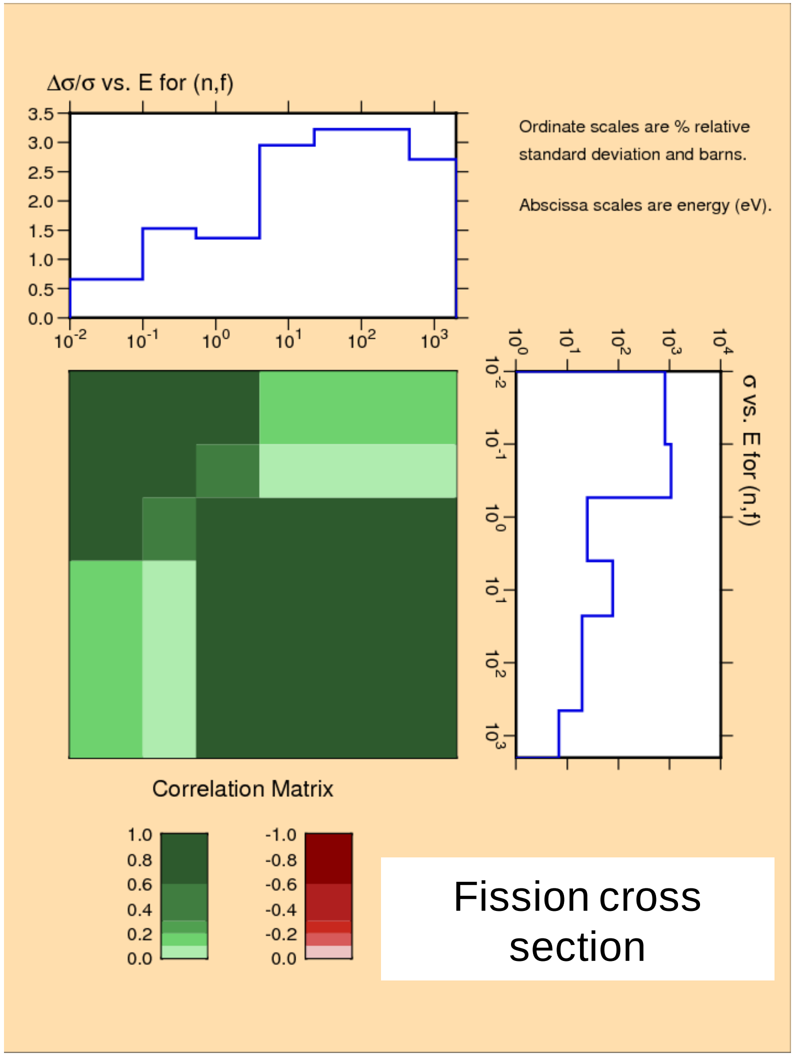


CONRAD is a nuclear reaction model code developped at CEA Cadarache

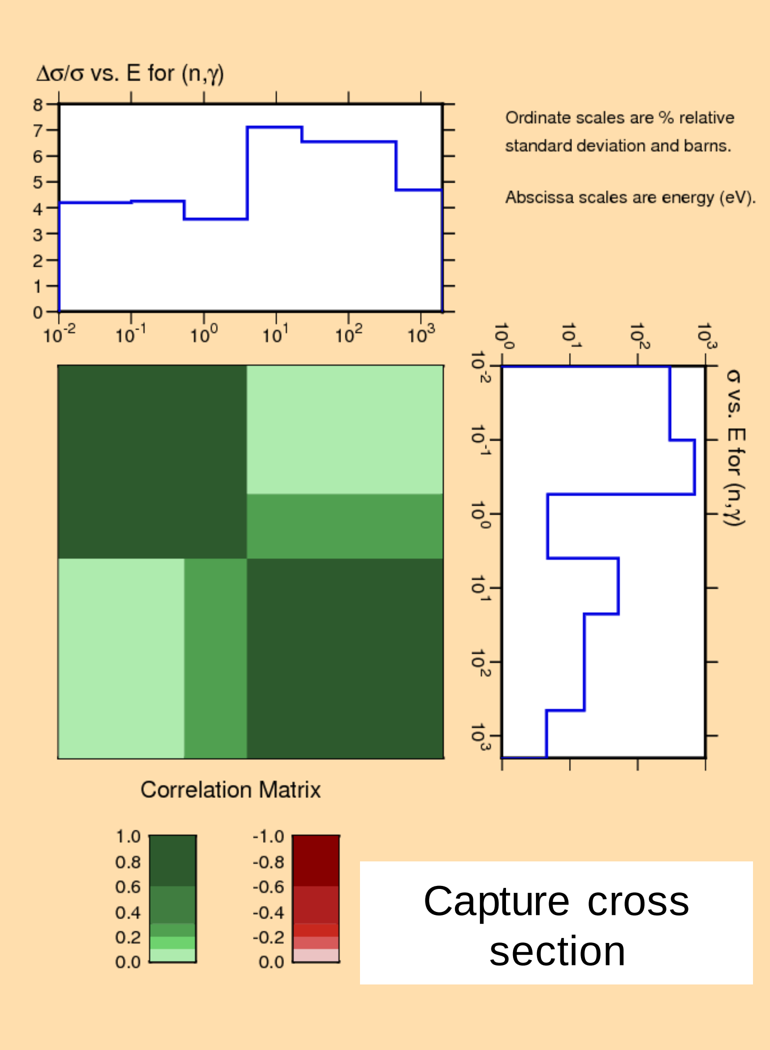
Resonance Parameter Covariance Matrix (**RPCM**) are determined by using the Marginalization technique to propagate the incertainties of the experimental corrections (normalization, background, resolution function, temperature, sample composition)

New evaluation of the resonance range of Pu239

Multi-group covariance matrices obtained from the RPCM



Fission cross section

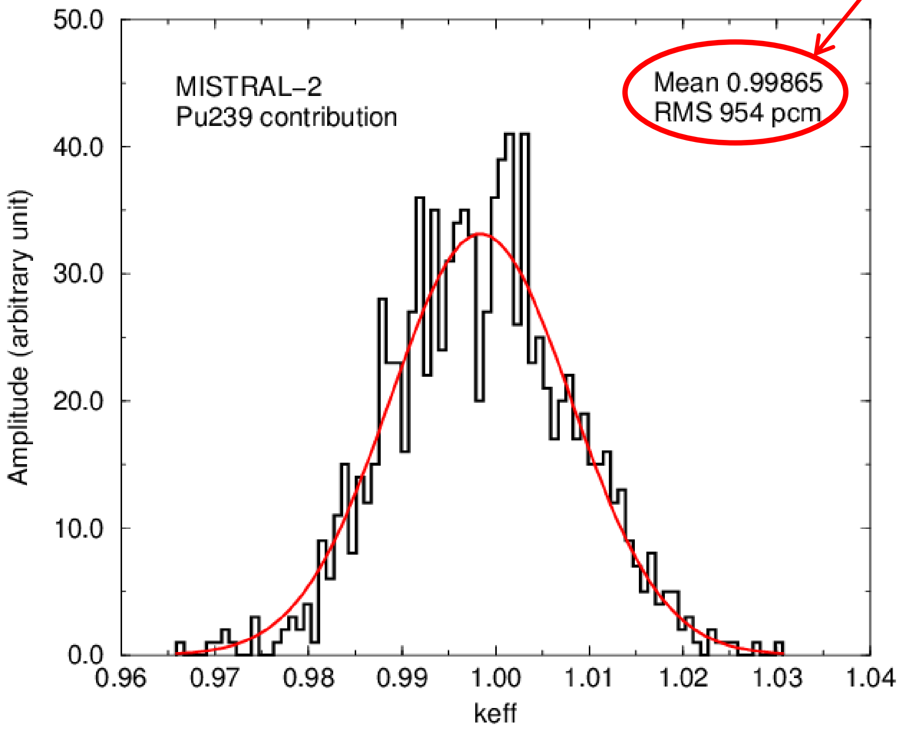


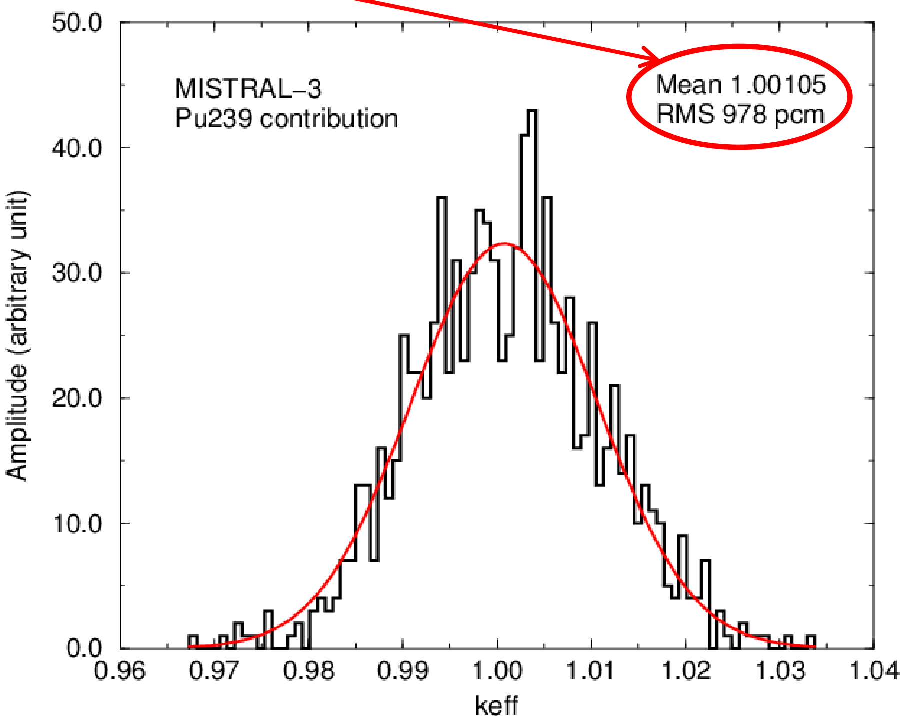
Capture cross section

Propagation of the Pu239 uncertainties

Propagation of the Pu239 resonance parameter uncertainties on EOLE benchmarks

Final uncertainty (**≈1000 pcm**) ⇒ dominated by the Pu239 capture cross section uncertainties



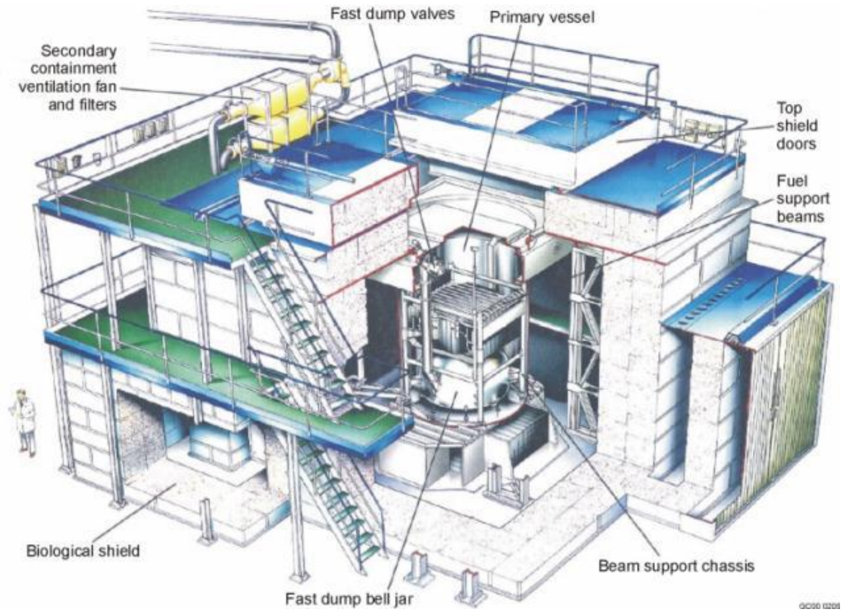


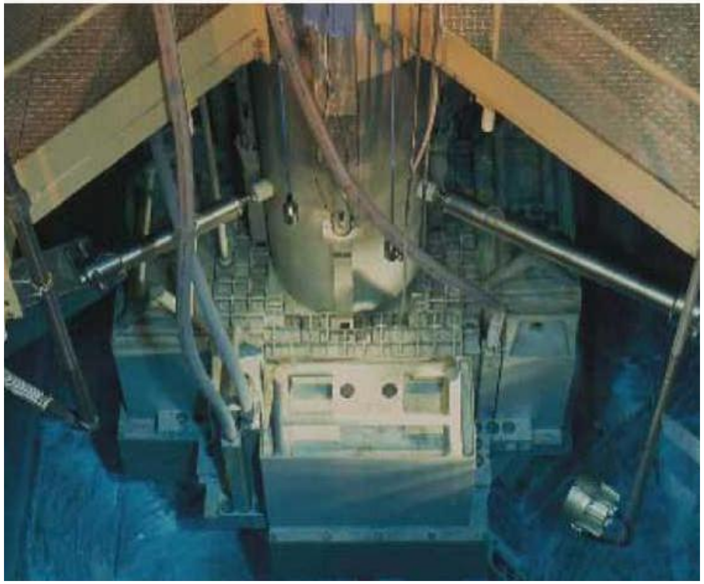
⇒ How to reduce these huge uncertainties ?

Experimental validation of the new Pu239 evaluation

Experimental program CERES (P. Leconte, PHYSOR 2014)

Oscilation measurements of 12 MOX samples in the **DIMPLE** (AEA Winfrith) and **MINERVE** (CEA Cadarache) reactors (from 1992 to 1995)





Experimental validation of the new Pu239 evaluation

@ P. Leconte

Sample	Résultats CERES Full Monte Carlo Method C/E-1 (%)	
	Assembly-I	**Assembly-III**
MOX1	3.0±4.9	10.9±6.6
MOX2	0.7±3.2	5.7±4.0
MOX3	0.8±1.7	2.9±1.8
MOX4	-0.7±1.5	0.1±1.3
MOX5	-1.1±1.3	-1.3±1.1
MOX6	-1.5±1.2	
Pu0403	-2.4±1.5	-2.9±2.4
Pu0413	-2.9±1.7	-5.6±2.6
Pu0426	-6.6±1.6	-8.1±3.1
Pu2003	1.1±1.4	-0.7±1.3
Pu2013	0.4±1.4	-0.8±1.3
Pu2026	1.1±1.4	2.1±1.5
Mean Value	**-1.0 ± 0.5**	**-0.4 ± 0.5**

Reactivity breakdown (TOT=100)

^{239}Pu	Σ_a	-9.0
	$\nu\Sigma_f$	**98.5**
^{240}Pu	Σ_a	-0.9
^{241}Pu	Σ_a	-1.0
	$\nu\Sigma_f$	12.3

^{239}Pu	Σ_a	-17.8
	$\nu\Sigma_f$	**118.0**
^{240}Pu	Σ_a	-0.3
^{241}Pu	Σ_a	
	$\nu\Sigma_f$	0.1

^{239}Pu	Σ_a	-17.9
	$\nu\Sigma_f$	**118.7**
^{240}Pu	Σ_a	-2.1
^{241}Pu	Σ_a	-0.3
	$\nu\Sigma_f$	2.0

Reactivity breakdown (TOT=100)

^{239}Pu	Σ_a	**-88.2**
	$\nu\Sigma_f$	**185.5**
^{240}Pu	Σ_a	-7.9
^{241}Pu	Σ_a	-10.3
	$\nu\Sigma_f$	23.5

^{239}Pu	Σ_a	**-100.8**
	$\nu\Sigma_f$	**202.4**
^{240}Pu	Σ_a	-1.6
^{241}Pu	Σ_a	
	$\nu\Sigma_f$	0.1

^{239}Pu	Σ_a	**-113.6**
	$\nu\Sigma_f$	**228.2**
^{240}Pu	Σ_a	-13.9
^{241}Pu	Σ_a	-1.8
	$\nu\Sigma_f$	3.8

Intégral trend
$\nu\Sigma_f$

Good agreement between C and E

Integral trend
K1=$\nu\Sigma_f$-Σ_a

New Pu239 evaluation
Included in JEFF-3.2

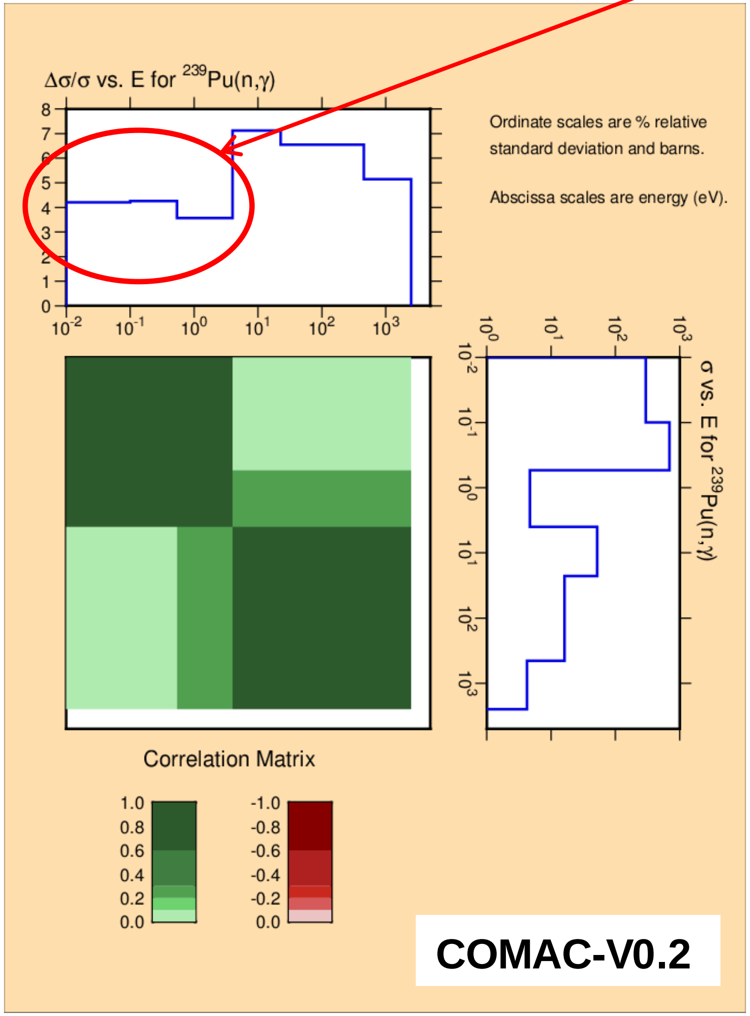
Integral Data Assimilation of the MINERVE benchmarks

Integral Data Assimilation with CONRAD ⇒ provide a sizeable reduction of the uncertainty of the Pu239 capture cross section in the low energy range

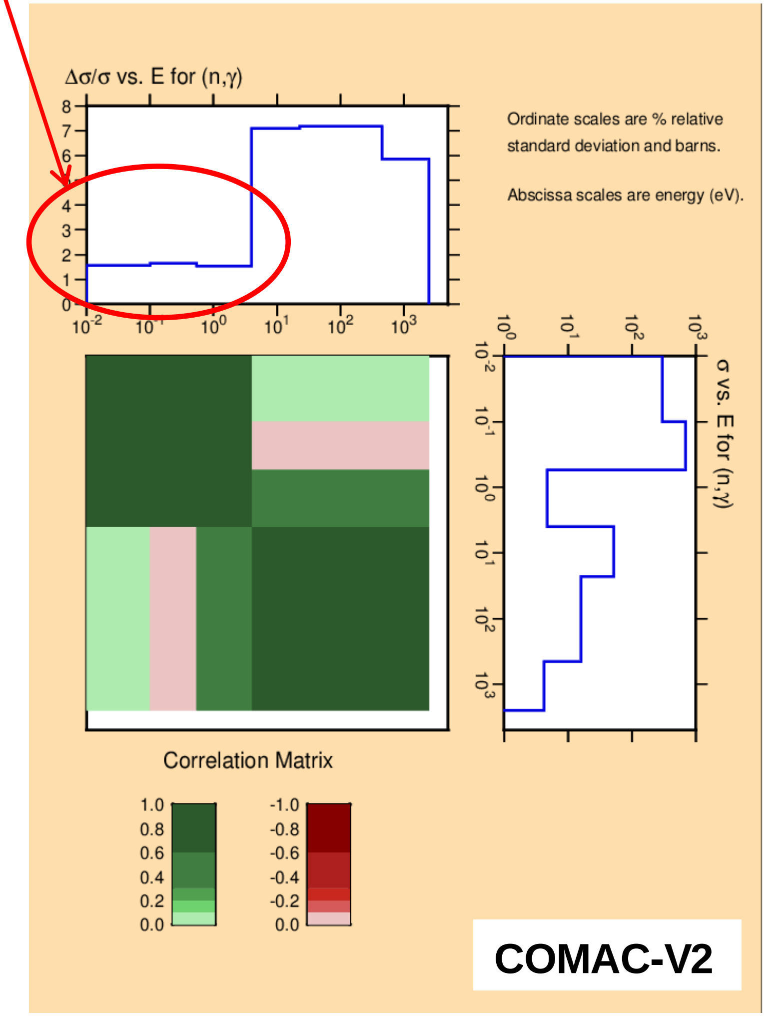
Groupes d'énergie		Incertitudes *a priori* COMAC-V0.2	Incertitudes *a posteriori* Analyse CONRAD		
			CERES-Pu JEFFDOC-1582	**CNA**	**177 PST (ICSBEP)**
11	[22.6 – 454 eV]	6.2%	5.3%	6.4%	3.9%
12	[4 - 22.6 eV]	6.4%	5.4%	7.0%	4.3%
13	[0.53 – 4 eV]	**2.5%**	**0.9%**	**1.9%**	**1.0%**
14	[0.1 – 0.53 eV]	**3.8%**	**1.4%**	**2.4%**	**1.2%**
15	[0.0001 – 0.1 eV]	**4.0%**	**1.5%**	**2.5%**	**1.2%**

Integral Data Assimilation of the MINERVE benchmarks

Significant reduction of the Pu239 capture cross section uncertainties at low neutron energy



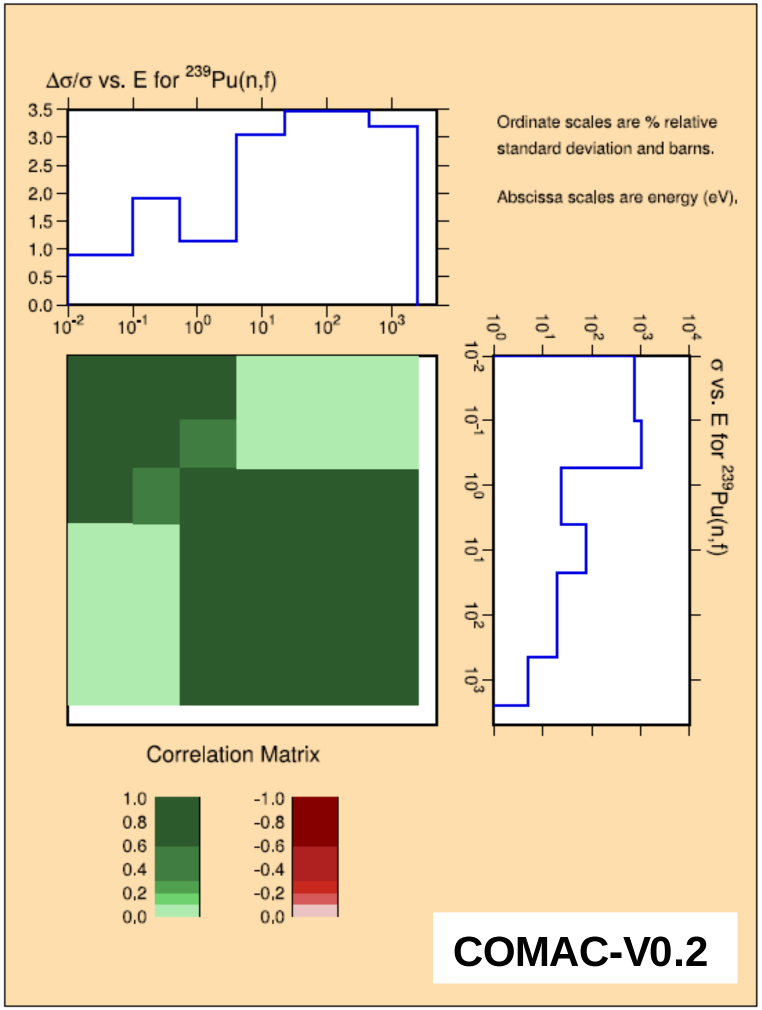
COMAC-V0.2



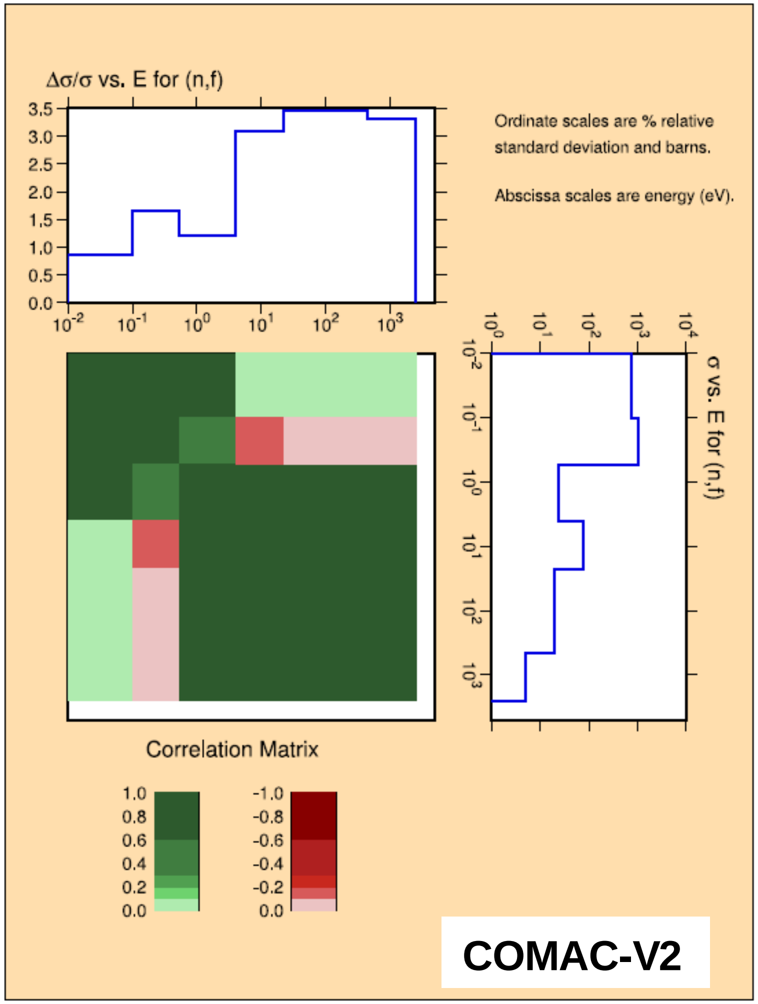
COMAC-V2

Integral Data Assimilation of the MINERVE benchmarks

No modification of the Pu239 fission cross section uncertainties



COMAC-V0.2



COMAC-V2

Integral Data Assimilation of the MINERVE benchmarks

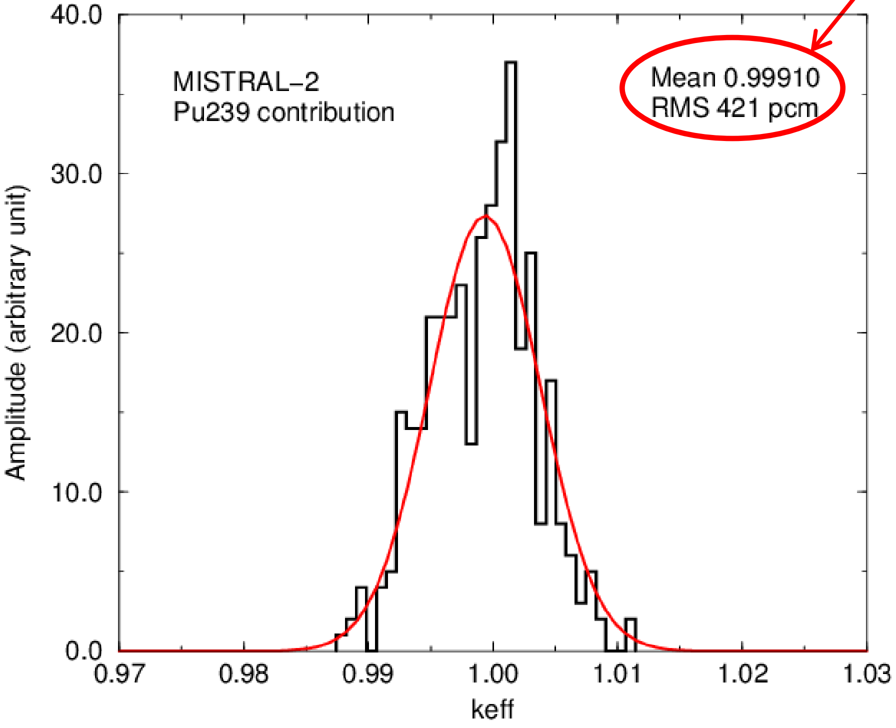
Final uncertainties after the Integral Data Assimilation of the CERES program

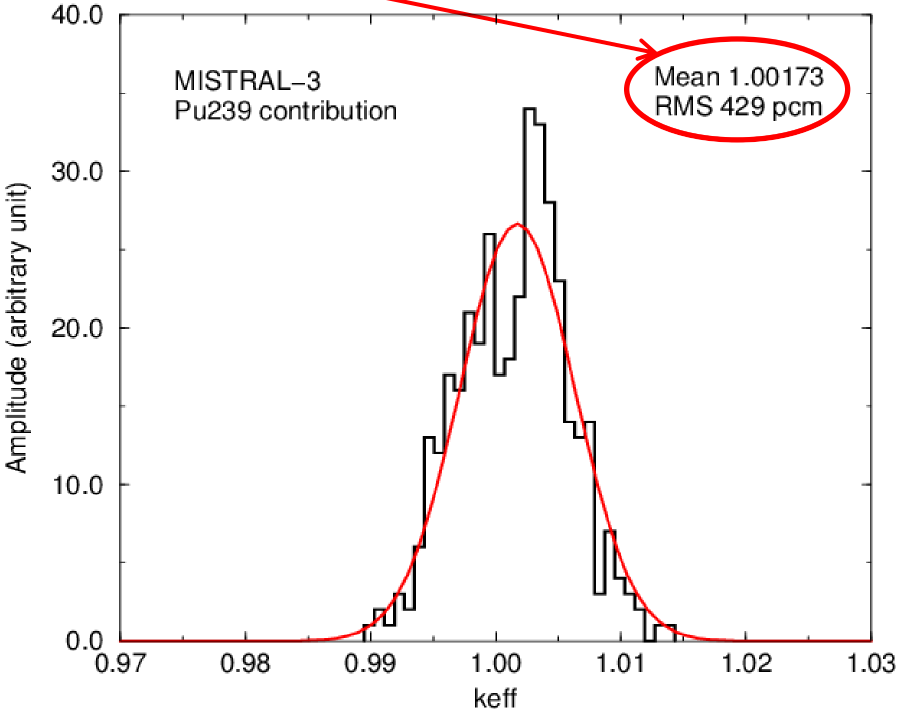
	JEFF-3.2 (=SG34)	Relative uncertainty	
		JEFF-3.2	**COMAC-V2**
σ_f	747.2 barns	0.9%	0.7%
σ_γ	**270.1 barns**	**4.4%**	**1.6%**
I_f	308.8 barns	2.3%	2.3%
I_γ	180.1 barns	5.7%	5.7%
K1	1161.5 barns	1.7%	0.9%

Propagation of the Pu239 uncertainties

Propagation of the final Pu239 resonance parameter uncertainties on EOLE benchmarks

Final uncertainty (**≈400 pcm**) after the Integral Data Assimilation of CERES





Outlines

- ☐ Long standing issue
- ☐ « Clean integral experiment »
- ☐ Generic mathematical framework in CONRAD
- ☐ New Pu239 evaluation

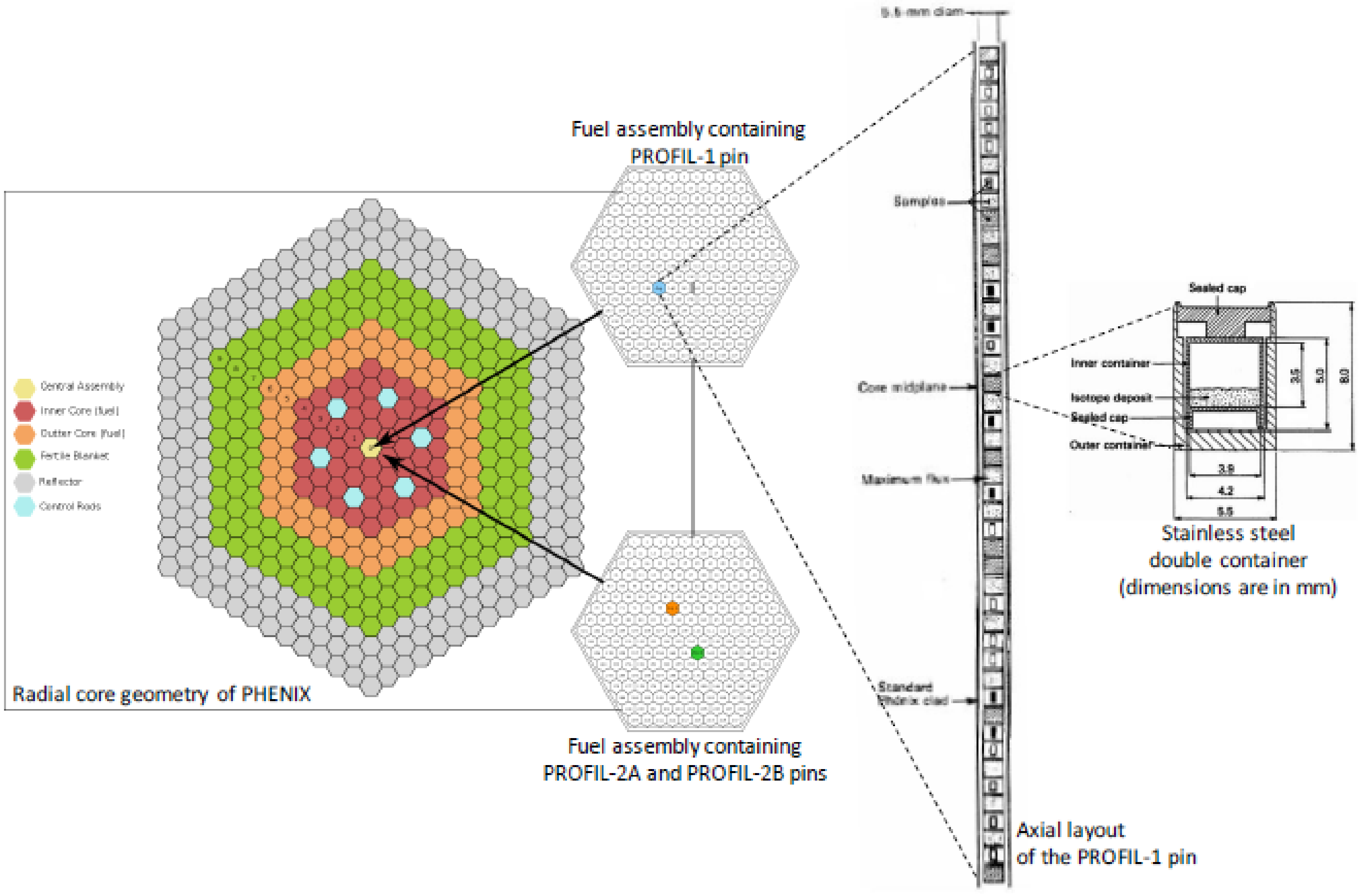
⇒ **Integral Data Assimilation of the CERES data for MOX fuel calculations**

- ☐ Evaluation of the cumulative fission yields as a function of the neutron energy

⇒ **Integral Data Assimilation of the PROFIL trends**

PROFIL experiments

Interpretation of the PROFIL experiments performed in the fast reactor PHENIX provides valuable integral trends on neutron cross sections and fission yields (Tommasi,NSE 160,232, 2008)



Fuel assembly containing PROFIL-1 pin

Radial core geometry of PHENIX

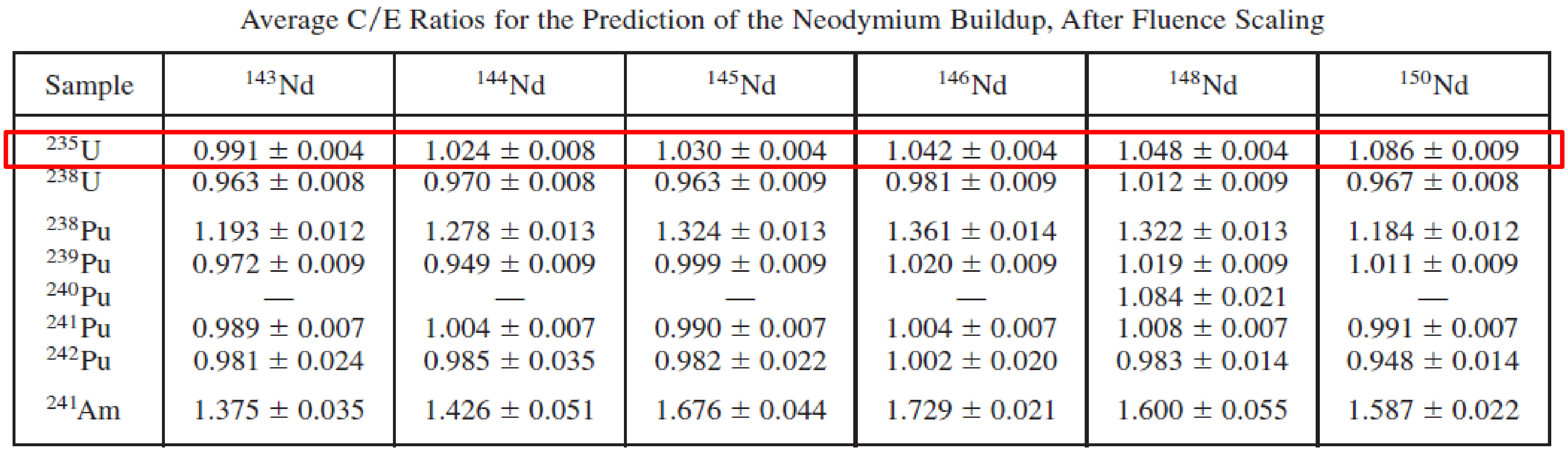
Fuel assembly containing PROFIL-2A and PROFIL-2B pins

Stainless steel double container (dimensions are in mm)

Axial layout of the PROFIL-1 pin

PROFIL experiments

Interpretation of the PROFIL experiments performed in the fast reactor PHENIX provides valuable integral trends on neutron cross sections and fission yields (Tommasi,NSE 160,232, 2008)



Average C/E Ratios for the Prediction of the Neodymium Buildup, After Fluence Scaling

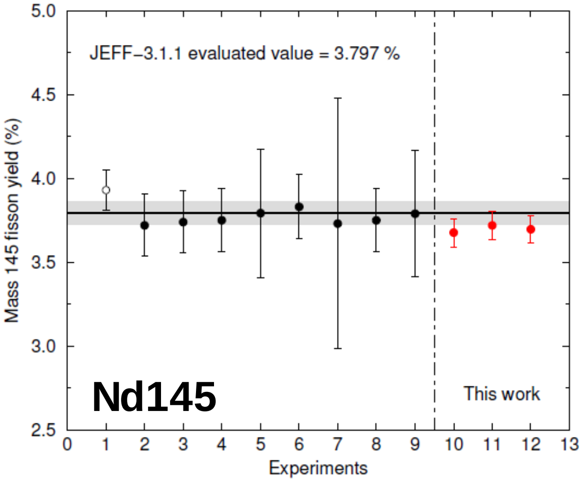
Sample	^{143}Nd	^{144}Nd	^{145}Nd	^{146}Nd	^{148}Nd	^{150}Nd
^{235}U	0.991 ± 0.004	1.024 ± 0.008	1.030 ± 0.004	1.042 ± 0.004	1.048 ± 0.004	1.086 ± 0.009
^{238}U	0.963 ± 0.008	0.970 ± 0.008	0.963 ± 0.009	0.981 ± 0.009	1.012 ± 0.009	0.967 ± 0.008
^{238}Pu	1.193 ± 0.012	1.278 ± 0.013	1.324 ± 0.013	1.361 ± 0.014	1.322 ± 0.013	1.184 ± 0.012
^{239}Pu	0.972 ± 0.009	0.949 ± 0.009	0.999 ± 0.009	1.020 ± 0.009	1.019 ± 0.009	1.011 ± 0.009
^{240}Pu	—	—	—	—	1.084 ± 0.021	—
^{241}Pu	0.989 ± 0.007	1.004 ± 0.007	0.990 ± 0.007	1.004 ± 0.007	1.008 ± 0.007	0.991 ± 0.007
^{242}Pu	0.981 ± 0.024	0.985 ± 0.035	0.982 ± 0.022	1.002 ± 0.020	0.983 ± 0.014	0.948 ± 0.014
^{241}Am	1.375 ± 0.035	1.426 ± 0.051	1.676 ± 0.044	1.729 ± 0.021	1.600 ± 0.055	1.587 ± 0.022

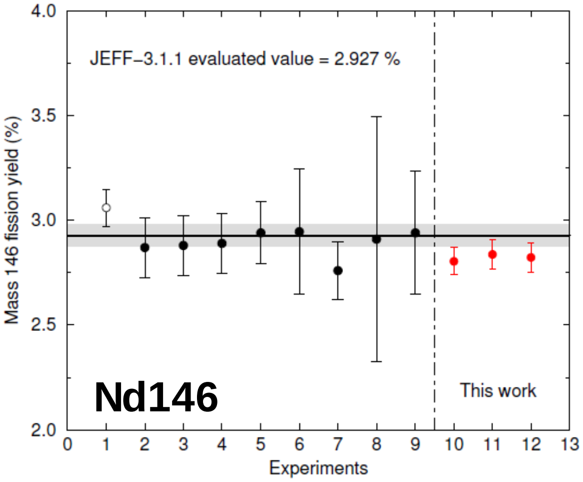
Presentation of the results obtained for the nuclear system U235+n

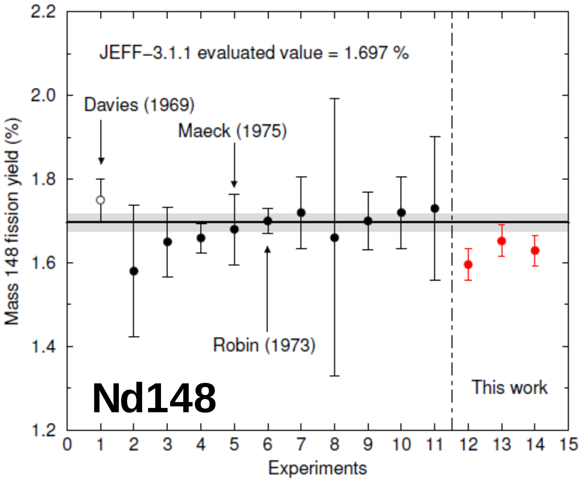
Effective cumulative fission yields : fast energy range

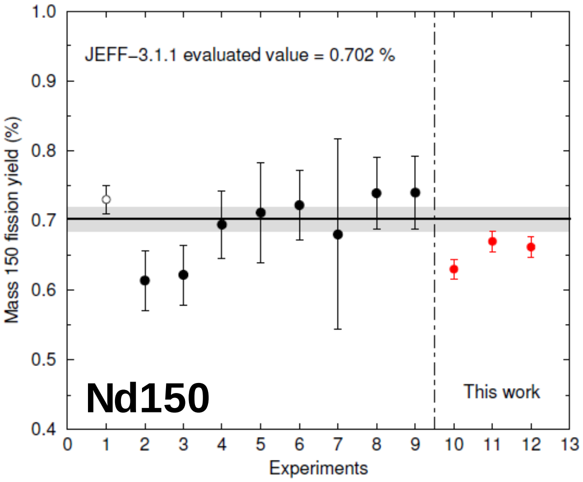
Effective cumulated FY obtained from PROFIL (**in red**) and compared to data used in the evaluation procedure of the JEFF-FY library ⇒ **included in the new evaluation of the FY for JEFF-3.3**

U235+n





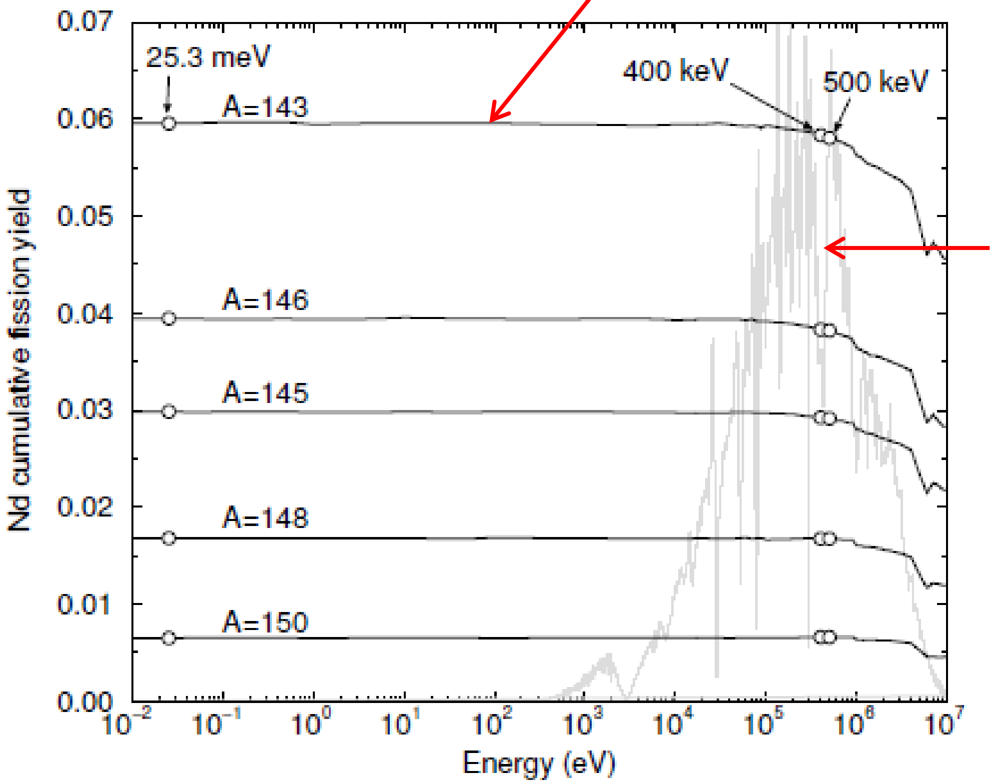




GEF caculations : U235+n

Analysis of the PROFIL trends with GEF results:

⇒ New analysis based on the PROFIL and GEF results



Flux spectrum representative of the PROFIL irradiation experiments

Prior covariance matrix for Y_c(Nd148) vs. Energy

Prior covariance matrix are needed for using the integral data assimilation technique

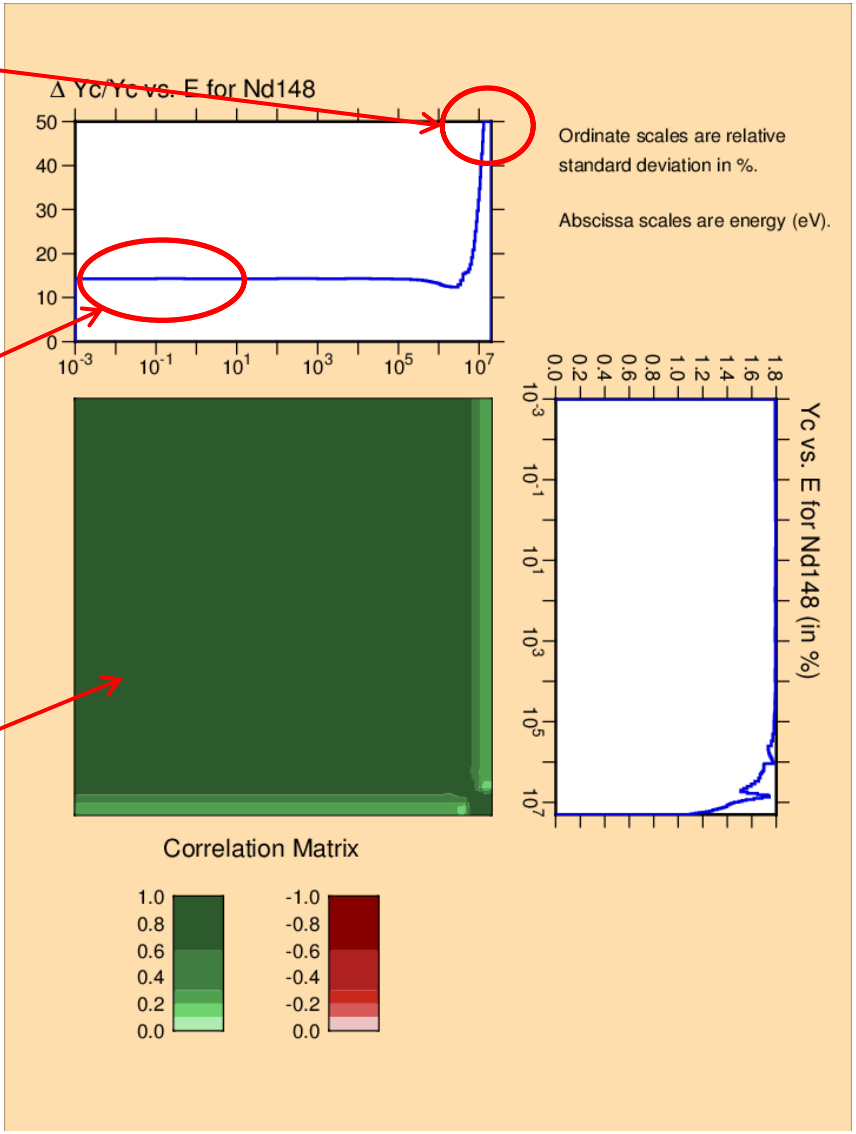
- Random files were calculated with GEF at PSI by D. Rochman
- They were converted in covariance matrices
- « Defect model » were added to account for observed biais ⇒ differences between GEF and JEFF-3.1.1)

Prior covariance matrix for Y_c(Nd148) vs. Energy

Large uncertainty
arround 14 MeV

Relative uncertainty of 14.3% in
the thermal energy range

The fast and thermal energy
ranges are strongly correlated



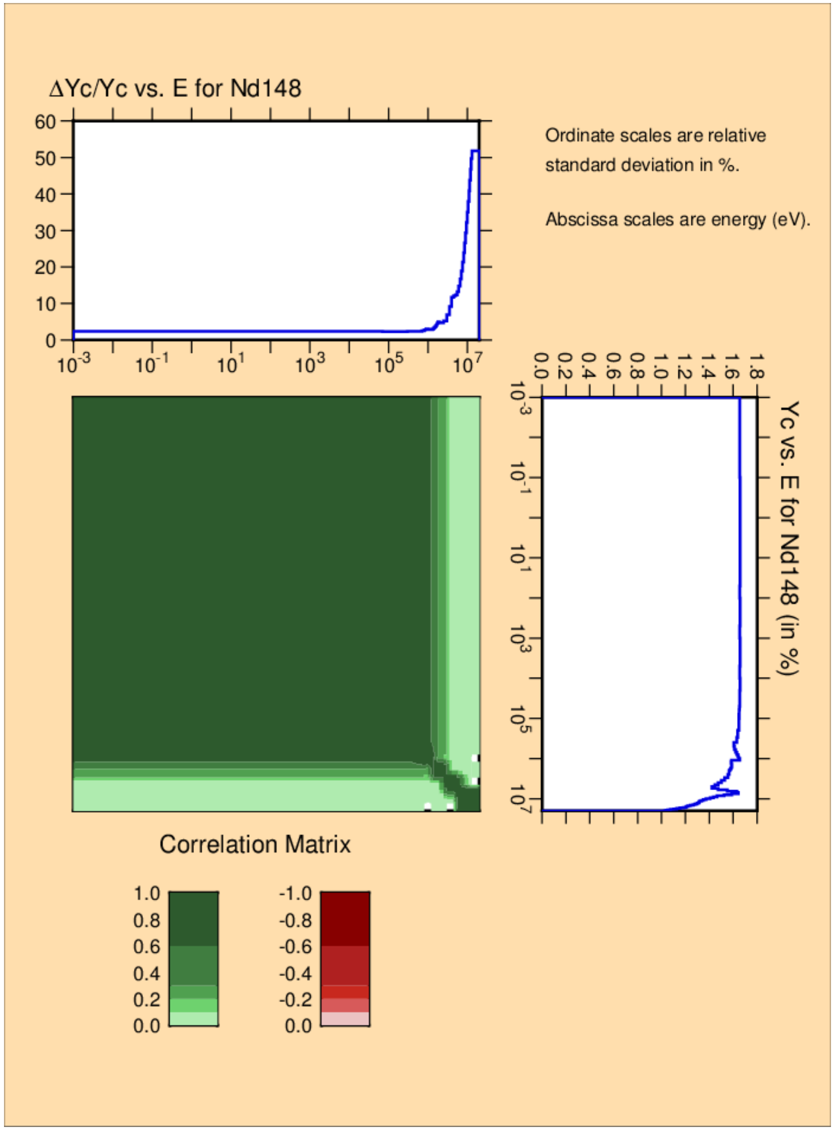
Posterior covariance matrix for Y_c(Nd148) vs. Energy

Covariance matrix obtain after the assimilation of the PROFIL data with CONRAD

- Uncertainty in the thermal energy range close to 2.4%

- Strong correlations still exists

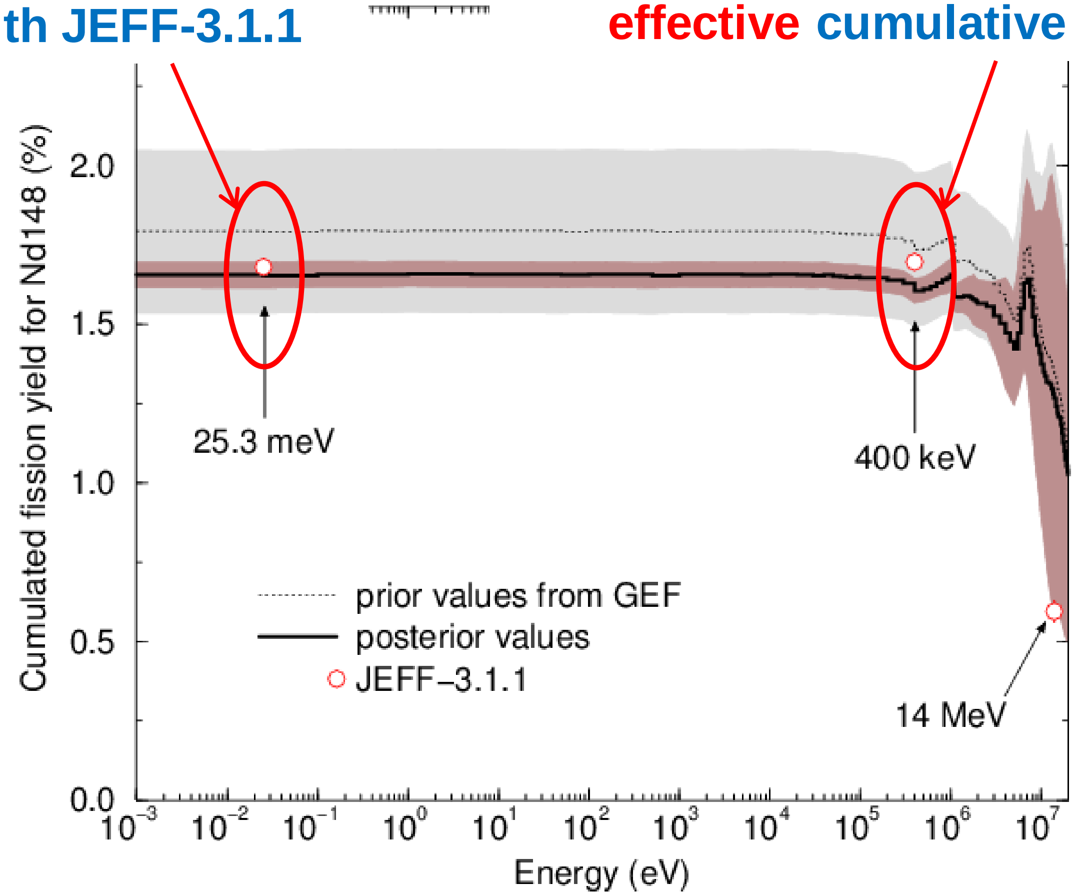
- No impact arround 14 MeV



Posterior uncertainties for Y_c(Nd148) vs. Energy

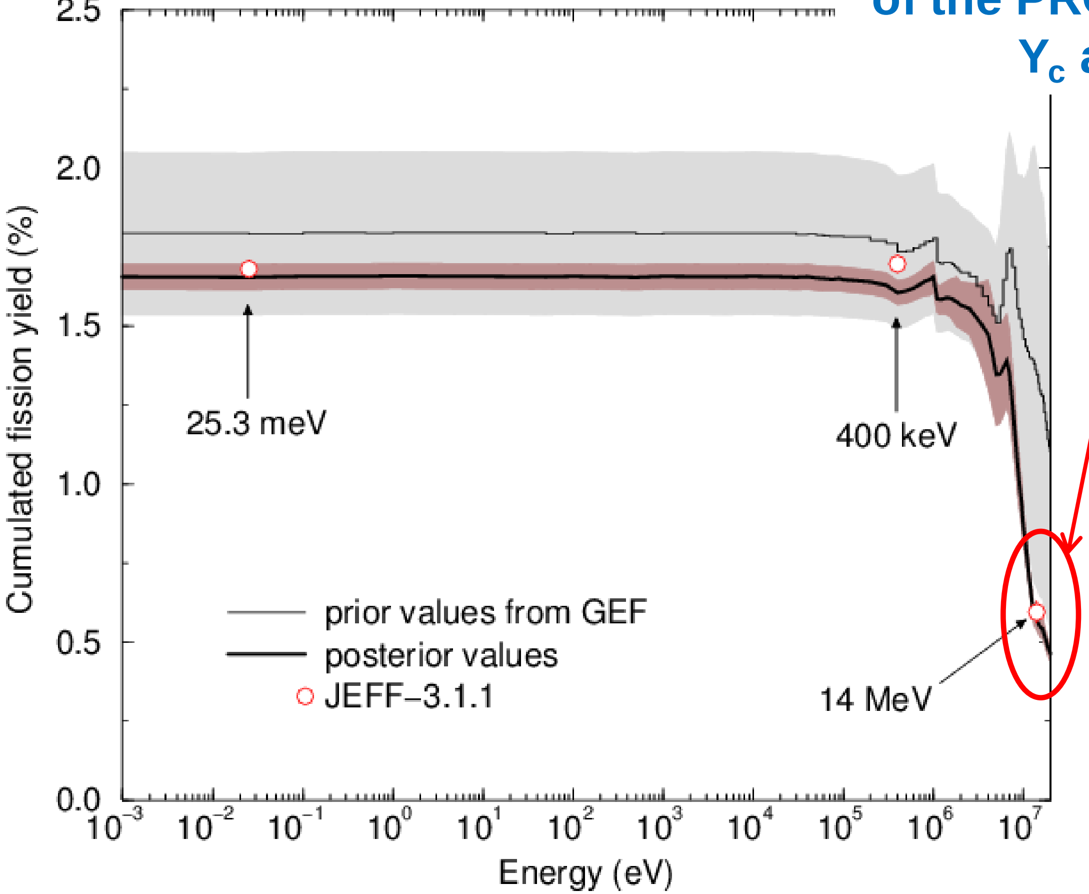
A good agreement is observed with JEFF-3.1.1

Not fully comparable because the JEFF evaluation is based on experimental effective cumulative fission yields



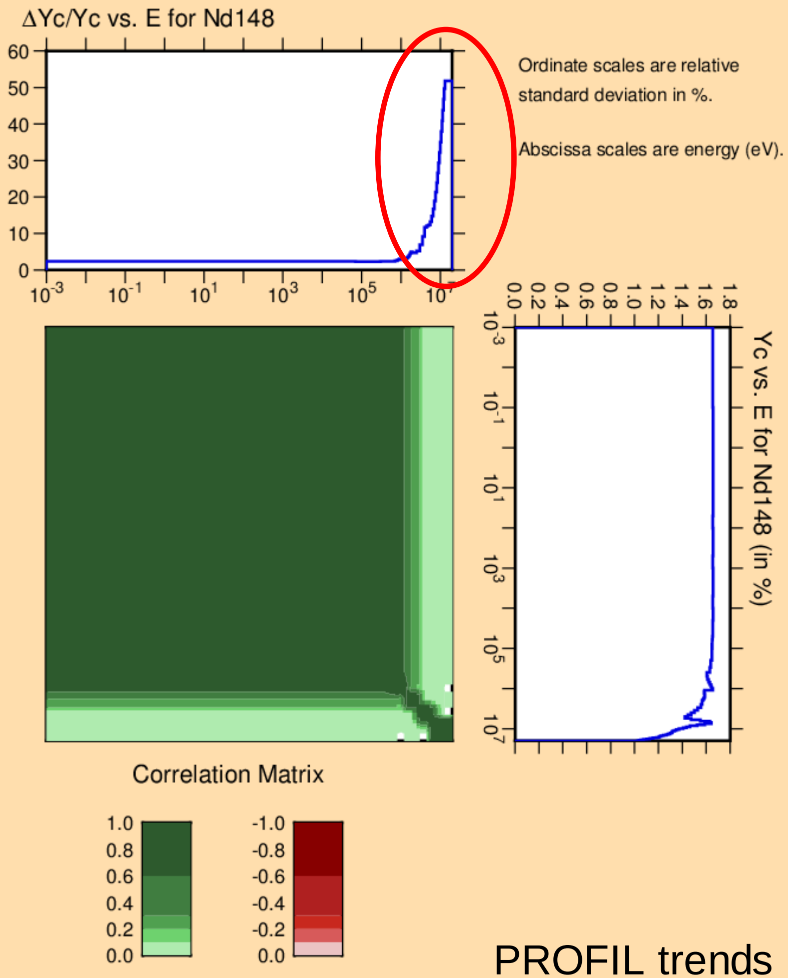
Posterior uncertainties for Y_c(Nd148) vs. Energy

Simultaneous adjustement of the PROFIL trends and Y_c at 14 MeV

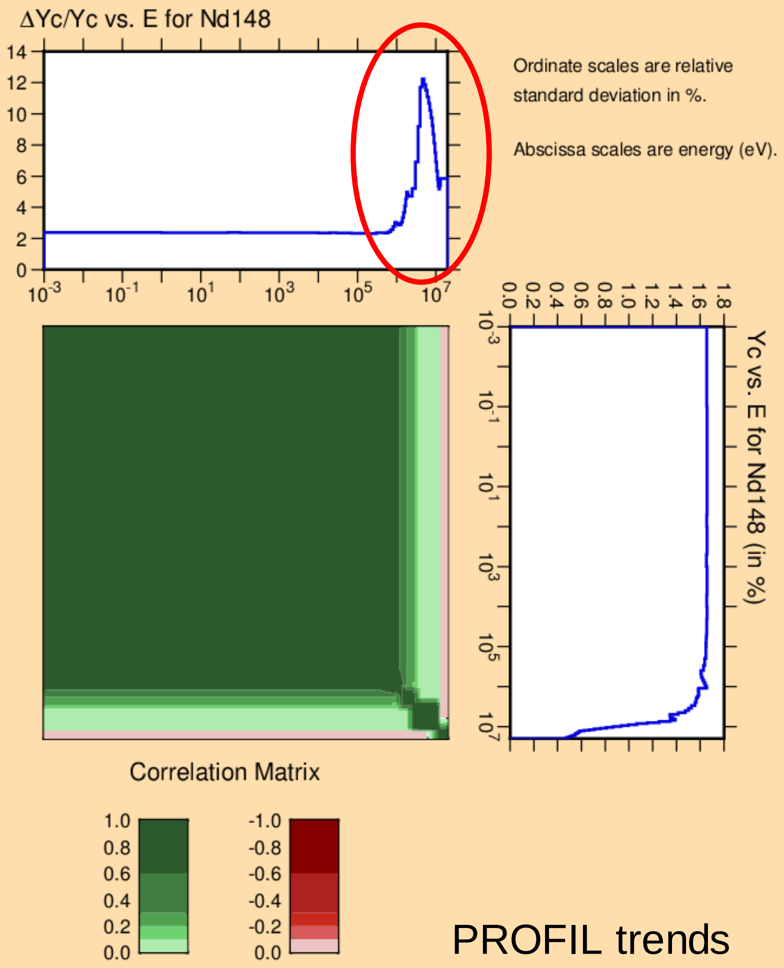


Posterior covariance matrix for Y_c(Nd148) vs. Energy

Sizeable reduction of the uncertainties



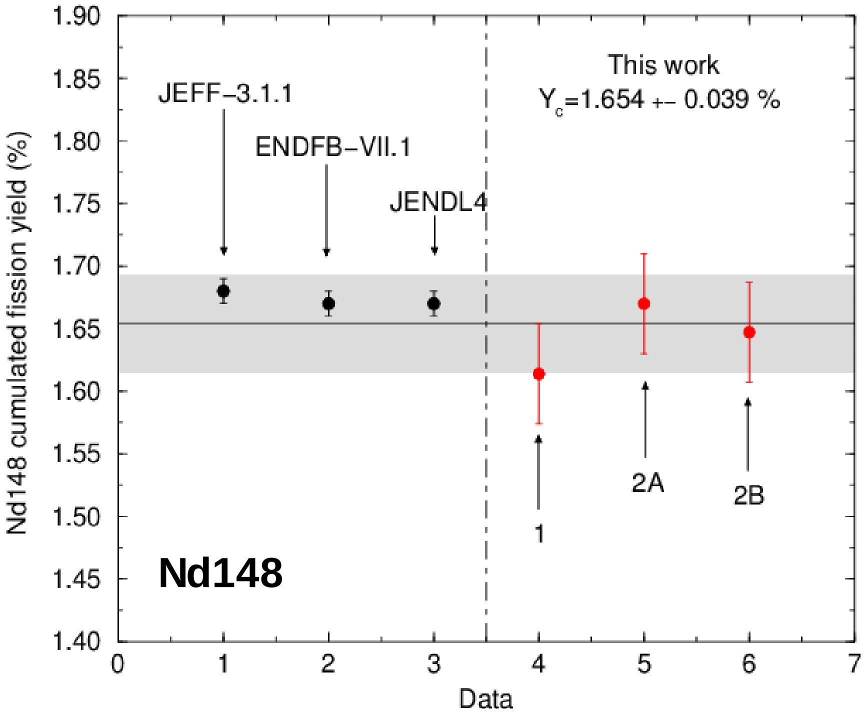
PROFIL trends alone



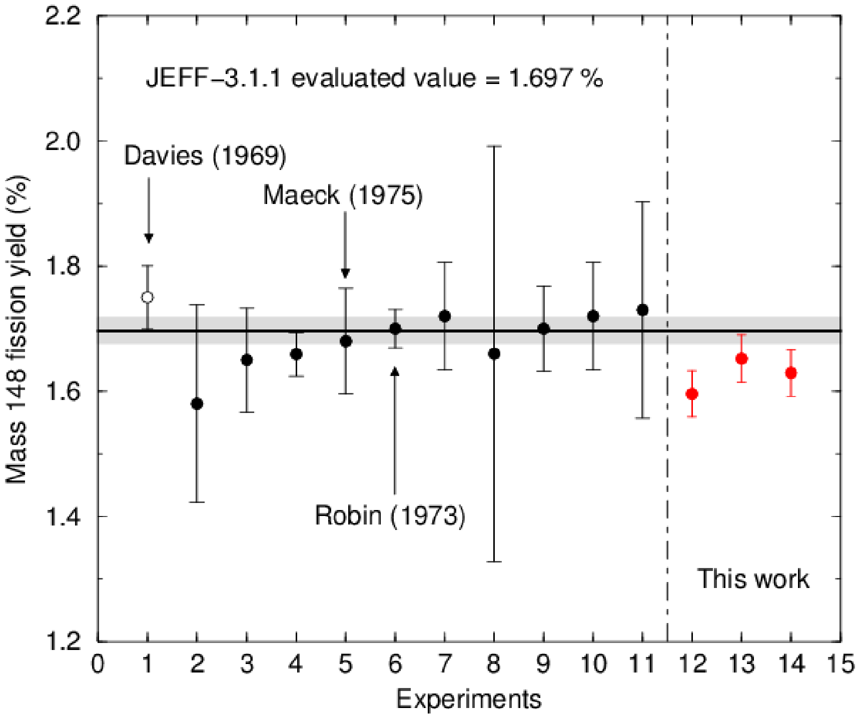
PROFIL trends and Y_c at 14 MeV

Comparison with evaluated values

Thermal energy range



Fast energy range



PROFIL provides Fission Yields in the thermal and fast energy ranges in good agreement with the evaluated values

Conclusions

❑ Integral Data Assimilation is a usefull tool to identify major problems in existing nuclear data.

❑ Mathematics algorithms and tools are mature for producing covariance matrices via the IDA procedure

❑ However, IDA should be use with care for producing results compatible with a « general purpose library »

❑ Two exemples were presented in order to show that separation between « general purpose library » and « application library » is not so obvious