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Workshop on
Nuclear Reaction Data and Nuclear Reactors: Physics, Design and Safety

16 February - 12 March 2004

Analysis and Correlation of Experimental Data

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Analysis and Correlation of Experimental Data

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Introduction

➢ **Objective**

- Explain **why** physics experiments are needed
- Describe **how** they are performed
- Illustrate **what** we get out of them

➢ **Presentation**

- Is intentionally kept at a non-specialist level
- Is based on words rather than equations $\rightarrow$ see references for details
- Largely refers to CEA + French activities, but has broader significance

Accuracy of neutronics simulations

- Despite the diversity and complexity of nuclear systems, current neutronics code systems are capable of yielding **accurate predictions** of conventional reactor characteristics in **short running times**

- Considerable progress has been made in neutronics over the past 15 years or so → **Example**: Core calculations

	Diffusion theory code	Transport theory code	Monte Carlo code
1970	2-D		
1980	3-D	1-D	Marginal
1990	3-D	2-D	Marginal
2000	3-D	3-D, homog. subassemblies	Frequent
2010	3-D	3-D, cell-by-cell	Systematic

Accuracy of neutronics calculations

➢ **Typical PWR calculation accuracies** (uncertainties at 1 σ)

	Computed quantity	1990	2000	2010
Control	• Boron concentration • Axial offset	 ± 4%	 ± 3%	± 5 ppm ± 2% Pn
Flux and power distributions	• Bundle power peak • Bundle mean power • Rod power	± 8% ± 5% ± 7%	± 5% ± 4% ± 5%	± 2% ± 2% ± 3%
Reactivity effects	• Boron concentration • Integral rod worth • Doppler coefficient • Beta effective	± 40 ppm ± 10% ± 13% ± 5%	± 20 ppm ± 8% ± 10% ± 5%	± 10 ppm ± 5% ± 5% ± 5%
Fuel burn-up and inventory	• Bundle mean burn-up • Fuel rod burn-up • Local burn-up • Main nuclide conc. • Other nuclide conc.	± 5% ± 6% ± 7% ± 5% ± 15%	± 4% ± 5% ± 6% ± 3% ± 10%	± 2% ± 3% ± 5% ± 2% ± 10%

Motivation for improved code predictions

- **Further improvements** are requested for
  - ✓ **margin gains**
  - ✓ **better predictive power**
  - ✓ **broader range of application**

- **Impact of improved code predictions**

  - ✓ **Needs / target accuracies actually depend on the "client"**
    - ❑ R&D organization
    - ❑ Reactor designer
    - ❑ Reactor operator
    - ❑ Safety Authority
    - ❑ …

  - ✓ **Example of potential gains for a plant operator (source EdF)**
    For a park made of 50 nuclear reactors of 1000 MWe each, in base load operation:
    - ❑ Cycle length underest. by 1% $\Rightarrow$ **~ ½ core fuel reload / year**
    - ❑ Gain of 1% on P_{lin} max $\Rightarrow$ **capacity increase by ~ ½ plant / year**
    - ❑ Decay heat : **1 day of shutdown ~ ½ core fuel reload / year**

Neutronics simulations in practice

- **Sources of errors impacting the code predictions (incl. Monte Carlo codes)**

  - The quality of the input **nuclear data** (which depend on temperature)
  - The number of available **modelling options** (geometry, boundary conditions, solid state effects, …)
  - The **approximations** made in solving the equations (nonlinearities,…)
  - The **simplifications** made in practice in representing / modelling the system, for example in defining the system boundaries or the radiation sources, in modelling fuel depletion, in modelling the migration of gaseous fission products,…
  - Technological / **manufacturing uncertainties** in the system constituents, e.g., in the fabricated fuel pellets
  - The errors arising from the calculation of **other characteristics** impacting the neutron flux density during operation, e.g., fuel pellet physical state, fuel and moderator temperature distributions,…
  - Etc.

Neutronics simulations in practice

- **Requirements (output)**
  → **Recommendations on how to use the neutronics code** in order to meet the users' needs :
  - A well defined **parametric domain** within which the code can be validly applied
  - **Recommended calculation options** for each requested characteristics
  - **Errors and uncertainties ΔC** to be assigned to the code predictions ΔC (calculating C alone is **not** sufficient)

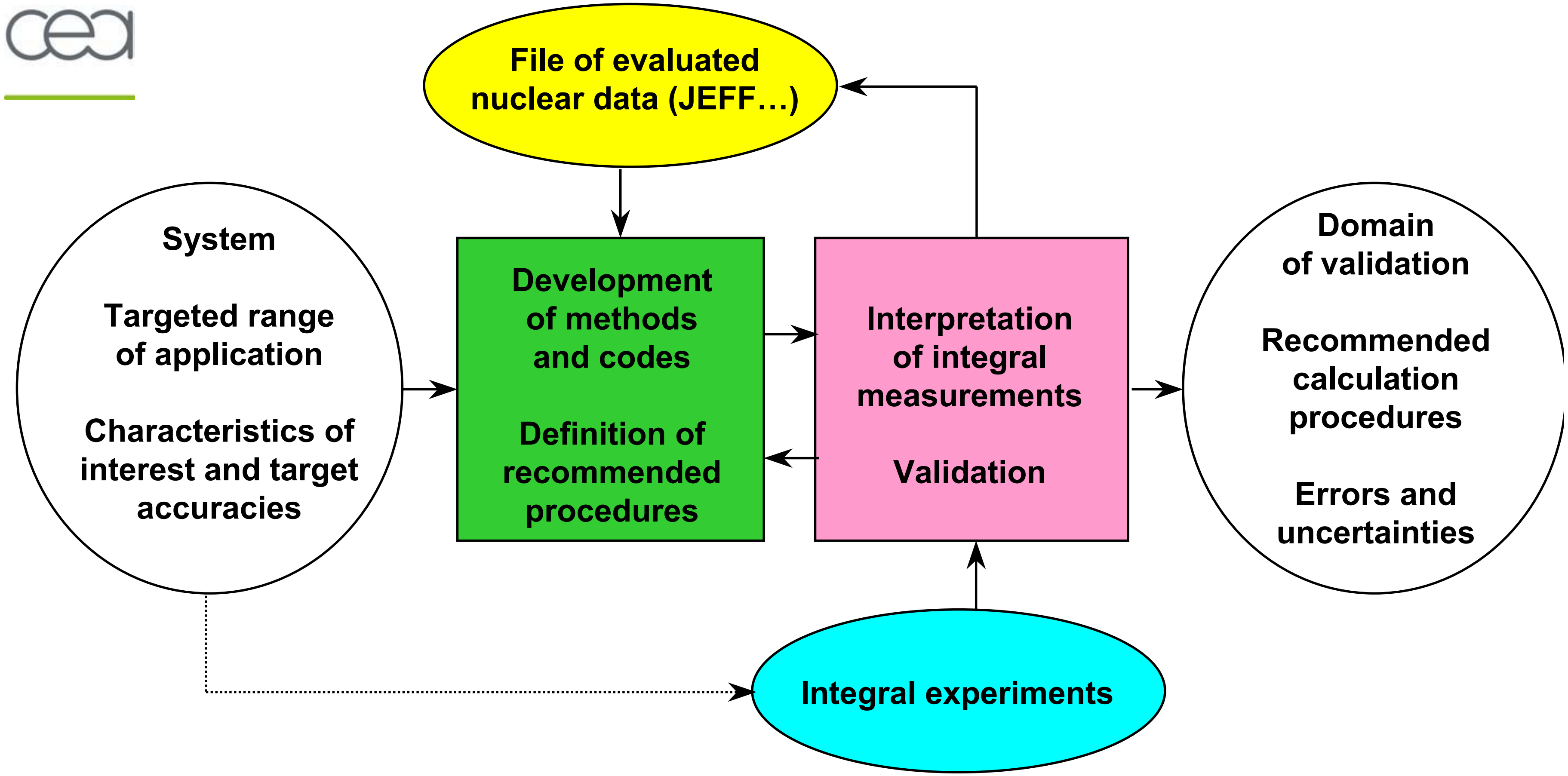
- **Main ingredients (input)**
  - **Data libraries** ← **nuclear data** (JEFF, ENDF/B, … files)
  - **Calculation procedures** ← **physics models and associated calculations modules**
  - **Validation data** ← **experiments** in reactors

The validation process

- **Consists in**
  - Assessing the errors and uncertainties of the code system for a given field of application
- **Applies to**
  - Reference calculation procedures and associated nuclear data
- **Relies on**
  - Nuclear data sets (with their associated uncertainties)
  - Calculation modules
  - Experimental validation data (with their associated uncertainties)
- **Leads to**
  - Recommended calculation procedures
  - Errors and uncertainties to be associated to the code outputs

The validation process

- **Schematic flow diagram**



Example of a lattice code validation range for UOX fuel applications

Characteristics	R_{mod}	T_{fuel}	T_{mod}	U-235 in %	BU range in GWd/t
Initial reactivity	0.5-5.5	20	20	3.0-4.8	-
Power distribution	1.4	20	20	3.7	-
Fuel inventory	PWR	700	300	3.1-4.5	10-62
Burn-up swing	PWR	20	20	3.1-4.5	10-62
Fuel temp. coeff.	1.4	20-1000	20	0.2-5.0	-
Mod. Temp. coeff.	1.3-1.7	isothermal	10-300	3.1-3.7	-
Boron coeff.	1.7	20	20	3.7	-
Void coeff.	1.2	20	20	3.7	-
Absorbers worth	1.2-1.7	20	20	3.7	-
Control rod antireactivity	1.4	20	20	3.7	-
Gd poison depletion	1.5	300	30	3.3	0-12
Gd poison worth	1.2-1.6	20	20	3.5-3.7	
Water reflector	1.2	20	20	3.7	-

The validation process

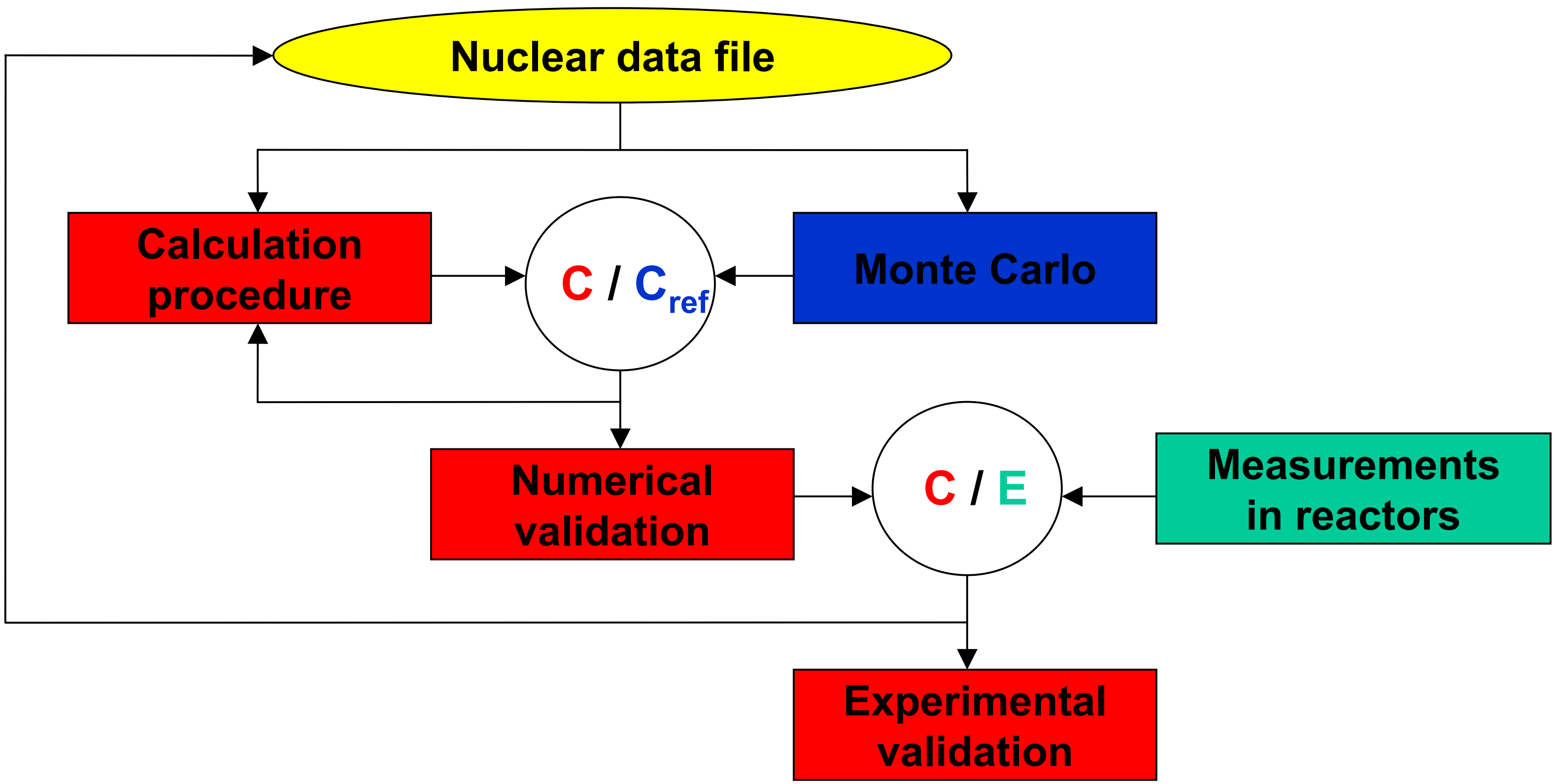
- **Distinguish**
  - ✓ **numerical validation** = calculation-vs.-**calculation** comparisons using the same nuclear data

    Reference results may be provided by a Monte Carlo code
  - ✓ **experimental validation** = calculation-vs.-**measurement** comparisons

- **Methodology**
  - ✓ Allows in principle to **separate** (and hence avoid compensations between)
    - ❑ Errors arising from the **nuclear data**
    - ❑ Errors arising from the **methods** / procedures

    and to suggest improvements on each of these
  - ✓ Has been **systematically used** at CEA for the past 10 years
  - ✓ Is possible because of progress in computing power → **Monte Carlo** calculations are becoming routine → method biases under control
  - ✓ In practice, separation is achieved to a great extent but not fully

- **Schematic flow diagram**



Experimental validation

- **Experimental validation** ⇒ integral measurements are needed as input
- **These physics experiments must be**
  - ✓ **Specific** → representative of the targeted application range
  - ✓ **Analytic** → phenomena can be analysed individually
  - ✓ **As simple as possible**: in terms of geometrical arrangement, constituents, … ⇒ no need of corrections for parasitic phenomena, code modelling,… ⇒ reduced error and uncertainty on C
  - ✓ **Sufficiently accurate**: measurement accuracy better than C / E discrepancies ⇒ need of well-controlled experimental conditions ⇒ reduced error and uncertainty on E
  - ✓ **Sufficiently diverse** → if integral measurements database is large enough, more predictive capability + possibility to do statistical adjustment studies to infer trends in nuclear data

- **Experimental validation**

  - ✓ Not only implies **C / E comparisons**
  - ✓ **But also**
    - ❑ **A detailed analysis of the C / E** discrepancies by means of **sensitivity calculations (perturbation theory, see lecture by A. Gandini) → trends in nuclear data**
    - ❑ **Transposing the results** obtained in an experimental reactor to actual reactor conditions
    - ❑ **Recommending calculation options and associated uncertainties**
    - ❑ Evaluating the input parameter **domain** within which those recommendations hold

Experimental validation

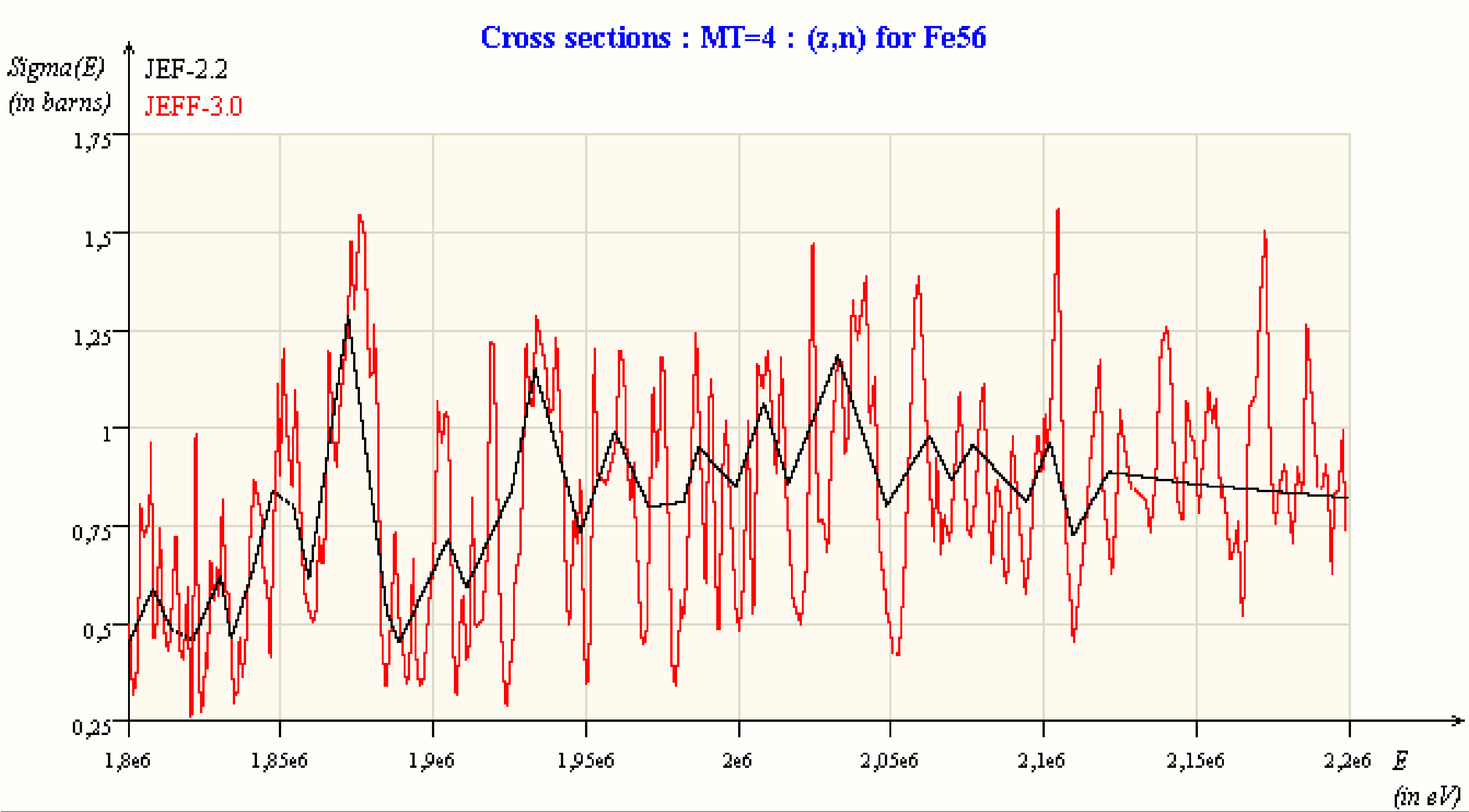
- **The experimental validation makes it possible to establish**

  - ✓ If the quality of the nuclear data is sufficient to meet the application needs

  - ✓ To identify those data that require improvements and by how much

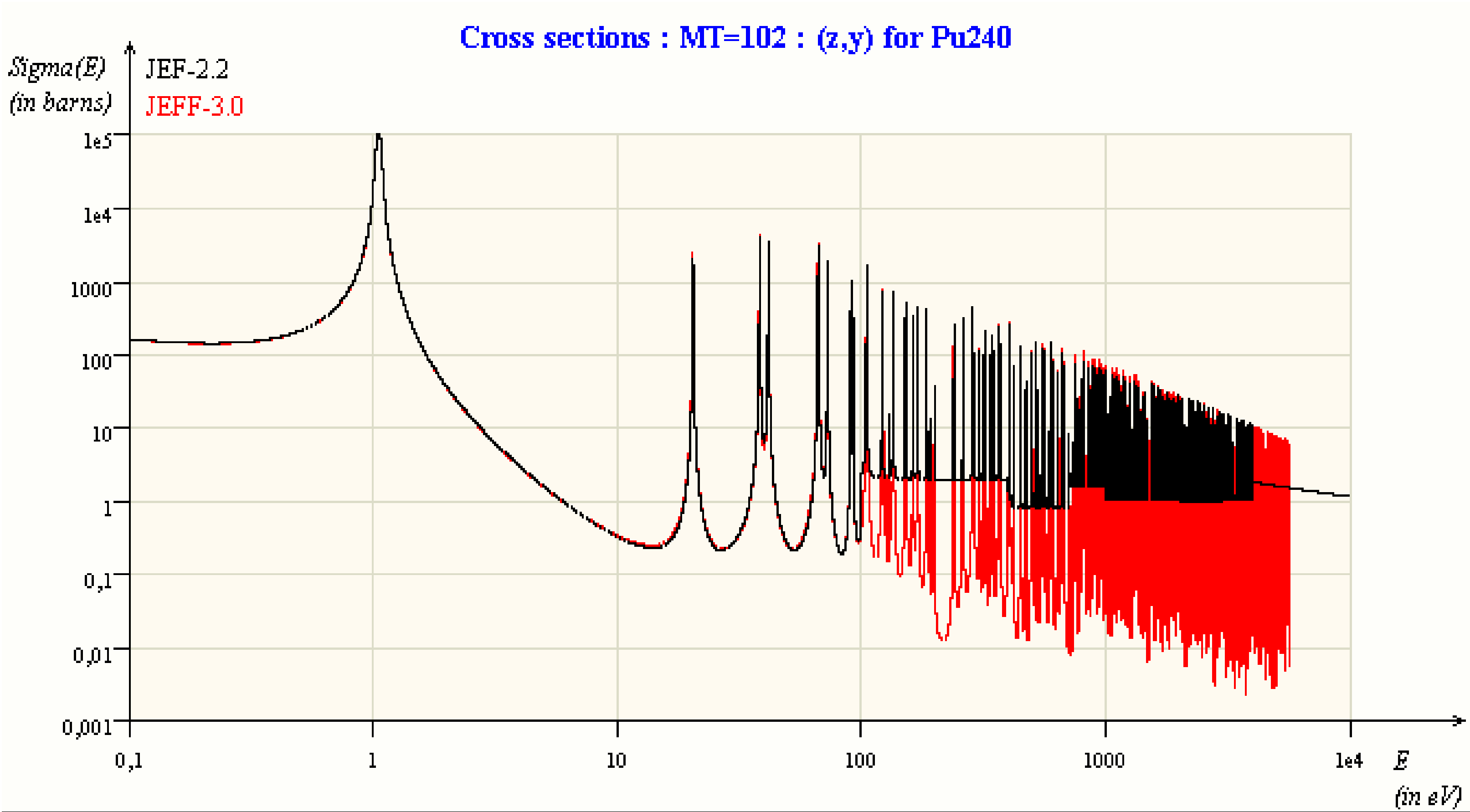
- **This is important** as the errors and *a priori* uncertainties affecting nuclear data are still quite **large** today

Examples of recent cross section re-evaluations



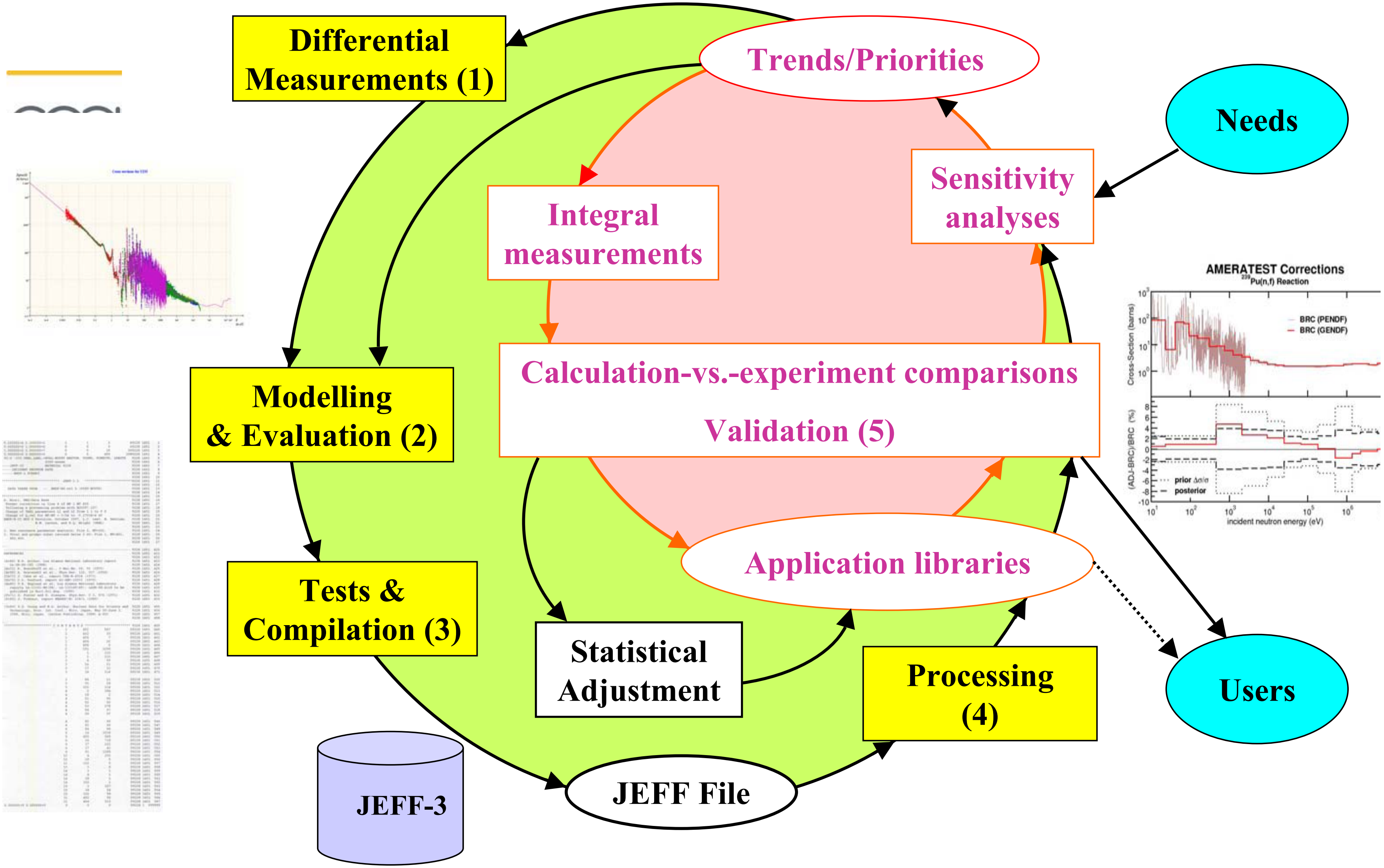
JEFF-3.0 vs. JEF-2.2 Inelastic Scattering Cross Section of Fe-56

Examples of recent cross section re-evaluations



JEFF-3.0 vs. JEF-2.2 Radiative Capture Cross Section of Pu-240

Flow diagram of the nuclear data production and validation process



Physics measurements in reactors

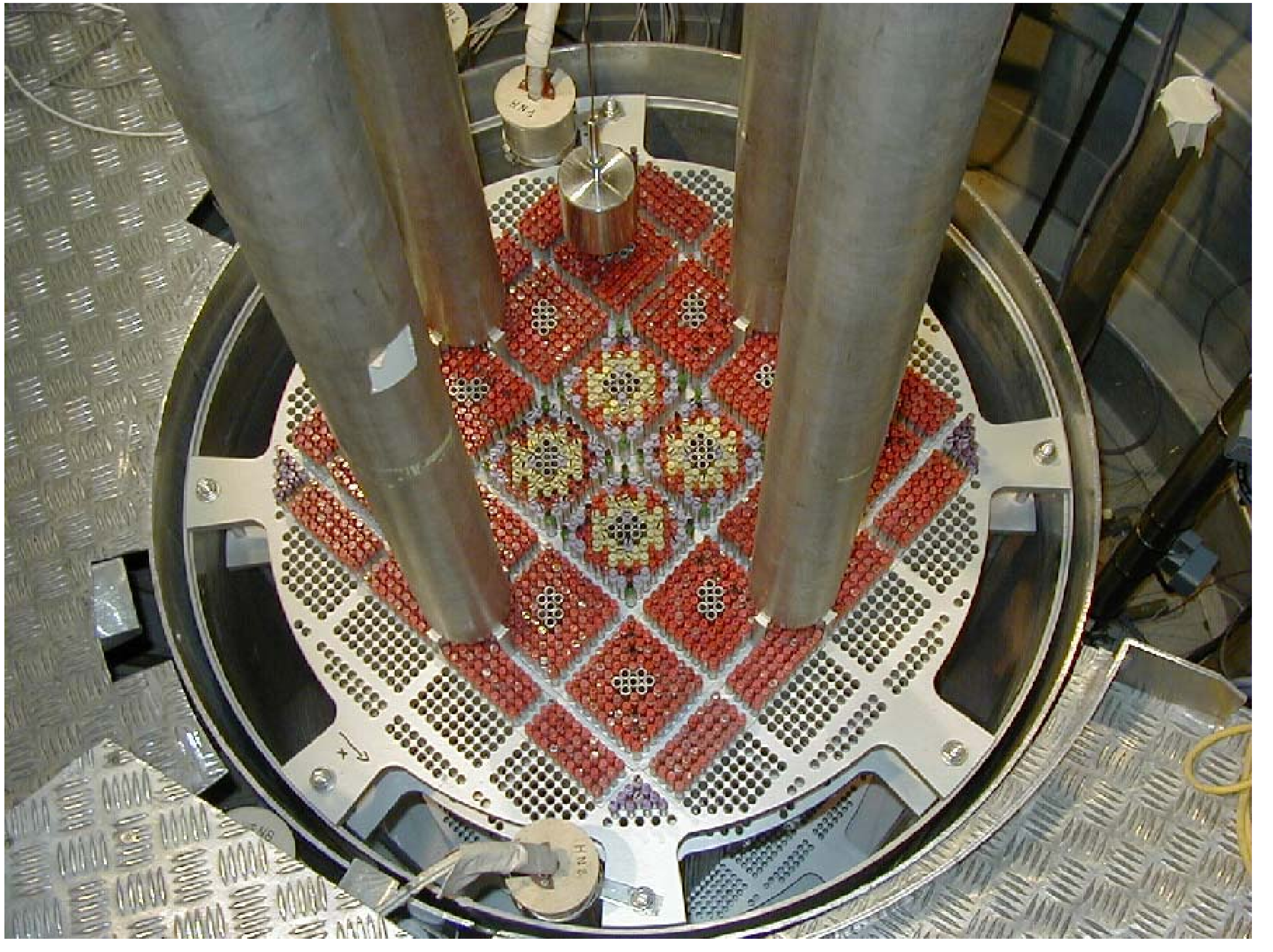
- **Integral and differential measurements** are **complementary** from the standpoint of validating nuclear data evaluations
  - ✓ Differential measurements provide information of high energy/angle resolution but generally inaccurate in level
  - ✓ Integral measurements provide information of poor resolution but of usually very good accuracy in level

- **Distinguish physics measurements**
  - ✓ in **near-zero-power critical facilities** such as
    - ❑ EOLE, MINERVE, MASURCA
    - ❑ PROTEUS
    - ❑ VENUS
    - ❑ KUCA
    - ❑ FCA
    - ❑ BFS
    - ❑ …
  - ✓ in **power reactors** → irradiations experiments

Physics measurements in reactors

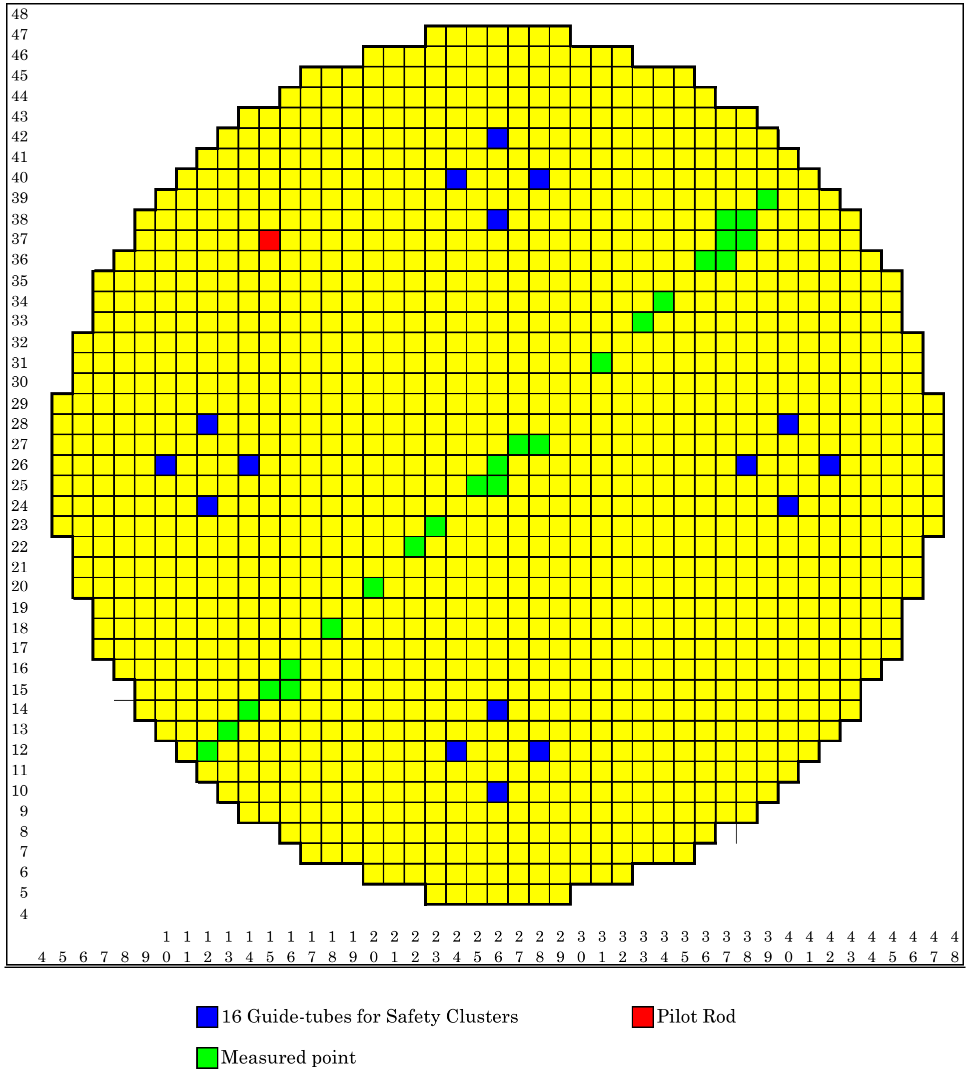
- **Zero-power reactors**
  - ✓ are characterised by well-known constituents and a high degree of **flexibility** in terms of core loading, geometrical arrangements, operation
  - ✓ allow measurements that are difficult or impossible to do in power reactors
  - ✓ are easy to model
- **Power reactors**
  - ✓ Provide full-scale and actual operating conditions
  - ✓ Provide information on capture cross sections and fuel inventory
  - ✓ Require more effort in modelling
- Both types of measurements are **needed and complementary**

Examples of experimental facilities

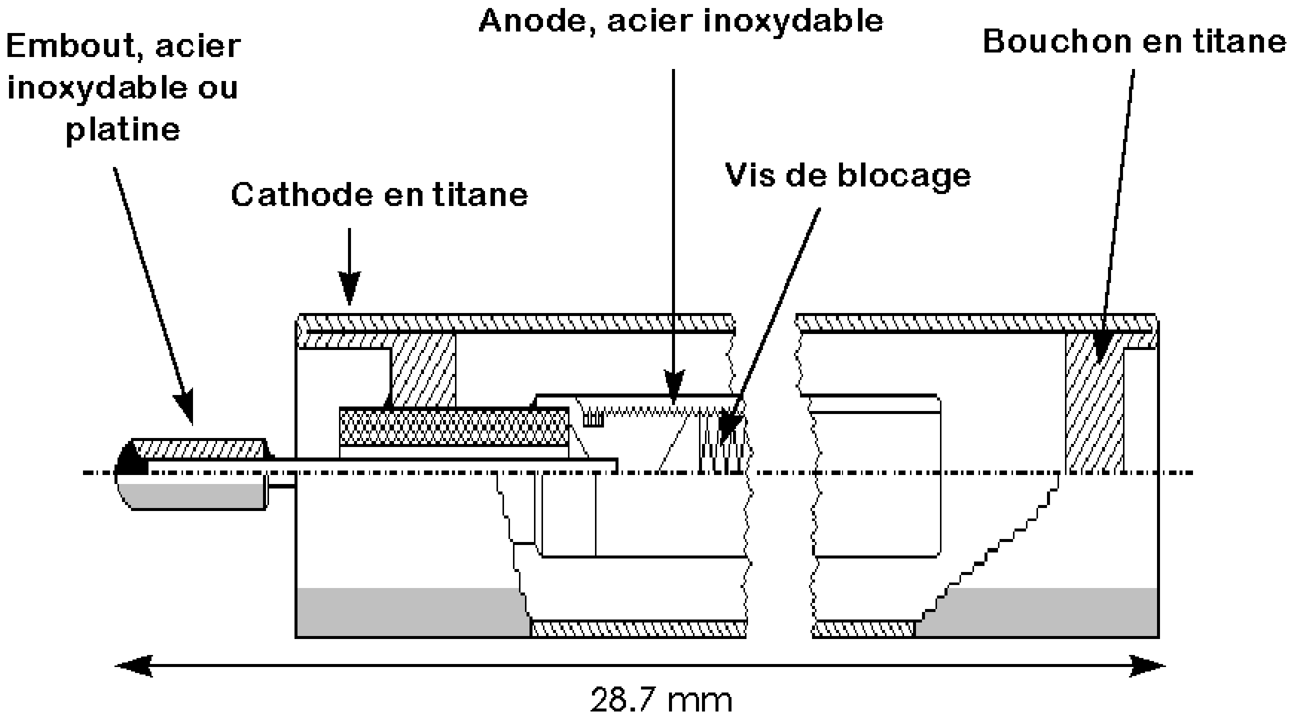


Top view of the EOLE reactor core

Examples of experimental facilities



X-Y Model of the EOLE/MISTRAL3 100% MOX Core

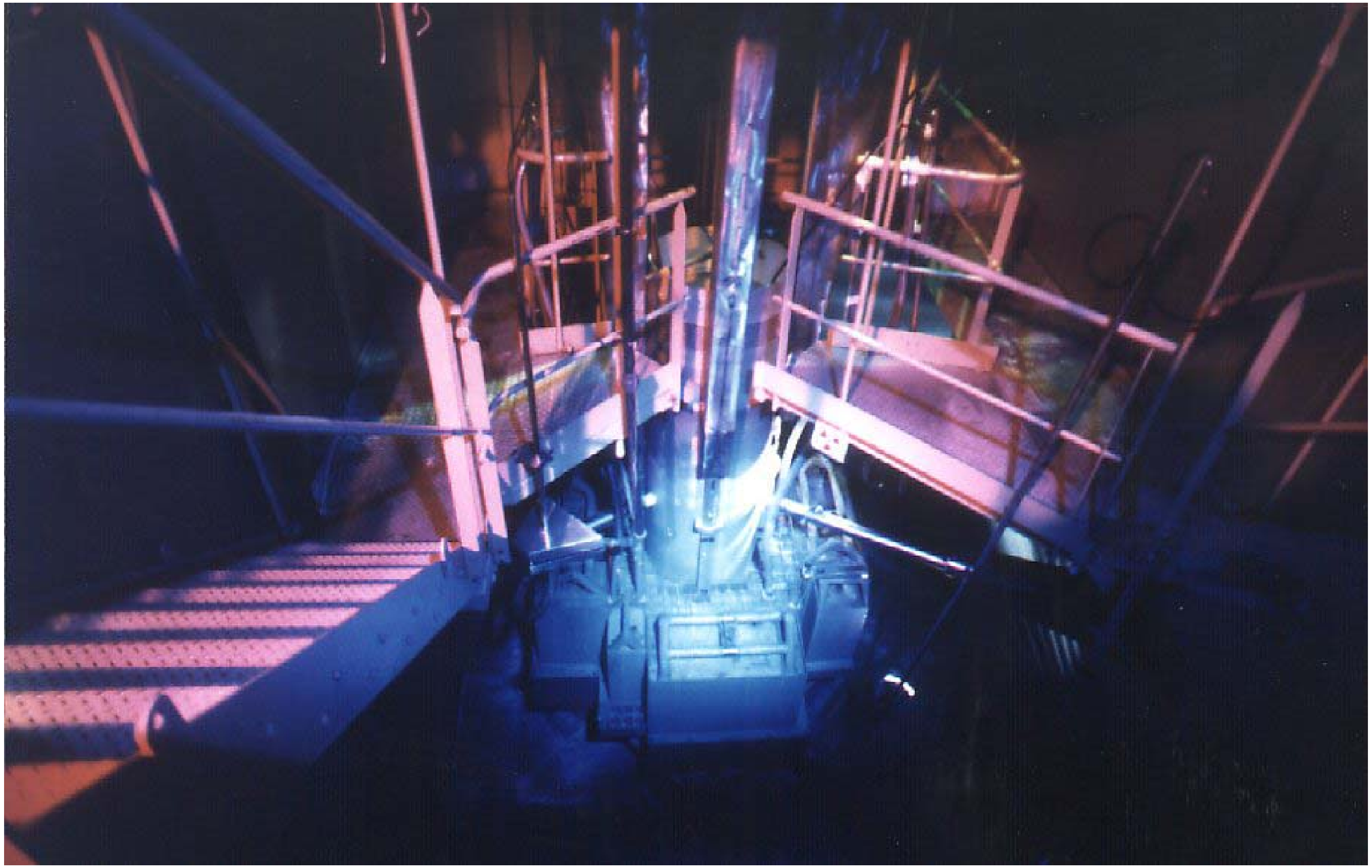


Fission chamber used in EOLE

Measurements Performed in EOLE as part of the MISTRAL Programme

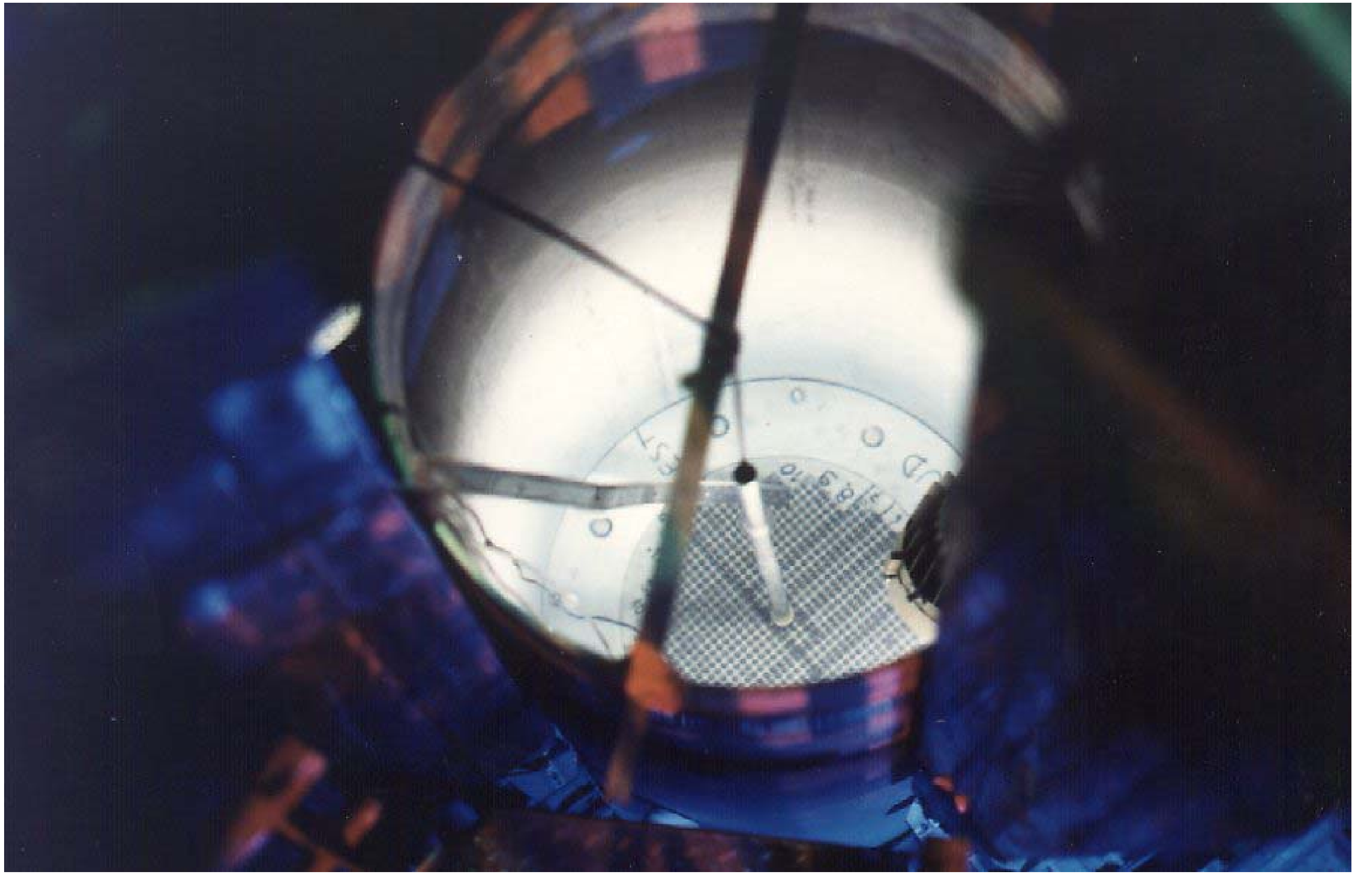
	MISTRAL-1	MISTRAL-2	MISTRAL-3	MISTRAL-4
Critical mass	O	O	O	O
Buckling	O	O	O	
Spectral indices	O	O	O	
P(r)				O
P(z)				O
Temperature coeff.	O	O	O	
Soluble boron worth	O	O	O	O
Single absorbers worth	O	O	O	
Rod cluster worth				O
2D void worth			O	
β_{eff}	O	O		
Central heterogeneity	O	O	O	
Norm. UOX/MOX				O

Examples of experimental facilities



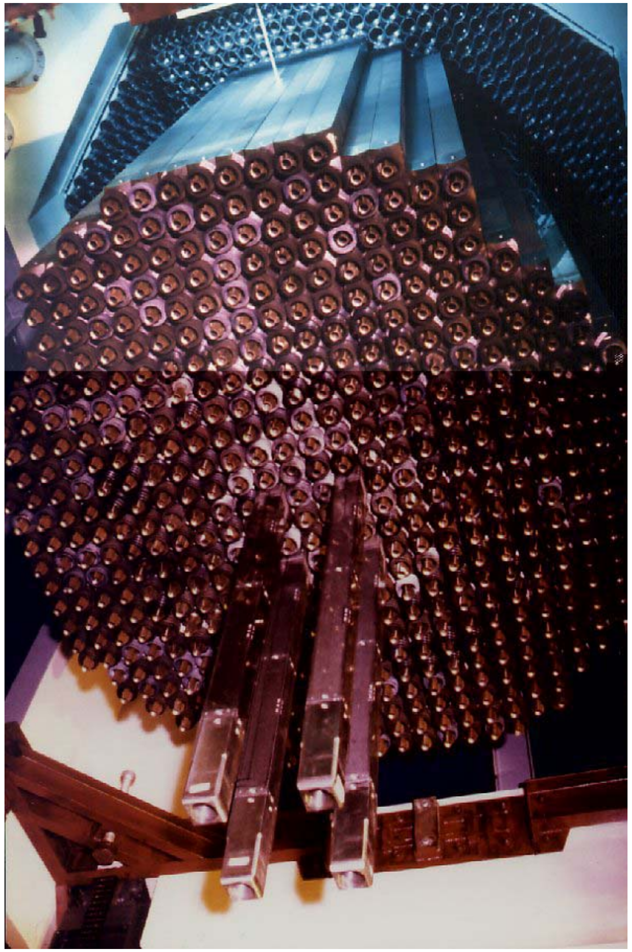
View of the MINERVE reactor

Examples of experimental facilities



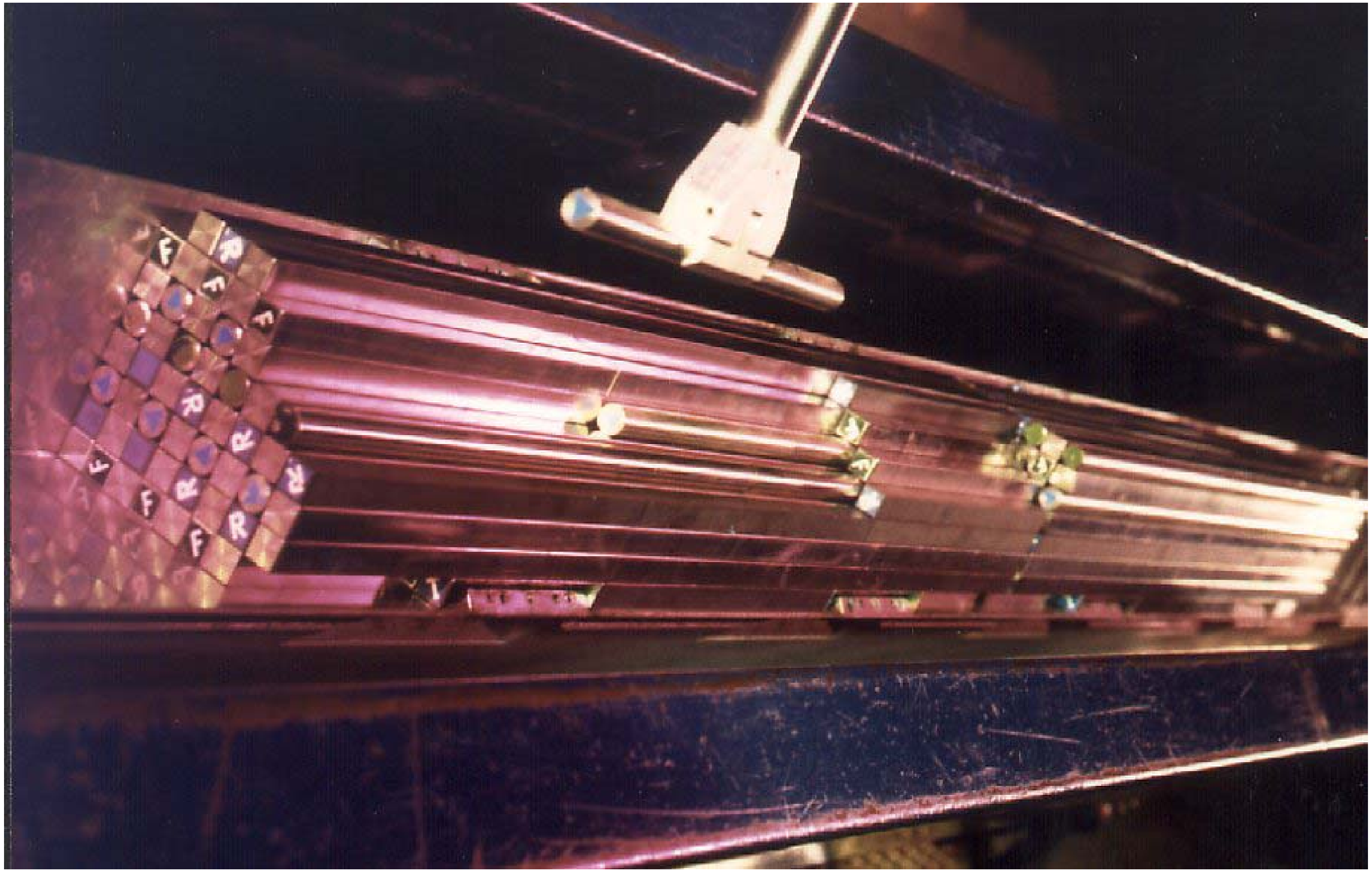
Top view of the MINERVE reactor core

Examples of experimental facilities



Bottom View of the MASURCA reactor core

Examples of experimental facilities

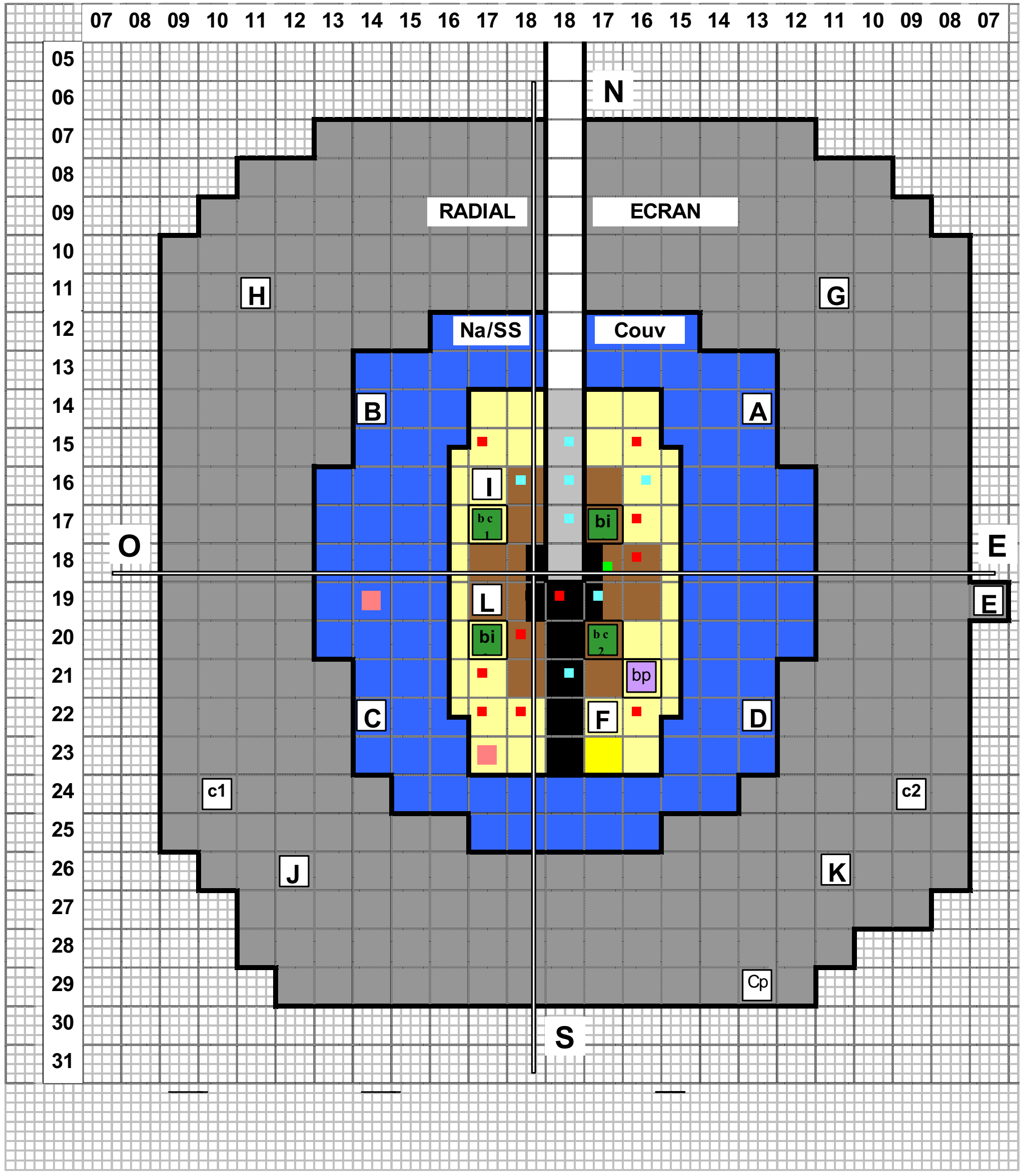


Cut-away View of a MASURCA Subassembly Showing Fuel Rodlets

Examples of experimental facilities

X-Y Model of the MUSE-4/SC3 Configuration in MASURCA

- Black zone = lead
- Brown zone = lead + fuel
- Yellow zone = fuel + sodium
- Blue zone = steel-sodium refl.
- Grey zone = shield



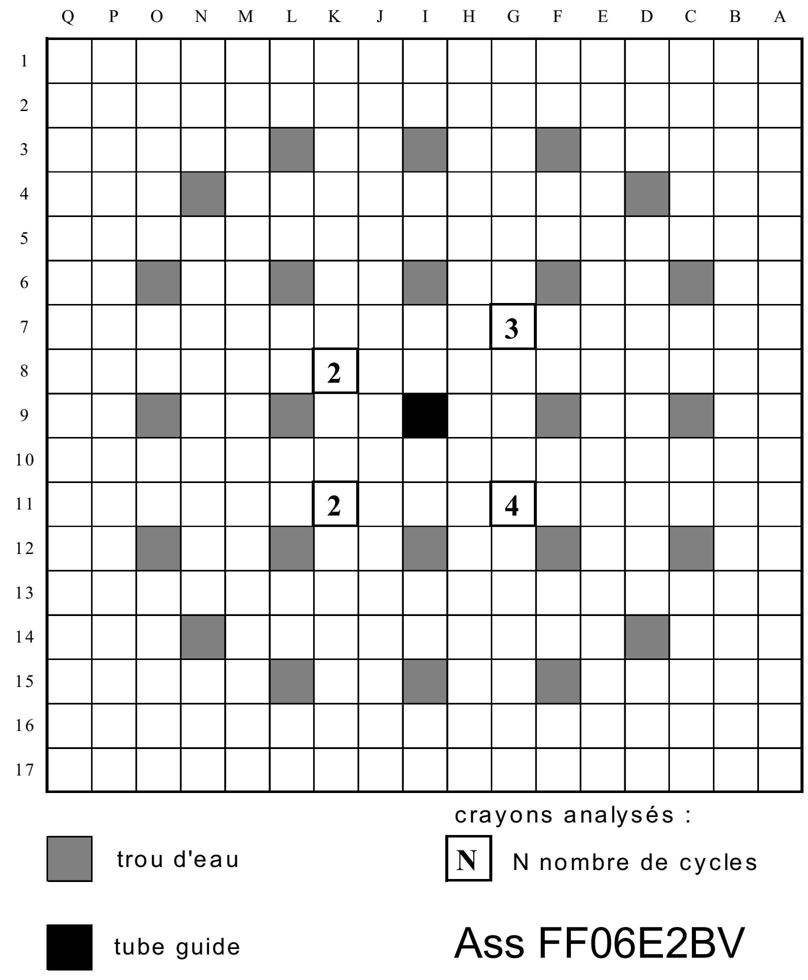
PWR C/E Errors and Uncertainties Obtained with APOLLO2.5 + JEF-2.2 Data

Characteristics	UOX	MOX
Keff	+ 270 ± 150 pcm	+ 100 ± 250 pcm
Power peak	- 0.4 ± 0.7 % + 0.2 ± 1.5 % (Ass Gd)	+ 1.3 ± 1.5 %
dK/dT_{comb} (Doppler coeff)	+ 1.9 ± 4 %	+ 11 ± 4 %
dK/dT_{mod} T = 20°C-80°C	- 0.0 ± 0.3 pcm/°C	- 1.5 ± 0.3 pcm/°C
dK/dT_{mod} T ≅ 300°C	- 0.9 ± 1.0 pcm/°C	+ 3.0 ± 2.2 pcm/°C
dK/dC_B (soluble boron coeff)	+ 3 ± 8 %	+ 0 ± 3 %
$\Delta K/\Delta V_{mod}$ (void coeff)	+ 0 ± 3 %	+ 2.0 ± 2.0 % (60% vide) + 1.3 ± 1.0 %(100% vide)
βeff	+ 2.4 ± 1.6 %	+ 0.1 ± 1.6 %
$\rho^{\text{Control rod cluster}}$	+ 1 ± 1 %	-
Pyrex 24poisons $\rho_{initial}$	+ 2.7 ± 1.5 %	-
UO_2-Gd_2O_3 $\rho_{initial}$	+ 0.2 ± 0.6 %	-

Analysis of irradiated fuel rods

Ex: **Gravelines UOX 4.7%**

NB: Detailed modelling required



JEF-2.2 Trends Derived from the Analysis of Irradiated Fuel Rods

Isotope		20GWj/t	40GWj/t	50GWj/t	60GWj/t
	e=3.1%	3.5	2.3		
U234	e=4.5%	0.4	0.8	-1.7	1.5
	Incertitudes	±1.1	±1.4	±1.6	±2.0
	e=3.1%	0.5	1.7		
U235	e=4.5%	1.0	1.8	2.1	3.0
	Incertitudes	±1.1	±2.0	±2.7	±3.5
	e=3.1%	-3.5	-3.3		
U236	e=4.5%	-4.6	-4.5	-4.6	-4.2
	Incertitudes	±1.3	±0.9	±0.7	±0.6
	e=3.1%	-10.2	-2.0		
Np237	e=4.5%	-3.8	-4.1	-5.0	-6.0
	Incertitudes	±3.0	±2.8	±2.8	±2.7
	e=3.1%	-7.8	-6.0		
Pu238	e=4.5%	-10.8	-9.0	-8.2	-8.4
	Incertitudes	±4.0	±3.9	±3.8	±3.7
	e=3.1%	-0.1	1.8		
Pu239	e=4.5%	-1.7	-0.4	0.3	0.6
	Incertitudes	±0.9	±1.1	±1.2	±1.3
	e=3.1%	-0.9	-0.6		
Pu240	e=4.5%	-3.5	-2.4	-1.0	-0.8
	Incertitudes	±1.9	±1.5	±1.3	±1.1
	e=3.1%	-3.2	-1.5		
Pu241	e=4.5%	-6.3	-5.0	-3.8	-3.1
	Incertitudes	±2.3	±1.8	±1.6	±1.6
	e=3.1%	-6.7	-7.0		
Pu242	e=4.5%	-10.5	-9.7	-8.8	-8.6
	Incertitudes	±4.0	±3.4	±3.1	±2.8

I_γ U235 underestimated by 10% ←

^{239}Pu predicted to within ±1% ←

^{242}Pu underestimated by 8% ←

JEF-2.2 Trends Derived from FP Sample Oscillations in MINERVE

σ_{Sm149} underestimated by 5%

σ_{Nd143} underestimated by 5%

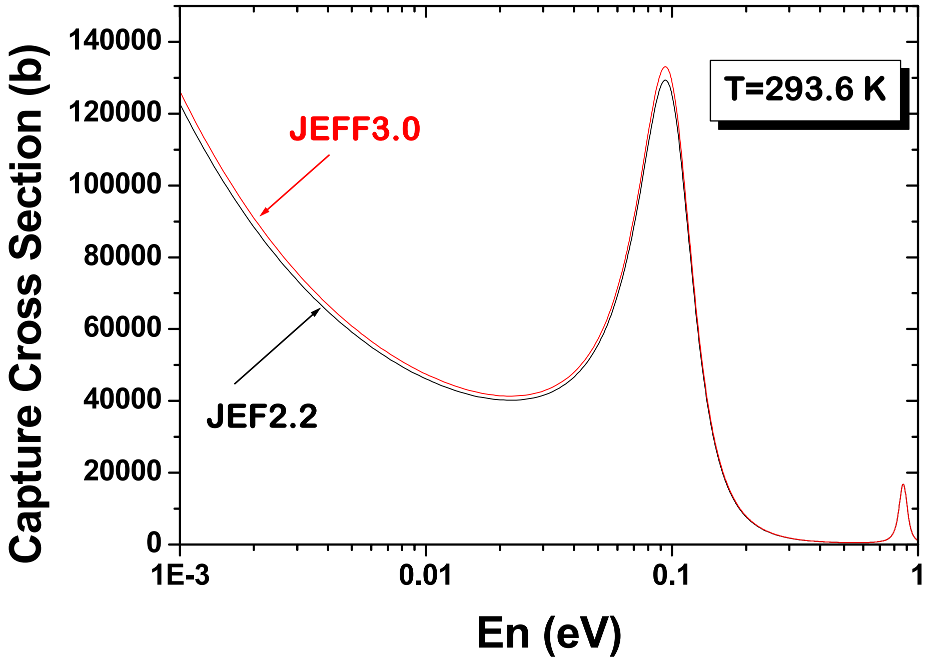
σ_{Rh103} overestimated by +10%

Fission Product	R1-UO_2 thermal core (C-E)/E in %	R1-UO_2 thermal core 1σ exp. Unc. (%)	R2-UO_2 very thermal core (C-E)/E in %	R2-UO_2 very thermal core 1σ exp. Unc. (%)
Sm	- 4.5	2.9	- 3.3	3.6
^{149}Sm	- 6.0	2.9	- 4.9	3.6
^{147}Sm	+ 1.3	4.3	+ 2.7	4.7
^{152}Sm	- 1.6	2.9	- 1.8	3.7
Nd	+ 0.4	3.0	- 3.3	3.7
^{143}Nd	- 7.1	3.1	- 8.5	3.8
^{145}Nd	+ 0.4	3.8	+ 1.1	4.4
^{155}Gd	- 2.5	2.9	- 6.1	4.0
^{103}Rh	+ 11.0	4.0	+ 8.0	4.2
^{103}Rh	-	-	+ 14	9.0
^{109}Ag	- 3.6	4.3	- 4.5	4.3
^{109}Ag	- 4.6	9.0	+ 2.8	6.9
Ag	- 4.7	4.2	+ 0.3	4.7
Mo	+ 1.5	3.2	+ 2.1	3.8
^{133}Cs	- 0.6	3.8	- 2.4	4.3
^{133}Cs	+ 4.1	8.5	+ 9.1	7.3

Analysis of integral Measurements

JEFF-3.0 vs. JEF-2.2 Radiative Capture Cross Section of Sm-149

E_{res} (eV)	Spin	G_{tot} (meV)	G_n (meV)	G_g (meV)	Comment
-0.285	3	62.17	0.16914	62.00	unchanged
+0.0973	4	61.05	0.549	60.5	G_n increased
+0.872	4	60.54	0.7422	59.8	unchanged



3% increase *in the first resonance G_n, compatible with the measurement performed by Pattenden*

	Thermal Value (b)	Resonance Integral (b)
JEF-2.2	40446	3487
JEFF-3.0	41617 (+2.9%)	3490 (+0.1%)

Fast Core C/E Errors and Uncertainties obtained with ERANOS using a JEF-2-based Unadjusted Cross Section Library

JEF2.2 ECCO Library		Average (C-E)/E	Standard Deviation
critical mass	M_c	+ 323 pcm	1460 pcm
buckling	B^2_m	- 210 pcm	1200 pcm
K-infinity	K^+	- 50 pcm	2200 pcm
Spectral Indices	f(Pu-239) / f(U-235)	1.1 %	2.6 %
	f(U-238) / f(U-235)	- 1.0 %	3.7 %
	c(U-238) / f(U-235)	1.4 %	2.2 %
	f(Pu-240) / f(U-235)	- 4.0 %	8.6 %
	f(Pu-241) / f(U-235)	- 1.4 %	5.0 %
	f(Pu-241) / f(U-235)	- 5.2 %	8.0 %
	c(B-10) / f(U-235)	- 2 %	2.3 %

Statistical Adjustment Procedure

Multigroup data σ
+ covariances C_σ

Calculated integral values Q_C
and
sensitivities $S = \sigma/Q \; \partial Q/\partial\sigma$

Measured
integral data Q_E
+ covariances C_Q

Statistical Adjustment Procedure

Min χ^2; $\chi^2 = (\sigma - \sigma_0)^T C_\sigma^{-1} (\sigma - \sigma_0) + [Q(\sigma) - Q_E]^T C_Q^{-1} [Q(\sigma) - Q_E]$

subject to $Q(\sigma) - Q(\sigma_0) = S_0 (\sigma - \sigma_0)$ + **consistency test** on χ^2

Adjusted multigroup data $\sigma*$

Adjusted calculated integral data Q_C*

Fast Core C/E Errors and Uncertainties obtained with ERANOS Using a JEF-2-based Adjusted Cross Section Library

ERALIB1 ECCO Library		Average (C-E)/E	Standard Deviation
critical mass	M_c	+ 83 pcm	100 pcm
buckling	B_m^2	- 260 pcm	150 pcm
K-infinity	K^+	123 pcm	240 pcm
Spectral Indices	f(Pu-239) / f(U-235)	0.3 %	0.5 %
	f(U-238) / f(U-235)	- 1.0 %	0.8 %
	c(U-238) / f(U-235)	1.0 %	0.5 %
	f(Pu-240) / f(U-235)	- 1.3 %	1.5 %
	f(Pu-241) / f(U-235)	0.5 %	1.2 %
	f(Pu-241) / f(U-235)	- 1.6 %	1.3 %
	c(B-10) / f(U-235)	- 1.3 %	0.8 %

- The statistical adjustment is based on **over 350 integral data** obtained in various facilities worldwide, consistently analysed with the same data + code system (ERANOS)

Measured β_{eff} values and associated uncertainties

Facility	Configuration	Experimental technique	β_{eff} (*pcm*)	Unc. in P_m (%)	Unc. in P_c (%)	Std. dev. (%)
MASURCA	R2	Cf Source	723.5 (739, 708)	3.2	1.2	3.4
		Frequencies	726.4	3.2	2.5	2.3
		α-Rossi	745.0	2.5	2.5	1.8
	ZONA2	Cf Source	353.7 (358.6, 348.7)	3.5	1.2	3.5
		Frequencies	350.0	3.2	2.9	2.2
SNEAK	7A	Cf Source	395.0	2.4	1.5	2.8
	7B		429.0	2.4	1.5	2.8
	9C1		748.0	3.9	1.5	4.2
	9C2		416.0	4.3	1.5	4.6
ZPR	Cref	Covariances	383.6	3.5	2.6	2.2
	PuCSS		223.4	3.5	3.0	2.3
	RSR		337.3	3.5	2.7	2.2
	U9		712.2	3.5	2.2	2.1
	UFeRef		670.8	3.5	2.4	2.1
	UFeLeak		675.8	3.5	2.4	2.1
FCA	XIX-1	Frequencies	742 (742, 742)	4.0	2.6	2.4
	XIX-3		249.1 (252, 246.2)	4.0	3.1	2.5
EOLE	MISTRAL-1	Frequencies	789.7	2.3	2.0	1.6
	MISTRAL-2		372.5	2.3	2.0	1.6
	SHE-8	Kinetics	696.0			4.6

Broad Group Trends Identified in JEF-2.2 Delayed Neutron Yields

Group / Isotope	1 0-10 keV	2 10-500 keV	3 0.5-4 MeV
U-235	-2.0% ± 1.3%	0.4.% ± 1.6%	0.3% ± 3.5%
U-238	0.0% ± 5.9%	0.0% ± 5.9%	0.2.% ± 2.4%
Pu-239	0.4% ± 1.7%	1.9% ± 2.6%	1.2% ± 4.1%

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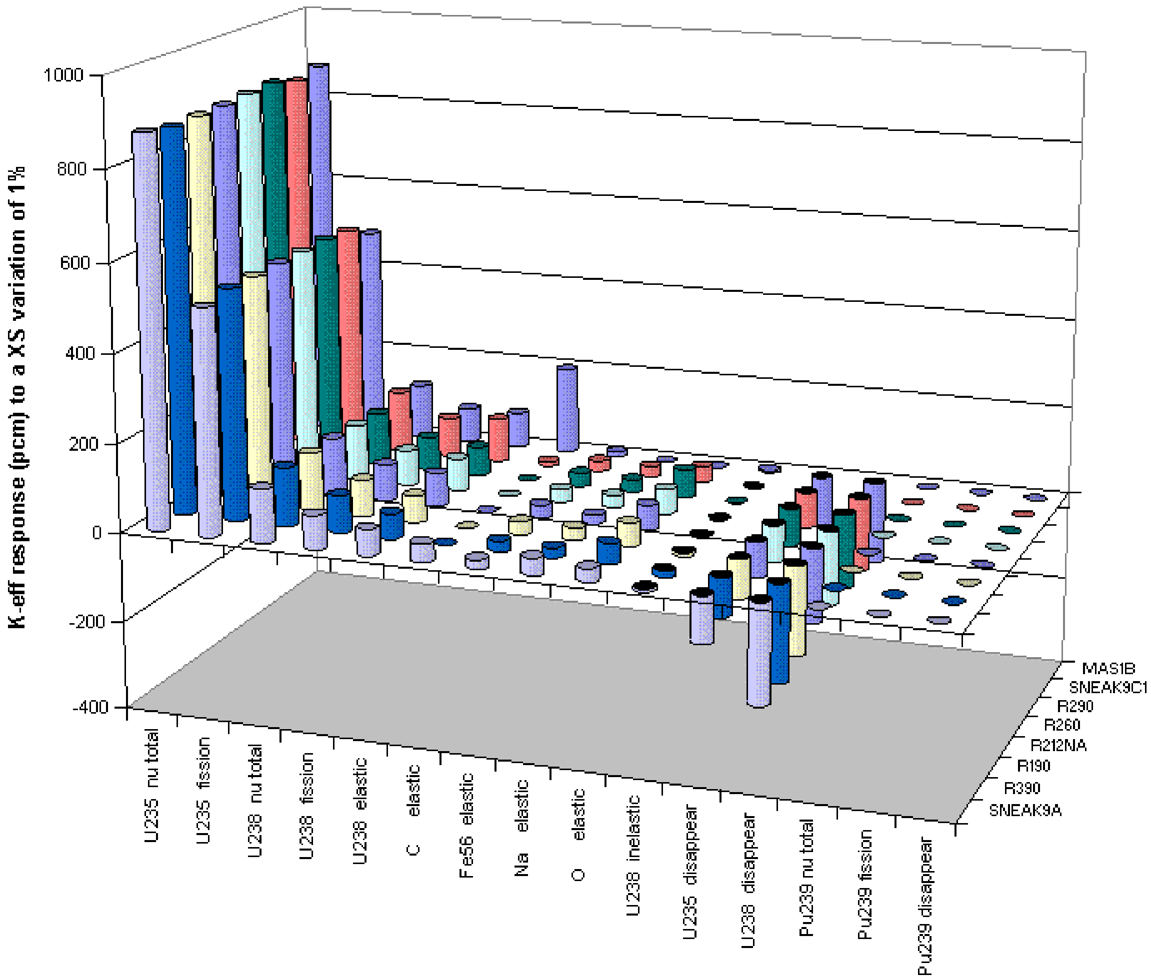
Changes in k_{eff} induced by the change from JEF-2.2 to JEFF-3.0 in ^{235}U and ^{238}U data (from JEFDOC-956)

Fast Uranium Cores	k_{eff} (JEFF-3.0) – k_{eff} (JEF-2.2) [pcm]			
	Δk (library)	Δk (^{235}U) [2]	Δk (^{238}U) [2]	$\Delta k(^{235}U)+\Delta k(^{238}U)$
R390	-623	-455	-121	-576
SNEAK9A	-743	***-740***	-77	-817
R190	-659	-425	-109	-534
MAS1B	***-907***	***-803***	-64	-867
R212Na	-658	-314	-87	-401
R260	-633	-340	-71	-411
R290	-660	-355	-63	-418
SNEAK9C1	***-508***	-297	-72	-369

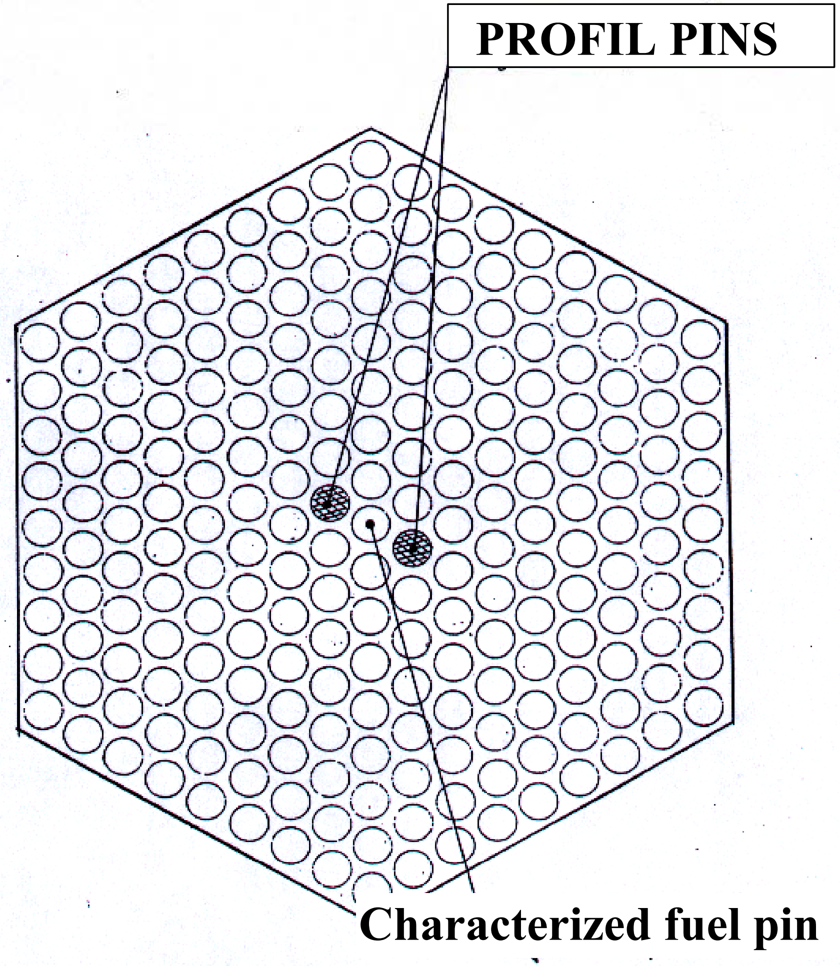
Analysis of integral Measurements

k_{eff} Sensitivities to the Main Nuclear Data for Fast U cores (from JEFDOC-956)



Analysis of integral Measurements

PROFIL Experiments in PHENIX



Analysis of integral Measurements

PROFIL Experiments in PHENIX

- **PROFIL-1**
  - ✓ One experimental pin, in the central fuel S/A of Phénix
  - ✓ Burnup from 01/1974 to 01/1975 for 179 efpd
  - ✓ **46 samples**
  - ✓ U235, U238, Pu238, Pu239, Pu240, Pu241, Pu242, Am241
  - ✓ Mo95, Mo97, Ru101, Pd105, Cs133, Nd145, Sm149, B, Li

- **PROFIL-2**
  - ✓ Two experimental pins in a fuel S/A next to the central one
  - ✓ Burnup from 07/1979 to 09/1980 for 316 efpd
  - ✓ **2 x 42 samples**
  - ✓ Th232, U233, U234, U235, U238, Np237, Pu238, Pu239, Pu240, Pu241, Pu242, Am241, Am243, Cm244
  - ✓ Zr92, Pd106, Nd143, Nd144, Sm147, Sm151, Eu153

Results of the PROFIL-1 and –2 Analysis with JEF-2.2 Data

C/E ± uncertainty(%)

Nuclide	Reaction	PROFIL-1	PROFIL-2
U233	capture	—	0.93 ± 3.0
U234	capture	—	0.99 ± 3.0
U235	capture	0.93 ± 1.7	0.92 ± 1.7
	n,2n	0.95 ± 5.0	0.96 ± 5.0
U238	capture	0.98 ± 2.3	0.99 ± 2.3
	fission	1.00 ± 1.4	—
Np237	capture	—	0.97 ± 3.6
	n,2n	—	1.20 ± 4.7
Pu238	capture	0.97 ± 4.0	0.99 ± 4.0
Pu239	capture	0.96 ± 3.0	0.96 ± 3.0
	n,2n	0.63 ± 15.0	0.58 ± 15.0
Pu240	capture	1.10 ± 2.2	1.13 ± 2.2
	n,2n	1.13 ± 20.0	0.88 ± 20.0
Pu241	Capture	1.24 ± 4.1	1.18 ± 5.9
	fission	0.98 ± 3.3	—
	n,2n	1.04 ± 4.1	—
Pu242	capture	1.18 ± 3.5	1.12 ± 4.3
	fission	0.94 ± 8.6	—
Am241	capture	1.03 ± 1.7	1.03 ± 1.7
Am243	capture	0.96 ± 5.0	—

Contribution of the PROFIL-1 and –2 Data to the JEF-2.2 Statistical Adjustment

E/C-1 (%) ± uncertainty (%)

Nuclide	*Reaction*	*Before adjustment*	*After adjustment (without PROFIL)*	*After adjustment (with PROFIL)*
U235	capture	+ 7.5 ± 1.9	+ 6.0 ± 1.9	+ 2.1 ± 1.9
U238	capture	+ 1.0 ± 2.2	+0.8 ± 2.2	+ 1.0 ± 2.2
Pu239	capture	+ 4.2 ± 2.6	+ 3.1 ± 2.6	+ 0.9 ± 2.6
Pu240	capture	- 9.9 ± 2.2	- 6.1 ± 2.2	- 1.1 ± 2.2
Pu241	capture	- 18.0 ± 3.9	- 17.7 ± 3.9	- 2.4 ± 3.9
Pu242	capture	- 13.0 ± 3.2	- 13.8 ± 3.2	- 2.1 ± 3.2

Conclusion

- **Modern deterministic neutron data** and code systems are capable of predicting nuclear reactor core characteristics **with very good accuracy** for conventional LWR's and FR's, especially in view of other potential sources of errors

- This has been **demonstrated** via
  - ✓ A large number of **numerical validation** calculations, particularly comparisons with Monte Carlo codes
  - ✓ The analysis of a large number of **physics experiments** performed in critical facilities and power reactors

- **Improved performance** is nonetheless required to achieve
  - ✓ **Additional margin gains**
  - ✓ **Better predictive power**
  - ✓ **A broader range of application**

- **This will require,** among other things,
  - ✓ Further **improvements in evaluated nuclear data**
  - ✓ **Additional integral data**, more diverse and more accurate