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Sensitivity and Uncertainty Analysis for the Neutronic Design of Fast Reactor Systems

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Background

- In general, the uncertainty analysis performed using current covariance data shows that the present integral parameters uncertainties resulting from the assumed uncertainties on nuclear data are probably acceptable in the early phases of design feasibility studies.
- However, in the successive phase of preliminary conceptual designs and in later design phases of selected reactor and fuel cycle concepts, there is the need for improved data and methods, in order to reduce margins, both for economic and safety reasons.
- It is then important to define as soon as possible priority issues, i.e. which are the nuclear data (isotope, reaction type, energy range) that need improvement, in order to quantify target accuracies and to select a strategy to meet the requirements needed (e.g. by some selected new differential measurements and by the use of integral experiments).

Background

- The ultimate goal is a design that has as low as possible uncertainties. Industry and utilities want reduced uncertainty for economical reasons: design and operation), while safety authorities want “guaranteed margins” that they can trust.
- There are two main sources of uncertainties: input data, and modeling
  - Example of input physical data: cross sections, fabrication data, etc.
  - Modeling uncertainties: coming from approximations made in the computational methodology used in the design process.
- High-fidelity simulation can provide a major benefit if it can reduce to the smallest amount the impact of uncertainties coming from the modeling of the physical processes.
- A scientific based approach can allow a reliable propagation of uncertainties and a correct evaluation of the impact of the uncertainty coming from the input data.

Background

- Input data uncertainties can be divided in two main categories:
  - Epistemic Uncertainties

    These are uncertainties related to lack of knowledge. Typical examples include thermal-physical properties for thermal-hydraulics and fuel behavior simulations, and cross sections in neutronics. Reduction of the epistemic uncertainty can be performed when useful and relevant experimental information is available.
  - Aleatory Uncertainties (Random Phenomena)

    These are intrinsic and in principle not reducible uncertainties associated to random phenomena. However, they have to be identified and characterized (with appropriate probability distributions) and quantified. Typical examples include: geometrical description, environment conditions (e.g. operating conditions, change from initial configurations, etc.), tolerances in specifications, impurities, boundary conditions, etc. The aleatory uncertainty can be rigorously treated for propagation in the computer model.

Background

- Calculation methods (approximations), and modeling uncertainties. They can be divided further:
  - The governing equations are well established and known. In this case, advanced simulation can be pushed to minimize the source of errors, or a hierarchical approach (if methodology is deterministic), or a probabilistic one (if the methodology is stochastic), can either extrapolate to the "right solution", or quantify the magnitude of the errors. The residual systematic errors introduced by the modelling and associated numerical treatment can be also taken into account in a rigorous manner within an assimilation/adjustment methodology.
  - The governing equations are not yet very well known. In some cases this is equivalent to an epistemic uncertainty. A typical example is the description of fuel behavior under irradiation. In this case only the experimental information can drive to some acceptable solution (i. e. compensate for the incorrectness of the model). Frequently, provided that the calculation method is in a domain range of applicability, a "tuning" methodology is employed. Examples in thermal-hydraulics and fuel behavior where correlation factors are derived from experiments and tuned within the model.

Sample of Target Reactor Performance (Fast Reactor)

Parameter	Current Uncertainty [a]				Targeted Uncertainty
	Input data origin (a priori)	Input data origin [b] (av. exper.)	Modeling origin	Total	
Neutronics [c]					
Core					
Multiplication factor, K_{eff} (Δk/k)	1%	0.2%	0.5%	0.5%	0.3%
Power peak	1%	1%	3%	3%	2%
Power distribution[d]	1%	1%	6%	6%	3%
Conversion ratio (absolute value in %)	5%	5%	2%	5%	2%
Control rod worth: Element	5%	4%	6%	7%	5%
Control rod worth: Total	5%	4%	4%	5%	2%
Burnup reactivity swing (Δk/k)	0.7%	0.5%	0.5%	0.7%	0.3%

Sample of Target Reactor Performance (Fast Reactor)

Parameter	Current Uncertainty [a]				Targeted Uncertainty
	Input data origin (a priori)	Input data origin [b] (av. exper.)	Modeling origin	Total	
Neutronics [c]					
Core					
Reactivity coefficients: total	7%	5%	15%	15%	7%
Reactivity coefficients: component	20%	10%	20%	20%	10%
Fast flux for damage	7%	6%	3%	7%	3%
Kinetics parameters	10%	10%	5%	10%	5%
Local nuclide densities: Major	5%	4%	3%	5%	2%
Local nuclide densities: Minor	30%	20%	10%	20%	10%
Fuel decay heat at shutdown	10%	10%	3%	10%	5%

Sample of Target Reactor Performance (Fast Reactor)

Parameter	Current Uncertainty [a]				Targeted Uncertainty
	Input data origin (a priori)	Input data origin [b] (av. exper.)	Modeling origin	Total	
Neutronics [c]					
Shielding					
Out of Core Coolant Activation	70%	70%	70%	100%	50
Shield Dimensioning (Total Flux)	70%	40%	30%	50%	20%
Structural Damage Out of Core (Total Flux)	40%	40%	30%	50%	20%

Sample of Target Reactor Performance (Fast Reactor)

Parameter	Current Uncertainty [a]				Targeted Uncertainty
	Input data origin (a priori)	Input data origin [b] (av. exper.)	Modeling origin	Total	
Neutronics [c]					
Fuel Cycle					
Neutron Dose at Fuel Fabrication	15%	15%	15%	20%	10%
Decay Heat of Spent Fuel at Repository	50%	50%	15%	50%	20%
Radiotoxicity at repository	50%	50%	15%	50%	20%

Background

- Sensitivity and uncertainty analyses are the main instruments for dealing with the sometimes scarce knowledge of the input parameters used in simulation tools.
- Sensitivity coefficients are the key quantities that have to be evaluated. They are determined and assembled, using different methodologies, in a way that when multiplied by the variation of the corresponding input parameter they will quantify the impact on the targeted quantities whose sensitivity is referred to.
- Sensitivity coefficients can be used for different objectives:
  - uncertainty estimates,
  - design optimization,
  - determination of target accuracy requirements,
  - adjustment of input parameters,
  - evaluations of the representativity of an experiment with respect to a reference design configuration.

Uncertainty Evaluation

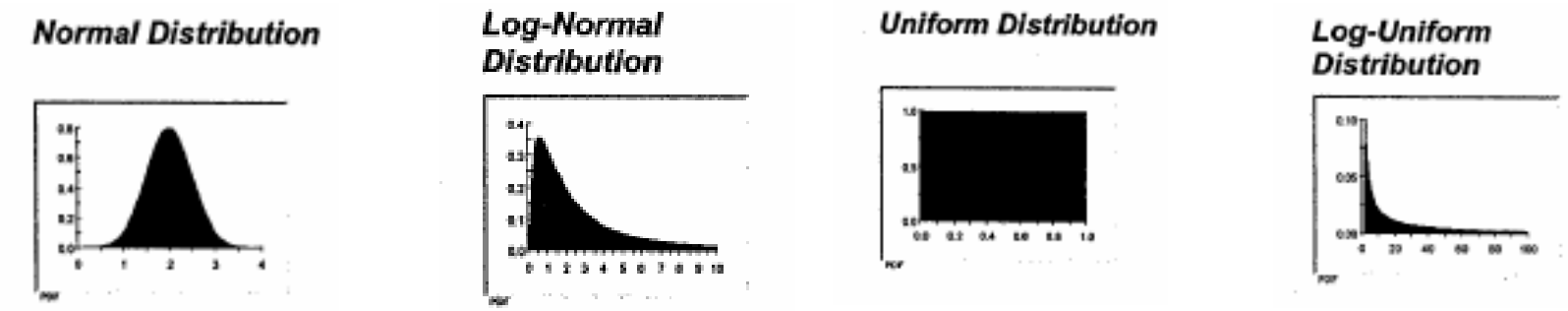
- In uncertainty assessment, the sensitivity coefficients are multiplied by the uncertainties of the input parameters in order to obtain the uncertainty of the targeted parameter of interest.
- The origin and quality of the uncertainties of the input parameters can be different and vary quite a lot:
  - In some cases, they are provided by the expert judgment of qualified designer.
  - In some other cases more useful information is available, for instance from experimental values, and they are cast in more rigorous formalism (e.g. covariance matrix for neutron cross sections, where correlations in energy and among the different input parameters, like reactions and isotopes, are also provided).

$$\Delta R^2 = S_R^+ D S_R$$

where ΔR is the uncertainty, S_R are the sensitivity coefficients arrays, and D is the covariance matrix.

Uncertainty Evaluation (cont.)

- Uncertainty evaluation can be performed also without the help of sensitivity coefficients. Uncertainties on input parameters can be propagated either using a stochastic approach (Monte Carlo methods type) or by some regression techniques.
- In the case of the Monte Carlo methodology, several runs of the same problems are performed with different random input values, taken within the range of the specified uncertainty and associated distribution law, and then at the end the final results are statistically combined in order to determine the average value and the associated standard deviation.



- Smarter sampling techniques (e.g. Latin Hypercube) for Monte Carlo simulations are developed in order to minimize the total number of direct calculations.

Design Optimization

- Design optimization can take advantage of sensitivity coefficients by using them in optimization algorithms.
- The main problem in this case is related to the fact that in most cases the sensitivity coefficients are calculated with linear approximation: they need to be determined repeatedly to take into account the nonlinear effects.
- There is also the problem of taking into account multi-physics effects.
- Sensitivity coefficients are evaluated only relative to one field (e.g. neutronics or thermal-hydraulics).

Comparison of ALPSMAP-generated Loading Patterns with Actual Design (courtesy of T. K. Kim)

Design Loading Pattern on NDR

33.4	16.2	17.7	24W 0.0	22.8	16W 0.0	17.7	36.4
	22.3	24W 0.0	23.9	16W 0.0	21.1	- 0.0	39.8
		22.6	24W 0.0	18.8	16W 0.0	23.2	
			22.23	16W 0.0	- 0.0	38.8	
				16.2	37.9		

Number of BP
Burnup(MWD/T)

Initial Loading Pattern

33.4	16.2	17.7	24W 0.0	22.8	16W 0.0	17.7	36.4
	22.3	24W 0.0	23.9	16W 0.0	38.8	- 0.0	39.8
		22.6	24W 0.0	18.8	16W 0.0	- 0.0	
			22.3	16W 0.0	23.2	21.1	
				16.2	37.9		

Optimum Loading Pattern

33.4	22.3	21.1	24W 0.0	22.8	16W 0.0	23.9	36.4
	22.6	24W 0.0	17.7	16W 0.0	18.8	- 0.0	39.8
		21.1	24W 0.0	16.2	16W 0.0	23.2	
			22.3	16W 0.0	- 0.0	38.8	
				23.9	37.9		

Twice burned Once burned Fresh

	Initial LP	Optimum LP	NDR
EOC SB [1)]	1 ppm	134 ppm	121 ppm
PPPF [2)]	1.481	1.522	1.486
CPU time [3)]	-	11708sec	-

1) EOC Burnup = 16400 MWD/T

2) Design limitation =1.527

2) Computer =IBMPC with Pentium-II Processor

Current Methodologies

- There are two main methodologies developed for sensitivity and uncertainty analysis:
  - the forward (direct) calculation method
  - the adjoint method

- The forward approach is preferable when there are few input parameters that can vary and many output parameters of interest.

- The adjoint methodology is preferable when there are a limited number of object parameters and a very large number of input parameters that are uncertain.

Forward Methods

- **Stochastic (probabilistic with Monte Carlo method) has some drawbacks:**
  - **large number of direct calculations**
  - **only uncertainty can be evaluated and sensitivity coefficients cannot be directly obtained**

  The method has been widely used in other fields, (e. g. DAKOTA, GOLDSIM).
- **Automatic differentiation:**
  - ***codes are directly modified in order to evaluate derivatives, through direct calculations, for all input parameters that are deemed to vary***
  - ***it can be very computational intensive***
  - ***it requires direct intervention within the code***
  - ***Software exists that directly modifies a code to add automatic differentiation if the used language of programming is FORTRAN or C***

Adjoint Methods

- The adjoint methodologies are based on the perturbation theory originally developed in the quantum mechanics field.
- Classical perturbation theory that makes use of the adjoint function (also called importance), has been widely used in neutronics to calculate the variation of the fundamental eigenvalue.
- For the generalized perturbation theory (GPT) a generalized importance is calculated for each output parameter of interest by solving an inhomogeneous adjoint neutron transport equation that contains a source term depending on a specific output parameter.

Adjoint Methods

- The adjoint methodology type of approach has been extended to other fields including nuclide depletion calculations where the adjoint solution of the Bateman equation is used, and Depletion Perturbation Theory (DPT) calculates the importance functions for the coupled neutron and nuclide field.
- Oblow and others have extended the adjoint methodology to the thermal-hydraulics field. Cacuci, Park, and Gandini have developed adjoint methodologies for time-dependent transient problems for application to safety analysis or reactor operation optimization.
- Automatic differentiation tools employing the so-called reverse mode are able to compute a discrete adjoint; in practice, the reverse mode requires more user intervention than forward sensitivity computations.

Adjoint Methods

- The main drawback of the adjoint methodology is related to the number of adjoint functions that have to be calculated if there is a large number of objective parameters. In many cases, the memory requirements for the adjoint method are significant, as many intermediate states must be recorded. Also inconvenient is the fact that the adjoint solution has to be coded directly inside of the code.
- Among the existing codes that are widely used, mostly in neutronics, we can list: VARI3D and its DPT version at ANL, the sensitivity capability of FORMOSA system (mainly for thermal reactor applications) at the North Carolina State University, the TSUNAMI (limited only to K_{eff}) and FORSS system at ORNL, the sensitivity and uncertainty modules that are part of the European fast reactor code system ERANOS.

Historical Notes

- The perturbation theory has been introduced in reactor physics in the 50' and one can find a classical presentation in the Weinberg and Wigner book. This is the perturbation theory applied to the k_{eff} of the critical reactor and L. N. Usachev gave a comprehensive development in an article published at the Geneva conference of 1955.
- It is interesting to note that the the perturbation theory applied to reactor makes use of a definition of a function (the adjoint flux), that has a specific physical meaning if one is dealing with a non-conservative system as in the case of a nuclear reactor. This physical interpretation of the adjoint flux has been the focus of extensive studies, during the 60', in particular by J. Lewins.
- The perturbation theory, mostly developed and applied for reactivity coefficient studies, was readily used for an application, sensitivity studies, that had a spectacular development in the 70' and 80'. This development was made possible by a generalization of the perturbation theory (thanks again to Usachev), that deals with the general problem of a variation of any kind of a neutron flux functional. Usachev derived an explicit formulation that relates the functional variation to any change of the Boltzmann operator.

Historical Notes

- This development, and its further generalization by Gandini, to the case of any kind of linear and bilinear functional of the real and adjoint flux, opened a new territory for the perturbation theory. It was now possible to relate explicitly the variation of any type of integral parameter (multiplication factor, reaction rates, reactivity coefficients, source values, etc.) to any kind of change of the operator that characterizes the system.
- The application of the generalized perturbation theory to real life problems lead to new interesting developments that allowed to clarify specific characteristics of the new theory with implications for the computation of the generalized importance functions introduced by the theory.
- Starting from the early 70' the generalized perturbation methods, which were essentially developed and used in Europe, became popular also in the rest of the world and in particular with new developments in several U. S. laboratories, ANL and ORNL, and in Japan.

Historical Notes

- The perturbation methods, and their main application in the field of sensitivity analysis, have been used mostly in their first order formulation. Actually, as for any perturbation theory, the power of the method is particularly evident when one considers small perturbations (for instance for cross-sections σ) that therefore induce little changes of the functions (e. g. the neutron flux φ), that characterize the system, and for whom one can neglect the second order product (for instance δσδφ). However, there have been theoretical developments that take into accounts higher order effects without losing all the advantages typical of the first order formulations.
- Among the theoretical developments after the 70' that had significant practical impact, one has to mention the extension of the perturbation theory to the nuclide field that allows to study the burn up due to irradiation in the reactor at the first order, and to higher orders. Subsequently a new formulation, the "Equivalent Generalized Perturbation Theory" EGPT, allowed to treat in a very simple and efficient way the perturbation and sensitivity analyses for reactivity coefficients.
- Among the most recent development it is worth to mention those related to the ADS (Accelerator Driven System) case with functionals that allow to calculate the sensitivity of the source importance (φ*) and the inhomogeneous reactivity.
- Finally, one should remind that, besides the neutronic field, there have been several studies for extending the perturbation theory developed for reactor physics to other domains (thermal-hydraulics, safety, etc.) with very interesting theoretical developments.

Theory

Homogeneous and inhomogeneous Boltzmann equations:

$$A\Phi = \frac{F\Phi}{K_{eff}} \quad \textbf{(1)} \quad A^*\Phi^* = \frac{F^*\Phi^*}{K_{eff}} \quad \textbf{(2)} \quad A\Phi_s = F\Phi_s + S \quad \textbf{(3)}$$

Where in multigroup notation:

$$A_g = \Omega\Delta\psi_g + \sigma_t\psi_g - \sum_g \sigma_{g\to g'}\Phi_{g'} \quad \textbf{(4)}$$

$$F_g = \chi_g \sum_{g'} \nu\sigma_f^{g'}\Phi_{g'} \quad \textbf{(5)}$$

Theory

$$\Phi = \int \psi \, d\,\Omega \qquad (6)$$

The inhomogeneous multiplication factor is defined as:

$$k_S = \frac{\langle F\Phi_S \rangle}{\langle A\Phi_S \rangle} \qquad (7)$$

$$S_m = \frac{S}{1 - k_S} \qquad (8)$$

The inhomogeneous reactivity is defined as:

$$\rho_S = 1 - \frac{1}{k_s} = \frac{\langle F\Phi_S \rangle - \langle A\Phi_S \rangle}{\langle F\Phi_S \rangle} = -\frac{\langle S \rangle}{\langle F\Phi_S \rangle} \qquad (9)$$

Theory

Classical Perturbation Theory

We consider the perturbed equation:

$$A'\Phi' = \frac{F'\Phi'}{K'_{eff}} \qquad (10)$$

Multiplying Eq. (2) by Φ' and Eq. (10) by Φ* and then subtracting, we obtain:

$$\delta\rho = \frac{1}{k'} - \frac{1}{k} = \frac{\left\langle \Phi^*, (\partial A - \partial F)\Phi' \right\rangle}{\left\langle \Phi^*, \frac{1}{k'} F'\Phi' \right\rangle} \qquad (11)$$

For the first order approximation we consider $\Phi' \approx \Phi$

Theory

Sensitivity Coefficients

The variations of any integral parameter Q due to variations of cross sections σ can be expressed using perturbation theories to evaluate sensitivity coefficients S:

$$\delta Q/Q = \sum_j S_j \frac{\delta \sigma_j}{\sigma_j} \qquad (12)$$

where the sensitivity coefficients S_j are formally given by:

$$S_j = \frac{\partial Q}{\partial \sigma_j} \cdot \frac{\sigma_j}{Q} \qquad (13)$$

For the Classical Perturbation Theory this gives:

$$S_j = \frac{\partial K}{\partial \sigma_j} \cdot \frac{\sigma_j}{K} \qquad (14)$$

Theory

In the the case of the Generalized Importance Theory (GPT), for practical purposes, in the expression of any integral parameter Q, the explicit dependence from some cross-sections (e.g. σ_i^e) and the implicit dependence from some other cross-sections (e.g. σ_j^{im}) are kept separated:

$$Q = f(\sigma_j^{im}, \sigma_i^e) \qquad (15)$$

As an example, we consider a reaction rate:

$$R = \left\langle \underline{\sigma}^e, \underline{\Phi} \right\rangle \qquad (16)$$

In Eq. (16), $\underline{\sigma}^e$ can be an energy dependent detector cross-section, R is "explicitly" dependent on the $\underline{\sigma}^e$ and "implicitly" dependent on the cross-sections which characterize the system, described by the flux $\underline{\Phi}$. In other terms, R depends on the system cross-sections via $\underline{\Phi}$.

Theory

Equation (12) can be rewritten as follows:

$$\delta Q / Q = \sum_j S_j \frac{\delta \sigma_j^{im}}{\sigma_j^{im}} + \left(\frac{\partial Q}{\partial \sigma^e} \cdot \frac{\sigma^e}{Q} \right) \cdot \frac{\delta \sigma^e}{\sigma^e} \qquad (17)$$

where we have the hypothesis of an explicit dependence of Q on only one σ^e. If we drop the index "*im*":

$$\delta Q / Q = \sum_j S_j \frac{\delta \sigma_j}{\sigma_j} + \left(\frac{\partial Q}{\partial \sigma^e} \cdot \frac{\sigma^e}{Q} \right) \cdot \frac{\delta \sigma^e}{\sigma^e} = I + D \qquad (18)$$

where the term I is generally called "indirect" effect, and the term D is called "direct" effect. While the direct effects can be obtained with explicit expressions of the derivatives of Q, the indirect effect (i.e. the sensitivity coefficients S), can be obtained with perturbation expression, most frequently at the first order.

Theory

For a reaction rate in a source driven system:

$$R_S = \left\langle \sigma_f \Phi_S \right\rangle \tag{19}$$

Sensitivity coefficients are calculated using GPT (Generalized Importance Theory). :

$$\frac{\sigma}{\left\langle \sigma_f \Phi_S \right\rangle} \frac{d\left\langle \sigma_f \Phi_S \right\rangle}{d\sigma} = \sigma \left\{ \frac{\left\langle \left(\frac{\partial \sigma_f}{\partial \sigma} \right) \Phi_S \right\rangle}{\left\langle \sigma_f \Phi_S \right\rangle} - \left\langle \tilde{\Psi}^*, \left(\frac{\partial A}{\partial \sigma} - \frac{\partial F}{\partial \sigma} \right) \Phi_S \right\rangle \right\} \tag{20}$$

Theory

The Generalized Importance Function Ψ* satisfies the equation:

$$\left(A^* - F^*\right)\tilde{\Psi}^* = \frac{\partial R_S}{\partial \Phi_S} = \sigma_f(r,E) \qquad (21)$$

Using Eq. (3) and Eq. (21) we can establish the principle of conservation of importance:

$$R_S = \left\langle \sigma_f \Phi_S \right\rangle = \left\langle S\tilde{\Psi}* \right\rangle \qquad (22)$$

When a perturbation is made:

$$A'\Phi'_s = F'\Phi'_s + S \qquad (23)$$

Theory

We can calculate the variation of the integral parameter for the indirect effect as:

$$\delta R_S = R'_S - R_S = \left\langle \tilde{\Psi}^*, (\partial A - \partial F)\Phi' \right\rangle \qquad (24)$$

In the case of a reaction rate ratio:

$$I_S = \frac{\left\langle \sigma_f \Phi_1 \right\rangle}{\left\langle \sigma_f \Phi_2 \right\rangle} \qquad (25)$$

The adjoint importance satisfies the following equation:

$$\left(A^* - F^*\right)\tilde{\Psi}^* = \frac{\sigma_f(r,E)}{\left\langle \sigma_f \Phi_1 \right\rangle} - \frac{\sigma_f(r,E)}{\left\langle \sigma_f \Phi_2 \right\rangle} \qquad (26)$$

Theory

For a critical system, only reaction rate ratio can be calculated, otherwise there is no solution to the generalized importance equation. The source has to be orthogonal to the direct flux and give no contribution to the total balance:

$$\left(A^* - \frac{F^*}{K} \right) \tilde{\Psi}^* = \frac{\sigma_f(r,E)}{\langle \sigma_f \Phi_1 \rangle} - \frac{\sigma_f(r,E)}{\langle \sigma_f \Phi_2 \rangle} \qquad (27)$$

If we, for instance, consider the power peak, this parameter can be expressed as the ratio:

$$R = \frac{\langle \Sigma_p \underline{\Phi} \rangle_{MAX}}{\langle \Sigma_p \underline{\Phi} \rangle_{Reactor}} = \frac{I_1}{I_2} \qquad (28)$$

with Σ_p the power cross-section, essentially represented by $E_f \bullet \Sigma_f$, where E_f is the average energy released per fission.

Theory

The direct effect sensitivity coefficient for Σ_p are (numerator and denominator) defined as:

$$S_j = \frac{\partial I_1}{\partial \Sigma_p} \cdot \frac{\Sigma_p}{I_1} - \frac{\partial I_2}{\partial \Sigma_p} \cdot \frac{\Sigma_p}{I_2} \qquad (28)$$

The indirect sensitivity coefficients are defined as:

$$S_j = \left\langle \underline{\Psi}^*, \sigma_j \underline{\Phi} \right\rangle \qquad (29)$$

and Ψ^* is the importance function solution of:

$$\left(A^* - \frac{F^*}{K} \right) \tilde{\Psi}^* = \frac{\Sigma_p(r,E)_{Max}}{I_1} - \frac{\Sigma_p(r,E)_{Reactor}}{I_2} \qquad (30)$$

Theory

Sensitivity Coefficients : The Case of Reactivity Coefficients (EGPT)

For the Equivalent Generalized Perturbation Theory the integral parameter is defined as:

$$I_s = \frac{1}{k'} - \frac{1}{k} = \frac{\left\langle \Phi^*, (\partial A - \partial F)\Phi' \right\rangle}{\left\langle \Phi^*, \frac{1}{k'} F'\Phi' \right\rangle} \qquad (31)$$

Where $(\partial A - \partial F)$ characterizes the reactivity coefficient and the sensitivity coefficients are calculated using the fact that changing the order of the derivatives does not change the results:

$$\frac{\sigma}{I_s}\frac{dI_s}{d\sigma} = \frac{\left\langle \Phi'^*, (\partial A' - \partial F')\Phi' \right\rangle}{\left\langle \Phi'^*, \frac{1}{k'} F'\Phi' \right\rangle} - \frac{\left\langle \Phi^*, (\partial A - \partial F)\Phi \right\rangle}{\left\langle \Phi^*, \frac{1}{k} F\Phi \right\rangle} \qquad (32)$$

Theory

A reactivity coefficient (like the Doppler effect) can be expressed as a variation of the reactivity of the unperturbed system (characterized by a value K of the multiplication factor, a Boltzmann operator M, a flux $\underline{\Phi}$ and an adjoint flux $\underline{\Phi}^*$) :

$$\Delta\rho = \left(1-\frac{1}{K_p}\right)-\left(1-\frac{1}{K}\right)=\frac{1}{K}-\frac{1}{K_p} \qquad (33)$$

where K_p corresponds to a variation of the Boltzmann operator such that :

$$\begin{array}{ll} M \to M_p\left(=M+\delta M_p\right) & \underline{\Phi} \to \underline{\Phi}_p\left(=\underline{\Phi}+\delta\underline{\Phi}_p\right) \\ \underline{\Phi}^* \to \underline{\Phi}^*_p\left(=\underline{\Phi}^*+\delta\underline{\Phi}^*_p\right) & K \to K_p\left(=K+\delta K_p\right) \end{array} \qquad (34)$$

The sensitivity coefficients (at first order) for $\Delta\rho$ to variations of the σ_j are given as :

$$S_j^{RO} = \frac{\partial(\Delta\rho)}{\partial\sigma_j}\cdot\frac{\sigma_j}{\Delta\rho} = \left\{\frac{1}{I_f^p}\left\langle\underline{\Phi}^*_p,\sigma_j\underline{\Phi}_p\right\rangle-\frac{1}{I_f}\left\langle\underline{\Phi}^*,\sigma_j\underline{\Phi}\right\rangle\right\} \qquad (35)$$

where $I_f=\left\langle\underline{\Phi}^*,F\underline{\Phi}\right\rangle$ and $I_f^p=\left\langle\underline{\Phi}^*_p,F\underline{\Phi}_p\right\rangle$

Theory

The external source importance is defined as:

$$\varphi^* = \frac{\bar{s}^*}{\bar{\chi}^*} = \frac{\langle \Phi^*, S \rangle}{\langle S \rangle} \Bigg/ \frac{\int dr \left[\Phi^* \chi\right]\left[\Sigma_f \Phi_s\right]}{\langle \Sigma_f \Phi_s \rangle} \qquad (36)$$

For the sensitivity analysis we introduce the function:

$$G = I_S - \langle \Psi^*, (A-F)\Phi_S - S \rangle - \left\langle \Psi, \left(A^* - \frac{1}{k} F^*\right)\Phi^* \right\rangle \qquad (37)$$

Requiring this function to be stationary with respect to the variations of Φ_s and Φ^* leads to the equations for the direct and adjoint importance functions:

Theory

$$\left(A^* - F^*\right)\tilde{\Psi}^* = \frac{1}{I_S}\frac{\partial I_S}{\partial \Phi_S} = -\frac{\left[\Phi^* \chi\right]\Sigma_f(r,E)}{\int dr\left[\Phi^* \chi\right]\left[\Sigma_f \Phi_S\right]} + \frac{\Sigma_f(r,E)}{\left\langle \Sigma_f \Phi_S \right\rangle} \tag{38}$$

$$\left(A - \frac{1}{k}F\right)\tilde{\Psi} = \frac{1}{I_S}\frac{\partial I_S}{\partial \Phi^*} = \frac{S(r,E)}{\left\langle \Phi^* S \right\rangle} - \frac{\left[\Sigma_f \Phi_S\right]\chi(r,E)}{\int dr\left[\Phi^* \chi\right]\left[\Sigma_f \Phi_S\right]} \tag{39}$$

The sensitivity coefficients for the source importance are calculated as:

$$\frac{\sigma}{I_S}\frac{dI_S}{d\sigma} = \frac{\sigma}{I_S}\left\{\frac{\partial I_S}{\partial \sigma} - \left\langle \tilde{\Psi}^*, \left(\frac{\partial A}{\partial \sigma} - \frac{\partial F}{\partial \sigma}\right)\Phi_S - \frac{\partial S}{\partial \sigma}\right\rangle - \left\langle \tilde{\Psi}, \left(\frac{\partial A}{\partial \sigma} - \frac{1}{k}\frac{\partial F}{\partial \sigma}\right)\Phi^* \right\rangle\right\} \tag{40}$$

Theory

The generalized importance for the inhomogeneous reactivity is calculated as:

$$\left(\mathbf{A}^{*}-\mathbf{F}^{*}\right)\tilde{\Psi}^{*}=\frac{1}{\mathbf{I}_{\mathbf{S}}}\frac{\partial \mathbf{I}_{\mathbf{S}}}{\partial \Phi_{\mathbf{s}}}=-\frac{\mathbf{S}(\mathbf{r},\mathbf{E})}{\Phi_{\mathbf{s}}(\mathbf{r},\mathbf{E})\langle \mathbf{S}\rangle}+\frac{\left[\nu\Sigma_{\mathbf{f}}\Phi_{\mathbf{s}}(\mathbf{r})\right]\chi(\mathbf{r},\mathbf{E})}{\Phi_{\mathbf{s}}(\mathbf{r},\mathbf{E})\langle \mathbf{F}\Phi_{\mathbf{s}}\rangle} \qquad (41)$$

Sensitivity coefficients are calculated as:

$$\frac{\sigma}{I_S}\frac{dI_S}{d\sigma}=\frac{\sigma}{I_S}\left\{\frac{\rho_s\langle\partial\nu\Sigma_f\Phi_s\rangle}{\langle F\,\Phi_s\rangle}-\left\langle\tilde{\Psi}^{*},\left(\frac{\partial A}{\partial\sigma}-\frac{\partial F}{\partial\sigma}\right)\Phi_s\right\rangle\right\} \qquad (42)$$

Theory

Sensitivity Coefficients : The Case of Nuclide Transmutation (e. g. nuclide densities at end of irradiation)

The generic nuclide K transmutation during irradiation can be represented as the nuclide density variation between time t_0 and t_F. If we denote n_F^K the "final" density, the appropriate sensitivity coefficients are given by :

$$S_j^K = \frac{\partial n_F^K}{\partial \sigma_j} \cdot \frac{\sigma_j}{n_F^K} = \frac{1}{n_F^K} \int_{t_0}^{t_F} \underline{n}^* \sigma_j \underline{n} \quad dt \qquad (43)$$

where the time dependent equations to obtain n* and n are the classical Bateman equation and its adjoint equation, with appropriate boundary conditions:

$$\frac{dn_k(t)}{dt} = \sum_{j=1}^{K-1} C_{kj} n_j(t) - C_{kk} n_k(t) \qquad (44)$$

Theory

Sensitivity Coefficients : The Case of the Reactivity Loss during Irradiation, $\Delta\rho^{cycle}$

At first order:

$$\Delta\rho^{cycle} = \sum_K \Delta n^K \rho_K \qquad \Delta n^K = n_F^K - n_0^K \tag{45}$$

and ρ_K is the reactivity per unit mass associated to the isotope K. The related sensitivity coefficients associated to the variation of a σ_j, are given by :

$$S_j^{cycle} = \frac{\sigma_j}{\Delta\rho^{cycle}} \frac{\partial \Delta\rho^{cycle}}{\partial \sigma_j} = \frac{\sigma_j}{\Delta\rho^{cycle}} \left(\sum_K \frac{\partial n^K}{\partial \sigma_j} \cdot \rho_K + \sum_K \Delta n_K \frac{\partial \rho_K}{\partial \sigma_j} \right) \tag{46}$$

or:

$$S_j^{cycle} = \frac{\sigma_j}{\Delta\rho^{cycle}} \frac{\partial \Delta\rho^{cycle}}{\partial \sigma_j} = \frac{\sigma_j}{\Delta\rho^{cycle}} \left(\sum_K \frac{\partial n^K}{\partial \sigma_j} \cdot \rho_K + \sum_K \Delta n_K \frac{\partial \rho_K}{\partial \sigma_j} \right) \tag{47}$$

Target Accuracy Assessments

- Target accuracy assessments are the inverse problem of the uncertainty evaluation.
- In order to establish priorities and target accuracies on data uncertainty reduction, a formal approach can be adopted by defining target accuracy on design parameter and finding out required accuracy on data.
- The unknown uncertainty data requirements can be obtained by solving a minimization problem where the sensitivity coefficients in conjunction with the existing constraints provide the needed quantities to find the solutions:

$$\sum_i \lambda_i / d_i^2 = \min \qquad i = 1 \ldots I \qquad (48)$$

$$\sum_i S_{ni}^2 d_i^2 < Q_n^T \qquad n = 1 \ldots N \qquad (49)$$

where d_i are the uncertainties to be found, S_{ni} are the sensitivity coefficients for the integral parameter Q_n, Q^T are the target accuracies on the N integral parameters, and λ_i are cost parameters.

Representativity

- A further use of sensitivity coefficients is, in conjunction with a covariance matrix, a representativity analysis of proposed or existing experiments.
- The calculation of correlations among the design and experiments allow to determine how representative is the latter of the former, and consequently, to optimize the experiments and to reduce their numbers.

$$\mathbf{r_{RE}} = \frac{\left(\mathbf{S_R^+ D S_E}\right)}{\left[\left(\mathbf{S_R^+ D S_R}\right)\left(\mathbf{S_E^+ D S_E}\right)\right]^{1/2}} \qquad \textbf{(50)}$$

- Formally one can reduce the estimated uncertainty on a design parameter by a quantity that represents the knowledge gained by performing the experiment:

$$\Delta \mathbf{R}_1^2 = \Delta \mathbf{R}_0^2 (1 - \mathbf{r}_{\mathbf{RE}}^2) \qquad \textbf{(51)}$$

Representativity

If more than one experiment is available, the Eq. (50) can be generalized. In the case of two experiments, characterized by sensitivity matrices S_{E1} and S_{E2} the following expression:

$$\Delta R_0^{'2} = \Delta R_0^2 \left[1 - \frac{1}{1 - r_{12}^2} \left(r_{R1} - r_{R2} \right)^2 - \frac{2}{1 + r_{12}} r_{R1} r_{R2} \right] \qquad (52)$$

$$r_{12} = \frac{\left(S_{E1}^{+} D S_{E2} \right)}{\left[\left(S_{E1}^{+} D S_{E1} \right) \left(S_{E2}^{+} D S_{E2} \right) \right]^{1/2}} \qquad (53)$$

$$r_{R1} = \frac{\left(S_{R}^{+} D S_{E1} \right)}{\left[\left(S_{R}^{+} D S_{R} \right) \left(S_{E1}^{+} D S_{E1} \right) \right]^{1/2}} \qquad (54)$$

$$r_{R2} = \frac{\left(S_{R}^{+} D S_{E2} \right)}{\left[\left(S_{R}^{+} D S_{R} \right) \left(S_{E2}^{+} D S_{E2} \right) \right]^{1/2}} \qquad (55)$$

Adjustments

- Sensitivity coefficients are also used in input parameter adjustments, where the coefficients are used within a fitting methodology (e.g. least square fit, Lagrange multipliers with most likelihood function, etc.) in order to reduce the discrepancies between measured and calculational results.
- The resulting adjusted input parameters can be subsequently used, sometimes in conjunction with bias factors, to obtain calculational results to which a reduced uncertainty will be associated.
- Adjustments allow to:
  - evaluate “a priori” uncertainties on reference design performance parameters
  - reduce these uncertainties using integral experiments (“a posteriori” uncertainties on performance parameters)
  - define “adjusted” nuclear data and associated “a posteriori” covariances

Global Statistical Adjustment Method

The method makes use of:

- "a priori" nuclear data covariance information,
- integral experiments analysis to define C/E values
- integral experiment uncertainties
- sensitivity coefficients

If we define: $y_j=(\sigma_j^{adj}-\sigma_j)/\sigma_j$ and $y_{Qi}^{exp}=(Q_i^{exp}-Q_i)/Q_i$, the y_i are given by:

$$y_i = \left(S^T D_Q^{-1} S + D^{-1}\right)^{-1} S^T D_Q^{-1}\, y_{Qi}^{exp}$$

where D_Q is the covariance matrix of the experiments, D the covariance matrix of the cross sections and S is the sensitivity vector. It will also result an adjusted covariance matrix for the nuclear data:

$$\left(D^{adj}\right)^{-1} = D^{-1} + S^T D_Q^{-1}\, S$$

Global Statistical Adjustment Method

If we define B_p the "a priori" nuclear data covariance matrix, S_B the sensitivity matrix of the performance parameters B (B=1…..BTOT) to the J nuclear data, the "a priori" covariance matrix of the performance parameters is given by:

$$B_B = S_B^T B_p S_B$$

It can be shown that, using a set of I integral experiments A, characterized by a sensitivity matrix S_A, besides a set of statistically adjusted cross-section data, a new ("a posteriori") covariance matrix can be obtained:

$$\tilde{B}_p = B_p - B_p S_A \left(S_A^T B_p S_A + B_A \right)^{-1} S_A^T B_p$$

where B_A is the integral experiment uncertainty matrix.

Global Statistical Adjustment Method

This matrix can then be used to define a new ("a posteriori") covariance matrix for the performance parameters B:

$$\tilde{B}_B = S_B^T \tilde{B}_p S_B = \{B_B -$$

$$- S_B^T B_p S_A \left(S_A^T B_p S_A + B_A\right)^{-1} S_A^T B_p S_B \} =$$

$$= B_B \{1 - (S_B^T B_p S_B)^{-1} (S_A^T B_p S_A + B_A)^{-1} \times$$

$$\times (S_A^T B_p S_B)^2 \}$$

From this expression, it results that in order to reduce the performance parameter "a priori" uncertainties, the most effective integral experiments are those:

- **with "representative" sensitivity profiles ($S_A \sim S_B$) and**
- **small experimental uncertainties ($B_A \sim 0$).**

Global Statistical Adjustment Method

If we consider only one performance parameter B and only one experiment "i", and if we put B_A =0, we obtain the expression of the "representativity" of one integral experiment:

$$r_{iB} = \frac{\left(S_i^T B_p S_B\right)}{\left[\left(S_i^T B_p S_i\right)\left(S_B^T B_p S_B\right)\right]^{1/2}}$$

Then, we can consider the previous equation as a generalized expression for the reference parameter uncertainty reduction. This generalized expression accounts for more than one experiment and allows estimating the impact of any new experiment in the reduction of the "a priori" uncertainty of the design performance parameters.

ERANOS Sensitivity Capabilities

- 1, 2, and 3D adjoint capability for calculation of adjoint flux and generalized importance function.
- Calculation of source term for functionals linear in the real or adjoint flux (e. g. reaction rate, reaction rate ratio, power density, etc.)
- Elimination of fundamental mode contamination, higher eigenfunctions calculation.
- Perturbation components, sensitivity coefficients in diffusion and transport theory for any possible change of cross sections, fission spectra, nuclide densities, or macroscopic variation.
- Sensitivity coefficients to bilinear functionals (e. g. reactivity coefficients, β_{eff}) using equivalent generalized perturbation theory.

ERANOS Sensitivity Capabilities (cont.)

- Direct and indirect effects calculations.
- Inhomogeneous solutions for ADS (φ*, inhomogeneous reactivity).
- Representativity factors between reference design and experiments.
- Special treatment of positive and negative source for S_n transport calculations.
- Time dependent perturbation theory in the nuclide evolution field for burn up and fuel cycle calculations (neutron sources, decay heat in the repository, radiotoxicity, etc.).
- Target accuracy assessment in connection with optimization code.

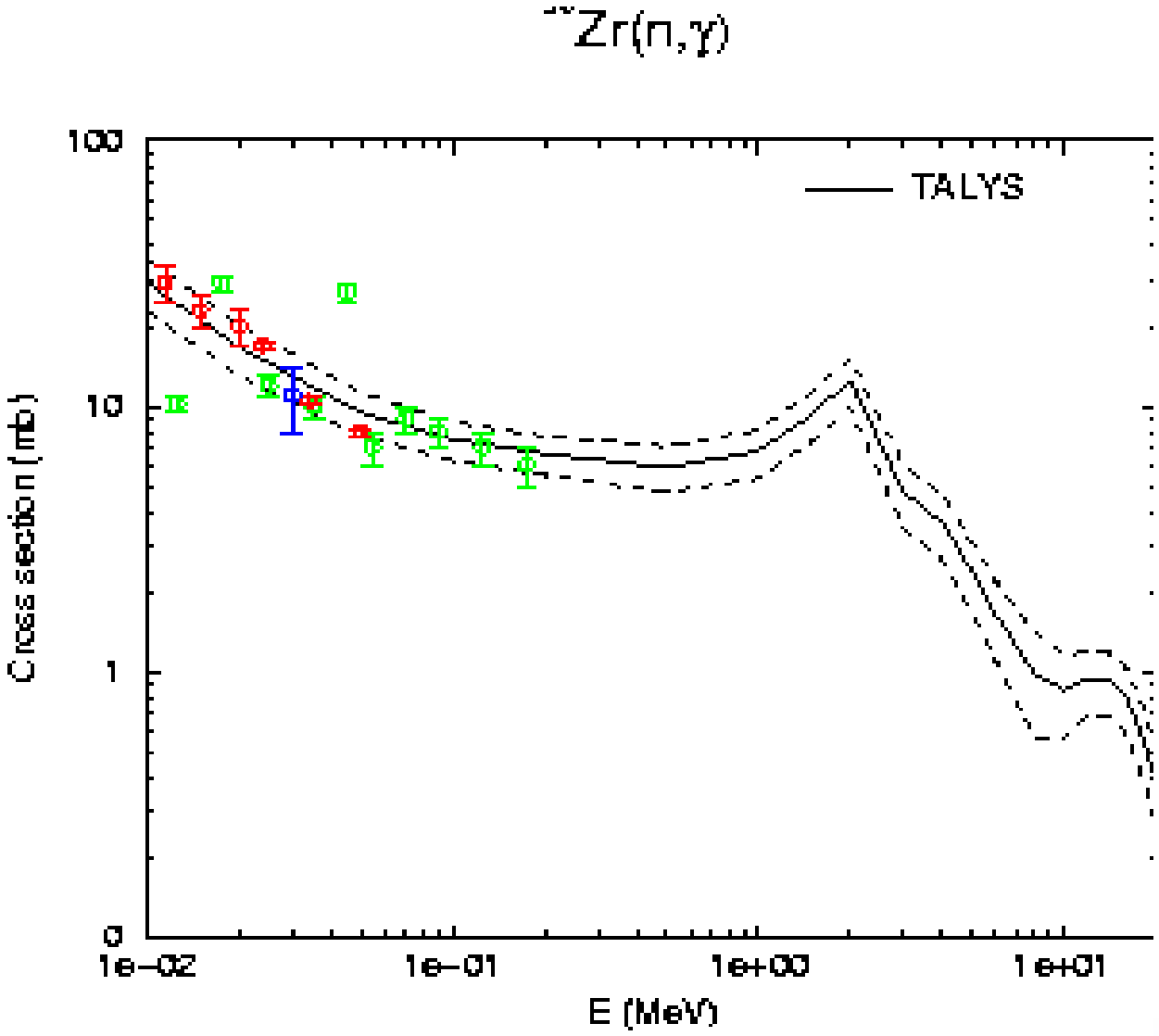
Covariance Data

- Covariance data have been scarce in the past. The first comprehensive effort was done for ENDF/B-IV at the end of the 70' and a code, PUFF, was written for processing this type of information
- After that no major effort was made until the 90' when scattered data were provided for few isotopes and reaction rates for different files (dosimetry, fusion, JEFF, ENDF)
- The Japanese were the only one that made a consistent and coherent issue for their files JENDL 3.2 and 3.3, but many very uncertainty low values for some important isotopes (U and Pu) have been questioned by the evaluator community
- In any case many questions are still open:
  - how good quality are these values?
  - are they scientifically based?
  - are they consistent with the basic data on the files?

However, nuclear physicists can provide much better, scientifically based, covariance data:

- **Needed: Systematical approach to provide covariance data with evaluated nuclear data files.**
- **Close interplay between experiments and nuclear models needed.**
- **A powerful approach : Propagate model parameter uncertainties to cross section uncertainties with Monte Carlo**
- **Random sampling of model parameters**
- **Full covariance file produced.**
- **Requirement 1: Parameter uncertainties and correlations should be physical**
- **Requirement 2: Produced uncertainties and correlations need to be credible, when compared with experimental covariances.**

And here an example of recent results.....



From A. Koenig NRG

From M. Herman BNL

Fast neutron region

Kalman: Bayesian, Generalized Least Squares approach

update #

exp. n x-sec

$$\mathbf{x}^{(n+1)} = \mathbf{x}^{(n)} + \mathbf{X}^{(n)}\mathbf{A}^{\mathrm{T}}\mathbf{Q}^{(n)}(\boldsymbol{\eta}^{(n)} - \boldsymbol{\sigma}(\mathbf{x}^{(n)}))$$

model parameters

model x-sec

$$\mathbf{X}^{(n+1)} = \mathbf{X}^{(n)} - \mathbf{X}^{(n)}\mathbf{A}^{\mathrm{T}}\mathbf{Q}^{(n)}\mathbf{A}\mathbf{X}^{(n)}$$

parameter covariance

exp. n covariance

$$\mathbf{Q} = (\mathbf{W} + \mathbf{V})^{-1}$$

error matrix

$$\mathbf{W} = \mathbf{A}\mathbf{X}\mathbf{A}^{\mathrm{T}}$$

sensitivity matrix

$$a_{i,j} = \frac{\partial\sigma(\mathbf{x}, E_i)}{\partial x_j}$$

Ingredients:

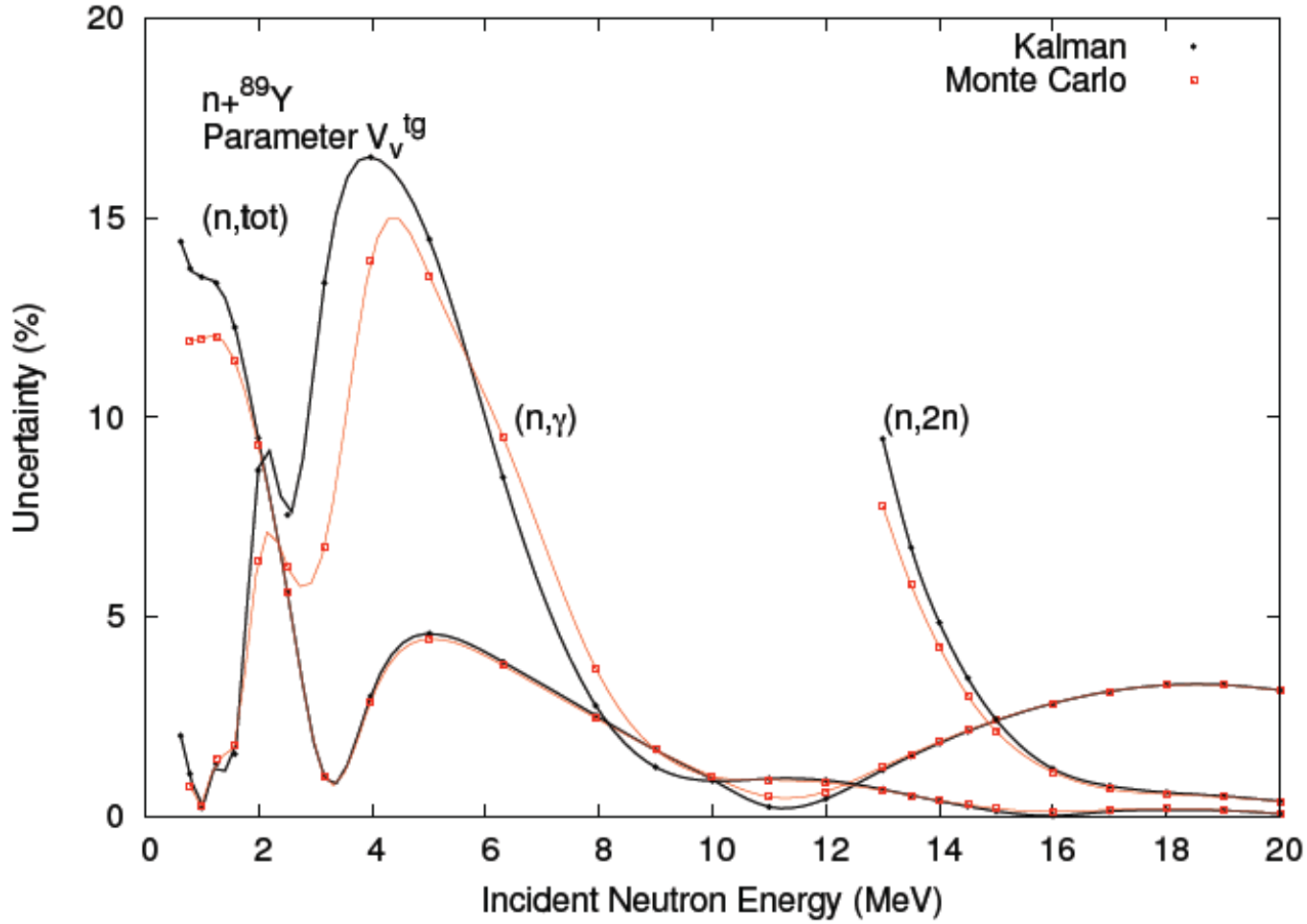
- model x-sections
- model parameter uncertainties
- x-sec. sensitivities to parameters
- experimental x-sections
- covariances for each experiment

Unifies model parameters, experiments, x-sections, covariances

From M. Herman BNL

n + ^{89}Y

comparison of uncertainties due to OMP real volume depth



BOLNA Covariance Matrix

Preliminary cross section covariances have been developed for a NEA- WPEC Subgroup at BNL for 45 out of 52 requested materials:

- **36 isotopes** (16O, 19F, 23Na, 27Al, 28Si, 52Cr, 56;57Fe, 58Ni, 90;91;92;94Zr,166;167;168;170Er, 206;207;208Pb, 209Bi, 233;234;236U, 237Np, 238;240;241;242Pu, 241;242m;243Am, 242;243;244;245Cm) were evaluated using the BNL-LANL methodology, based on the ENDF/B-VII.0 library, the Atlas of Neutron resonances, the nuclear model code EMPIRE and the Bayesian code Kalman

- **6 isotopes** (155; 156; 157; 158; 160Gd and 232Th) were taken from ENDF/BVII.0; and
- **3 isotopes** (1H, 238U and 239Pu) were taken from JENDL-3.3.

LANL has evaluated the covariance matrices for U235, U238 and Pu239, in the fast energy region

To complete these data, at ORNL resonance-parameter covariance evaluations were done for ^{235}U, ^{238}U, and ^{239}Pu with the computer code SAMMY

Finally, covariance data files for Pb isotopes have been produced at NRG by a purely stochastic approach

BOLNA Covariance Matrix

All the available BNL data have been used:

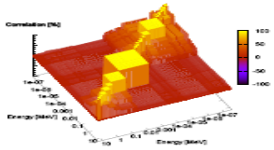
Preliminary Cross Section and ν-bar Covariances for WPEC Subgroup 26

by

D. Rochman, M. Herman, P. Obložinský and S. F. Mughabghab

January 2007

Report prepared for WPEC Subgroup 26
"Nuclear Data Needs for Advanced Reactor Systems"
Proposed by P.J. Finck, coordinated by M. Salvatores



National Nuclear Data Center
BROOKHAVEN

BNL Report: BNL-77407-2007-IR

- except the U-235, U-238 and Pu-239 data, which have been taken from the combined LANL/ORNL evaluation,
- and the Pb isotope data, taken from the NRG evaluation.
- Missing data have been taken from the ANL estimated covariance data

Energy correlations have been used, but practically no reaction cross-correlations

BOLNA Diagonal Values

		U235						U238					
Gr	E [MeV]	ν	σ_f	σ_{inel}	σ_{el}	σ_{capt}	$\sigma_{n,2n}$	ν	σ_f	σ_{inel}	σ_{el}	σ_{capt}	$\sigma_{n,2n}$
1	19.6	0.89	0.5	21.73	9.6	61.13	20.35	1.26	0.57	29.28	13.3	21.41	5.32
2	6.07	0.69	0.47	6.79	4.15	36.99	8.86	1.17	0.55	19.75	14.5	13.5	0
3	2.23	0.56	0.48	6.41	4.54	19.14	0	1.34	0.6	20.58	18.7	6.05	0
4	1.35	0.55	0.46	7.55	3.56	16.1	0	1.3	2.91	11.56	5.35	2.27	0
5	4.98e-1	0.61	0.5	11.32	2.87	22.13	0	2	5.26	4.19	1.92	1.41	0
6	1.83e-1	0.66	0.53	15.01	2.38	30.64	0	2	5.14	10.96	2.12	1.67	0
7	6.74e-2	0.66	0.5	14.72	2.63	32.89	0	2	5.14	11.12	3.76	1.64	0
8	2.48e-2	0.66	0.58	50	3.24	34.03	0	2	50.31	0	1.52	9.43	0
9	9.12e-3	0.66	3.18	48.48	5.16	33.92	0	2	214.62	0	0.67	3.11	0
10	2.03e-3	0.66	0.77	0	2.07	4.56	0	2	9.69	0	0.72	2.1	0
11	4.54e-4	0.66	0.44	0	1.33	0.63	0	2	2.38	0	2.39	1.71	0
12	2.26e-5	0.69	0.62	0	1.52	0.65	0	2	5.82	0	5.97	1.03	0
13	4.00e-6	0.69	0.4	0	1.78	1.36	0	2	51.89	0	0.82	2.45	0
14	5.40e-7	0.71	0.3	0	3.42	1.55	0	2	55.19	0	0.92	1.66	0
15	1.00e-7	0.71	0.25	0	4.9	1.73	0	2	55.42	0	0.94	1.64	0

BOLNA Diagonal Values

		Np237						Pu239					
Gr	E [MeV]	ν	σ_f	σ_{inel}	σ_{el}	σ_{capt}	$\sigma_{n,2n}$	ν	σ_f	σ_{inel}	σ_{el}	σ_{capt}	$\sigma_{n,2n}$
1	19.6	1.94	5.58	42.85	2.39	41.47	9.51	0.5	0.63	23.06	6.94	37.08	8.53
2	6.07	2.19	7.9	6.54	3.7	36.48	0	0.17	0.69	22.18	9.36	37.8	4.34
3	2.23	1.47	7.63	22.35	4.12	17.62	0	0.17	0.89	19	10.3	26.56	0
4	1.35	0.66	5.82	28.6	3.62	10.34	0	0.12	0.64	29.01	10.29	18.18	0
5	4.98e-1	0.6	5.79	44.99	3.47	5.79	0	0.19	0.68	34.01	5.66	11.55	0
6	1.83e-1	0.6	5.79	54.97	4.07	2.08	0	0.54	0.85	46.06	3.98	9.04	0
7	6.74e-2	0.6	5.79	36.27	4.37	6.66	0	0.58	0.72	40.04	2.37	10.12	0
8	2.48e-2	0.6	5.79	0	4.48	5.25	0	0.58	0.96	28.52	2.16	7.39	0
9	9.12e-3	0.6	5.79	0	3.93	5.25	0	0.65	0.62	8.64	4.04	15.46	0
10	2.03e-3	0.6	5.77	0	2.44	5.54	0	0.2	1.2	0	0.74	1.39	0
11	4.54e-4	0.6	7.54	0	2.41	1.7	0	0.2	1.24	0	1.2	1.25	0
12	2.26e-5	0.6	4.64	0	2.31	0.55	0	0.2	0.47	0	0.24	0.61	0
13	4.00e-6	0.6	5.58	0	2.23	0.7	0	0.2	1.43	0	0.3	1.22	0
14	5.40e-7	0.6	14.74	0	2.18	2.41	0	0.2	0.88	0	0.44	1.36	0
15	1.00e-7	0.6	4.55	0	2.03	1.55	0	0.2	1.11	0	0.68	1.6	0

BOLNA Diagonal Values

		Pu240						Pu241					
Gr	E [MeV]	ν	σ_f	σ_{inel}	σ_{el}	σ_{capt}	$\sigma_{n,2n}$	ν	σ_f	σ_{inel}	σ_{el}	σ_{capt}	$\sigma_{n,2n}$
1	19.6	1.09	9.56	37.11	2.34	52.16	54.09	0.45	24.09	25.15	4.45	55.39	39.68
2	6.07	2.65	4.8	9.65	5.19	32.47	0	0.27	14.16	19.47	3.74	54.1	33.43
3	2.23	2.69	5.65	10.09	5.42	19.74	0	0.27	21.26	18.38	4.39	38.41	0
4	1.35	3.74	5.82	7.79	4.76	16.28	0	0.28	16.62	19.78	5.38	31.66	0
5	4.98e-1	4.81	3.91	9.78	5.53	14.29	0	0.29	13.54	20.92	5.16	20.51	0
6	1.83e-1	4.81	5.7	42.55	5.76	13.79	0	0.29	19.87	30.09	4.69	11.29	0
7	6.74e-2	4.81	7.45	48.58	5.8	11.31	0	0.29	8.74	37.51	3.92	4.43	0
8	2.48e-2	4.81	7.45	0	5.05	10.21	0	0.29	11.29	0	9.14	7.79	0
9	9.12e-3	4.81	8.01	0	2.08	4.35	0	0.29	10.44	0	9.29	7.73	0
10	2.03e-3	4.81	21.62	0	1.26	1.47	0	0.29	12.68	0	10.96	7.74	0
11	4.54e-4	4.81	4.72	0	1.64	1.63	0	0.29	19.38	0	10.87	7.43	0
12	2.26e-5	4.81	8.91	0	3.25	5.5	0	0.29	4.21	0	10.66	8.38	0
13	4.00e-6	4.81	1.22	0	0.48	0.44	0	0.29	26.83	0	11.49	6.37	0
14	5.40e-7	4.81	29.76	0	4.58	3.23	0	0.29	2.94	0	9.91	6.84	0
15	1.00e-7	4.81	48.46	0	5.64	4.79	0	0.29	3.27	0	11.32	3.59	0

BOLNA Diagonal Values

		Am241						Am242m					
Gr	E [MeV]	ν	σ_f	σ_{inel}	σ_{el}	σ_{capt}	$\sigma_{n,2n}$	ν	σ_f	σ_{inel}	σ_{el}	σ_{capt}	$\sigma_{n,2n}$
1	19.6	1.88	12.74	55.29	3.51	28.83	10.03	10.43	21.37	55.82	8.36	84.91	31.77
2	6.07	1.98	11.67	15.2	3.77	15.38	0	0.91	23.36	17.32	12.05	63.01	37.23
3	2.23	1.91	9.81	29.63	5.12	9.16	0	0.66	19.7	23.84	11.15	43.35	0
4	1.35	0.98	8.25	24.45	4.52	6.9	0	0.68	16.51	26.47	12.06	39.41	0
5	4.98e-1	1	8.29	23.03	5.5	5.29	0	0.7	16.57	27.1	13.66	29	0
6	1.83e-1	1	8.29	48.53	5.2	6.79	0	0.7	16.57	33.65	13.91	19.39	0
7	6.74e-2	1	7.39	51.78	4.81	7.96	0	0.7	14.43	31.15	12.76	18.01	0
8	2.48e-2	1	13.71	0	11.54	6.85	0	0.7	11.8	50	18.89	19.17	0
9	9.12e-3	1	13.51	0	12.35	6.66	0	0.7	12.36	0	19.36	20.23	0
10	2.03e-3	1	13.41	0	9.7	6.59	0	0.7	12.2	0	19.42	20.08	0
11	4.54e-4	1	8.08	0	14.53	3.67	0	0.7	10.39	0	16.68	11.39	0
12	2.26e-5	1	5.15	0	14.03	1.82	0	0.7	10.38	0	19.95	13.25	0
13	4.00e-6	1	6.72	0	14.2	5.54	0	0.7	7	0	20.61	13.57	0
14	5.40e-7	1	8.93	0	13.81	1.26	0	0.7	8.83	0	17.64	19.87	0
15	1.00e-7	1	3.02	0	13.03	1.8	0	0.7	8.06	0	21.78	19.6	0

BOLNA Diagonal Values

		Am243						Cm244					
Gr	E [MeV]	ν	σ_f	σ_{inel}	σ_{el}	σ_{capt}	$\sigma_{n,2n}$	ν	σ_f	σ_{inel}	σ_{el}	σ_{capt}	$\sigma_{n,2n}$
1	19.6	1.88	14.44	61.97	7.51	60.42	26.63	10.55	17.86	38.26	10.5	89.19	40.91
2	6.07	1.98	11.03	17.87	4.64	41.5	0	11.08	31.25	22.67	10.2	53.78	0
3	2.23	1.91	5.97	35.3	7.49	21.66	0	10.68	43.8	15.1	5.56	36.49	0
4	1.35	1.09	9.18	42.15	4.11	14.18	0	5.5	50.01	18.18	10.7	20.8	0
5	4.98e-1	1.2	9.62	40.98	5.9	8.92	0	5.6	36.53	29.09	9.33	22.54	0
6	1.83e-1	1.2	9.62	79.53	7.84	6.6	0	5.6	47.56	63.31	8.38	17.71	0
7	6.74e-2	1.2	7.12	80.77	4.41	4.57	0	5.6	26.26	59.72	9.21	17.43	0
8	2.48e-2	1.2	13.79	0	9.13	6.77	0	5.6	19.03	0	14.9	19.32	0
9	9.12e-3	1.2	13.54	0	9.6	6.64	0	5.6	11.92	0	14.0	12.14	0
10	2.03e-3	1.2	13.41	0	7.68	6.58	0	5.6	5.27	0	7.72	4.47	0
11	4.54e-4	1.2	9.64	0	8.96	2.31	0	5.6	5.7	0	3.61	4.6	0
12	2.26e-5	1.2	5.95	0	8.22	1.74	0	5.6	17.09	0	7.75	6.64	0
13	4.00e-6	1.2	4.81	0	7	3.43	0	5.6	21.99	0	6.62	11.79	0
14	5.40e-7	1.2	2.25	0	12.4	3.75	0	5.6	26.4	0	6.16	12.16	0
15	1.00e-7	1.2	2.12	0	11.4	3.58	0	5.6	27.18	0	6.12	12.51	0

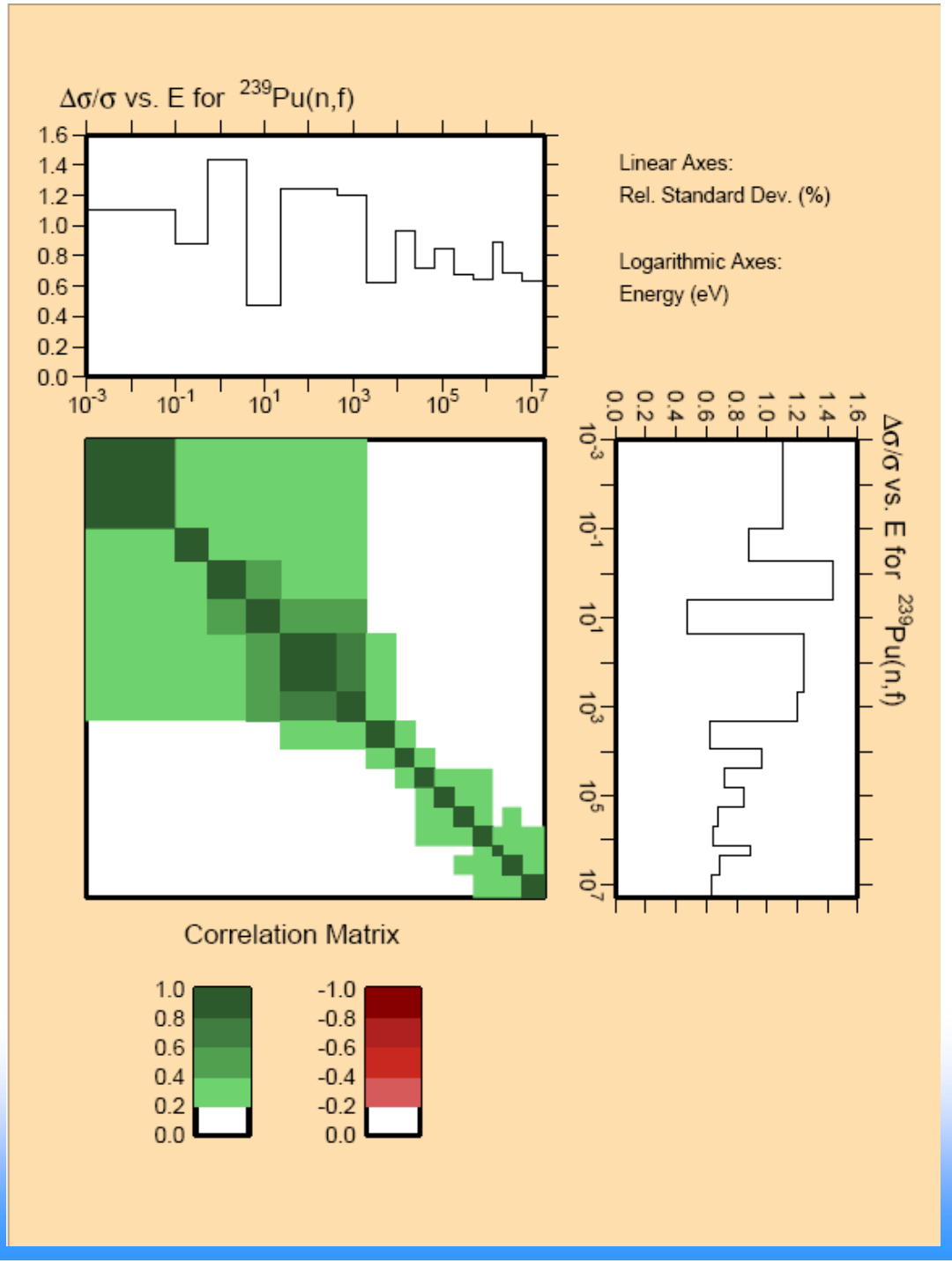
BOLNA Diagonal Values

		Pb207				Bi				Fe56				Zr90			
Gr	E [MeV]	σ_{inel}	σ_{el}	σ_{capt}	$\sigma_{n,2n}$	σ_{inel}	σ_{el}	σ_{capt}	$\sigma_{n,2n}$	σ_{inel}	σ_{el}	σ_{capt}	$\sigma_{n,2n}$	σ_{inel}	σ_{el}	σ_{capt}	$\sigma_{n,2n}$
1	19.6	17.01	4	61.28	5.69	5.25	0.83	47.58	9.39	12.97	4.61	46.24	7.05	11.32	0.44	46.36	6.74
2	6.07	4.98	5.81	24.25	0	2.44	1.02	27.74	0	7.23	8.14	31.69	0	17.96	0.92	18.59	0
3	2.23	13.77	4.43	21.56	0	34.07	2.06	17.56	0	25.4	5.89	23.48	0	18.52	3.96	9.14	0
4	1.35	11.31	4.78	19.46	0	41.77	4.59	11.35	0	16.12	0.64	7.43	0	50	3.39	6.26	0
5	4.98e-1	0	2.44	16.41	0	0	2.22	8.32	0	0	1.71	4.02	0	0	2.77	5.16	0
6	1.83e-1	0	3.73	15.94	0	0	1.8	8.79	0	0	2.08	10.77	0	0	2.02	3.13	0
7	6.74e-2	0	6.35	15.96	0	0	1.88	6.05	0	0	2.05	13.19	0	0	3.09	5.2	0
8	2.48e-2	0	8.85	15.05	0	0	2.41	3.85	0	0	4.6	8.81	0	0	4.43	7.89	0
9	9.12e-3	0	12	14.27	0	0	1.82	0.71	0	0	3.98	8.56	0	0	5.93	6.96	0
10	2.03e-3	0	16.6	20.01	0	0	1.93	0.43	0	0	4.16	11.23	0	0	6.83	10.55	0
11	4.54e-4	0	10.69	8.6	0	0	1.85	1.47	0	0	4.28	11.25	0	0	6.73	5.95	0
12	2.26e-5	0	0	0	0	0	1.82	1.82	0	0	4.31	11.25	0	0	6.7	2.75	0
13	4.00e-6	0	0	0	0	0	1.8	1.86	0	0	4.31	11.25	0	0	6.7	2.56	0
14	5.40e-7	0	0	0	0	0	1.8	1.87	0	0	4.31	11.25	0	0	6.7	2.53	0

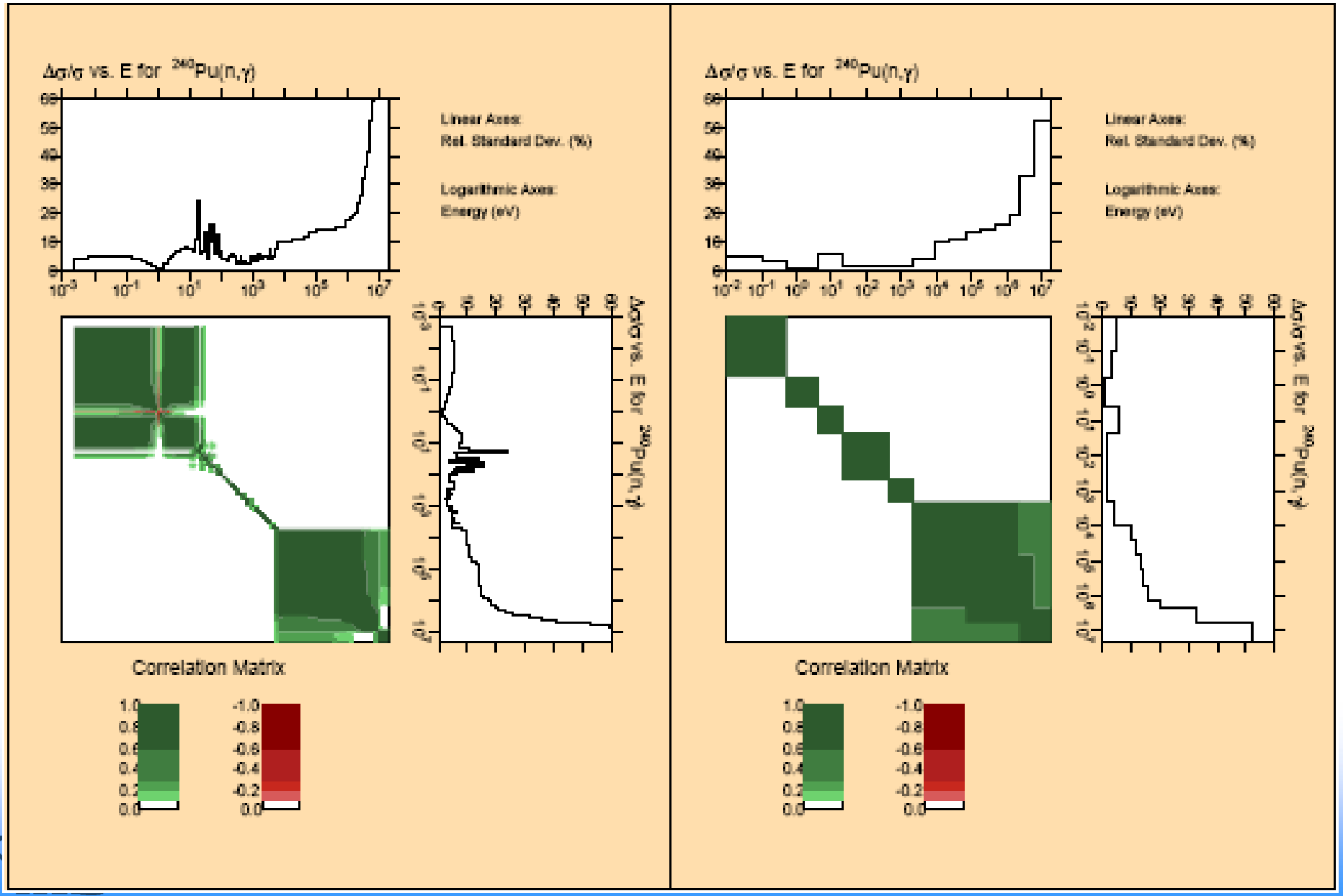
BOLNA Diagonal Values

		Si				O				Na			
Gr	E [MeV]	σ_{inel}	σ_{el}	σ_{capt}	$\sigma_{n,2n}$	σ_{inel}	σ_{el}	σ_{capt}	$\sigma_{n,2n}$	σ_{inel}	σ_{el}	σ_{capt}	$\sigma_{n,2n}$
1	19.6	21.38	0.69	52.87	50	100	84.6	100	100	18.79	1.8	46.44	11.07
2	6.07	13.54	2.77	11.12	0	0	54.9	100	0	8.87	4.62	24.33	0
3	2.23	50	1.66	10.07	0	0	12.1	100	0	12.56	3.72	1.7	0
4	1.35	0	1.43	6.77	0	0	1.43	100	0	28	3.01	7.44	0
5	4.98e-1	0	1.08	3.86	0	0	1.68	81.81	0	50	3.31	6.81	0
6	1.83e-1	0	2.97	5.65	0	0	1.68	69.63	0	0	3.25	23.59	0
7	6.74e-2	0	4.3	11.19	0	0	2.36	47.27	0	0	2.38	6.79	0
8	2.48e-2	0	4.18	8.93	0	0	2.35	28.21	0	0	2.87	6.63	0
9	9.12e-3	0	3.62	8.71	0	0	2.24	12.1	0	0	3.23	1.18	0
10	2.03e-3	0	3.23	5.12	0	0	2.23	9.36	0	0	4.93	2.28	0
11	4.54e-4	0	3.03	3.57	0	0	2.22	10.42	0	0	4.76	2.3	0
12	2.26e-5	0	2.97	3.25	0	0	2.22	11.29	0	0	4.73	2.29	0
13	4.00e-6	0	2.97	3.23	0	0	2.23	10.62	0	0	4.71	2.29	0
14	5.40e-7	0	2.97	3.22	0	0	2.23	11.03	0	0	4.7	2.29	0
15	1.00e-7	0	2.9	2.96	0	0	2	8	0	0	4.59	2.07	0

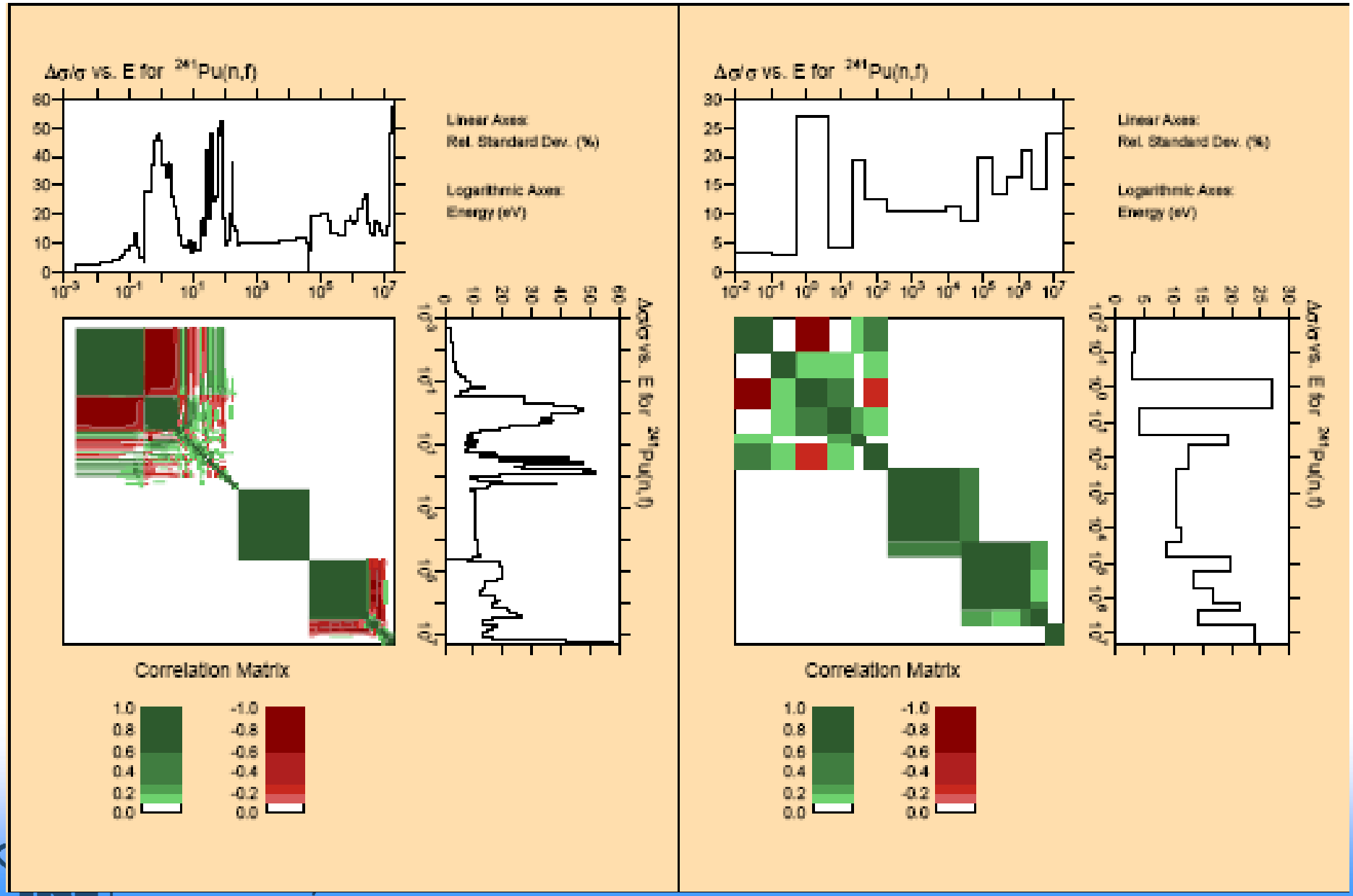
15 Groups BOLNA Covariance Matrix for the Pu239 (n,f)



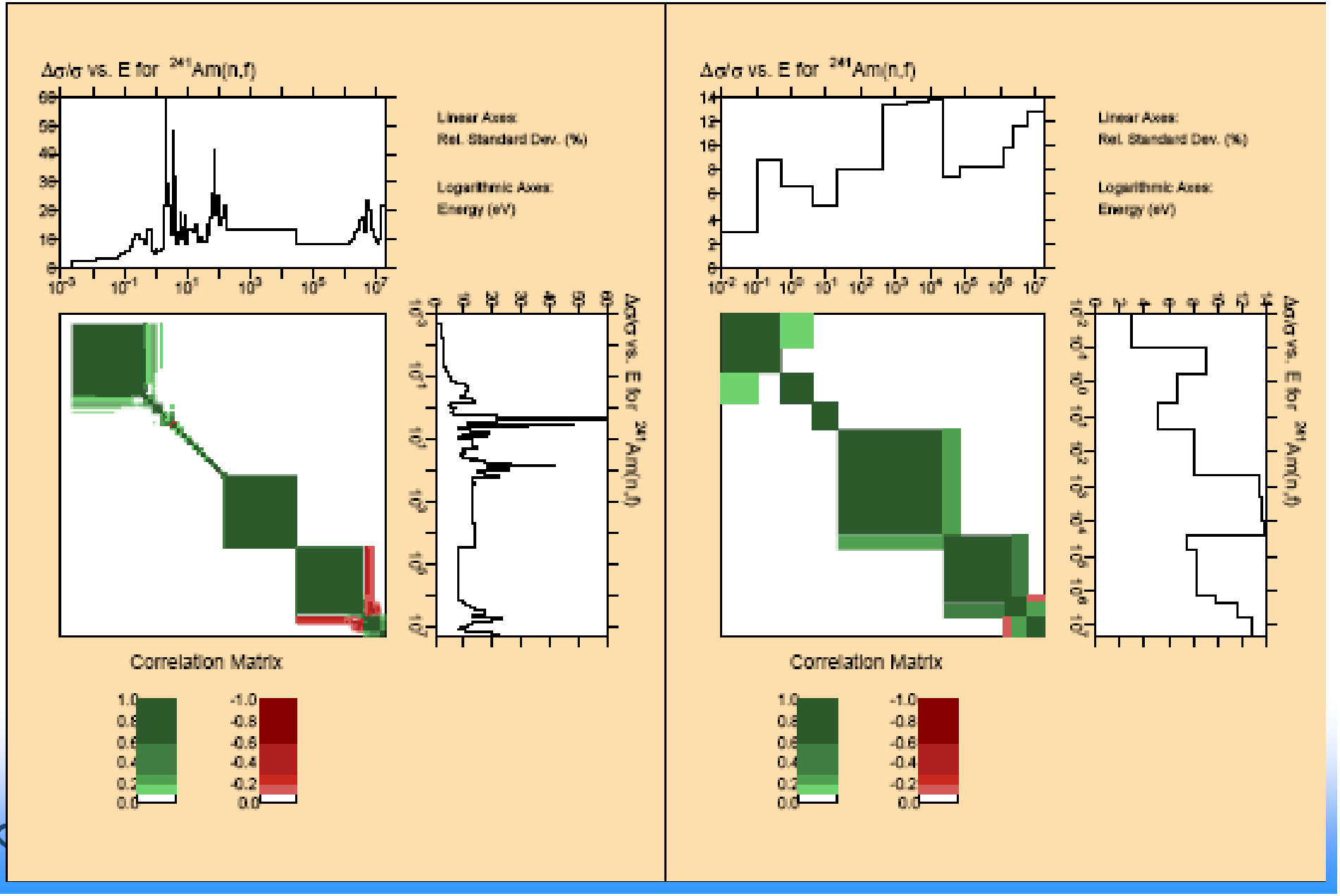
BOLNA Covariance Matrix in 187 (Left) and 15 (Right) Groups for the Pu-240(n,γ) Reaction



BOLNA Covariance Matrix in 187 (Left) and 15 (Right) Groups for the Pu-241(n,f) Reaction



BOLNA Covariance Matrix in 187 (Left) and 15 (Right) Groups for the Am-241(n,f) Reaction

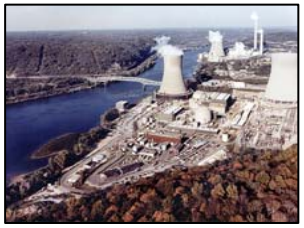


Sensitivity and uncertainty analysis: the case of GENERATION IV and Innovative Fuel Cycles

Nuclear energy systems deployable no later than 2030 in both developed and developing countries, for generation of electricity and other energy products

Generation I

Early Prototype Reactors



- Shippingport
- Dresden, Fermi I
- Magnox

Generation II

Commercial Power Reactors

- LWR-PWR, BWR
- CANDU
- VVER/RBMK

Generation III

Advanced LWRs

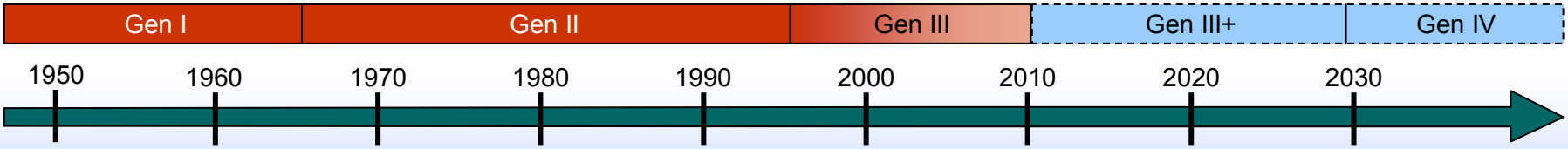
- ABWR
- System 80+
- AP600
- EPR

Near-Term Deployment

Generation III+ Evolutionary Designs Offering Improved Economics

Generation IV

- **Highly Economical**
- **Enhanced Safety**
- **Minimal Waste**
- **Proliferation Resistant**

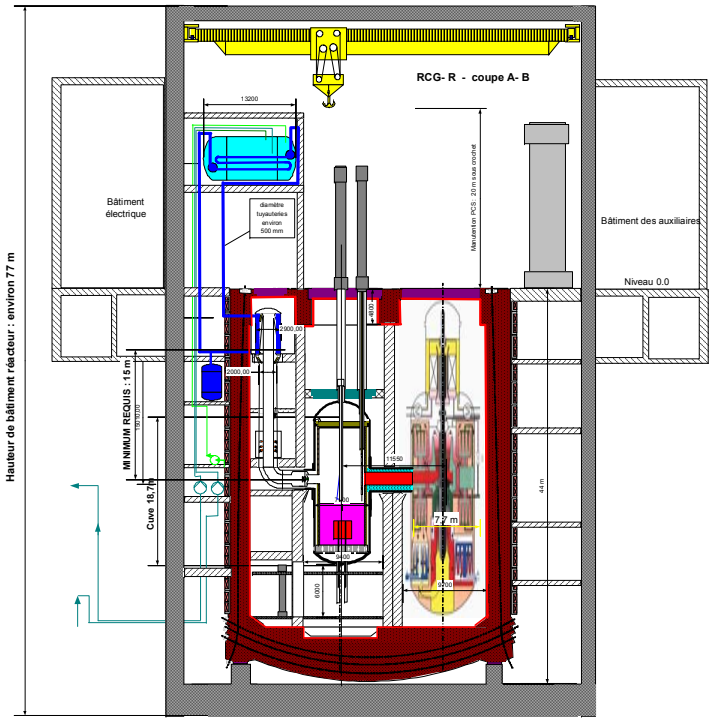


Idaho National Laboratory

Features of the Investigated Fast Systems

System	Coolant	Fuel Type	TRU/(U+TRU)	MA/(U+TRU)	Power [MW_{th}]
ABTR	Na	Metal	0.162	~ 0	250
SFR	Na	Metal	0.605	0.106	840
EFR	Na	MOX	0.237	0.012	3600
GFR	He	Carbide	0.217	0.050	2400
LFR	Pb	Metal	0.233	0.024	900

Fast Reactors



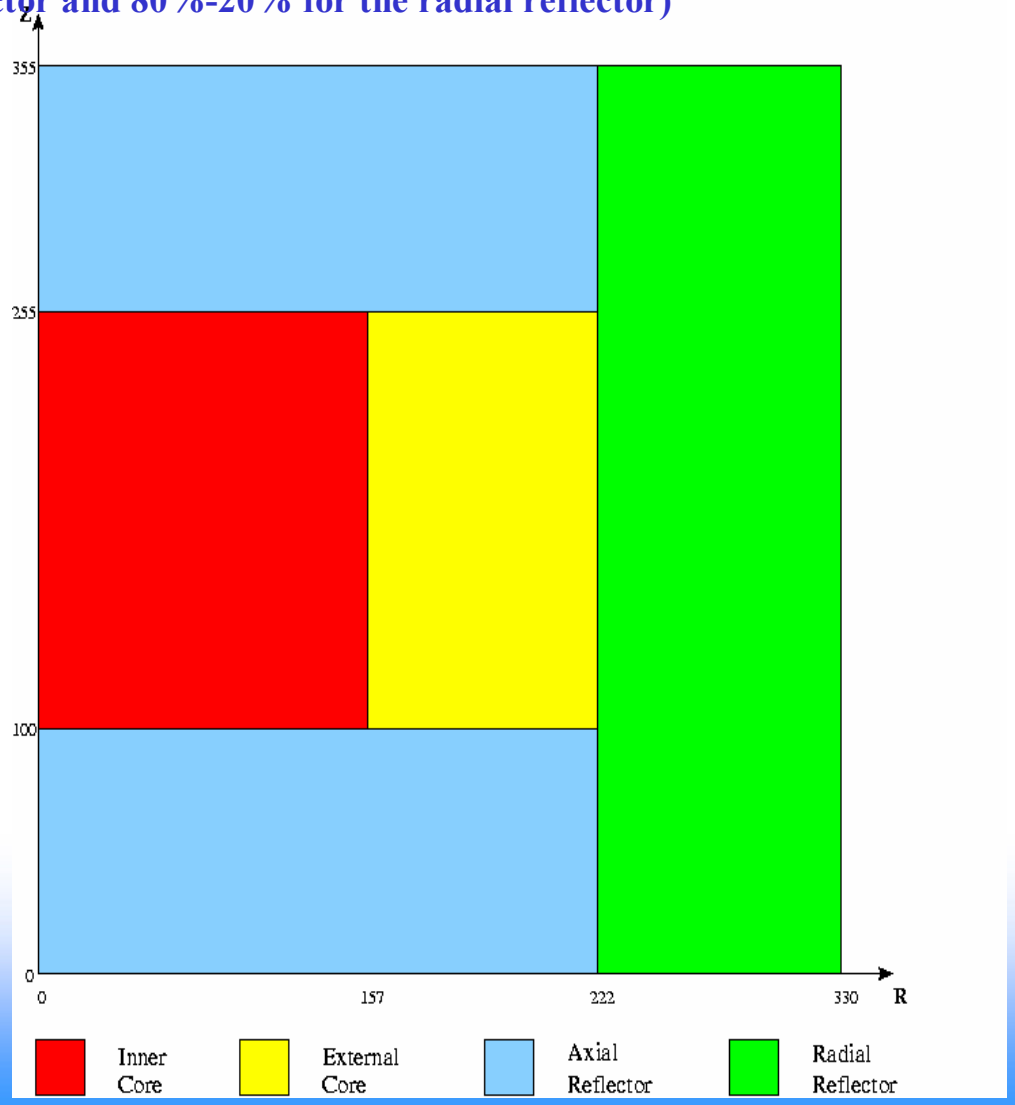
RCG-R Cavité béton précontraint coupe A-A

Breeder or Burner

Fuels, structures, and reflectors might contain new materials (Zr, Si)

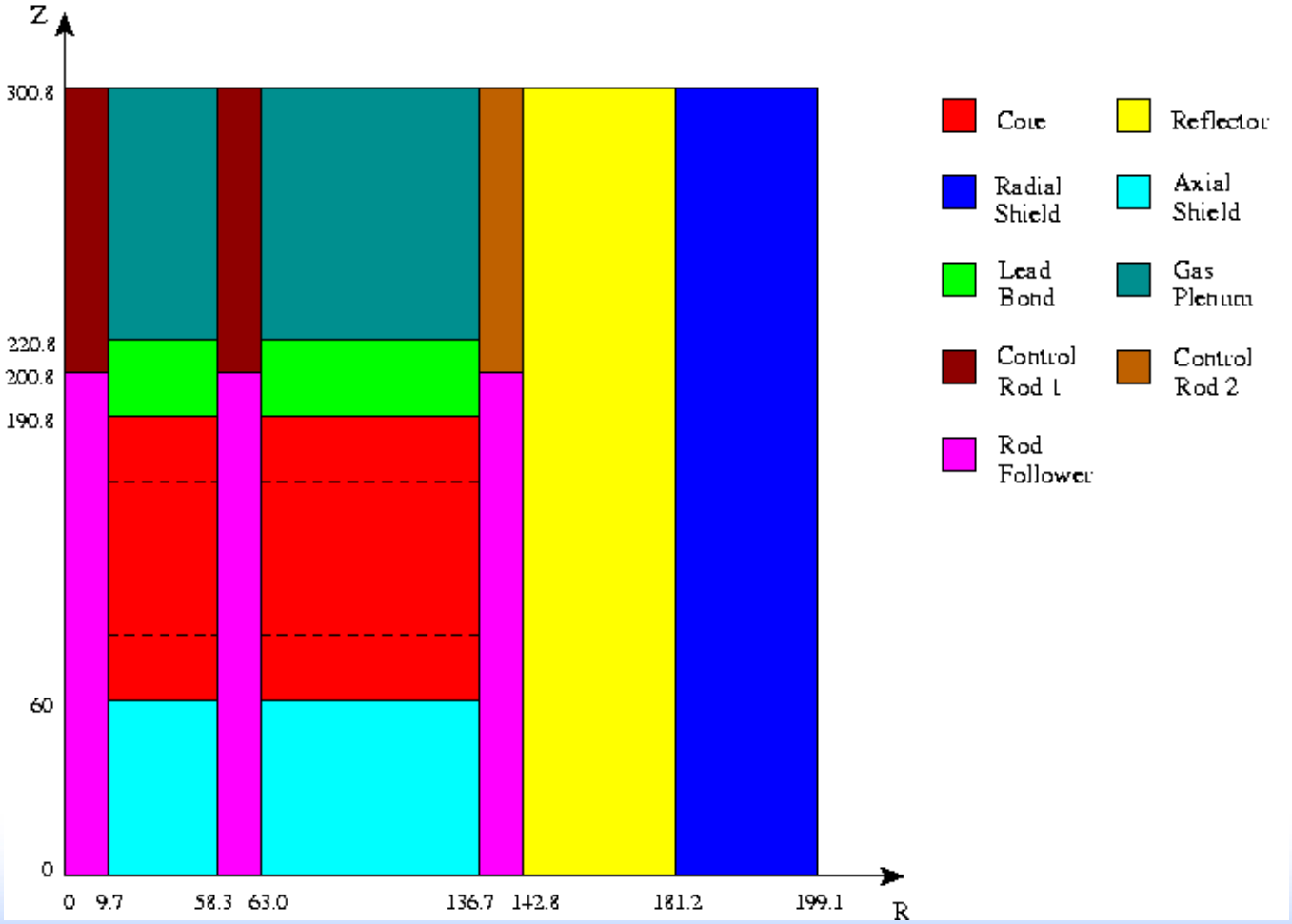
GFR

The gas cooled fast reactor contains CERCER fuel which is a mixture (56%-44%) of a ceramic matrix material SiC and a ceramic heavy metal carbide fuel with 5% of Minor Actinides (MA). The materials of the core region are structure (20%), coolant (40%) and fuel (40%) and the average enrichment (PUC/(UC+PuC)) is 17%. The coolant is helium and the reflector is a mixture of Zr_3Si_2 and coolant (60%-40% for the axial reflector and 80%-20% for the radial reflector)



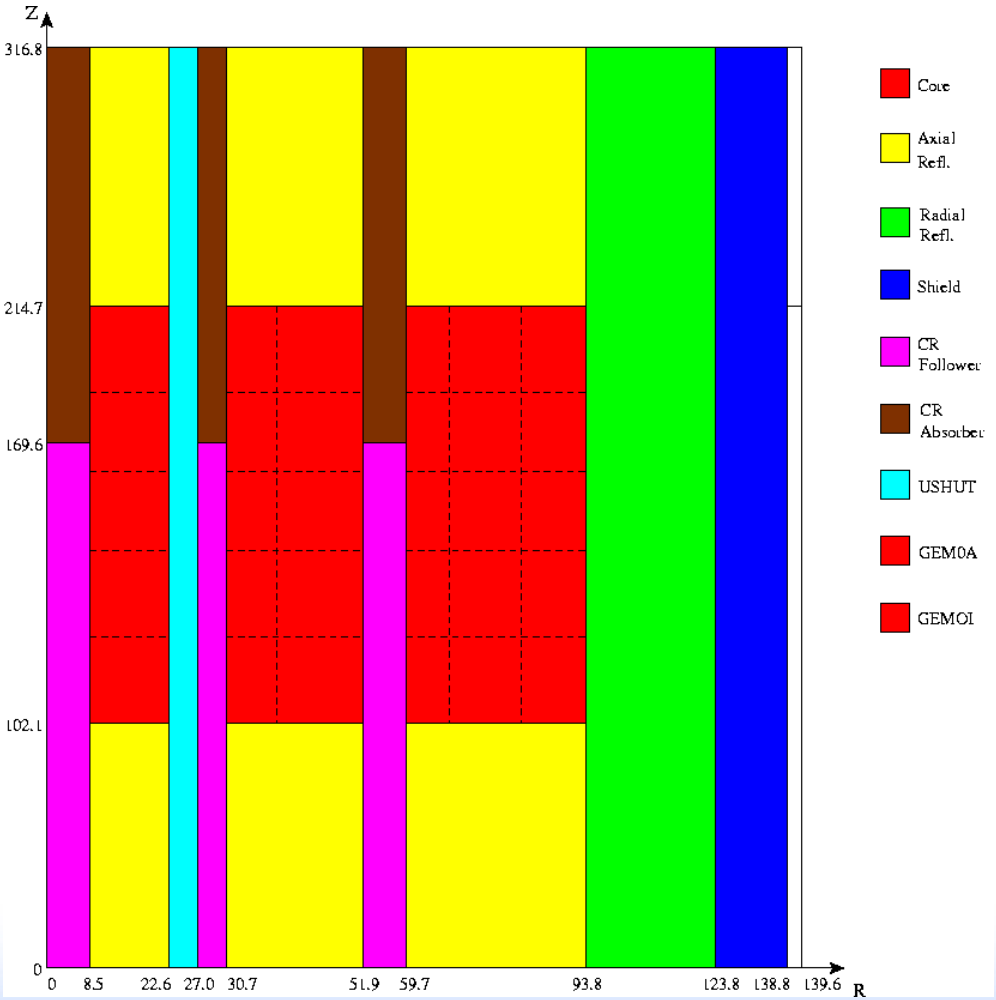
LFR

The lead cooled fast rector, that is being also investigated in the frame of a benchmark problem prepared by KAERI and also adopted by IAEA, is a 900 MWth reactor loaded with U-TRU-Zr metallic alloy fuels (2% of MA). The core contains 192 hexagonal ductless fuel assemblies and it is surrounded by ducted lead reflector and steel shields.



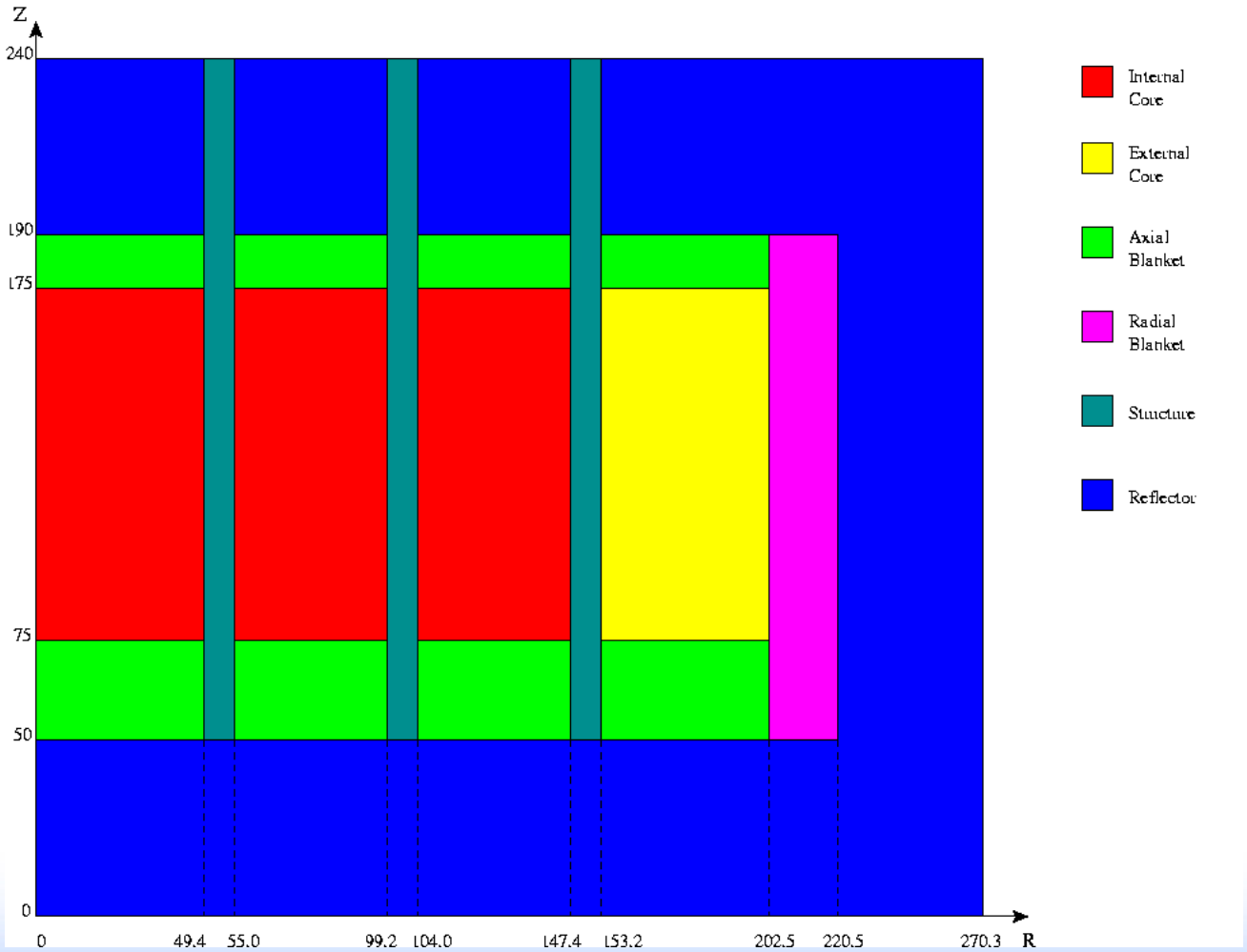
SFR

The small size transmuter sodium cooled fast reactor is an 840 MWth reactor loaded with U-TRU-Zr metallic alloy (10% of MA) and very low conversion ratio (<0.25).



EFR

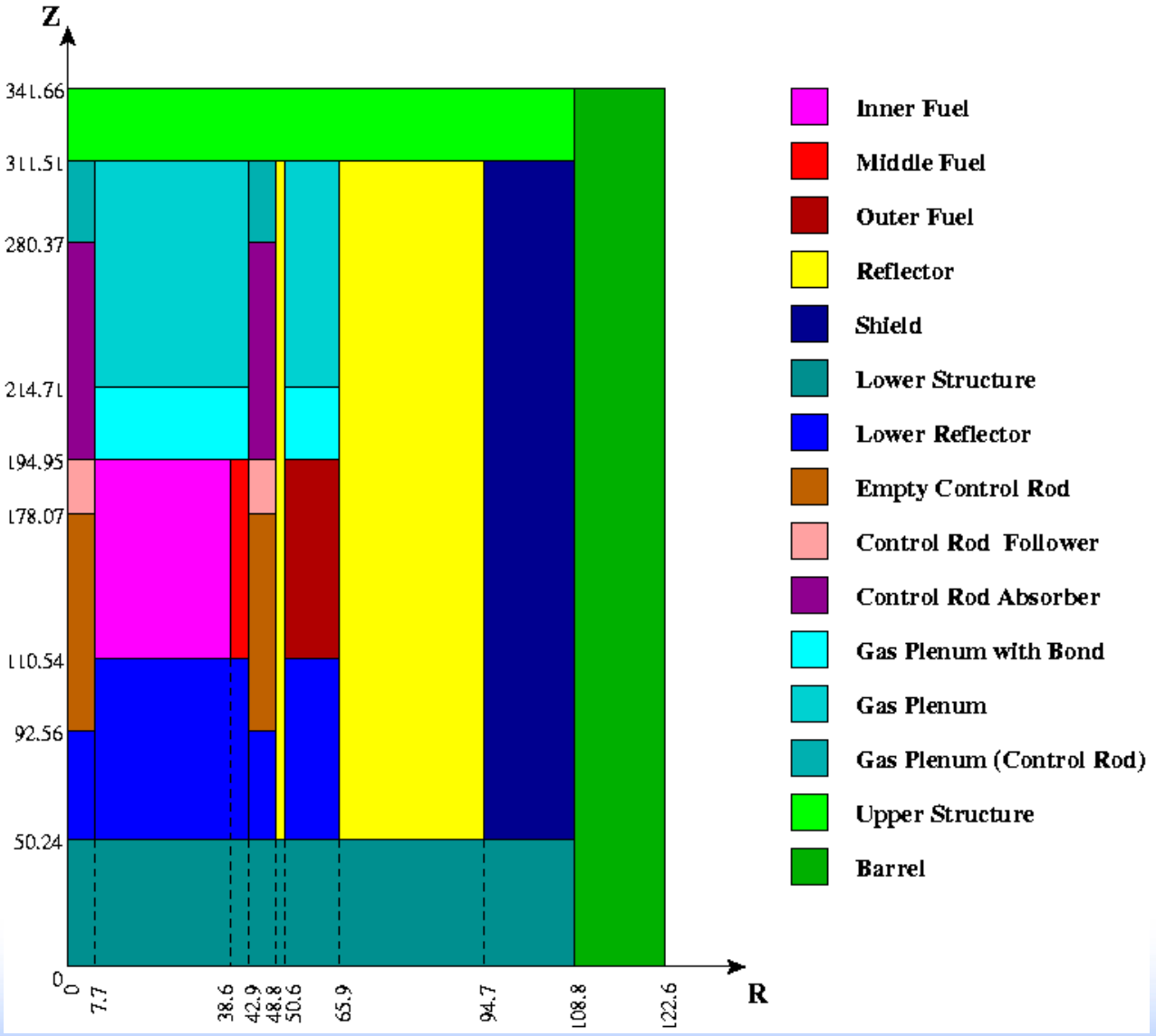
The large size sodium cooled reactor, whose specifications have been provided by the CEA, is a 3600 MWth reactor loaded with U-TRU oxide fuel (1% of MA). The core is surrounded by a blanket.



ABTR

- The primary mission of the ABTR (Advanced Burner Test Reactor) is to demonstrate the transmutation of transuranics (TRU) recovered from the LWR spent fuel, and hence the benefits of the fuel cycle closure to nuclear waste management. This requires:
  - to demonstrate reactor-based transmutation of TRU as part of an advanced fuel cycle,
  - to qualify the TRU-containing fuels and advanced structural materials needed for a full-scale ABR,
  - to support the research, development and demonstration required for certification of an ABR standard design by the U.S. Nuclear Regulatory Commission.
- Based on these objectives, core design and fuel cycle studies have been performed to develop ABTR core designs and associated fuel cycle strategies.

ABTR (Metallic Fuel: U,PU, Zr)



GFR

2400 MWe – He Cooled

SiC – (U-TRU)C Fuel

Zr_3Si_2 Reflector

Enrichment: 17%

MA: 5%

Irradiation Cycle: 415 d

LFR

900 MW_{th} – Pb Cooled

U-TRU-Zr Metallic Alloy

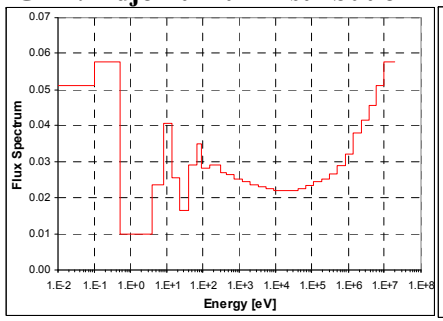
Pb Reflector

Enrichment: 21%

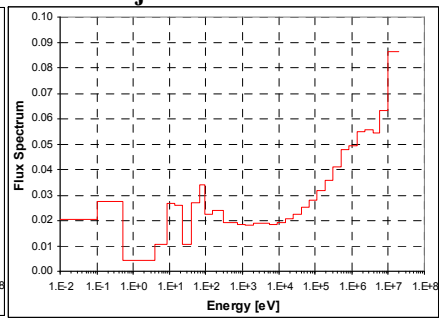
MA: 2%

Irradiation Cycle: 310 d

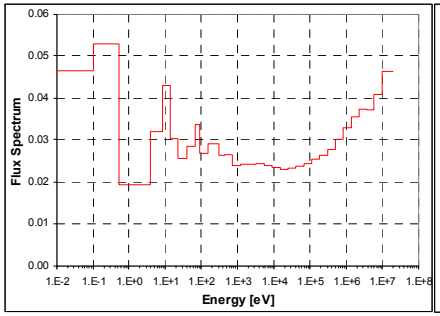
GFR: Adjoint Flux Distribution



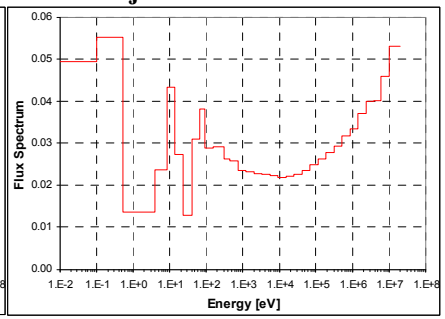
LFR: Adjoint Flux Distribution



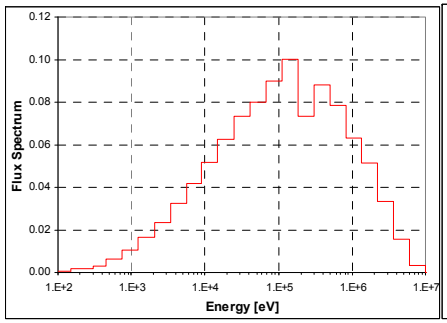
SFR: Adjoint Flux Distribution



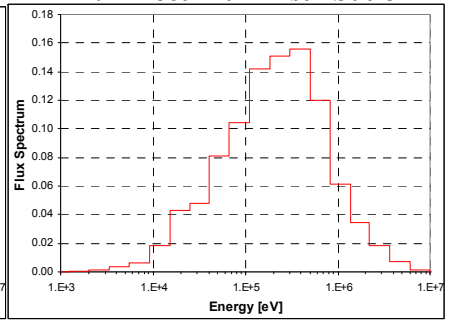
EFR: Adjoint Flux Distribution



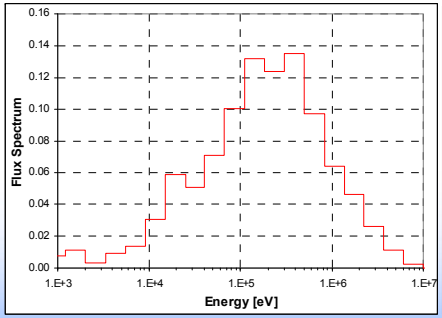
GFR: Direct Flux Distribution



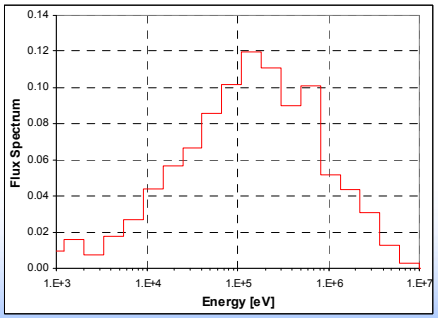
LFR: Direct Flux Distribution



SFR: Direct Flux Distribution



EFR: Direct Flux Distribution



SFR (Burner: CR = 0.25)

840 MW_{th} – Na Cooled

U-TRU-Zr Metallic Alloy

SS Reflector

Enrichment: 56%

MA: 10%

Irradiation Cycle: 155 d

EFR

3600 MW_{th} – Na Cooled

U-TRU Oxide

U - Blanket

Enrichment: 22.7%

MA: 1%

Irradiation Cycle: 1700 d

Representativity Studies

Sensitivity coefficients are calculated on:

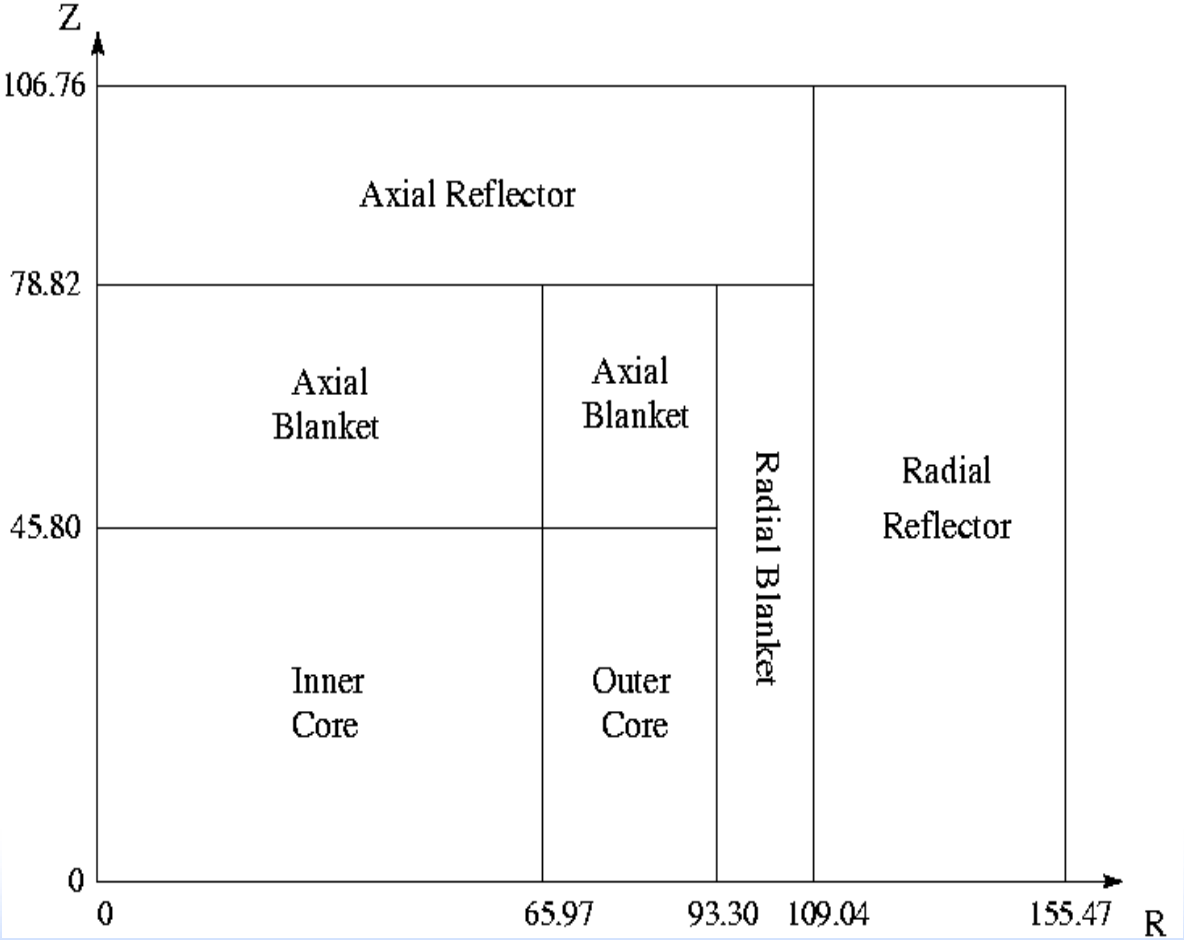
- K_{eff}
- $^{238}U\ \sigma_f$ /$^{239}Pu\ \sigma_f$ at core center
- $\eta = \upsilon\Sigma_f/\Sigma_a$ (being representative of the adjoint)
- β_{eff} (if available)
- $^{238}U\ \sigma_f$ slope close to reflector or blanket
- $^{239}Pu\ \sigma_f$ slope close to reflector or blanket
- Control rod reactivity (if available)
- Coolant void reactivity (if available)

Selected Experiments

- ZPPR-2, ZPPR-3, ZPPR-9, and ZPPR-10 as representative of large size sodium cooled fast reactor (EFR)
- ZPPR-15 for the small size sodium cooled fast transmuter reactor (SFR)
- ZPR3-48, -53, -54, 55, ZPR9-28, -31 for the gas cooled fast reactor (GCFR)

ZPPR-15

The ZPPR-15A experiment was performed to support the DOE innovative design initiatives in August 1985. The assembly is based on sodium cooled, metallic fueled, homogeneous, two-enrichment-zone core of about 330MWe size. With respect to the ZPPR-15A, the EFR has a bigger size and an oxide fuel loading; the SFR has a comparable size, metallic fuel, but a larger amount of minor actinides, and, additionally, the SFR has no blanket.



ZPPR-15
Representativity

	R = EFR E = ZPPR-15A	R = SFR E = ZPPR-15A	R = EFR E = ZPPR-15A	R = SFR E = ZPPR-15A
Integral Parameter	K_{eff}	K_{eff}	$\frac{\langle \sigma_{f,U8} \Phi \rangle_{pos1}}{\langle \sigma_{f,Pu9} \Phi \rangle_{pos1}}$	$\frac{\langle \sigma_{f,U8} \Phi \rangle_{pos1}}{\langle \sigma_{f,Pu9} \Phi \rangle_{pos1}}$
Absolute Value in R:	**1.108481**	**1.052802**	**0.025**	**0.025**
Absolute Value in E:	**.986312**	**.986312**	**0.020**	**0.020**
Total Uncertainty in R:	**1.02**	**1.10**	**4.84**	**4.75**
Total Uncertainty in E:	**1.42**	**1.42**	**7.36**	**7.37**
Representativity factor:	**0.931**	**0.613**	**0.235**	**0.148**
Reduced Uncertainty in R:	**0.37**	**0.87**	**4.71**	**4.69**

ZPPR-15
Representativity

	R = EFR E = ZPPR-15A	R = SFR E = ZPPR-15A	R = EFR E = ZPPR-15A	R = SFR E = ZPPR-15A
Integral Parameter	Void coefficient $(\rho_{void} - \rho_{ref})$	Void coefficient $(\rho_{void} - \rho_{ref})$	$\frac{\langle \sigma_{f,U8} \Phi \rangle_{pos2}}{\langle \sigma_{f,U8} \Phi \rangle_{pos3}}$	$\frac{\langle \sigma_{f,U8} \Phi \rangle_{pos2}}{\langle \sigma_{f,U8} \Phi \rangle_{pos3}}$
Absolute Value in R:	**1934.5 pcm**	**1831 pcm**	**3.139**	**3.043**
Absolute Value in E:	**1652.9 pcm**	**1652.9 pcm**	**4.196**	**4.196**
Total Uncertainty in R:	**8.40**	**17.75**	**3.81**	**5.46**
Total Uncertainty in E:	**20.43**	**20.43**	**4.12**	**4.12**
Representativity factor:	**0.685**	**0.566**	**0.932**	**0.928**
Reduced Uncertainty in R:	**6.12**	**14.64**	**1.38**	**2.03**

ZPPR-15
Representativity

	R = EFR E = ZPPR-15A	R = SFR E = ZPPR-15A	R = EFR E = ZPPR-15A	R = SFR E = ZPPR-15A
Integral Parameter	$\frac{\langle \nu\Sigma_f \Phi \rangle}{\langle \Sigma_a \Phi \rangle}$	$\frac{\langle \nu\Sigma_f \Phi \rangle}{\langle \Sigma_a \Phi \rangle}$	$\frac{\langle \sigma_{f,Pu9} \Phi \rangle_{pos2}}{\langle \sigma_{f,Pu9} \Phi \rangle_{pos3}}$	$\frac{\langle \sigma_{f,Pu9} \Phi \rangle_{pos2}}{\langle \sigma_{f,Pu9} \Phi \rangle_{pos3}}$
Absolute Value in R:	**2.94**	**3.03**	**0.586**	**0.045**
Absolute Value in E:	**2.92**	**2.92**	**0.579**	**0.579**
Total Uncertainty in R:	**0.04**	**0.05**	**1.43**	**2.59**
Total Uncertainty in E:	**0.03**	**0.03**	**2.02**	**2.02**
Representativity factor:	**0.856**	**0.696**	**0.836**	**0.853**
Reduced Uncertainty in R:	**0.02**	**0.04**	**0.78**	**1.35**

ZPPR-15
Representativity

	R = EFR E = ZPPR-15A	R = SFR E = ZPPR-15A
Integral Parameter	β_{eff}	β_{eff}
Absolute Value in R:	**206.2 pcm**	**207.5 pcm**
Absolute Value in E:	**213.6 pcm**	**213.6 pcm**
Total Uncertainty in R:	**0.70**	**0.64**
Total Uncertainty in E:	**0.67**	**0.67**
Representativity factor:	**0.950**	**0.593**
Reduced Uncertainty in R:	**0.22**	**0.51**

NUCLEAR DATA UNCERTAINTY IMPACT ON:

Reactor parameters….

- Criticality (multiplication factor)
- Doppler Reactivity Coefficient
- Coolant Void Reactivity Coefficient
- Reactivity Loss during Irradiation
- Transmutation Potential (i.e. nuclide concentration at the end of irradiation)
- Peak Power Value

….and fuel cycle parameters:

- MA Decay Heat in a Repository (100 years after disposal)
- Radiation Source at Fuel Discharge
- Radiotoxicity in a Repository (100000 years after disposal)

Fast Neutron Systems:
Total Uncertainties (%)

Reactor		k_{eff}	Power Peak	Doppler	Void	Burnup [pcm]	Decay Heat	Dose	Neutron Source
ABTR	PEC	1.96	0.6	6.4	12.5	97	0.1	0.1	0.5
	BOLNA	0.92	0.3	4.4	6.0	52	0.2	0.1	0.5
SFR	PEC	1.66	0.5	6.0	23.4	234	0.3	0.2	0.9
	BOLNA	1.82	0.4	5.6	17.1	272	0.4	0.3	1.0
EFR	PEC	1.57	1.1	5.1	12.1	989	2.3	1.7	6.0
	BOLNA	1.18	1.2	3.8	7.8	871	2.4	1.2	6.6
GFR	PEC	1.90	1.8	5.5	7.1	384	0.5	0.6	1.8
	BOLNA	1.88	1.7	5.5	7.7	381	0.4	0.5	1.4
LFR	PEC	2.26	1.0	7.8	20.6	258	0.5	0.5	1.1
	BOLNA	1.43	0.6	4.3	7.2	198	0.6	0.4	1.1

ABTR Uncertainties (%) - Breakdown by Isotope

	k_{eff}	Doppler	Power Peak	Void	Burnup [pcm]
U235	0.01	0.04	-	0.06	0.8
U238	0.77	3.40	0.16	3.74	20.1
Pu238	0.02	0.03	0.02	0.04	2.1
Pu239	0.36	1.62	0.11	2.14	30.2
Pu240	0.13	0.32	0.03	0.44	12.1
Pu241	0.12	0.34	0.07	0.27	13.3
Fe56	0.27	1.60	0.18	0.78	9.2
Cr52	0.06	0.45	0.08	0.17	1.0
Ni58	-	0.12	0.01	0.03	0.1
Zr90	0.04	0.13	0.05	0.22	1.0
Na23	0.08	1.51	0.13	4.10	3.1
Total	0.92	4.42	0.34	6.03	51.8

ABTR K_{eff} Uncertainties (%) - Breakdown by Isotope and Reaction

Isotope	σ_{cap}	σ_{fiss}	ν	σ_{el}	σ_{inel}	$\sigma_{n.2n}$	Total
U238	0.26	0.04	0.14	0.20	0.69	-	0.77
Pu238	-	0.01	0.01	-	-	-	0.02
Pu239	0.23	0.24	0.13	0.03	0.06	-	0.36
Pu240	0.06	0.09	0.08	-	0.01	-	0.13
Fe56	0.07	-	-	0.08	0.24	-	0.27
Zr90	0.01	-	-	0.01	0.03	-	0.04
Na23	0.02	-	-	0.05	0.07	-	0.08
B10	0.04	-	-	-	-	-	0.04
Total	0.36	0.29	0.20	0.23	0.73	0.01	0.92

ABTR Uncertainties (%) – Energy Breakdown for Selected Isotope/Reaction

Parameter		K_{eff}	K_{eff}	Dopp.	Void	Void	Void	Con. Rod
Gr.	Energy	U238 σ^{inel}	Pu239 σ^{fiss}	Fe56 σ^{inel}	U238 σ^{capt}	Na23 σ^{elas}	Na23 σ^{inel}	B10 $\sigma^{n,\alpha}$
1	19.6 MeV	0.079	0.012	0.20	0.01	0.09	0.75	0.09
2	6.07 MeV	0.479	0.046	0.36	0.02	0.39	1.30	0.35
3	2.23 MeV	0.446	0.053	0.70	0.06	0.58	1.78	0.38
4	1.35 MeV	0.183	0.099	0.81	0.35	0.70	2.79	0.70
5	498 KeV	0.021	0.126	0.00	0.15	1.03	0.15	2.53
6	183 KeV	0.049	0.128	0.00	0.27	1.03	0.00	2.35
7	67.4 KeV	0.014	0.077	0.00	0.46	0.49	0.00	1.38
8	24.8 KeV	0.000	0.059	0.00	3.28	0.48	0.00	1.02
9	9.12 KeV	0.000	0.021	0.00	0.52	0.06	0.00	0.61
10	2.03 KeV	0.000	0.020	0.00	0.59	0.20	0.00	0.05
11	454 eV	0.000	0.009	0.00	0.09	0.03	0.00	0.02
12	22.6 eV	0.000	0.002	0.00	0.01	0.02	0.00	0.01
13	4.00 eV	0.000	0.002	0.00	0.01	0.01	0.00	0.00
14	0.54 eV	0.000	0.000	0.00	0.00	0.00	0.00	0.00
15	0.10 eV	0.000	0.000	0.00	0.00	0.00	0.00	0.00
Total (%)		0.686	0.240	1.14	3.44	1.76	3.63	4.00

SFR Uncertainties (%) - Breakdown by Isotope (Diagonal Values)

	k_{eff}	Power Peak	Doppler	Void	Burnup Total [pcm]
U238	0.16	0.05	0.60	1.65	10.5
Pu238	0.34	0.01	0.86	2.72	45.6
Pu239	0.13	0.02	0.49	1.39	20.6
Pu240	0.38	0.03	0.96	3.83	32.2
Pu241	0.52	0.02	1.70	4.34	89.8
Pu242	0.26	0.02	0.74	2.65	24.4
Np237	0.03	0.01	0.23	0.40	1.2
Am241	0.07	0.01	0.34	0.62	3.4
Am242m	0.37	0.02	1.08	3.06	50.4
Am243	0.05	0.01	0.31	0.53	5.8
Cm242	0.02	-	0.06	0.14	8.6
Cm243	0.01	-	0.02	0.05	2.3
Cm244	0.27	0.01	0.66	2.84	42.6
Cm245	0.19	0.01	0.49	1.28	31.5
Fe56	0.37	0.13	1.89	4.44	31.4
Cr52	0.04	0.01	0.27	0.47	2.2
Zr90	0.03	0.02	0.10	0.24	2.3
Na23	0.23	0.10	1.25	12.29	19.6
B10	0.12	0.24	0.22	1.16	8.7
Total	1.04	0.31	3.62	15.66	152.1

SFR Uncertainties (%) - Breakdown by Isotope (With Corr.)

	k_{eff}	Power Peak	Doppler	Void	Burnup Total [pcm]
U238	0.24	0.07	0.94	2.43	16.0
Pu238	0.64	0.02	1.50	3.00	83.2
Pu239	0.19	0.04	0.71	1.75	29.3
Pu240	0.66	0.05	1.60	3.86	56.9
Pu241	0.96	0.02	2.77	4.12	170.2
Pu242	0.41	0.03	1.15	3.37	37.5
Np237	0.06	0.01	0.31	0.51	2.1
Am241	0.11	0.01	0.55	0.91	5.6
Am242m	0.73	0.02	1.84	3.73	100.7
Am243	0.07	0.01	0.49	0.78	8.8
Cm242	0.04	-	0.10	0.13	15.5
Cm243	0.02	-	0.04	0.03	4.5
Cm244	0.40	0.02	1.00	3.01	64.5
Cm245	0.39	0.01	0.95	1.00	62.2
Cm246	0.04	-	0.14	0.28	4.1
Fe56	0.55	0.20	2.48	4.47	47.0
Cr52	0.06	0.01	0.38	0.51	2.9
Zr90	0.03	0.03	0.12	0.29	2.5
Na23	0.25	0.13	1.85	13.53	21.6
B10	0.17	0.36	0.35	1.53	12.8
Total	1.82	0.45	5.57	17.11	271.9

SFR K_{eff} Uncertainties (%) – Energy Breakdown for Selected Isotope/Reaction

Group	Energy	Pu-238 $\sigma_{fission}$	Pu-240 $\sigma_{capture}$	Pu-241 $\sigma_{fission}$	Am-242m $\sigma_{fission}$
1	19.6 MeV	0.01	0.00	0.02	0.02
2	6.07 MeV	0.18	0.03	0.10	0.12
3	2.23 MeV	0.23	0.05	0.26	0.15
4	1.35 MeV	0.31	0.11	0.40	0.28
5	498 keV	0.28	0.14	0.47	0.39
6	183 keV	0.12	0.16	0.58	0.39
7	67.4 keV	0.07	0.13	0.29	0.28
8	24.8 keV	0.06	0.13	0.16	0.12
9	9.12 keV	0.03	0.05	0.10	0.08
10	2.03 keV	0.03	0.01	0.08	0.10
11	454 eV	0.00	0.00	0.03	0.02
12-15	22.6 eV	0.00	0.00	0.00	0.00
Total		0.53	0.31	0.96	0.73

GFR Uncertainties (%) - Breakdown by Isotope

	k_{eff}	Power Peak	Doppler	Void	Burnup Total [pcm]
U235	0.07	0.05	0.14	0.17	26.2
U238	1.48	1.57	4.13	5.56	73.9
Pu238	0.25	0.07	0.53	0.32	85.6
Pu239	0.30	0.07	0.86	0.92	150.7
Pu240	0.35	0.12	0.73	0.53	16.3
Pu241	0.82	0.16	2.21	1.66	240.6
Pu242	0.27	0.09	0.68	0.68	18.2
Np237	0.06	0.04	0.24	0.16	10.6
Am241	0.34	0.22	1.37	0.90	57.8
Am242m	0.01	0.01	0.03	0.03	75.4
Am243	0.07	0.05	0.30	0.21	16.4
Cm242	-	-	-	-	85.6
Cm243	0.01	0.01	0.02	0.01	9.5
Cm244	0.13	0.09	0.32	0.27	35.7
Cm245	0.12	0.10	0.28	0.17	25.2
Cm246	-	-	0.01	-	0.5
C	0.31	0.29	1.91	1.65	8.3
He4	0.02	0.01	0.05	4.38	0.3
Si28	0.27	0.27	0.75	0.65	6.3
Zr90	0.02	0.20	0.07	0.07	1.7
Total	1.88	1.68	5.51	7.67	380.7

GFR Helium Void Uncertainties (%) - Breakdown by Isotope and Reaction

Isotope	σ_{cap}	σ_{fiss}	ν	σ_{el}	σ_{inel}	Total
U238	1.23	0.03	0.08	0.46	5.40	5.56
Pu239	0.73	0.39	0.24	-	0.31	0.92
Pu240	0.30	0.33	0.27	-	0.10	0.53
Pu241	0.12	1.65	0.09	-	0.04	1.66
Pu242	0.52	0.40	0.10	-	0.16	0.68
Am241	0.82	0.34	0.07	-	0.17	0.90
Am243	0.19	0.06	0.02	-	0.08	0.21
Cm244	0.03	0.27	0.05	-	0.01	0.27
Cm245	0.01	0.17	0.05	-	0.01	0.17
C	-	-	-	1.65	0.05	1.65
He4	-	-	-	4.38	-	4.38
Si28	0.02	-	-	0.22	0.61	0.65
Total	1.78	1.86	0.43	4.71	5.46	7.67

EFR Uncertainties (%) - Breakdown by Isotope

	k_{eff}	Power Peak	Doppler	Void	Burnup Total [pcm]
U238	0.88	0.90	2.79	3.10	82.8
Pu238	0.15	0.04	0.36	0.22	68.1
Pu239	0.32	0.09	1.08	1.49	439.7
Pu240	0.43	0.13	1.02	0.79	109.5
Pu241	0.36	0.07	1.13	1.34	634.0
Pu242	0.09	0.03	0.28	0.13	15.9
Am241	0.06	0.03	0.31	0.24	30.8
Am242m	0.03	0.02	0.06	0.08	30.8
Am243	0.01	0.01	0.08	0.07	19.0
Cm242	-	-	-	-	23.2
Cm243	0.01	0.01	0.02	0.02	11.6
Cm244	0.06	0.02	0.14	0.18	51.5
Cm245	0.06	0.04	0.13	0.10	52.5
Fe56	0.26	0.27	0.86	0.95	64.0
Ni58	0.08	0.05	0.48	0.22	12.1
Na23	0.14	0.25	0.88	6.72	39.8
O16	0.27	0.65	1.05	0.88	93.3
Total	1.18	1.18	3.80	7.83	871.0

EFR Doppler Uncertainties (%) - Breakdown by Isotope and Reaction

Isotope	σ_{cap}	σ_{fiss}	ν	σ_{el}	σ_{inel}	Total
U238	0.38	0.08	0.22	0.46	2.71	2.79
Pu238	0.06	0.34	0.11	-	0.01	0.36
Pu239	0.79	0.54	0.22	-	0.45	1.08
Pu240	0.22	0.81	0.55	0.01	0.23	1.02
Pu241	0.11	1.12	-	-	0.02	1.13
Pu242	0.17	0.22	0.04	-	0.03	0.28
Am241	0.30	0.10	0.02	-	0.02	0.31
Cm243	-	0.02	-	-	-	0.02
Cm244	0.02	0.14	0.02	-	0.01	0.14
Cm245	-	0.13	-	-	-	0.13
Fe56	0.14	-	-	0.28	0.80	0.86
Cr52	0.02	-	-	0.11	0.02	0.12
Ni58	0.09	-	-	0.46	0.10	0.48
Na23	0.02	-	-	0.71	0.52	0.88
O16	0.33	-	-	0.99	0.07	1.05
Total	1.05	1.56	0.64	1.41	2.92	3.80

EFR Uncertainty (%) on Pu Isotope Density at End of Cycle

Uncertainty on → / ↓ due to		Pu238	Pu239	Pu240	Pu241	Pu242
U238	capture	-	1.1	0.2	0.1	-
Pu238	capture	1.7	0.1	-	-	-
	fission	4.6	-	-	-	-
Pu239	capture	-	0.8	1.3	0.7	0.1
	fission	-	0.2	-	-	-
Pu240	capture	0.2	-	1.5	6.0	1.0
	fission	-	-	0.8	0.4	-
Pu241	capture	-	-	-	0.8	1.5
	fission	0.2	-	-	5.0	0.7
Pu242	capture	-	-	-	-	3.9
	fission	-	-	-	-	2.2
Am241	capture	1.3	-	-	-	0.2
	fission	0.2	-	-	-	-
Total		5.1	1.3	2.1	7.9	4.9

EFR Uncertainty (%) on Selected MA Isotope Density at End of Cycle

Uncertainty on → / ↓ due to		Am241	Am242m	Am243	Cm244	Cm245
Pu240	capture	1.6	0.6	0.2	-	-
	fission	0.1	-	-	-	-
Pu241	capture	0.2	0.1	0.4	0.1	-
	fission	1.2	0.4	0.1	-	-
Pu242	capture	-	-	9.3	4.1	1.5
	fission	-	-	0.6	0.2	-
Am241	capture	3.1	2.0	0.1	-	-
	fission	0.9	0.5	-	-	-
Am242m	capture	-	1.6	0.3	0.2	0.1
	fission	-	7.4	0.1	-	-
Am243	capture	-	-	1.9	1.9	1.0
	fission	-	-	0.5	0.2	0.1
Cm244	capture	-	-	-	1.8	7.2
	fission	-	-	-	6.0	2.8
Cm245	capture	-	-	-	-	0.9
	fission	-	-	-	-	15.6
Total		3.8	7.8	9.5	7.8	17.6

LFR Uncertainties (%) - Breakdown by Isotope

	k_{eff}	Power Peak	Doppler	Void	Burnup Total [pcm]
U238	0.78	0.08	2.59	4.73	8.5
Pu238	0.42	0.02	0.85	0.60	36.4
Pu239	0.31	0.05	1.35	0.87	106.1
Pu240	0.56	0.03	1.18	0.70	21.9
Pu241	0.61	0.02	1.44	1.37	145.1
Pu242	0.19	0.02	0.68	0.37	5.5
Np237	0.04	0.01	0.17	0.07	1.7
Am241	0.08	0.01	0.31	0.18	4.3
Am242m	0.07	-	0.16	0.16	14.4
Cm242	0.02	-	0.04	0.03	11.4
Cm244	0.16	0.01	0.38	0.12	14.3
Cm245	0.22	0.01	0.49	0.44	33.5
Fe56	0.25	0.08	1.09	1.51	5.4
Pb206	0.20	0.08	0.88	3.10	4.3
Pb207	0.17	0.08	0.86	2.61	3.4
Pb208	0.14	0.22	1.15	2.17	2.3
B10	0.44	0.57	1.02	1.14	6.6
Total	1.43	0.64	4.35	7.18	198.2

LFR Pb Void Uncertainties (%) - Breakdown by Isotope and Reaction

Isotope	σ_{cap}	σ_{fiss}	ν	σ_{el}	σ_{inel}	$\sigma_{n.2n}$	Total
U238	1.09	0.08	0.37	1.06	4.46	0.01	4.73
Pu238	0.10	0.45	0.39	-	0.02	-	0.60
Pu239	0.53	0.52	0.36	0.13	0.25	-	0.87
Pu240	0.62	0.11	0.30	0.03	0.06	-	0.70
Pu241	0.04	1.37	0.05	-	0.02	-	1.37
Pu242	0.32	0.18	0.04	-	0.04	-	0.37
Am241	0.14	0.11	0.03	-	0.03	-	0.18
Am242m	0.01	0.16	0.01	-	-	-	0.16
Cm244	0.04	0.10	0.06	-	0.01	-	0.12
Cm245	0.01	0.43	0.06	-	-	-	0.44
Fe56	0.16	-	-	0.20	1.49	-	1.51
Zr90	0.02	-	-	0.11	0.61	-	0.62
Pb204	0.42	-	-	0.03	0.17	-	0.46
Pb206	1.31	-	-	0.49	2.77	-	3.10
Pb207	0.77	-	-	0.35	2.47	0.01	2.61
Pb208	0.37	-	-	2.01	0.73	0.02	2.17
B10	1.14	-	-	0.06	0.01	-	1.14
Total	2.43	1.62	0.72	2.37	6.07	0.03	7.18

LESSON DRAWN FROM UNCERTAINTY STUDY

1. A significant result is the strong impact of correlation data (i.e. off-diagonal elements) on the uncertainty assessment. Any credible uncertainty analysis should include the best available covariance data accounting for energy correlations and possibly for cross-correlations among reactions (a typical case would be the inter-relation among total, elastic and inelastic cross-sections) and even for cross-correlation among isotopes, if needed e.g. to account for normalization issues

2. One other important point seems to be the shift of priority from the three major actinide fission data to their inelastic (in particular for U-238) and capture data (for Pu-239, and, to a lesser extent, for U-238; the case of U-235 capture data in the keV region is presently under investigation). The shift of priority is related to the relatively small uncertainty values associated to the fission cross-sections of Pu-239.

3. Higher priority should also be given to higher Pu isotopes (and in particular to their fission data) and to selected coolant/structural material inelastic cross-sections (e.g., Fe-56 and Na-23). Minor actinide data play a significant role only for dedicated burner reactors (SFR) with low Conversion Ratio CR<0.25 and high content of MA in the fuel. For more conventional burners (Pu/MA~5) and down to CR~0.25, only selected MA data require significant improvements.

Fuel Cycle Parameters: SFR

Breakdown

Isotope	Decay Heat	Dose	Neutron Source
Pb210	-	28.01	-
Ra226	-	10.76	-
Ac227	-	0.06	-
Th229	-	2.78	-
Th230	-	7.95	-
Pa231	-	0.05	-
U233	-	0.28	-
U234	0.02	3.20	-
U236	-	0.18	-
Np237	-	1.38	-
Pu238	46.51	-	0.59
Pu239	1.37	34.71	0.01
Pu240	6.82	0.07	0.13
Pu242	0.03	10.52	0.06
Am241	26.61	0.01	0.10
Am242m	0.17	-	-
Am243	0.65	0.02	0.01
Cm242	12.91	-	1.53
Cm243	0.10	-	0.01
Cm244	4.59	-	87.12
Cm245	0.10	0.01	-
Cm246	0.10	-	10.44
Cm248	-	-	0.01
Total	100.00	100.00	100.00

Uncertainty

		Decay	Dose	N.
	capture	0.06	0.04	-
Pu238	fission	0.26	0.23	-
	capture	-	0.02	-
Pu239	fission	-	0.01	-
	capture	0.11	0.01	-
Pu240	fission	0.01	-	-
	capture	0.02	0.01	-
Pu241	fission	0.16	0.02	-
	capture	-	0.02	0.02
Pu242	fission	-	0.03	-
	capture	0.03	0.05	0.02
Am241	fission	0.02	-	-
	capture	0.06	0.03	-
Am242m	fission	0.29	0.18	0.01
	capture	0.01	0.01	0.24
Am243	fission	-	0.01	-
	capture	-	-	-
Cm242	fission	0.02	-	0.01
	capture	0.01	-	0.18
Cm244	fission	0.05	-	0.91
	capture	-	-	0.02
Cm245	fission	-	0.01	-
	capture	-	-	0.03
Cm246	fission	-	-	0.07
Total		0.45	0.31	0.96

Fuel Cycle Parameters: EFR

Breakdown

Isotope	Decay Heat	Dose	Neutron Source
Pb210	-	11.83	-
Ra226	-	4.54	-
Ac227	-	0.13	-
Th229	-	1.90	-
Th230	-	3.36	-
Pa231	-	0.10	-
U233	-	0.19	-
U234	0.02	1.35	-
U235	-	0.01	-
U236	-	0.23	-
U238	-	0.01	-
Np237	-	0.94	-
Pu238	34.15	-	0.67
Pu239	5.58	70.45	0.04
Pu240	14.72	0.08	0.39
Pu241	0.01	-	-
Pu242	0.03	4.87	0.07
Am241	39.38	-	0.14
Am242m	0.03	-	-
Am243	0.39	0.01	-
Cm242	1.96	-	1.95
Cm243	0.21	-	0.02
Cm244	3.42	-	85.82
Cm245	0.05	-	-
Cm246	0.06	-	7.38
Total	100.00	100.00	100.00

Uncertainty

		Decay	Dose	N.
Pu238	capture	0.50	0.22	0.01
	fission	1.34	0.71	0.03
Pu239	capture	0.37	0.49	0.01
	fission	0.03	0.16	-
	n,2n	0.01	0.01	-
Pu240	capture	1.79	0.23	0.07
	fission	0.26	0.02	0.01
Pu241	capture	-	-	0.10
	fission	-	0.02	0.05
Pu242	capture	0.20	0.04	3.51
	fission	0.01	0.11	0.15
Am241	capture	0.17	0.24	0.08
	fission	0.21	0.06	0.02
Am242m	capture	0.05	0.01	0.12
	fission	0.26	0.09	0.04
Am243	capture	0.06	0.04	1.60
	fission	0.01	0.01	0.16
Cm244	capture	0.07	-	1.50
	fission	0.23	-	5.11
Cm245	capture	-	-	0.16
	fission	-	-	0.17
Cm246	capture	-	-	0.22
	fission	-	-	0.37
Total		2.40	1.24	6.60

Fuel Cycle Parameters: VHTR

Breakdown

Isotope	Decay Heat	Dose	Neutron Source
Pb210	-	8.80	-
Ra226	-	3.38	-
Ac227	-	0.74	-
Th229	-	6.17	-
Th230	-	2.50	-
Pa231	-	0.58	-
U233	-	0.62	-
U234	0.01	1.02	-
U235	-	0.05	-
U236	-	0.54	-
U238	-	0.11	-
Np237	0.01	3.04	-
Pu238	18.77	-	1.32
Pu239	3.06	63.00	0.08
Pu240	4.11	0.04	0.38
Pu241	0.02	-	-
Pu242	0.03	9.42	0.30
Am241	72.45	-	0.29
Am243	0.35	0.01	0.01
Cm242	0.09	-	5.23
Cm243	0.03	-	0.01
Cm244	1.06	-	92.14
Cm245	0.01	-	-
Cm246	-	-	0.23
Total	100.00	100.00	100.00

Uncertainty

		Decay	Dose	N. Sr
	capture	0.18	0.17	0.01
U235	fission	0.03	0.03	-
	capture	0.07	0.07	0.01
U236	fission	-	-	-
	capture	0.77	0.75	0.92
U238	fission	-	-	-
	n,2n	0.01	0.01	-
	capture	0.17	0.13	0.01
Np237	fission	-	-	-
	capture	0.03	0.02	-
Pu238	fission	0.01	0.01	-
	capture	0.80	0.14	1.06
Pu239	fission	0.53	0.40	0.51
	capture	0.05	0.03	0.23
Pu240	fission	-	-	-
	capture	0.45	0.45	4.39
Pu241	fission	0.28	0.28	0.86
	capture	0.05	0.02	2.93
Pu242	fission	-	-	0.01
	capture	0.01	0.02	0.04
Am241	fission	-	-	-
	capture	0.02	0.02	2.06
Am243	fission	-	-	-
	capture	-	-	0.38
Cm244	fission	-	-	0.03
Total		1.37	1.04	5.94

Fuel Cycle Parameters: . n_f Uncertainties (%)

	ABTR		SFR		EFR		GFR		LFR		VHTR		PWR	
	Diag.	Full	Diag.	Full	Diag.	Full	Diag.	Full	Diag.	Full	Diag.	Full	Diag.	Full
U235	0.07	0.16	0.31	0.59	2.57	5.14	0.42	0.84	0.36	0.72	0.25	0.44	0.46	0.87
U238	0.01	0.01	0.02	0.02	0.24	0.26	0.04	0.04	0.03	0.03	0.05	0.06	0.04	0.06
Np237	0.25	0.26	0.11	0.22	2.51	3.04	0.25	0.44	0.18	0.33	1.03	1.38	1.05	1.49
Pu238	0.21	0.36	0.42	0.74	2.71	4.91	0.64	1.15	0.56	0.95	1.28	1.55	1.05	1.45
Pu239	0.04	0.05	0.06	0.09	1.13	1.33	0.37	0.41	0.20	0.23	0.96	1.22	0.88	1.22
Pu240	0.13	0.22	0.10	0.19	1.25	2.20	0.31	0.44	0.16	0.31	1.25	1.56	1.11	1.49
Pu241	0.35	0.72	0.62	1.23	3.99	8.01	0.83	1.55	1.19	2.44	2.04	2.33	2.02	2.55
Pu242	0.13	0.22	0.25	0.40	2.75	4.66	0.51	0.84	0.35	0.55	4.75	5.03	3.87	4.47
Am241	0.08	0.13	0.18	0.30	2.12	3.78	0.35	0.60	0.27	0.45	2.13	2.40	2.50	2.94
Am242m	0.47	0.86	0.70	1.40	4.26	8.36	2.17	3.85	1.17	2.22	5.63	7.21	5.41	7.50
Am243	0.35	0.55	0.47	0.76	6.41	10.87	1.07	1.81	0.91	1.46	5.58	5.93	4.48	5.04
Cm242	1.13	1.91	1.38	2.36	1.58	2.87	2.43	4.27	2.49	4.11	1.88	2.15	1.87	2.36
Cm243	1.30	2.53	1.90	3.77	10.59	20.63	3.91	6.51	3.29	6.37	8.37	11.80	5.50	7.56
Cm244	0.38	0.59	0.73	1.10	4.83	7.67	0.91	1.54	0.88	1.28	6.01	6.42	4.77	5.33
Cm245	0.90	1.69	1.47	2.91	9.37	18.15	1.83	3.60	2.35	4.54	8.02	8.76	6.84	7.51
Cm246	0.51	0.89	0.47	0.77	4.48	7.45	1.35	2.56	0.57	0.89	-	-	-	-

Summary Fuel Cycle Performance

- Yucca Mountain repository capacity is dominated by long term heat load (Pu238, Pu239, Am241)
- Neutron source of recycle materials has strong implication for fabrication costs and non-proliferation (Cm244)
- Radiotoxicity is key measure for environmental benefits (Pu239)

Uncertainties on fuel cycle parameters are small:

- **Decay heat and dose less than a percent in many cases**
- **Neutron source a few percents for EFR, VHTR and PWR**
- **Minor Actinide final density uncertainties significant only in few cases**

Target Accuracy Assessment

Targeted Accuracies for Main design Parameters

Multiplication factor (BOL)	300 pcm
Power peak (BOL)	2%
Burnup reactivity swing	300 pcm
Reactivity coefficients (Coolant void and Doppler - BOL)	7%
Major nuclide density at end of irradiation cycle	2%
Other nuclide density at end of irradiation cycle	10%

Integral parameter Uncertainities Uncertainties (%)

	ABTR	SFR	EFR	GFR	LFR	ADMAB	VHTR	PWR
k_{eff} BOC	0.62	1.04	0.79	1.24	0.88	1.95	0.37	0.36
k_{eff} EOC	-	-	-	-	-	-	0.41	0.64
Power Peak BOC	0.32	0.31	0.81	1.18	0.45	14.22	0.85	-
Power Peak EOC	-	-	-	-	-	-	0.90	-
Doppler BOC	2.86	3.62	2.46	3.62	2.85	-	4.27	1.53
Doppler EOC	-	-	-	-	-	-	2.77	2.01
Void	5.11	15.66	6.68	5.46	4.97	13.11	-	-
Burnup [pcm]	37.4	152.1	584.9	254.2	127.7	602.9	487.0	684.6
$N_{f,U235}$	0.07	0.31	2.57	0.42	0.36	-	0.25	0.46
$N_{f,U238}$	0.01	0.02	0.24	0.04	0.03	-	0.05	0.04
$N_{f,Np237}$	0.25	0.11	2.51	0.25	0.18	0.20	1.03	1.05
$N_{f,Pu238}$	0.21	0.42	2.71	0.64	0.56	1.13	1.28	1.05
$N_{f,Pu239}$	0.04	0.06	1.13	0.37	0.20	0.12	0.96	0.88
$N_{f,Pu240}$	0.13	0.10	1.25	0.31	0.16	0.26	1.25	1.11
$N_{f,Pu241}$	0.35	0.62	3.99	0.83	1.19	0.90	2.04	2.02
$N_{f,Pu242}$	0.13	0.25	2.75	0.51	0.35	0.54	4.75	3.87
$N_{f,Am241}$	0.08	0.18	2.12	0.35	0.27	0.31	2.13	2.50
$N_{f,Am242m}$	0.47	0.70	4.26	2.17	1.17	1.72	5.63	5.41
$N_{f,Am243}$	0.35	0.47	6.41	1.07	0.91	0.27	5.58	4.48
$N_{f,Cm242}$	1.13	1.38	1.58	2.43	2.49	2.78	1.88	1.87
$N_{f,Cm243}$	1.30	1.90	10.59	3.91	3.29	3.04	8.37	5.50
$N_{f,Cm244}$	0.38	0.73	4.83	0.91	0.88	1.07	6.01	4.77
$N_{f,Cm245}$	0.90	1.47	9.37	1.83	2.35	2.48	8.02	6.84
$N_{f,Cm246}$	0.51	0.47	4.48	1.35	0.57	3.12	-	-

Target Accuracy Assessment: ABTR

Isotope	Cross-Section	Energy Range	Uncertainty (%)		
			Initial	Required	
				λ=1	λ≠1 case B [(a)]
U238	σ_{inel}	6.07 - 2.23 MeV	19.8	3.3	5.8
		2.23 - 1.35 MeV	20.6	3.6	6.3
		1.35 - 0.498 MeV	11.6	6.5	11.4
U238	σ_{capt}	24.8 - 9.12 keV	9.4	2.9	1.6
Pu239	σ_{capt}	498 - 183 keV	11.6	5.7	3.2
		183 - 67.4 keV	9.0	5.0	2.8
		67.4 - 24.8 keV	10.1	5.8	3.2
		9.12 - 2.04 keV	15.5	7.4	4.1
Pu241	σ_{fiss}	183 - 67.4 keV	19.9	8.8	7.0
Fe56	σ_{inel}	2.23 - 1.35 MeV	25.4	5.6	9.9
		1.35 - 0.498 MeV	16.1	7.5	13.1
Na23	σ_{inel}	1.35 - 0.498 MeV	28.0	10.1	17.7

	λ=1	λ≠1 case A	λ≠1 case B
$\lambda_{capt,fiss,\nu}$ (U235,U238,Pu239)	1	1	1
$\lambda_{capt,fiss,\nu}$ (other fissiles)	1	2	2
λ_{capt} (structurals)	1	1	1
λ_{el} (fissiles and structurals)	1	1	1
λ_{inel} (fissiles and structurals)	1	3	10

Target Accuracy Assessment: SFR

Isotope	Cross-Section	Energy Range	Uncertainty (%)	
			Initial	Required (λ=1)
Pu241	σ_{fiss}	1.35 - 0.498 MeV	16.6	3.4
		498 - 183 keV	13.5	2.6
		183 - 67.4 keV	19.9	2.6
		24.8 - 9.12 keV	11.3	3.5
		2.04 - 0.454 keV	12.7	4.4
Fe56	σ_{inel}	2.23 - 1.35 MeV	25.4	3.3
		1.35 - 0.498 MeV	16.1	3.2
Na23	σ_{inel}	1.35 - 0.498 MeV	28.0	4.0
Cm244	σ_{fiss}	1.35 - 0.498 MeV	50.0	5.1
Am242m	σ_{fiss}	1.35 - 0.498 MeV	16.5	4.2
		498 - 183 keV	16.6	3.1
		183 - 67.4 keV	16.6	3.1
		67.4 - 24.8 keV	14.4	4.0
		24.8 - 9.12 keV	11.8	4.2
		2.04 - 0.454 keV	12.2	5.1
Pu240	σ_{fiss}	1.35 - 0.498 MeV	5.8	1.8
Pu240	ν	1.35 - 0.498 MeV	3.7	1.5
Pu238	σ_{fiss}	2.23 - 1.35 MeV	33.8	5.6
		1.35 - 0.498 MeV	17.1	3.3
		498 - 183 keV	17.1	3.6
Pu238	ν	1.35 - 0.498 MeV	7.0	2.7
Pu242	σ_{fiss}	2.23 - 1.35 MeV	21.4	4.9
		1.35 - 0.498 MeV	19.0	3.5
Cm245	σ_{fiss}	183 - 67.4 keV	47.5	6.7
Pu242	σ_{capt}	24.8 - 9.12 keV	38.6	8.4
U238	σ_{capt}	24.8 - 9.12 keV	9.4	4.3
Fe56	σ_{capt}	2.04 - 0.454 keV	11.2	5.3

Target Accuracy Assessment: EFR

Isotope	Cross-Section	Energy Range	Uncertainty (%)	
			Initial	Required (λ=1)
U238	σ_{inel}	6.07 - 2.23 MeV	19.8	3.7
		2.23 - 1.35 MeV	20.6	4.0
		1.35 - 0.498 MeV	11.6	5.0
U238	σ_{capt}	24.8 - 9.12 keV	9.4	2.9
O16	σ_{capt}	19.6 - 6.07 MeV	100.0	14.2
		6.07 - 2.23 MeV	100.0	10.9
Fe56	σ_{inel}	2.23 - 1.35 MeV	25.4	6.6
		1.35 - 0.498 MeV	16.1	8.4
Pu241	σ_{fiss}	1.35 - 0.498 MeV	16.6	8.0
		498 - 183 keV	13.5	6.7
		183 - 67.4 keV	19.9	5.7
		67.4 - 24.8 keV	8.7	6.2
		24.8 - 9.12 keV	11.3	6.8
		9.12 - 2.04 keV	10.4	7.6
		2.04 - 0.454 keV	12.7	6.9
Pu240	σ_{fiss}	1.35 - 0.498 MeV	5.8	3.5
Pu239	σ_{capt}	183 - 67.4 keV	9.0	7.0
		67.4 - 24.8 keV	10.1	6.7
		24.8 - 9.12 keV	7.4	6.1
		9.12 - 2.04 keV	15.5	5.6
Na23	σ_{inel}	1.35 - 0.498 MeV	28.0	7.9
Pu240	σ_{capt}	498 - 183 keV	14.3	8.9
		183 - 67.4 keV	13.8	6.7
		67.4 - 24.8 keV	11.3	6.1
		24.8 - 9.12 keV	10.2	6.5

Target Accuracy Assessment: GFR

Isotope	Cross-Section	Energy Range	Uncertainty (%)	
			Initial	Required (λ=1)
U238	σ_{inel}	6.07 - 2.23 MeV	19.8	1.6
		2.23 - 1.35 MeV	20.6	1.8
		1.35 - 0.498 MeV	11.6	2.4
U238	σ_{capt}	24.8 - 9.12 keV	9.4	1.6
		9.12 - 2.04 keV	3.1	1.4
Pu241	σ_{fiss}	1.35 - 0.498 MeV	16.6	3.5
		498 - 183 keV	13.5	3.1
		183 - 67.4 keV	19.9	2.5
		67.4 - 24.8 keV	8.7	2.5
		24.8 - 9.12 keV	11.3	2.6
		9.12 - 2.04 keV	10.4	2.2
		2.04 - 0.454 keV	12.7	2.8
Pu239	σ_{capt}	9.12 - 2.04 keV	15.5	2.8
Si28	σ_{capt}	19.6 - 6.07 MeV	52.9	5.6
Si28	σ_{inel}	6.07 - 2.23 MeV	13.5	3.0
		2.23 - 1.35 MeV	50.0	5.8
C	σ_{el}	1.35 - 0.498 MeV	5.0	1.7
Pu242	σ_{fiss}	1.35 - 0.498 MeV	19.0	4.0
Pu240	σ_{fiss}	1.35 - 0.498 MeV	5.8	2.2
Am241	σ_{fiss}	6.07 - 2.23 MeV	11.7	3.3

Target Accuracy Assessment: LFR

Isotope	Cross-Section	Energy Range	Uncertainty (%)	
			Initial	Required (λ=1)
U238	σ_{inel}	6.07 - 2.23 MeV	19.8	2.8
		2.23 - 1.35 MeV	20.6	2.3
		1.35 - 0.498 MeV	11.6	2.1
Pu241	σ_{fiss}	1.35 - 0.498 MeV	16.6	3.7
		498 - 183 keV	13.5	2.6
		183 - 67.4 keV	19.9	2.6
B10	σ_{capt}	498 - 183 keV	15.0	2.4
		183 - 67.4 keV	10.0	2.3
		67.4 - 24.8 keV	10.0	2.7
U238	σ_{capt}	24.8 - 9.12 keV	9.4	2.0
Pu240	σ_{fiss}	1.35 - 0.498 MeV	5.8	1.6
Pu240	ν	1.35 - 0.498 MeV	3.7	1.3
Pu238	σ_{fiss}	1.35 - 0.498 MeV	17.1	3.3
		498 - 183 keV	17.1	3.4
Fe56	σ_{inel}	2.23 - 1.35 MeV	25.4	4.2
		1.35 - 0.498 MeV	16.1	3.6
Pb206	σ_{inel}	2.23 - 1.35 MeV	14.2	3.3
Pb207	σ_{inel}	1.35 - 0.498 MeV	11.3	3.0
Pu242	σ_{fiss}	1.35 - 0.498 MeV	19.0	3.9
Cm244	σ_{fiss}	1.35 - 0.498 MeV	50.0	6.4

Summary on Target Accuracy requirements

- As expected from the results of the uncertainty analysis, very tight requirements are shown for the σ_{inel} of U-238 (2-3%), Fe-56 (3-6%), Na-23 (4-10%) and even for Pb isotopes. Some of the required accuracies are probably beyond achievable limits with current experimental techniques. There are little margins to relax the requirements on σ_{inel} if one does not want to produce comparably difficult requirements on some Pu isotope σ_{fiss} and σ_{capt}.
- The accuracy requirements for Pu isotopes are very tight (very often <2-3%). As for σ_{capt}, the requirements for U-238 and Pu-239 aim to cut by more then a factor of 2 the current uncertainties. The requirement for improved accuracy of the higher Pu isotopes, and in particular the fission of Pu-241, is more stringent for the EFR, GFR and LFR cases.
- In the case of MA, uncertainties improvements for selected isotopes and reactions in some cases are very significant. However, this is the case when MA play an important role in the critical balance, as for MA dedicated burner with a fuel heavily loaded with MA (SFR). For these very specific cases, the accuracy requirement for σ_{fiss} of selected MA isotopes can range from 3 to 7%.
- A few specific requirements are indicated according to specificities of some cores, e.g., Si data requirements for the GFR and Pb data for the LFR

Adjustment

- Uncertainty and sensitivity analysis can be used in order to reduce priori uncertainties on performance parameters (like k_{eff} or reactivity coefficients) that characterize a reference design configuration.

- Several approaches (usually called “bias factor” methods) have been attempted. Moreover, statistical “data adjustment” methods have been developed and used (mostly in Europe). E.g. the use of adjusted data did allow to predict the SUPERPHENIX critical mass (~4t of Pu!) to <0.3% $\Delta k/k$.

- A general and consistent method can be defined and an application can be performed to show relevant features of the uncertainty reduction process.

Adjustment criteria

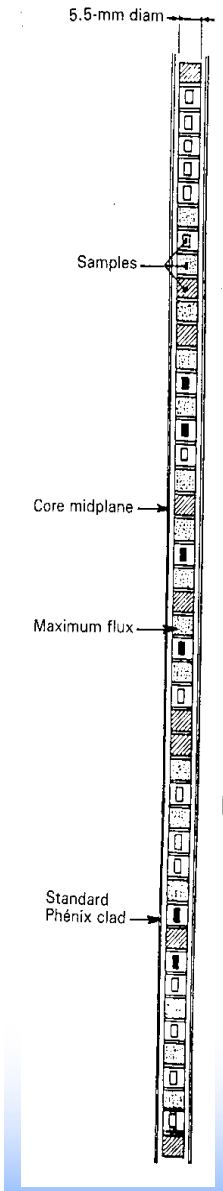
- **Reference system** is the Advanced Burner Reactor (metal and oxide fuel) as considered within GNEP
- The selected integral experiments meet a series of **requirements**: a) low and well documented experimental uncertainties; b) enabling to separate effects (e.g., capture and fission); and c) allowing validating global energy and space dependent effects.
  - irradiation experiment for b)
  - "representative experiments" for c)
  - specific spatial effects are singled out with appropriate experiments (e.g. experiments with or without blankets)
- **Global** statistical adjustment: $\tilde{B}_p = B_p - B_p S_A \left(S_A^T B_p S_A + B_A \right)^{-1} S_A^T B_p$
- **Four bands of energy**: 20 MeV, 0.5 MeV, 67 KeV, 2 KeV

List of integral experiments used in the adjustment

Experiment	Parameter to be analyzed			Fuel Type	Pu/(U+Pu)
	Critical mass	Reaction Rates	Irradiation Experiment		
GODIVA	Yes	Yes	-	U Metal	0.0
JEZEBEL^{239}	Yes	Yes	-	Pu Metal	1.0
JEZEBEL^{240}	Yes	-	-	Pu Metal	1.0
ZPR-3/53	Yes	Yes	-	PuC-UC	0.42
ZPR-3/54	Yes	Yes	-	PuC-UC	0.42
ZPPR-15	Yes	Yes		Pu-U Metal	0.13
COSMO[a]	-	Yes	-	PuO_2-UO_2	0.27
CIRANO[a]	Yes	-	-	PuO_2-UO_2	0.27
PROFIL[b]	-	-	Yes	PuO_2-UO_2	0.27
TRAPU[b]	-	-	Yes	PuO_2-UO_2	0.27

a) experiments performed in the MASURCA facility
b) irradiation experiments performed in the PHENIX reactor

Isotope sample irradiation: PROFIL irradiation in PHENIX



Plutonium and americium

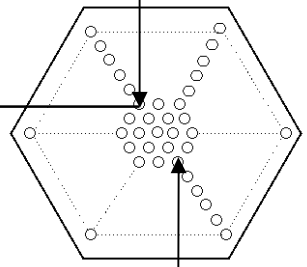
References

Uranium

Fission products

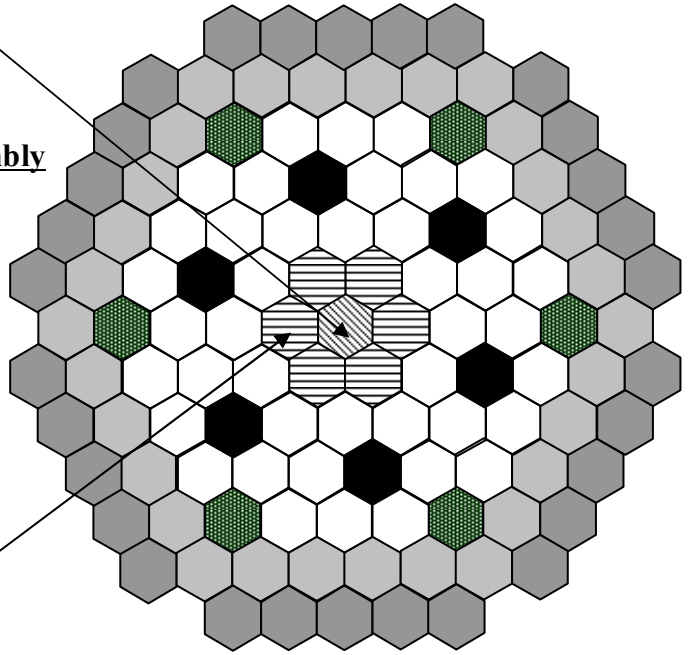
PROFIL-1 Irradiation Pin

PHENIX Subassembly



PROFIL-2 Irradiation Pin

PHENIX Core



Core 1

Core 2

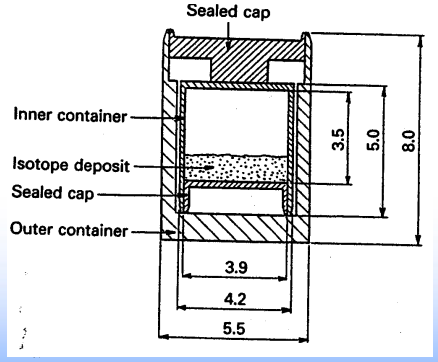
Core 3

Control Rod

Core 1

Core 1

PROFIL-1 Irradiation Pin



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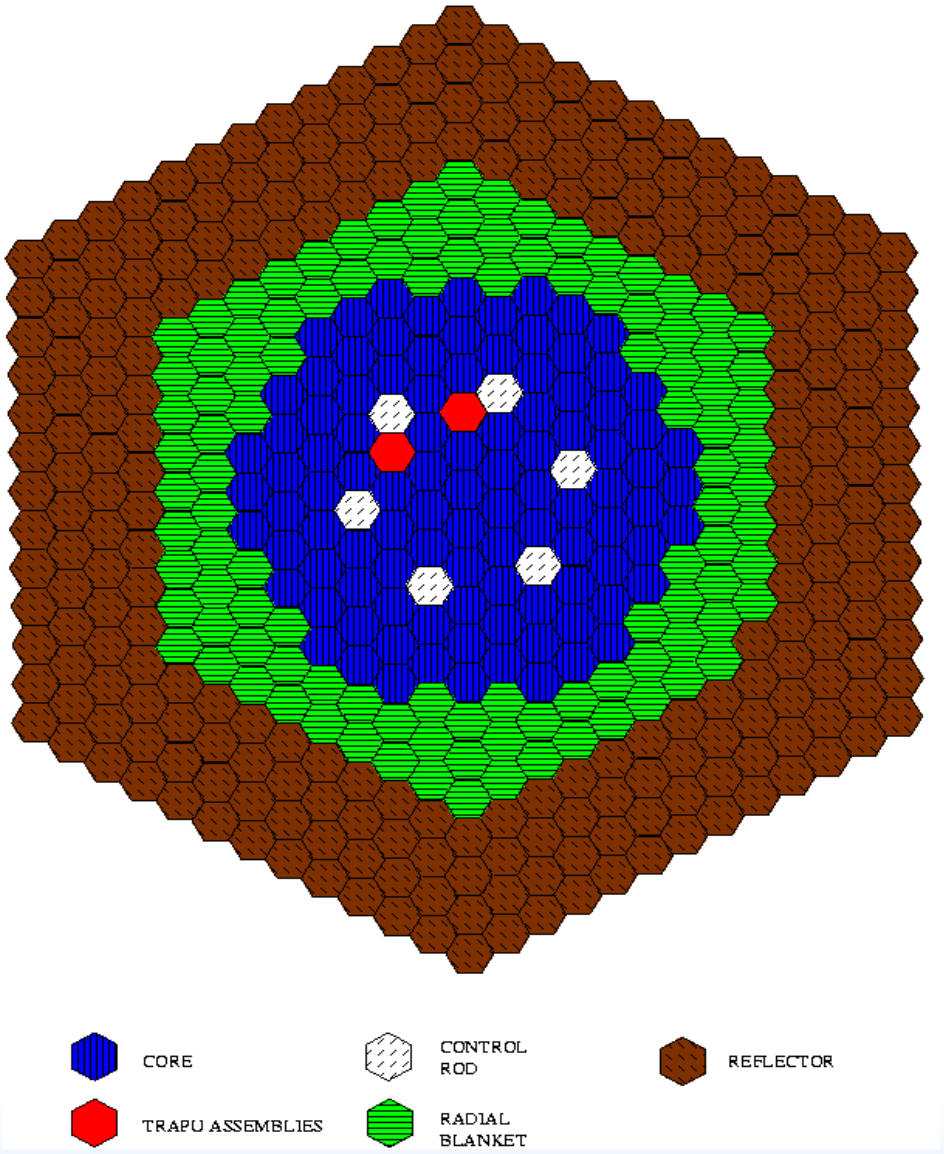
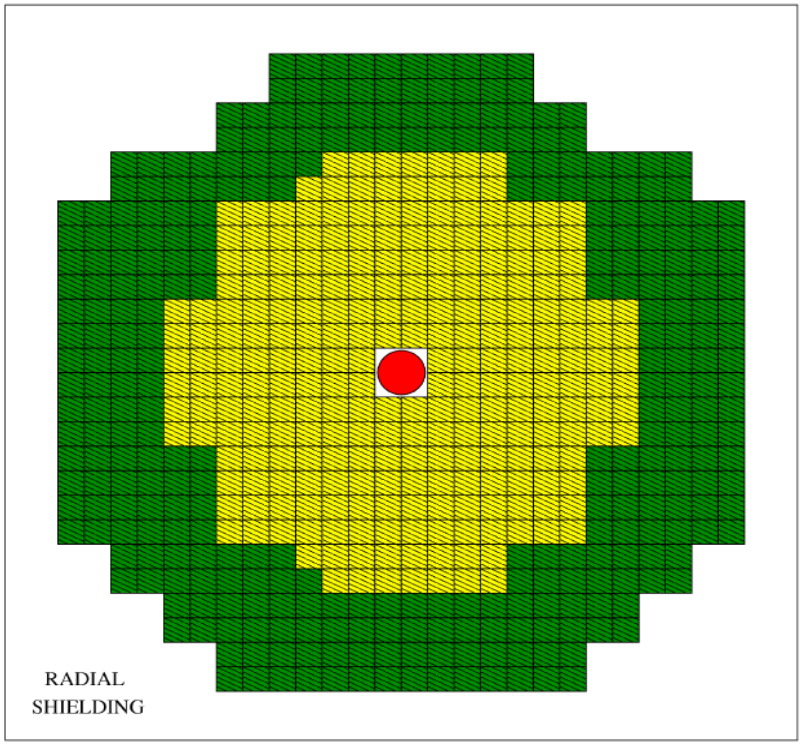
Fuel pin irradiations: TRAPU Experiment positions in Phénix





The MUSE experiment at MASURCA (critical mass and fission rates)….

The CIRANO experiment at MASURCA (critical mass)…

…..and 7 critical masses from GODIVA, JEZEBEL and Np sphere experiments.

C/E and Associated Uncertainties (σ) Before and After Adjustment

Type of Experiment	old C/E ± σ	new C/E ± σ	Type of Experiment	old C/E ± σ	new C/E ± σ
U235 Capture PROFIL1 [a]	0.977 ± 0.020	1.009 ± 0.009	**Cm244 TRAPU2 [b]**	**0.872± 0.023**	**0.978 ± 0.021**
U238 Capture PROFIL1 [a]	1.004 ± 0.023	1.005 ± 0.010	U238 Fission Rate COSMO[c]	0.988 ± 0.015	1.006 ± 0.010
Pu238 Capture PROFIL2 [a]	**1.744 ± 0.040**	**1.015 ± 0.036**	Np237 Fission Rate COSMO [c]	0.960 ± 0.015	0.979 ± 0.011
Pu239 (N,2N) PROFIL1 [a]	**0.752 ± 0.150**	**0.949± 0.133**	**Pu238 Fission Rate COSMO [c]**	**1.083 ± 0.025**	**1.005 ± 0.023**
Pu239 Capture PROFIL1 [a]	0.963 ± 0.030	1.021 ± 0.015	Pu239 Fission Rate COSMO [c]	0.983 ± 0.013	0.984 ± 0.003
Pu240 Capture PROFIL1 [a]	1.001 ± 0.022	0.995 ± 0.013	Pu240 Fission Rate COSMO [c]	1.034 ± 0.023	1.016 ± 0.016
Pu241 Capture PROFIL1 [a]	**0.847 ± 0.041**	**0.871± 0.013**	Pu241 Fission Rate COSMO [c]	0.998 ± 0.020	1.013 ± 0.017
Pu242 Capture PROFIL1 [a]	**1.092 ± 0.035**	**1.128 ± 0.019**	Pu242 Fission Rate COSMO [c]	1.000 ± 0.023	1.002 ± 0.022
Am241 Capture PROFIL1 [a]	1.000 ± 0.020	1.003± 0.015	**Am241 Fission Rate COSMO [c]**	**1.074 ± 0.023**	**1.003 ± 0.022**
Np237 Capture PROFIL2 [a]	0.988 ± 0.036	1.009± 0.022	Am243 Fission Rate COSMO [c]	1.059 ± 0.023	1.008 ± 0.021

C/E and Associated Uncertainties (σ) Before and After Adjustment (cont.)

Type of Experiment	old C/E ± σ	new C/E ± σ	Type of Experiment	old C/E ± σ	new C/E ± σ
U236 TRAPU2 (b)	0.965 ± 0.010	0.995 ± 0.009	k_{eff} GODIVA (d)	1.000 ± 0.001	0.999 ± 0.001
Np237 TRAPU2 (b)	**0.880 ± 0.033**	**0.954 ± 0.026**	U238 Fission Rate GODIVA (d)	0.955 ± 0.012	0.965 ± 0.004
Pu238 TRAPU2 (b)	0.942 ± 0.010	1.000 ± 0.006	Np237 Fission Rate GODIVA (d)	0.991 ± 0.016	1.003 ± 0.010
Pu239 TRAPU2 (b)	1.006 ± 0.005	1.001 ± 0.004	Pu239 Fission Rate GODIVA (d)	0.986 ± 0.017	0.987 ± 0.003
Pu240 TRAPU2 (b)	0.982 ± 0.006	1.000 ± 0.006	k_{eff} JEZEBEL9 (e)	1.000 ± 0.002	1.001 ± 0.001
Pu241 TRAPU1 (b)	1.005 ± 0.006	1.001 ± 0.003	U238 Fission Rate JEZEBEL9 (e)	0.974 ± 0.009	0.984 ± 0.004
Pu242 TRAPU1 (b)	0.998 ± 0.008	1.012 ± 0.004	Np237 Fission Rate JEZEBEL9 (d)	1.009 ± 0.017	1.021 ± 0.010
Am241 TRAPU2 (b)	0.985 ± 0.039	0.986± 0.005	k_{eff} JEZEBEL0 (e)	1.000 ± 0.002	0.999 ± 0.002
Am242 TRAPU2 (b)	1.029 ± 0.043	1.032 ± 0.013	**k_{eff} CIRANO (f)**	**1.007 ± 0.002**	**1.002 ± 0.001**
Am243 TRAPU1 (b)	0.939 ± 0.026	0.974 ± 0.020	k_{eff} ZPPR-15	0.999± 0.002	0.999± 0.001
Cm242 TRAPU1 (b)	1.003 ± 0.039	0.971 ± 0.013	**k_{eff} ZPR-3/53**	**1.009 ± 0.002**	**1.001 ± 0.001**
Cm243 TRAPU2 (b)	**0.462 ± 0.031**	**0.999 ± 0.031**	**k_{eff} ZPR-3/54**	**1.008 ± 0.002**	**1.000 ± 0.001**

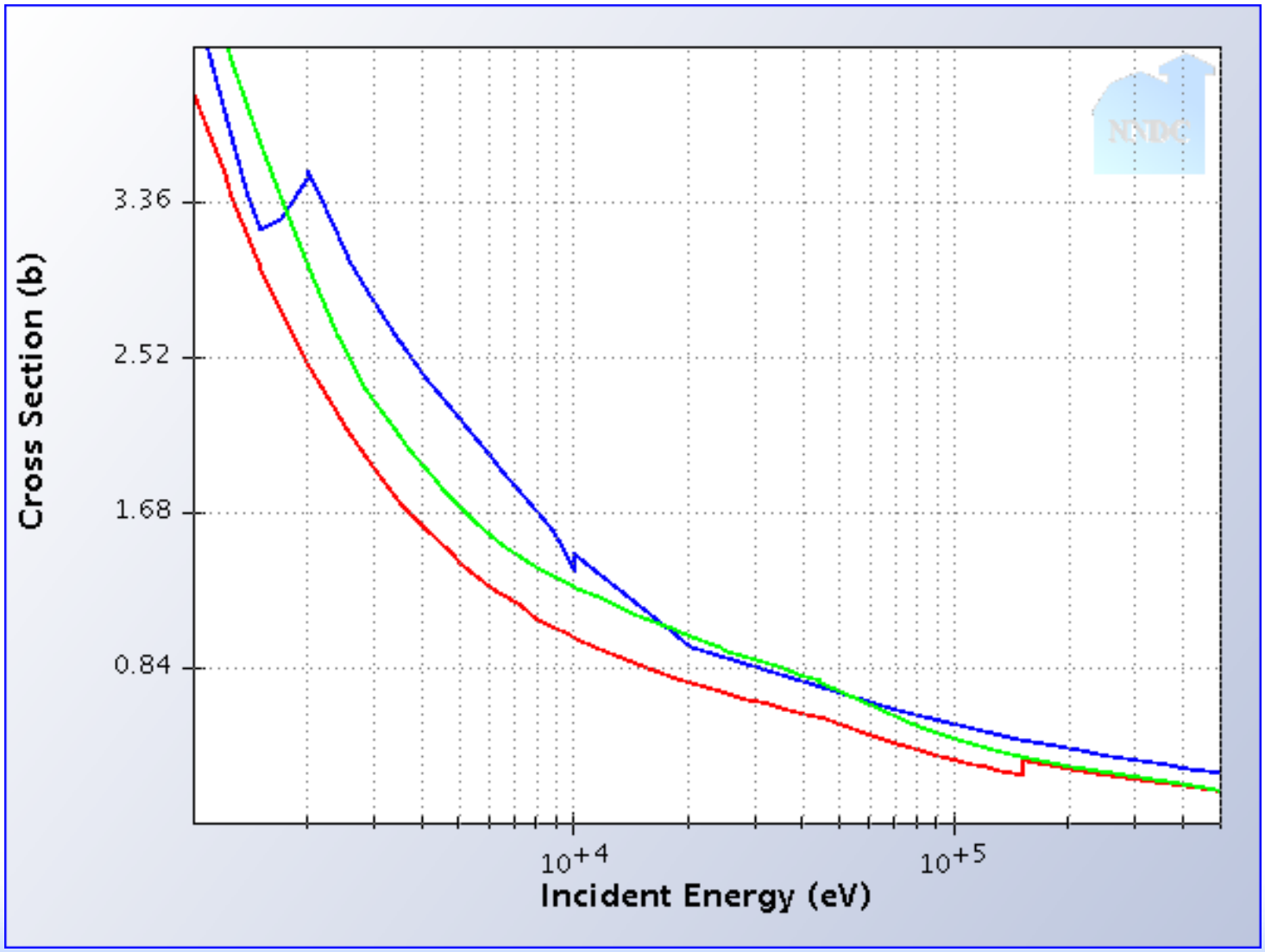
Calculated Adjusted Data Change and Original and Adjusted Standard Deviation (%)

Param.	Adjus. %	Stand. Deviat. %		Param.	Adjus. %	Stand. Deviat. %	
		Orig.	Adj.			Orig.	Adj.
Pu238 σ^{cap} gr. 2	**-61.9**	**50.0**	**22.7**	**Pu240 σ^{cap} gr. 2**	-0.7	**14.0**	**3.4**
Pu238 σ^{cap} gr. 3	**-67.4**	**50.0**	**12.0**	**Pu240 σ^{cap} gr. 3**	-0.4	**9.0**	**2.0**
Pu238 σ^{cap} gr. 4	**-60.7**	**50.0**	**24.3**	**Pu241 σ^{cap} gr. 2**	**8.9**	14.8	9.3
Pu238 σ^{fis} gr. 1	**-11.6**	**18.3**	**7.7**	**Pu241 σ^{fis} gr. 1**	2.9	**15.0**	**6.0**
Pu239 σ^{cap} gr. 3	**5.0**	8.9	5.8	**Pu241 σ^{fis} gr. 2**	2.7	**16.9**	**5.4**
Pu239 σ^{cap} gr. 4	**11.3**	12.6	7.6	**Pu242 σ^{cap} gr. 3**	**7.1**	**38.1**	**10.2**
				Pu242 σ^{fis} gr. 1	-0.6	**16.6**	**2.6**

Calculated Adjusted Data Change and Original and Adjusted Standard Deviation (%)(cont.)

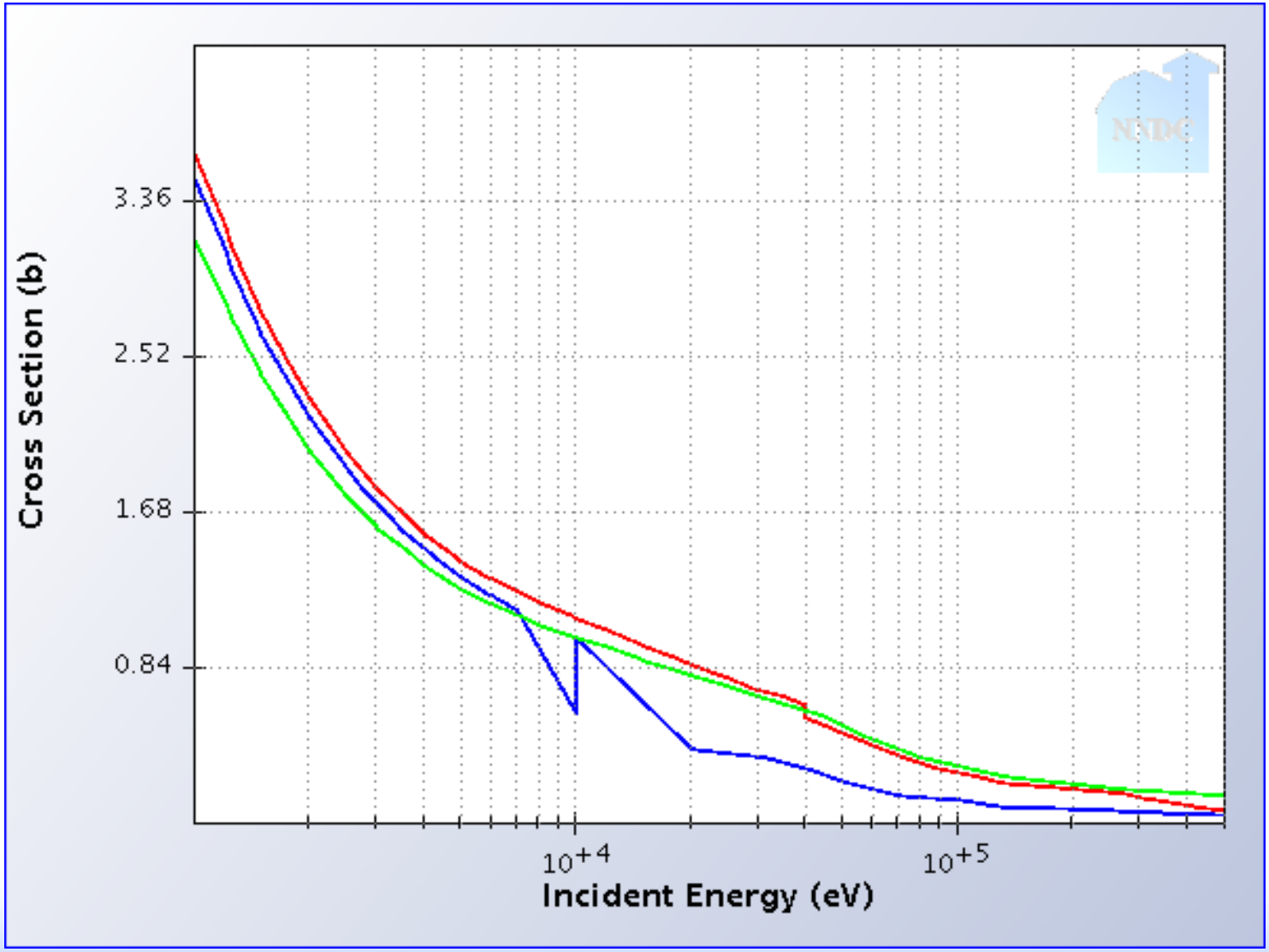
Param.	Adjus. %	Stand. Deviat. %		Param.	Adjus. %	Stand. Deviat. %	
		Orig.	Adj.			Orig.	Adj.
Cm242 σ^{cap} gr. 2	101.5	100.0	70.7	U238 σ^{n2n} gr. 1	9.6	5.0	3.1
Cm242 σ^{cap} gr. 3	139.5	100.0	24.5	Pu239 σ^{n2n} gr. 1	25.8	30.0	14.1
Cm242 σ^{cap} gr. 4	96.8	100.0	74.3				
Am241 σ^{fis} gr. 1	-7.7	8.3	2.4				
Am243 σ^{cap} gr. 2	5.2	6.0	5.6	U238 σ^{inel} gr. 1	3.5	17.1	8.5
Am243 σ^{cap} gr. 3	9.7	5.5	3.5	Fe56 σ^{inel} gr. 1	-7.9	10.5	8.4
Am243 σ^{cap} gr. 4	9.1	5.6	3.9	Na23 σ^{inel} gr. 1	-3.4	16.7	14.3
Am243 σ^{fis} gr. 1	-5.7	6.1	2.3				

Pu-238 (n,γ)



Blue ENDF/B-VII.0; Red JEFF-3.1; Green JENDL-3.3

Cm-242 (n,γ)



Blue ENDF/B-VII.0; Red JEFF-3.1; Green JENDL-3.3

ZPPR-15 k_{eff} correlation with other experiments after adjustment

Type of Experiment	A-posteriori Correlation
U-238 Capture, PROFIL1	-0.22
Pu-240 Capture, PROFIL1	0.11
Pu-239, TRAPU2	-0.21
U-238 Fission Rate, COSMO	0.18
Np-237 Fission Rate, COSMO	0.11
Pu-241 Fission Rate, COSMO	0.20
Np-237 Fission Rate, GODIVA	-0.07
Np-237 Fission Rate, JEZEBEL9	-0.07
k_{eff}, CIRANO	0.30
k_{eff}, ZPR-3/53	0.08
k_{eff}, ZPR-3/54	-0.05

K_{eff} Uncertainties [pcm] calculated with the BOLNA (BNL; ORNL; LANL; NRG; ANL) covariance matrix and adjusted covariance

Reactor	BOLNA 4 groups	Adjusted Covariance
ABR Oxide	1439	639
ABR Metal	1460	639

Conclusions

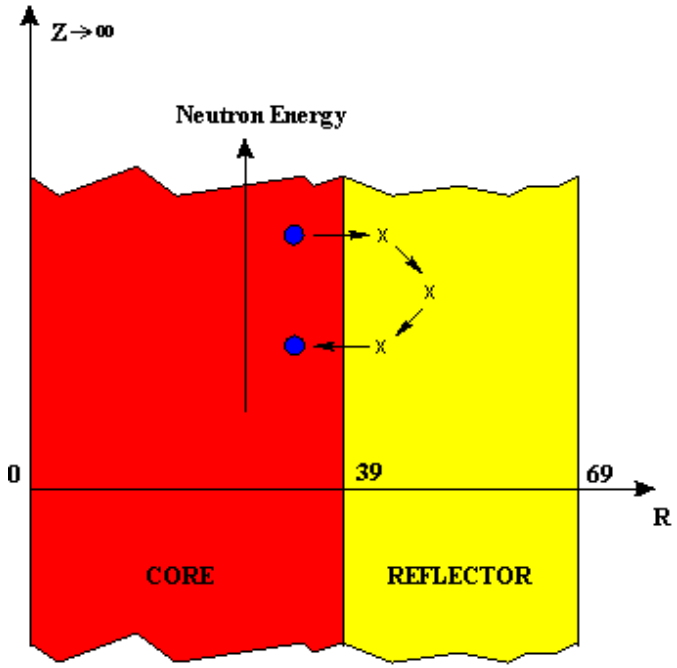
- A global validation approach has been defined that makes use of
  - ➢Sensitivity/uncertainty methods
  - ➢Statistical data adjustments
  - ➢Integral experiment analysis
  - ➢"Representativity" quantification with respect to a reference system
  - ➢Scientifically based cross section covariance data
  - ➢Well defined formal procedures allowing extrapolation beyond the validation domain
- This global approach has been applied to the uncertainty reduction on the criticality of the Advanced Burner Reactor, (both metal and oxide core versions) of the GNEP initiative.
- It is remarkable that already at this stage it has been possible to indicate a few significant improvements of the present ENDF/B-VII data file, that have as consequence:
  - ➢ to reduce by more than a factor of two the present uncertainty, e.g., of the ABR cores keff and
  - ➢ to improve significantly the prediction of TRU nuclide densities during the cycle.

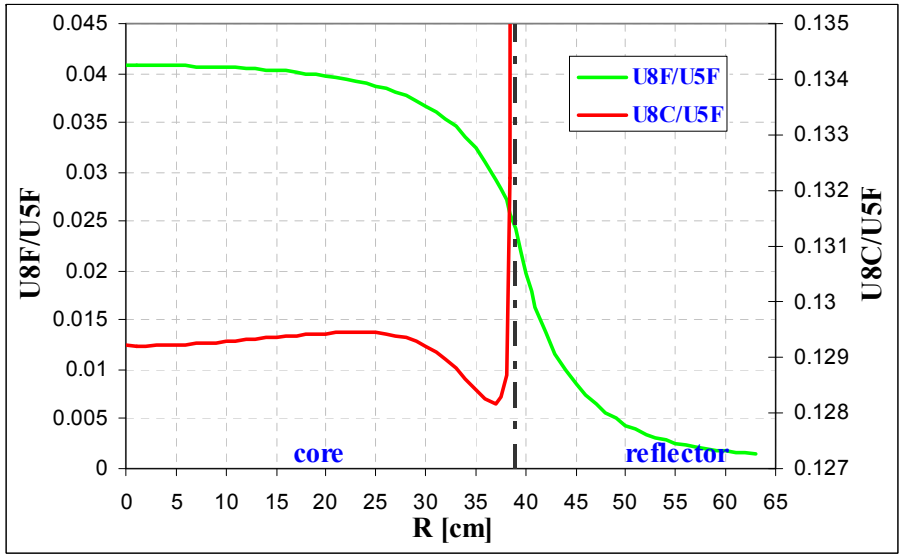
Sensitivity Analysis of Experimental Blanket/Reflector Interface Effects in Fast Reactors for Nuclear Data Improvement

Idaho National Laboratory

Background

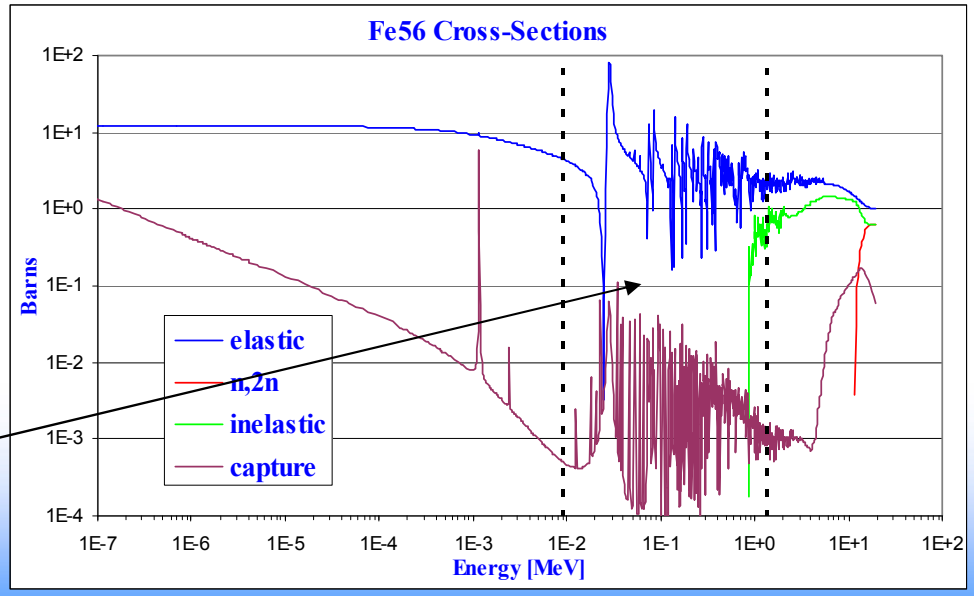
- Interface effects (e.g. core/reflector interfaces) are expected to play an important role in burner type of future fast reactors, in particular for power distribution assessment due to the existence of severe space and energy neutron flux distribution transients at interfaces.
- A detailed multigroup energy treatment to account for spectrum transient at interfaces dramatically improves the agreement with a reference continuous energy Monte Carlo calculation.
- However, when analyzing experimental configurations that were purposely conceived for studying these types of effects (replacing blankets with reflectors) still large discrepancies can be observed for reaction rates gradients in regions close to boundaries between core and reflector.
- An adjustment has been performed using the ENDF/B-VII data and the GNEP 1.0 covariance data in order to better understand the origin of the discrepancy and provide feedbacks to nuclear data evaluators.



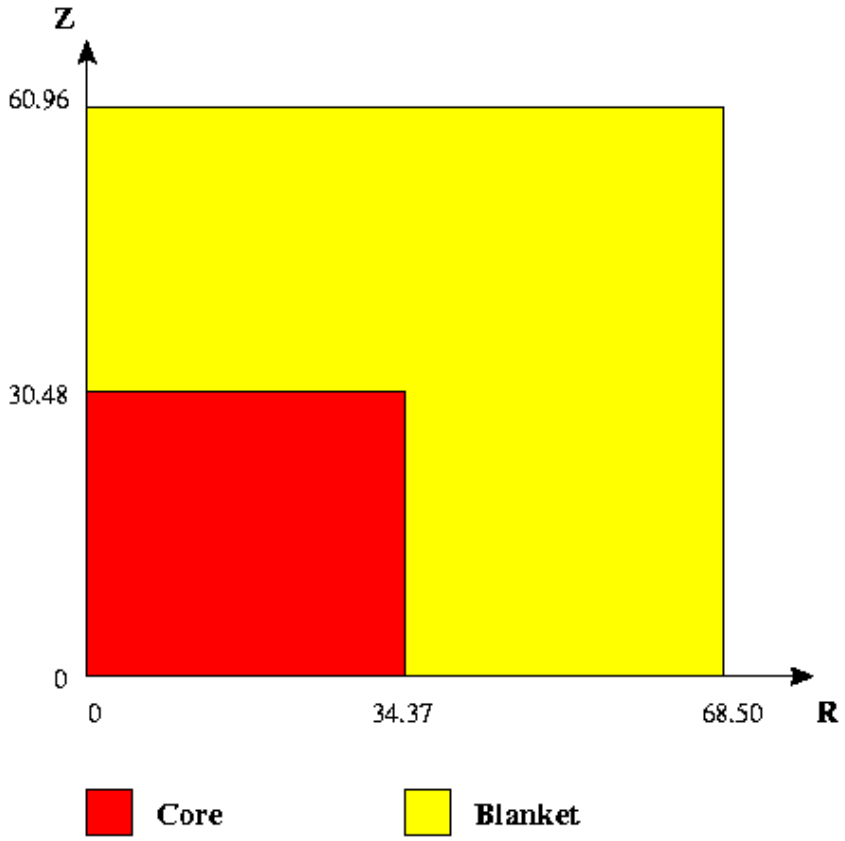


Spectral Indices vary strongly at interface

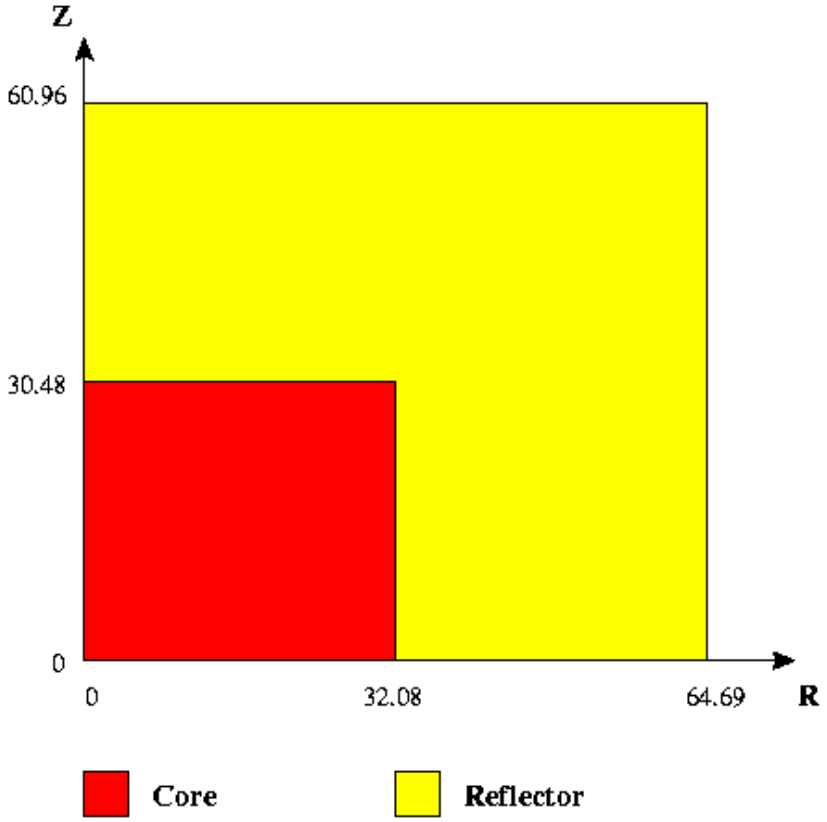
- *There is a severe spectrum transient at the interface that extends from 10 cm inside the core to 10 cm in the reflector, mostly due to the fine structure of iron cross section in the keV energy region.*



ZPR-3 Configurations

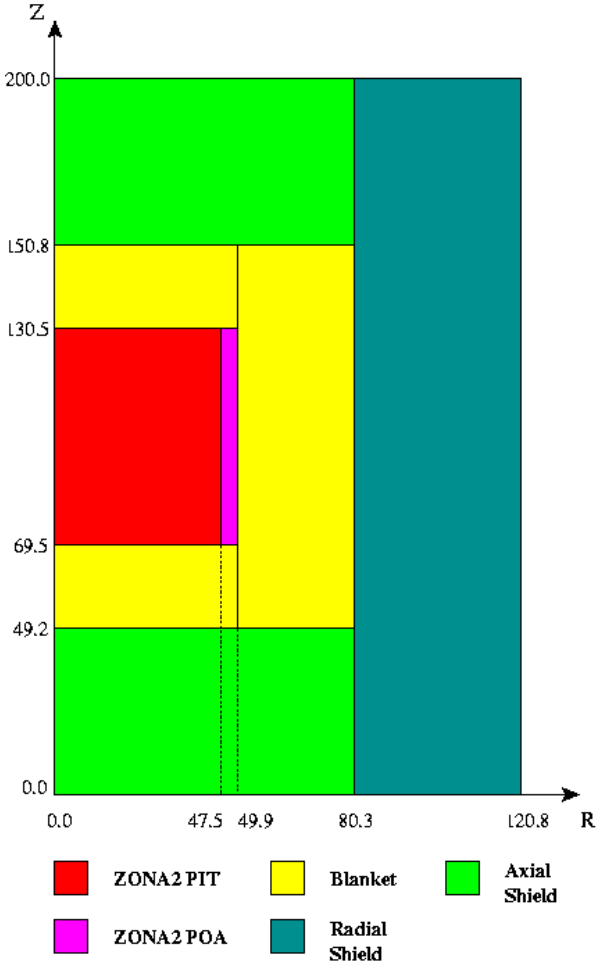


ZPR-3 Assembly 53

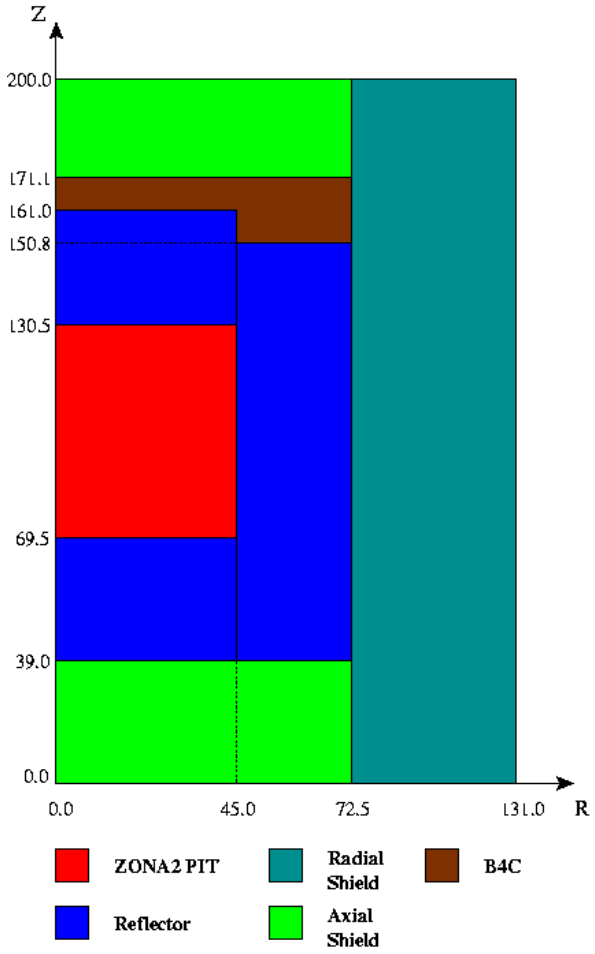


ZPR-3 Assembly 54

CIRANO Configurations

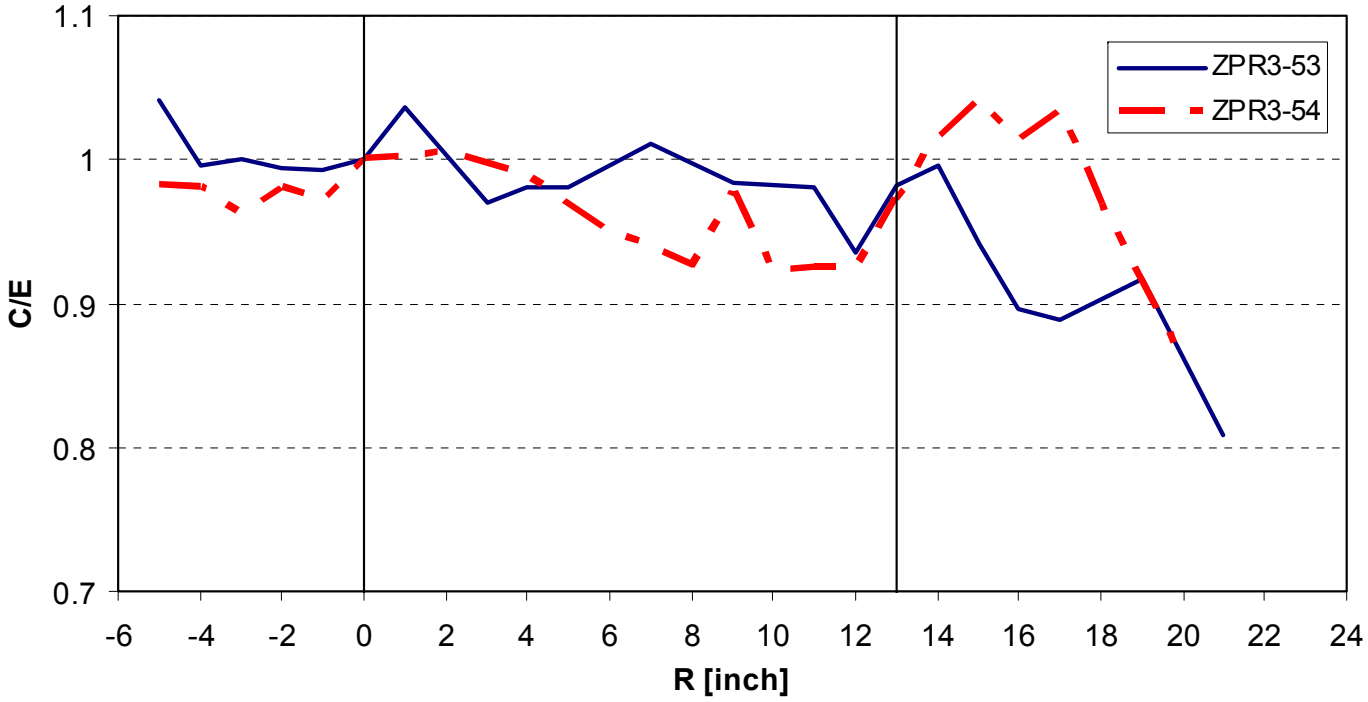


CIRANO 2A



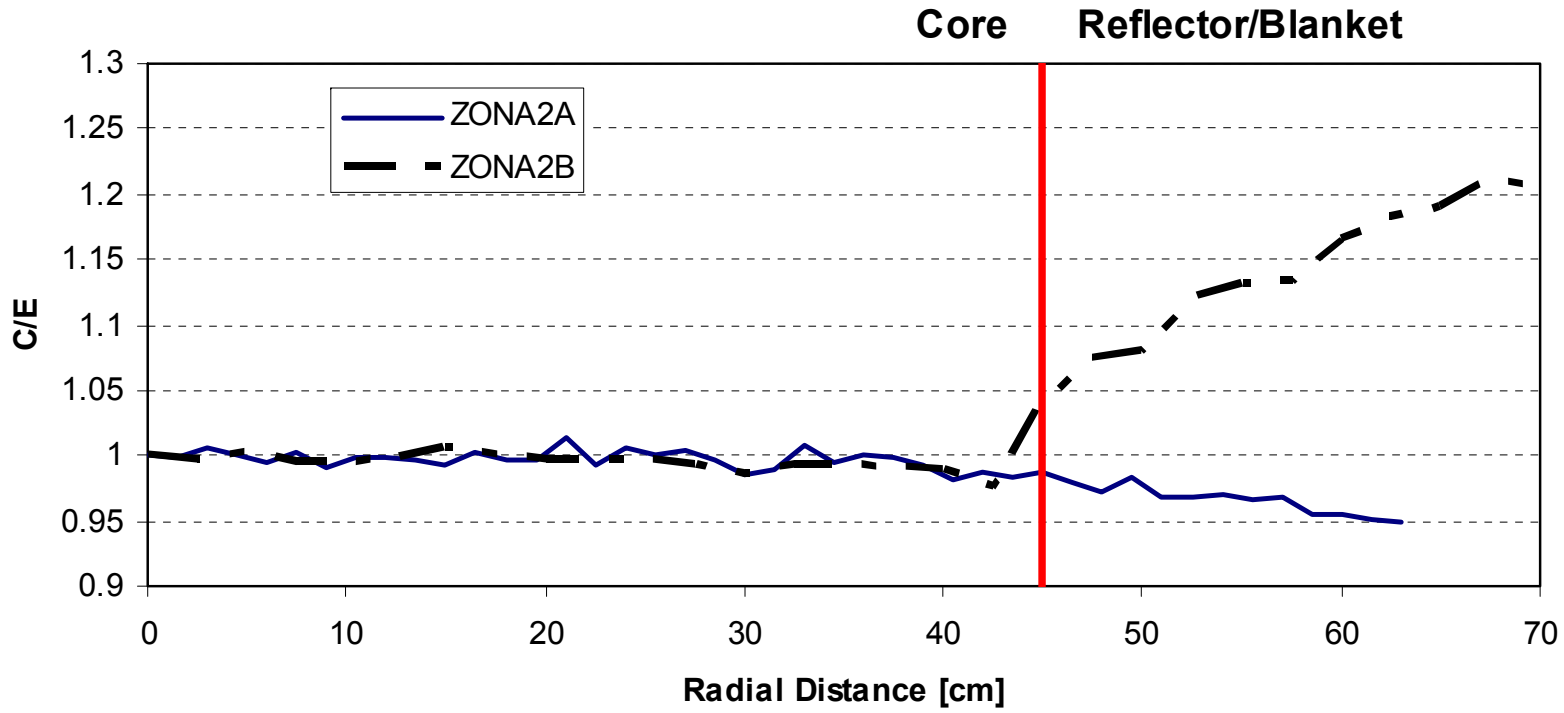
CIRANO 2B

Adjustment for Reflector Effect



C/E in Pu239 fission rate distribution in ZPR3-53 (Blanket) and ZPR3-54 (Reflector)

Adjustment for Reflector Effect



C/E in U235 fission rate distribution in CIRANO 2A (Blanket) and CIRANO 2B (Reflector)

C/E for Different Reaction Rate Slope in ZPR3 and CIRANO

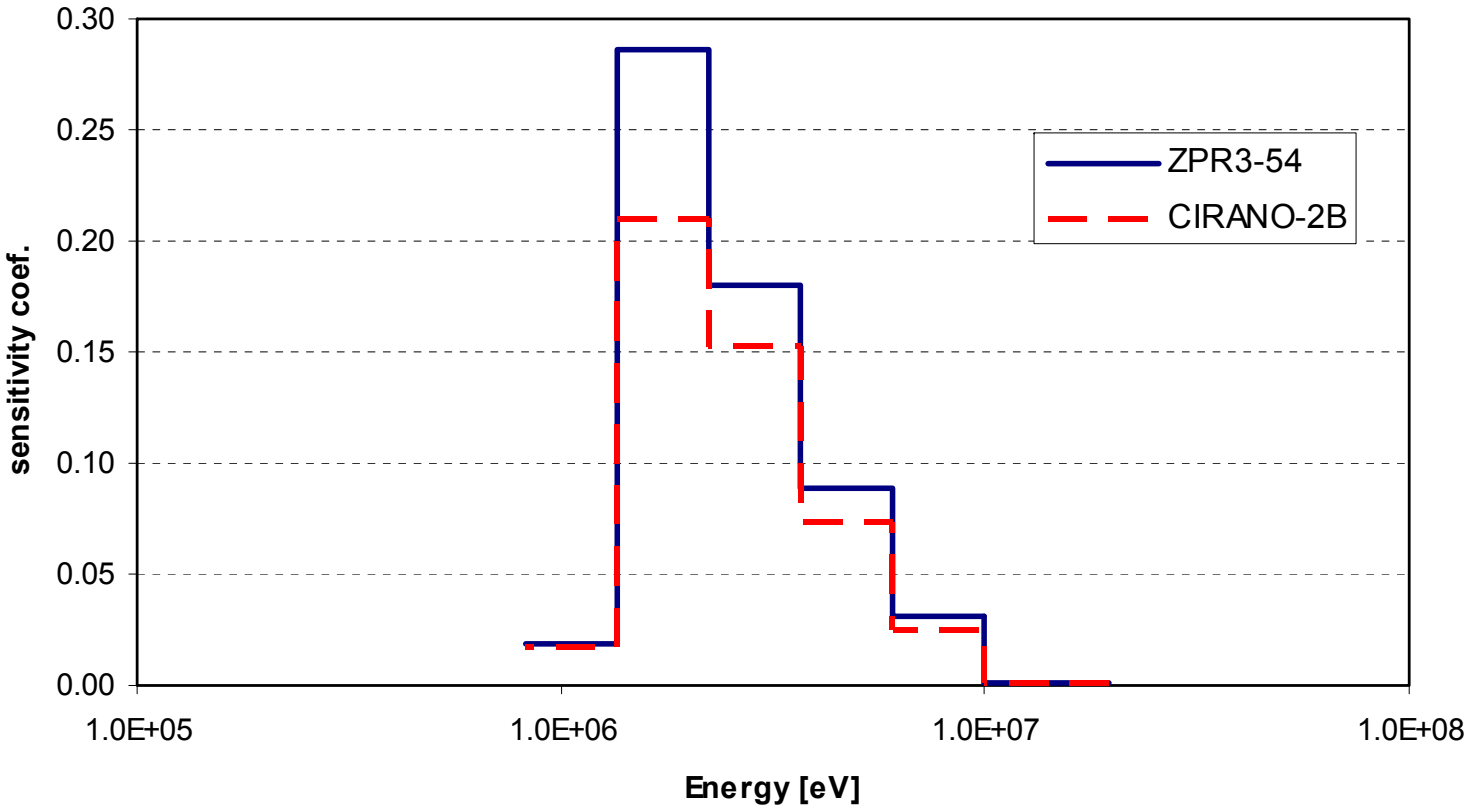
Experiment	A	B	C/E ± σ	Experiment	A	B	C/E ± σ
B10(n,α) Slope ZPR3-54	9”	15”	0.913 ± 0.030	U235 Fission Slope CIRANO 2B	37.5 cm	52.5 cm	0.893 ± 0.030
U235 Fission Slope ZPR3-54	9”	15”	0.989 ± 0.030	Np237 Fission Slope CIRANO 2B	37.5 cm	52.5 cm	1.076 ± 0.030
Pu239 Fission Slope ZPR3-54	9”	15”	0.937 ± 0.030	B10(n,α) Slope ZPR3-53	9”	16”	1.107 ± 0.030
U238 fission Slope ZPR3-54	9”	15”	1.202 ± 0.030	Pu239 Fission Slope ZPR3-53	9”	16”	1.098 ± 0.030
U238 Fission Slope CIRANO 2B	37.5 cm	52.5 cm	1.221 ± 0.030	U238 fission Slope ZPR3-53	9”	16”	1.386 ± 0.030
Pu239 Fission Slope CIRANO 2B	37.5 cm	52.5 cm	0.870 ± 0.0030	U235 Fission Slope CIRANO 2A	40.5 cm	54 cm	1.013 ± 0.030
U235 Fission Slope ZPR3-53	9”	16”	1.233 ± 0.0030	U238 Fission Slope CIRANO 2A	40.5 cm	51 cm	1.176 ± 0.030

A: Radial position at midcore plane of detector in core region for slope calculation.
B: Radial position at midcore plane of detector in blanket or reflector region for slope calculation.

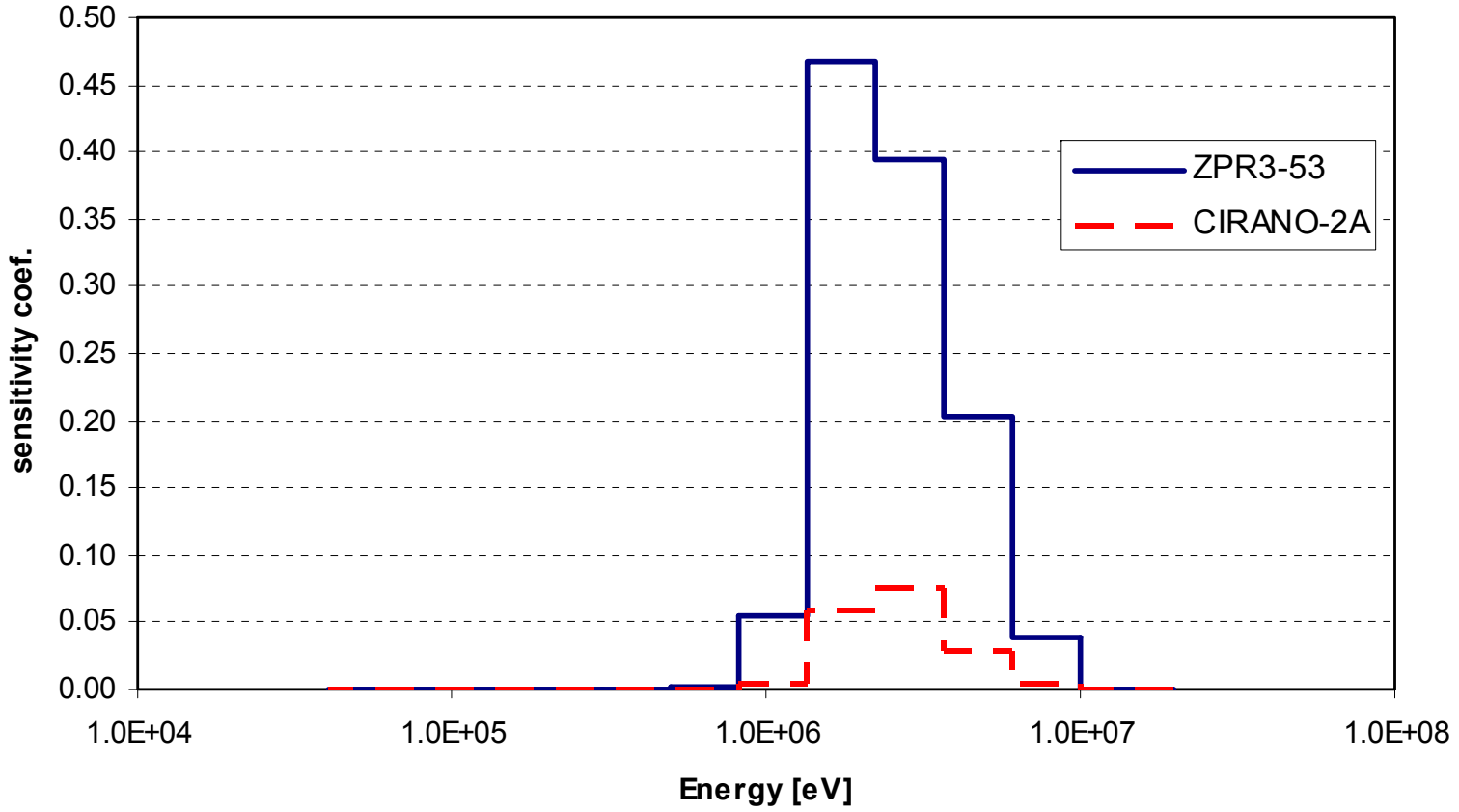
Integrated sensitivity coefficients

		F9 gradient		F8 gradient	
		ZPR3-53 (U blkt)	ZPR-54 (Fe refl)	ZPR3-53 (U blkt)	ZPR-54 (Fe refl)
Fe-56	capture	0.0	0.32	0.0	0.02
	inelastic	0.1	-0.02	0.0	0.61
	elastic	0.1	-0.77	0.04	0.05
U238	capture	0.38	0.0	0.02	0.0
	inelastic	0.19	0.0	1.18	0.02
	fission	-0.09	0.0	-0.11	0.0

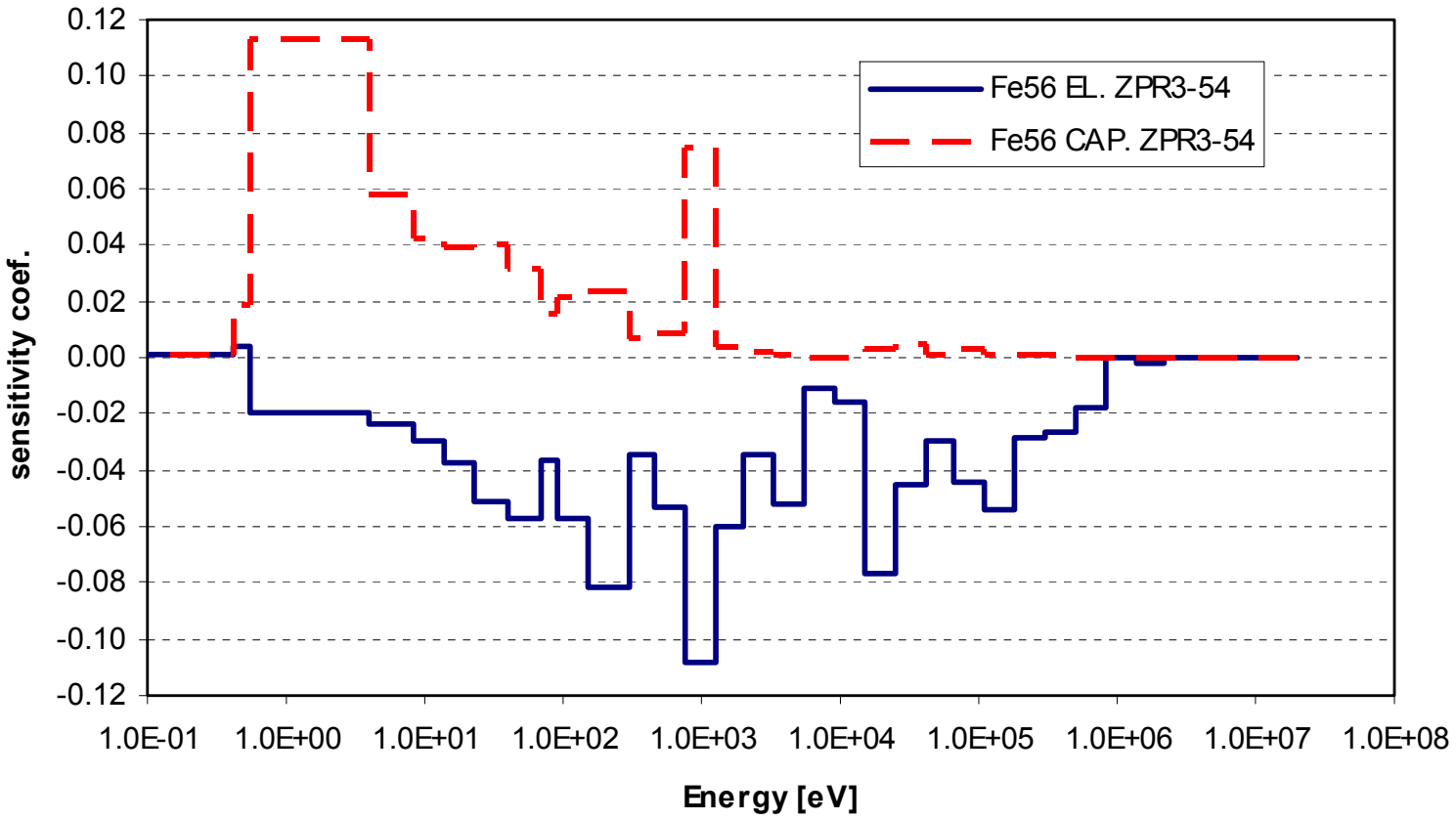
Sensitivity profile of Fe56 inelastic to U238 fission slope in ZPR3-54 and CIRANO-2B



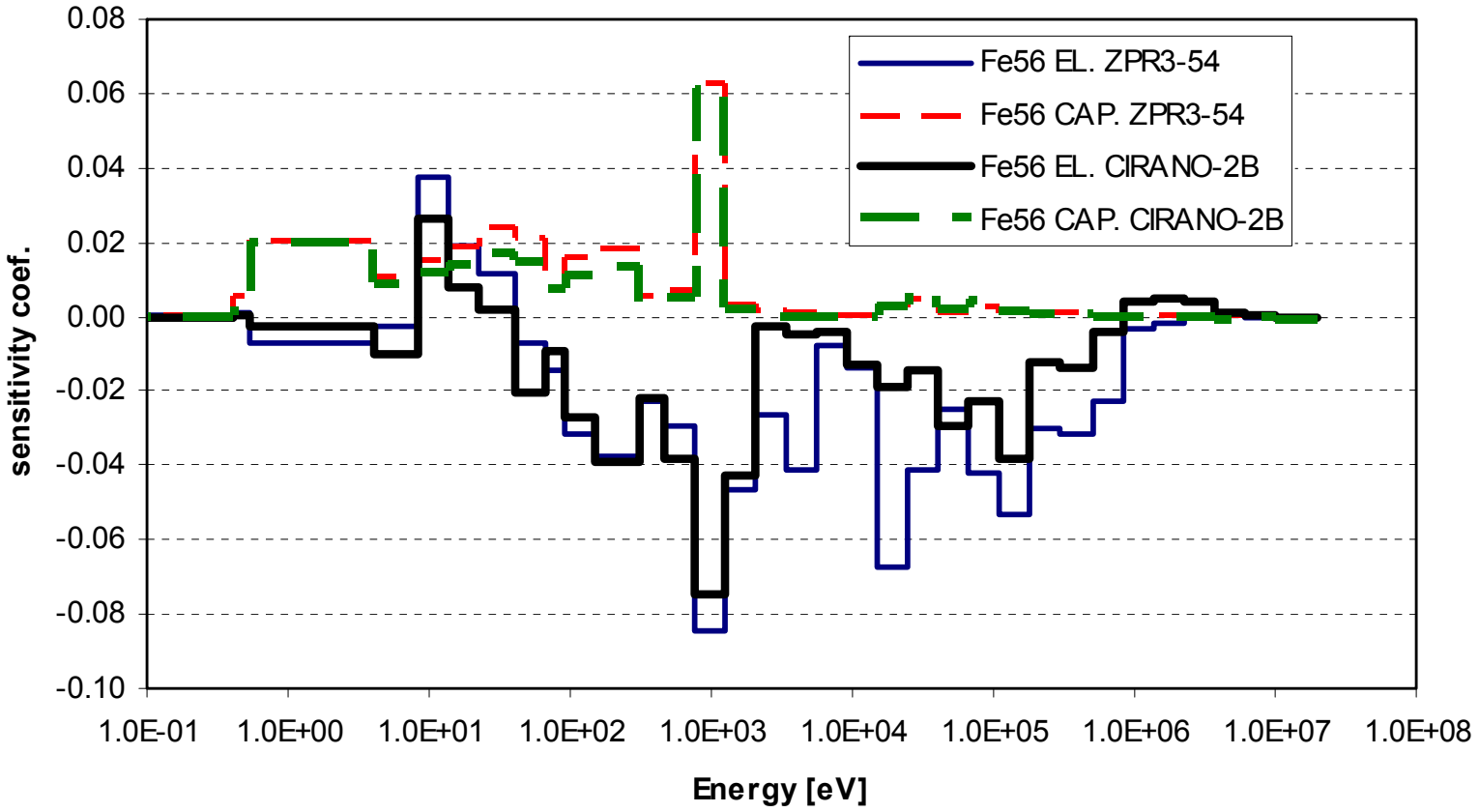
Sensitivity profile of U238 inelastic to U238 fission slope in ZPR3-53 and CIRANO-2A



Sensitivity profile of Fe56 elastic and capture to B10 n,α slope in ZPR3-54



Sensitivity profile of Fe56 elastic and capture for U235 fission slope in ZPR3-54 and CIRANO-2B



Initial and new C/E from adjustment using original covariance data.

Experiment	old C/E ± σ	new C/E ± σ	Experiment	old C/E ± σ	new C/E ± σ
B10(n,α) Slope ZPR3-54	0.913 ± 0.030	0.991 ± 0.019	U235 Fission Slope CIRANO 2B	0.893 ± 0.030	0.963 ± 0.015
U235 Fission Slope ZPR3-54	0.989 ± 0.030	1.045 ± 0.011	Np237 Fission Slope CIRANO 2B	1.076 ± 0.030	1.019 ± 0.009
Pu239 Fission Slope ZPR3-54	0.937 ± 0.030	0.990 ± 0.011	B10(n,α) Slope ZPR3-53	1.107 ± 0.030	1.084 ± 0.002
U238 fission Slope ZPR3-54	1.202 ± 0.030	1.019± 0.020	Pu239 Fission Slope ZPR3-53	1.098 ± 0.030	1.049 ± 0.004
U238 Fission Slope CIRANO 2B	1.221 ± 0.030	1.080 ± 0.015	U238 fission Slope ZPR3-53	1.386 ± 0.030	0.988 ± 0.029
Pu239 Fission Slope CIRANO 2B	0.870 ± 0.0030	0.940 ± 0.014	U235 Fission Slope CIRANO 2A	1.013 ± 0.030	1.005 ± 0.001

Adjusted multigroup cross section using original covariance data.

Param.	Adjus. %	Stand. Deviat. %		Param.	Adjus. %	Stand. Deviat. %		Param.	Adjus. %	Stand. Deviat. %	
		Initial	Adj.			Initial	Adj.			Initial	Adj.
C12 σ^{el} gr. 3	-0.2	2.5	2.5	C12 σ^{inel} gr. 2	-1.6	17.8	17.8	Cr52 σ^{cap} gr. 19	7.1	20.0	19.9
Cr52 σ^{el} gr. 7	-5.8	19.9	19.8	Cr52 σ^{el} gr. 10	-6.8	22.7	22.6	Fe54 σ^{el} gr. 14	-2.4	13.5	13.2
Fe54 σ^{el} gr. 15	-2.8	15.9	15.7	Fe56 σ^{cap} gr. 20	16.6	10.0	5.6	Fe56 σ^{cap} gr. 23	16.1	10.0	6.0
Fe56 σ^{cap} gr. 24	16.1	10.0	6.0	Fe56 σ^{cap} gr. 26	16.2	10.0	6.0	Fe56 σ^{cap} gr. 27	16.2	10.0	6.0
Fe56 σ^{cap} gr. 28	16.2	10.0	6.0	Fe56 σ^{cap} gr. 29	16.2	10.0	5.9	Fe56 σ^{cap} gr. 30	14.1	8.6	5.1
Fe56 σ^{cap} gr. 31	6.9	5.0	3.6	Fe56 σ^{el} gr. 4	8.7	3.8	3.1	Fe56 σ^{el} gr. 5	13.2	5.0	3.7
Fe56 σ^{el} gr. 14	-1.3	3.0	2.9	Fe56 σ^{el} gr. 20	-1.4	3.2	3.1	Fe56 σ^{el} gr. 23	-1.3	3.0	2.9
Fe56 σ^{inel} gr. 3	3.8	3.3	3.2	Fe56 σ^{inel} gr.4	-16.7	5.5	3.4	Fe56 σ^{inel} gr.5	-43.5	12.7	6.4
Fe56 σ^{inel} gr. 6	-49.7	15.0	8.2	Ni58 σ^{el} gr. 13	-52.3	24.0	19.2	Ni58 σ^{el} gr. 14	-60.3	27.6	22.1
Ni60 σ^{el} gr. 14	-8.4	30.2	30.1	U238 σ^{el} gr. 5	21.7	18.8	7.0	U238 σ^{inel} gr. 2	-22.5	30.3	26.9
U238 σ^{inel} gr. 3	-26.1	20.1	7.2	U238 σ^{inel} gr. 4	-25.6	19.4	4.0	U238 σ^{inel} gr. 5	-24.2	20.6	7.1
U238 σ^{inel} gr. 6	-20.0	16.9	7.1								

Major contributors to new C/E after adjustment using original covariance data

Experiment	Total Relative Change	Major Contributors					
		Parameter	Contribution	Parameter	Contribution	Parameter	Contribution
B10(n,α) Slope ZPR3-54	8.6	Fe56 σ^{inel} gr. 6	1.3	Fe56 σ^{cap} gr. 20	1.2	Ni58 σ^{el} gr. 14	0.9
U235 Fission Slope ZPR3-54	5.6	Fe56 σ^{inel} gr. 6	1.2	Fe56 σ^{cap} gr. 20	1.0	Ni58 σ^{el} gr. 14	0.8
Pu239 Fission Slope ZPR3-54	5.6	Fe56 σ^{inel} gr. 6	1.1	Fe56 σ^{cap} gr. 20	1.0	Ni58 σ^{el} gr. 14	0.8
U238 fission Slope ZPR3-54	-15.2	Fe56 σ^{inel} gr. 5	-12.4	Fe56 σ^{inel} gr. 5	-3.0	Fe56 σ^{inel} gr. 6	-1.0
U238 Fission Slope CIRANO 2B	-11.6	Fe56 σ^{inel} gr. 5	-9.1	Fe56 σ^{inel} gr. 4	-2.5	Fe56 σ^{inel} gr. 6	-0.9
Pu239 Fission Slope CIRANO 2B	8.1	Ni58 σ^{el} gr. 14	3.3	Fe56 σ^{cap} gr. 20	1.0	Ni58 σ^{el} gr. 13	0.7
U235 Fission Slope CIRANO 2B	7.9	Ni58 σ^{el} gr. 14	3.3	Fe56 σ^{cap} gr. 20	1.0	Ni58 σ^{el} gr. 13	0.8
Np237 Fission Slope CIRANO 2B	-5.3	Fe56 σ^{inel} gr. 6	-3.6	Fe56 σ^{inel} gr. 5	-2.0	Fe56 σ^{inel} gr. 4	-0.5
B10(n,α) Slope ZPR3-53	-2.1	U238 σ^{inel} gr. 5	-0.8	U238 σ^{inel} gr. 4	-0.8	U238 σ^{inel} gr. 3	-0.4
Pu239 Fission Slope ZPR3-53	-4.4	U238 σ^{inel} gr. 5	-1.6	U238 σ^{inel} gr. 4	-1.4	U238 σ^{inel} gr. 3	-0.8
U238 fission Slope ZPR3-53	-28.7	U238 σ^{inel} gr. 5	-11.3	U238 σ^{inel} gr. 4	-10.1	U238 σ^{inel} gr. 3	-5.3
U235 Fission Slope CIRANO 2A	-0.8	U238 σ^{inel} gr. 4	-0.3	U238 σ^{inel} gr. 5	-0.3	U238 σ^{inel} gr. 3	-0.1

Modified Covariance Data

- Because of a ~30% uncertainty on Ni58 elastic in the resonance region, there is a very large adjustment (close to 70%) that results in a significant contribution to the discrepancy reduction for low energy reaction rates gradients in the case of CIRANO 2B. The quite large uncertainty associated to the Ni58 is probably fairly artificial and due to a lack of modeling of the large resonances for structural materials.
- Moreover, the covariance data for Fe56 were derived directly from JENDL 3.3 data, and that are probably slightly optimistic for the elastic (~3% standard deviation) and inelastic (~ 5% standard deviation) cross sections.
- In order to avoid this rather artificial type of effects, we have tentatively modified a few uncertainty data (namely the standard deviations in all groups of: Ni58 elastic (set to 8%), Fe56 elastic (set to 8%), and Fe56 inelastic (set to 15%)). With these modified covariance data, a new adjustment was attempted .

Initial and new C/E from adjustment using modified covariance data.

Experiment	old C/E ± σ	new C/E ± σ	Experiment	old C/E ± σ	new C/E ± σ
B10(n,α) Slope ZPR3-54	0.913 ± 0.030	1.018 ± 0.022	U235 Fission Slope CIRANO 2B	0.893 ± 0.030	0.939 ± 0.010
U235 Fission Slope ZPR3-54	0.989 ± 0.030	1.054 ± 0.012	Np237 Fission Slope CIRANO 2B	1.076 ± 0.030	1.021 ± 0.009
Pu239 Fission Slope ZPR3-54	0.937 ± 0.030	1.002 ± 0.012	B10(n,α) Slope ZPR3-53	1.107 ± 0.030	1.084 ± 0.002
U238 fission Slope ZPR3-54	1.202 ± 0.030	0.998± 0.022	Pu239 Fission Slope ZPR3-53	1.098 ± 0.030	1.050 ± 0.004
U238 Fission Slope CIRANO 2B	1.221 ± 0.030	1.059 ± 0.016	U238 fission Slope ZPR3-53	1.386 ± 0.030	0.988 ± 0.029
Pu239 Fission Slope CIRANO 2B	0.870 ± 0.0030	0.920 ± 0.010	U235 Fission Slope CIRANO 2A	1.013 ± 0.030	1.005 ± 0.001

Adjusted multigroup cross section using modified covariance data.

Param.	Adjus. %	Stand. Deviat. %		Param.	Adjus. %	Stand. Deviat. %		Param.	Adjus. %	Stand. Deviat. %	
		Initial	Adj.			Initial	Adj.			Initial	Adj.
C12 σ^{el} gr. 3	-0.2	2.5	2.5	C12 σ^{inel} gr. 2	-1.4	17.8	17.8	Cr52 σ^{cap} gr. 19	10.2	20.0	19.8
Cr52 σ^{el} gr. 7	-8.0	19.9	19.9	Cr52 σ^{el} gr.10	-9.4	22.7	22.6	Fe54 σ^{el} gr.14	0.1	13.5	13.4
Fe54 σ^{el} gr.14	0.1	15.9	15.8	Fe56 σ^{cap} gr. 20	10.4	10.0	9.5	Fe56 σ^{cap} gr. 23	8.8	10.0	10.0
Fe56 σ^{cap} gr. 24	8.9	10.0	10.0	Fe56 σ^{cap} gr. 26	8.9	10.0	10.0	Fe56 σ^{cap} gr. 27	9.0	10.0	9.9
Fe56 σ^{cap} gr. 28	9.1	10.0	9.9	Fe56 σ^{cap} gr. 29	9.1	10.0	9.9	Fe56 σ^{cap} gr. 30	8.0	8.6	8.6
Fe56 σ^{cap} gr. 31	3.8	5.0	4.9	Fe56 σ^{el} gr.4	14.4	8.0	8.0	Fe56 σ^{el} gr.5	15.0	8.0	8.0
Fe56 σ^{el} gr.7	-2.2	8.0	7.9	Fe56 σ^{el} gr.8	-2.9	8.0	8.0	Fe56 σ^{el} gr.9	-2.9	8.0	8.0
Fe56 σ^{el} gr.10	-3.1	8.0	7.9	Fe56 σ^{el} gr.11	-3.5	8.0	7.9	Fe56 σ^{el} gr.12	-6.5	8.0	8.0
Fe56 σ^{el} gr.13	-7.9	8.0	7.9	Fe56 σ^{el} gr.14	-7.9	8.0	7.8	Fe56 σ^{el} gr.17	-7.9	8.0	7.9
Fe56 σ^{el} gr.18	-7.9	8.0	8.0	Fe56 σ^{el} gr.19	-7.9	8.0	7.9	Fe56 σ^{el} gr.20	-7.9	8.0	7.6
Fe56 σ^{el} gr.21	-7.9	8.0	7.9	Fe56 σ^{el} gr.22	-7.9	8.0	8.0	Fe56 σ^{el} gr.23	-7.9	8.0	7.9
Fe56 σ^{el} gr.24	-7.9	8.0	7.9	Fe56 σ^{el} gr.25	-7.9	8.0	8.0	Fe56 σ^{el} gr.26	-7.9	8.0	7.9
Fe56 σ^{el} gr.27	-7.9	8.0	7.9	Fe56 σ^{el} gr.28	-7.9	8.0	8.0	Fe56 σ^{el} gr.29	-7.9	8.0	7.9
Fe56 σ^{el} gr.31	-7.8	8.0	8.0	Fe56 σ^{inel} gr. 2	-6.4	15.0	15.0	Fe56 σ^{inel} gr. 3	4.7	15.0	14.6
Fe56 σ^{inel} gr. 4	-38.5	15.0	13.1	Fe56 σ^{inel} gr. 5	-37.7	15.0	10.3	Fe56 σ^{inel} gr. 6	-35.8	15.0	14.0
Ni58 σ^{el} gr.14	-5.7	8.0	7.9	Ni60 σ^{el} gr.14	-12.6	30.2	30.1	U238 σ^{el} gr. 5	22.1	18.8	18.7
U238 σ^{inel} gr. 2	-21.3	30.3	30.2	U238 σ^{inel} gr. 3	-25.7	20.1	19.2	U238 σ^{inel} gr. 4	-25.4	19.4	15.9
U238 σ^{inel} gr. 5	-24.5	20.6	14.2	U238 σ^{inel} gr. 6	-20.1	16.9	16.8				

Major contributors to new C/E after adjustment using modified covariance data

Experiment	Total Relative Change	Major Contributors					
		Parameter	Contribution	Parameter	Contribution	Parameter	Contribution
B10(n,α) Slope ZPR3-54	11.5	Fe56 σ^{inel} gr. 6	0.9	Fe56 σ^{el} gr. 20	0.9	Fe56 σ^{cap} gr. 20	0.8
U235 Fission Slope ZPR3-54	6.5	Fe56 σ^{inel} gr. 6	0.9	Fe56 σ^{el} gr. 20	0.7	Fe56 σ^{cap} gr. 20	0.6
Pu239 Fission Slope ZPR3-54	6.9	Fe56 σ^{inel} gr. 6	0.8	Fe56 σ^{el} gr. 20	0.6	Fe56 σ^{cap} gr. 20	0.6
U238 fission Slope ZPR3-54	-17.0	Fe56 σ^{inel} gr. 5	-10.8	Fe56 σ^{inel} gr. 4	-7.0	Fe56 σ^{el} gr. 5	0.8
U238 Fission Slope CIRANO 2B	-13.2	Fe56 σ^{inel} gr. 5	-7.9	Fe56 σ^{inel} gr. 4	-5.9	Fe56 σ^{el} gr. 5	0.9
Pu239 Fission Slope CIRANO 2B	5.7	Fe56 σ^{cap} gr. 20	0.6	Fe56 σ^{el} gr. 20	0.6	Fe56 σ^{inel} gr. 6	0.5
U235 Fission Slope CIRANO 2B	5.1	Fe56 σ^{cap} gr. 20	0.6	Fe56 σ^{el} gr. 20	0.6	Fe56 σ^{inel} gr. 6	0.4
Np237 Fission Slope CIRANO 2B	-5.1	Fe56 σ^{inel} gr. 6	-2.6	Fe56 σ^{inel} gr. 5	-1.7	Fe56 σ^{inel} gr. 4	-1.2
B10(n,α) Slope ZPR3-53	-2.0	U238 σ^{inel} gr. 5	-0.9	U238 σ^{inel} gr. 4	-0.8	U238 σ^{inel} gr. 3	-0.4
Pu239 Fission Slope ZPR3-53	-4.4	U238 σ^{inel} gr. 5	-1.6	U238 σ^{inel} gr. 4	-1.4	U238 σ^{inel} gr. 3	-0.8
U238 fission Slope ZPR3-53	-28.7	U238 σ^{inel} gr. 5	-11.4	U238 σ^{inel} gr. 4	-10.0	U238 σ^{inel} gr. 3	-5.2
U235 Fission Slope CIRANO 2A	-0.8	U238 σ^{inel} gr. 4	-0.3	U238 σ^{inel} gr. 5	-0.3	Fe56 σ^{inel} gr. 4	-0.1

Summary

- A global statistical adjustment has shown that using experimental reaction rate distributions, measured in different cores with different reflectors (ZPR3 53 and 54, and CIRANO assemblies), one can put in evidence nuclear data that should be improved in order to obtain better calculation-to-experiment agreement.
- The results obtained in the study offer some clear indication towards the improvement needed in the inelastic cross sections both of Fe-56 and of U-238. Another indication is related to the need of improvement of the Fe-56 capture cross section in the 1 keV energy range.
- Finally, the study has also underlined that in order to consolidate the present results and trend indications for data improvement, it is needed to further strengthen the covariance data that are used in the uncertainty analysis and subsequent statistical adjustment. In particular a new reevalaution of covariance matrices for Fe-56, Ni-58, and Cr-52 will be needed.

Consistent Data Assimilation

- This methodology is the first attempt to build up a link between the wealth of precise integral experiments and basic theory of nuclear reactions.
- Essential ingredients of such a procedure are covariances for model parameters and sensitivity matrices. The latter provide direct link between reaction theory and integral experiments.
- The result is a consistent data assimilation performed directly on the basic nuclear physics parameters that are being used in a variety of nuclear reaction mechanisms.
- Resulting improvement in their performance will consequently reduce related uncertainties when employed in reactor calculations.
- By using integral reactor physics experiments (meter scale), information is propagated back to the nuclear physics level (femtometers) covering a range of more than 13 orders of magnitude

Consistent (Multiscale) Data Assimilation
Going from Integral to Elemental Level

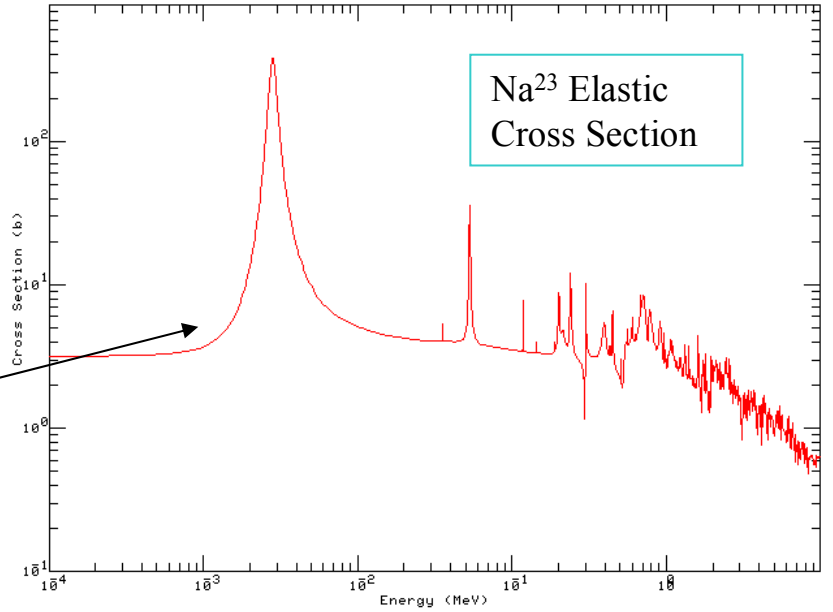
- A consistent data assimilation algorithm using integral experiment information and performed directly on the basic nuclear parameters present in the nuclear data files (ENDF/B-VII) will allow to reduce their related uncertainty when employed in reactor calculations.
- It represents a genuine attempt to use a first principle approach as it deals directly with the information coming from nuclear modeling calculation and evaluation.
- Sensitivity coefficients of basic nuclear parameters are calculating by folding sensitivities of nuclear parameters to multigroup cross sections (EMPIRE) with multigroup sensitivities of integral experiments (ERANOS)
- After the data assimilation is carried out, new data files are produced that can be used by cross section processing codes in any energy group structure that the reactor nuclear engineer deems to be useful.

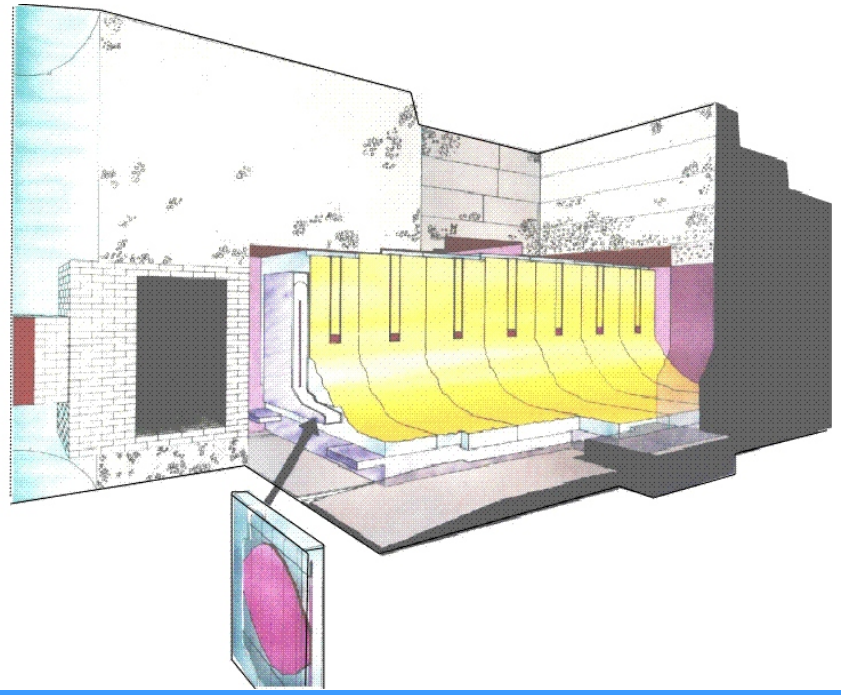
Consistent Data Assimilation

From Meters to Femtometers

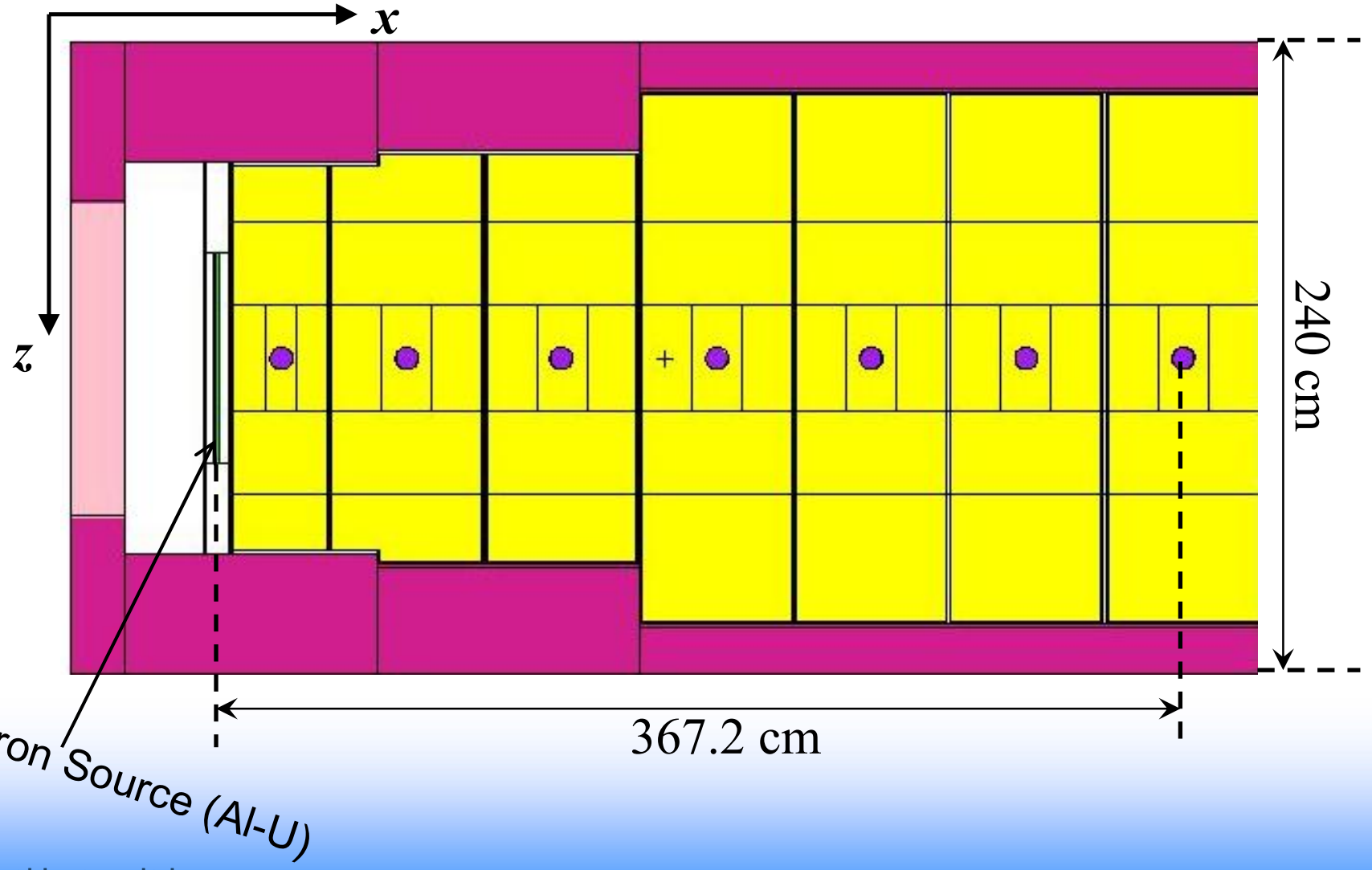
A first attempt of consistent data assimilation is under way using neutron propagation experiments.

The parameters characterizing 10 major resonances of the sodium elastic cross section and 16 basic nuclear parameters (optical model) will be tuned using the C/E of reaction rates measured in the EURACOS experiment (propagation in 3 meters of sodium).





EURACOS Na-Experimental Facility



Nuclear Parameters Considered

- For the Na^{23} isotope a total number of 39 nuclear parameters has been considered.

- Bound level and 5 resonances: 18 parameters. For each resonance the peak energy and neutron and radiative width has been taken into account.
- Optical model: 10 parameters
- Statistical Hauser-Feshbach model: 7 parameters
- Preequilibrium exciton model: 4 parameters

Parameter Rel. Stand. Dev.

Parameter	Std. Dev.	Parameter	Std. Dev.	Parameter	Std. Dev.
En_b	0.010	Gn_b	0.018	Gg_b	0.100
En_2k	0.001	Gn_2k	0.019	Gg_2k	0.021
En_52k	0.001	Gn_52k	0.032	Gg_52k	0.101
En_201k	0.001	Gn_201k	0.076	Gg_201k	0.325
En_239k	0.001	Gn_239k	0.018	Gg_239k	0.171
En_299k	0.001	Gn_299k	0.052	Gg_299k	0.167
UOMPRS	0.015	UOMPRW	0.034	UOMPVV	0.006
UOMPWS	0.014	UOMPWV	0.035	UOMPAS	0.015
UOMPAV	0.012	UOMPVV0002	0.034	TOTRED	0.014
FUSRED	0.032	ATILNO0000	0.100	ATILNO0001	0.015
ATILNO0002	0.050	ATILNO0100	0.015	TUNE0000	0.147
TUNE0002	0.090	TUNE0003	0.488	PCROSS	0.103
GTILNO0001	0.083	GTILNO0100	0.099	TUNEPE	0.100

Sensitivity to Nuclear Parameters

- The sensitivities of integral experiments to fundamental parameters p_k are defined as:

$$\frac{\Delta R}{\Delta p_k} = \sum_j \frac{\Delta R}{\Delta \sigma_j} \times \frac{\Delta \sigma_j}{\Delta p_k}$$

- R is an integral reactor physics parameter (e. g. K_{eff}, reaction rates, reactivity coefficient, etc.), and σ_j a multigroup cross section.
- In order to compute σ_j one use EMPIRE with an appropriate set of parameters p_k to generate an ENDF/B file for that specific isotope. and finally NJOY for producing multi-group cross sections.
- The multigroup sensitivity coefficients $\frac{\Delta R}{\Delta \sigma_j}$, are provided by

  reactor physics calculations, using the Generalized Perturbation Theory and the ERANOS code system.

Sensit. and Unc. S32 (n,p)

Paramet.	S32 Slope: 2/1		S32 Slope: 7/1	
	Sensit.	Unc.	Sensit.	Unc.
UOMPRS	-0.008	0.001	-0.080	0.009
UOMPRW	0.000	0.000	-0.003	0.000
UOMPVV	-0.018	0.000	-0.144	0.002
UOMPWS	-0.002	0.000	-0.017	0.001
UOMPWV	0.000	0.000	-0.001	0.000
UOMPAS	-0.007	0.000	-0.067	0.003
UOMPAV	-0.015	0.001	-0.130	0.012
UOMPVV0002	0.000	0.000	0.000	0.000
TOTRED	-1.544	0.023	-14.104	0.210
FUSRED	-0.841	0.028	-7.267	0.241
ATILNO0000	0.000	0.000	0.000	0.000
ATILNO0001	0.000	0.000	0.012	0.000
ATILNO0002	0.000	0.000	0.000	0.000
ATILNO0100	0.002	0.000	0.049	0.001
TUNE0000	0.000	0.000	0.002	0.000
TUNE0002	0.001	0.000	0.022	0.004
TUNE0003	0.000	0.000	0.004	0.003
PCROSS	0.005	0.000	0.056	0.005
GTILNO0001	0.012	0.001	0.143	0.011
GTILNO0100	-0.016	0.002	-0.194	0.019
TUNEPE	0.000	0.000	0.000	0.000
TOTAL	-2.432	0.036	-21.721	0.320

Sensit. and Unc. Au197 (n,γ)

Paramet.	Au197 Slope: 2/1		Au197 Slope: 7/1	
	Sensit.	Unc.	Sensit.	Unc.
En_b	0.326	0.003	-2.288	0.022
Gn_b	0.226	0.004	-1.583	0.028
Gg_b	-0.002	0.000	-0.008	0.001
En_2k	-0.301	0.000	2.952	0.004
Gn_2k	0.210	0.004	-1.646	0.031
Gg_2k	-0.075	0.002	-0.232	0.005
En_52k	-0.010	0.000	0.071	0.000
Gn_52k	0.010	0.000	-0.066	0.002
Gg_52k	0.000	0.000	-0.001	0.000
En_201k	-0.016	0.000	0.167	0.000
Gn_201k	0.001	0.000	-0.017	0.001
Gg_201k	0.000	0.000	0.000	0.000
En_239k	-0.001	0.000	0.014	0.000
Gn_239k	0.002	0.000	-0.035	0.001
Gg_239k	0.000	0.000	0.000	0.000
En_299k	0.000	0.000	0.007	0.000
Gn_299k	0.000	0.000	-0.001	0.000
Gg_299k	0.000	0.000	0.000	0.000
TOTRED	-0.049	0.001	-0.628	0.009
FUSRED	0.012	0.000	-0.274	0.009
TOTAL	0.334	0.007	-3.590	0.050

C/E using EMPIRE Na23

Slope	S32(n,p)				Au197(n,γ)			
	Exp.	C/E ENDF7	C/E EMP.	Unc.	Exp.	C/E ENDF7	C/E EMP.	Unc.
2/1	6.09E-02	0.890	0.877	10.11%	8.56E-01	0.988	0.991	9.03%
7/1	6.15E-08	1.433	0.996	16.21%	1.46E-03	1.182	1.103	11.51%

New C/E after data assimilation

Slope	S32(n,p)				Au197(n,γ)			
	Exp.	Old C/E	New C/E.	Unc.	Exp.	Old C/E	New C/E..	Unc.
2/1	6.09E-02	0.877	0.880	10.11%	8.56E-01	0.991	0.993	9.03%
7/1	6.15E-08	0.996	1.026	16.21%	1.46E-03	1.103	1.089	11.51%

Relative change of nuclear parameters after data assimilation

Parameter	Change	Parameter	Change	Parameter	Change
En_b	0.001	Gn_b	0.003	Gg_b	0.000
En_2k	0.000	Gn_2k	0.004	Gg_2k	0.001
En_52k	0.000	Gn_52k	0.000	Gg_52k	0.000
En_201k	0.000	Gn_201k	0.001	Gg_201k	0.000
En_239k	0.000	Gn_239k	0.000	Gg_239k	0.000
En_299k	0.000	Gn_299k	0.000	Gg_299k	0.000
UOMPRS	0.000	UOMPRW	0.000	UOMPVV	0.000
UOMPWS	0.000	UOMPWV	0.000	UOMPAS	0.000
UOMPAV	0.000	UOMPVV0002	0.000	TOTRED	-0.001
FUSRED	-0.003	ATILNO0000	0.000	ATILNO0001	0.000
ATILNO0002	0.000	ATILNO0100	0.000	TUNE0000	0.000
TUNE0002	0.000	TUNE0003	0.001	PCROSS	0.000
GTILNO0001	0.000	GTILNO0100	0.000	TUNEPE	0.000

Conclusions

- Sensitivity and uncertainty methodologies have been developed in the past for nuclear reactor applications and applied to a series of different problems: uncertainty analysis, optimization, target accuracy requirements, adjustment, and representativity.
- Most of the methodologies have been developed in the reactor physics field. Extension to other fields is needed: thermal-hydraulics, structural mechanics, fuel behavior, chemical separation, etc.
- A promising consistent approach can allow to link basic nuclear parameters to integral experimentally measured quantities.
- Credibility of uncertainty evaluation, target accuracy requirements, and adjustment of input parameters is mainly linked to the quality of the covariance information attached to those data.