Perspectives on basic nuclear cross section measurements for nuclear reactor applications

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What is evaluated nuclear data ?

Basic source of nuclear data

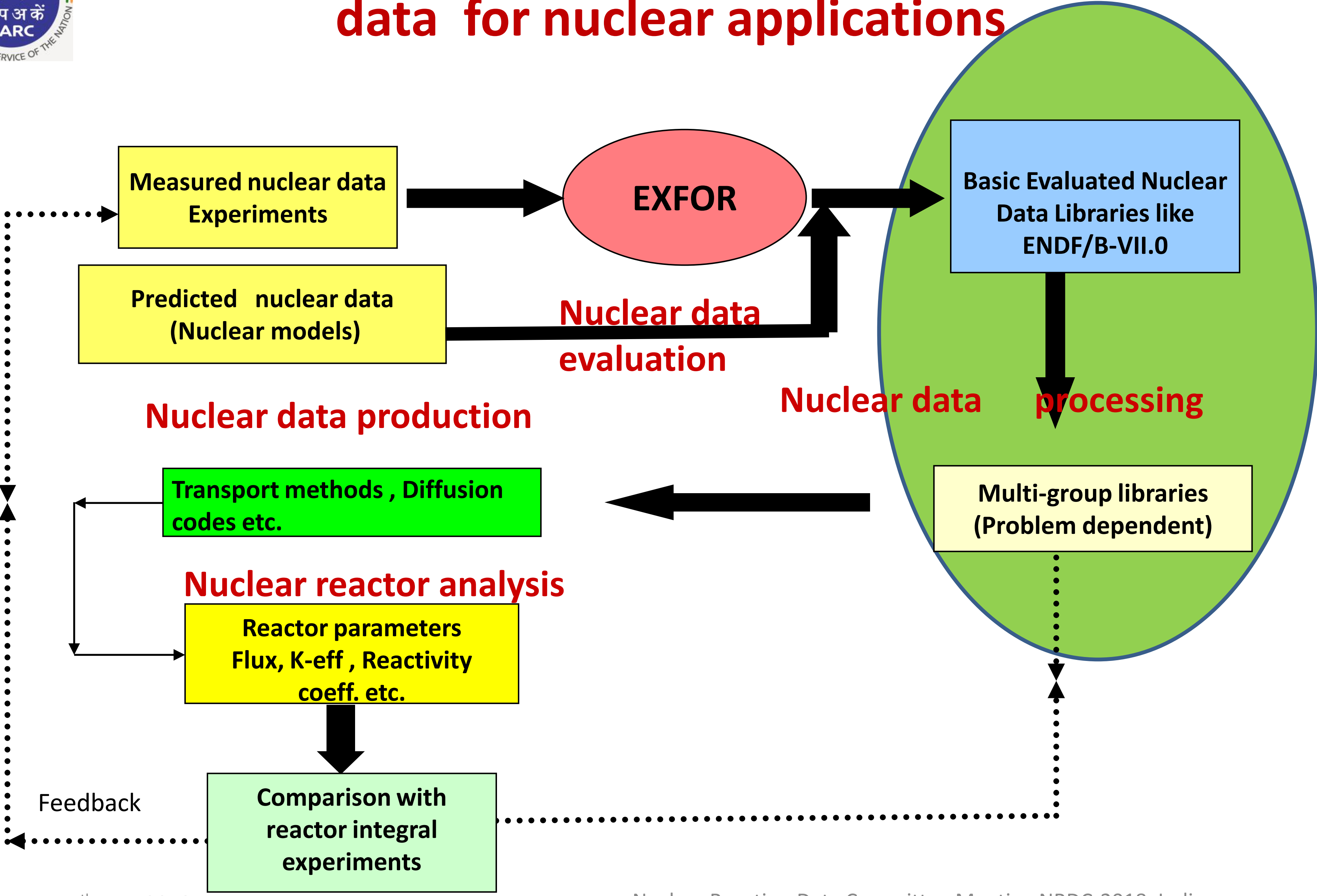
- Nuclear physics experiments _ Single energy behaviour/ limitted energy range
  - Carefully computerised ordered data (EXFOR)
  - Indexed data (w.r.t experimentor, lab, country etc.) (CINDA)

Critical evaluation process

- Supplemented with nuclear models (for eg. Where data does not exist)
- “Best estimate” approach where several data exist
- Critically reviewed for systematic errors
- Critical comparison, selection, re-normalisation, averaging and formatting

Set of highly ordered computer readable data-set grouped by materials and data types ----- **ENDF**

General scheme for generation of nuclear physics data for nuclear applications



Measured nuclear data Experiments

Predicted nuclear data (Nuclear models)

EXFOR

Nuclear data evaluation

Basic Evaluated Nuclear Data Libraries like ENDF/B-VII.0

Nuclear data production

Nuclear data processing

Multi-group libraries (Problem dependent)

Transport methods , Diffusion codes etc.

Nuclear reactor analysis

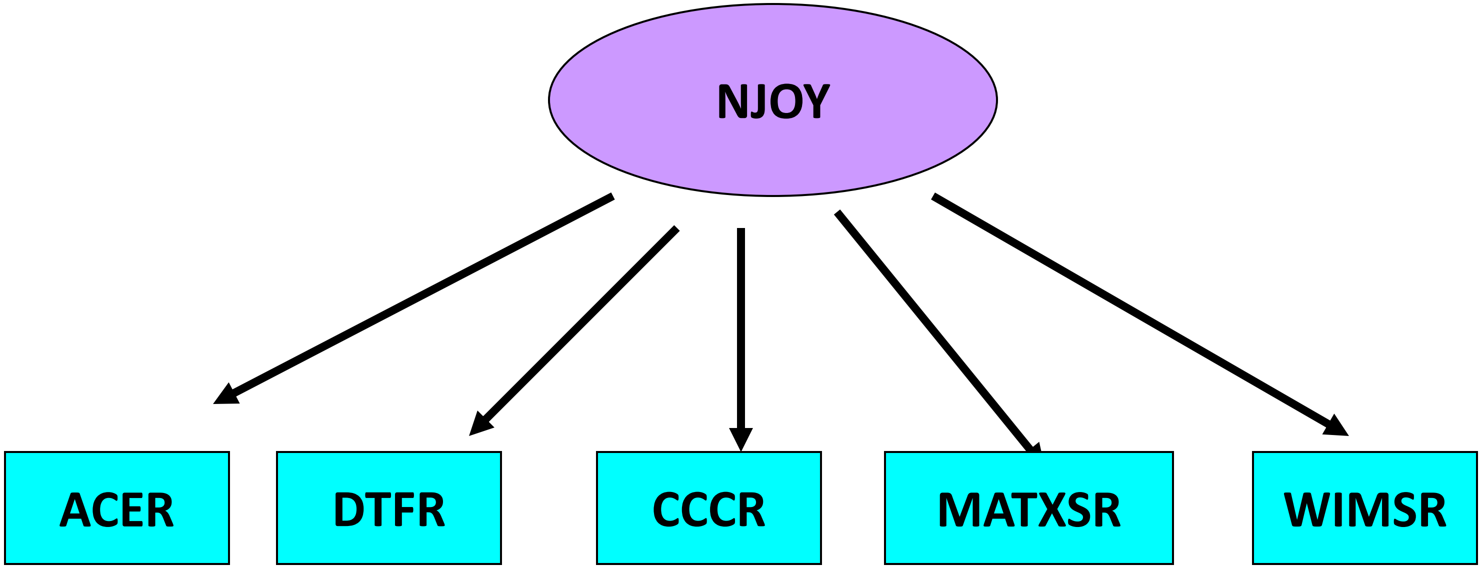
Reactor parameters Flux, K-eff , Reactivity coeff. etc.

Comparison with reactor integral experiments

Feedback

Processing of basic nuclear data

Codes like NJOY process the basic nuclear data and format it either in pointwise or multi-group sets for the required applications



NJOY

ACER

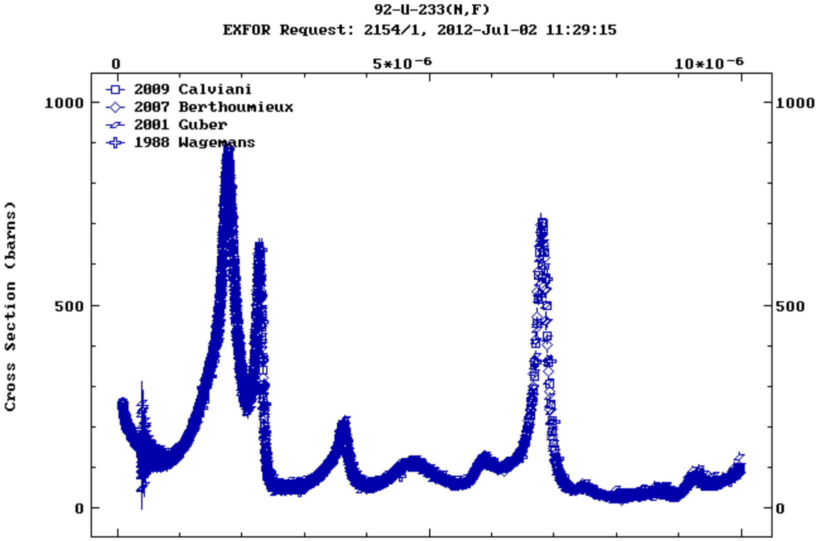
DTFR

CCCR

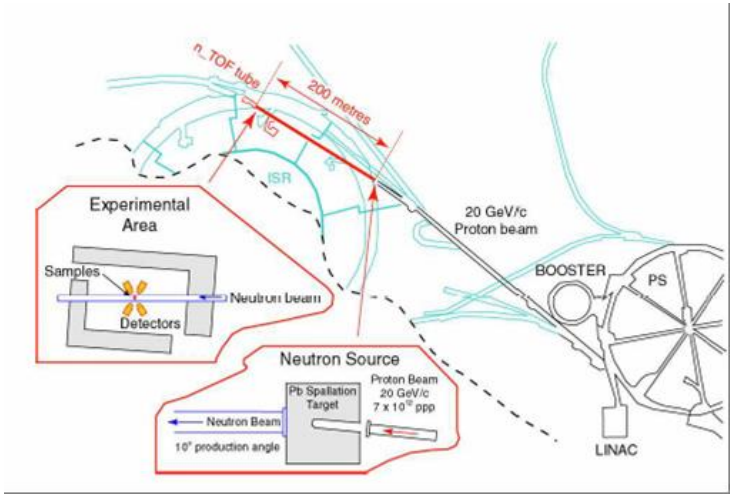
MATXSR

WIMSR

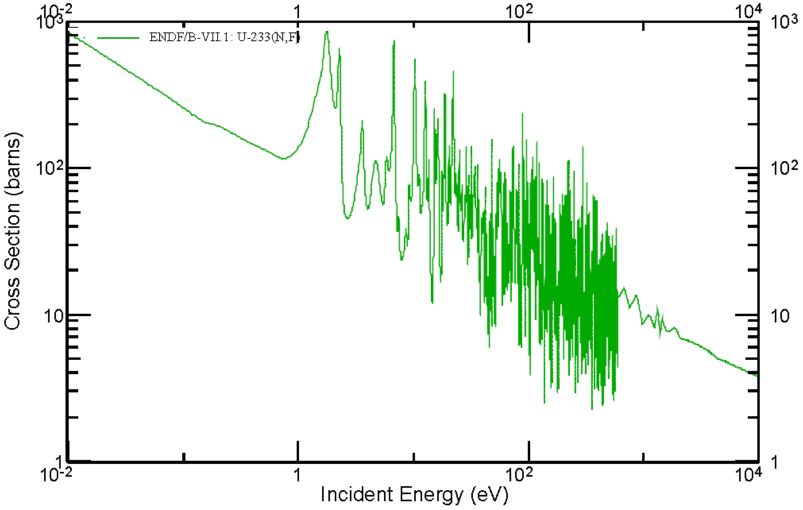
Nuclear data measurement to physical constants for potential applications



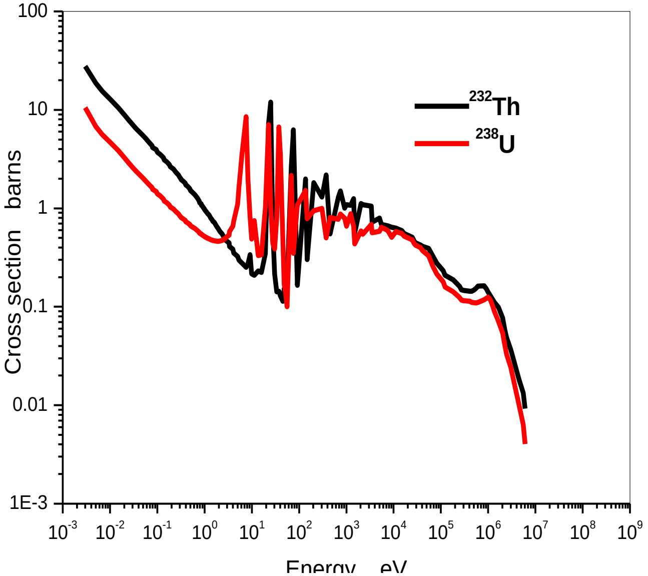
Variation of U-233 fission cross section with energy



The n_TOF facility at CERN



Capture cross section of ^{232}Th and ^{238}U



Processing of basic nuclear data for potential applications

- Which data is important in which energy range ?
- With what acccuracy ?
  - Material composition
  - Neutron spectrum

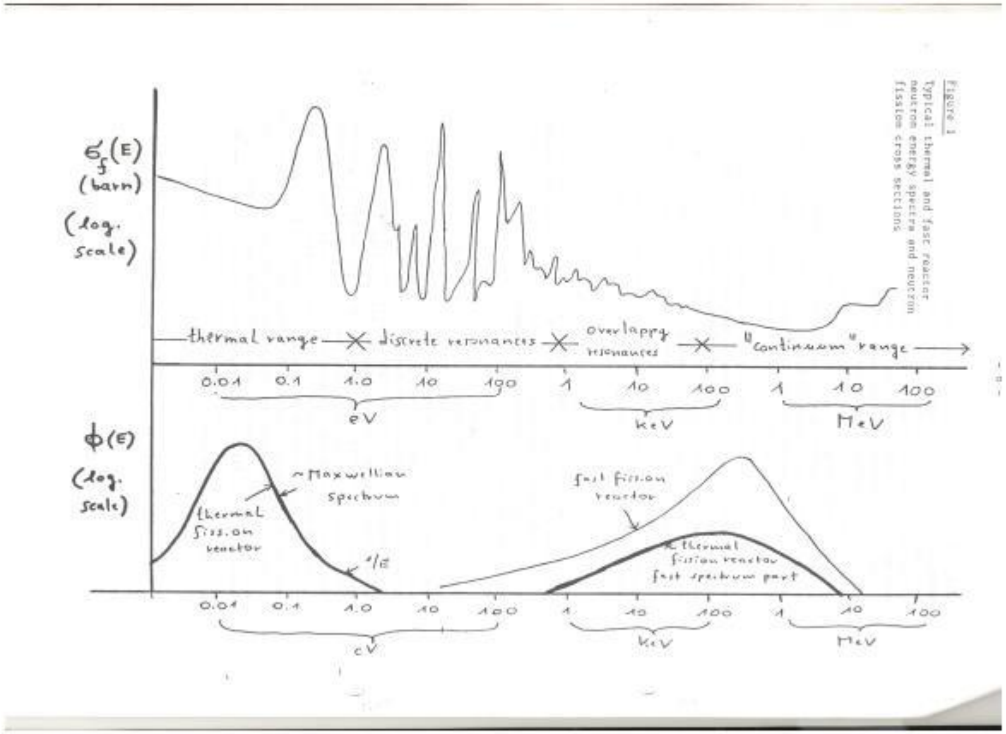


Figure 1

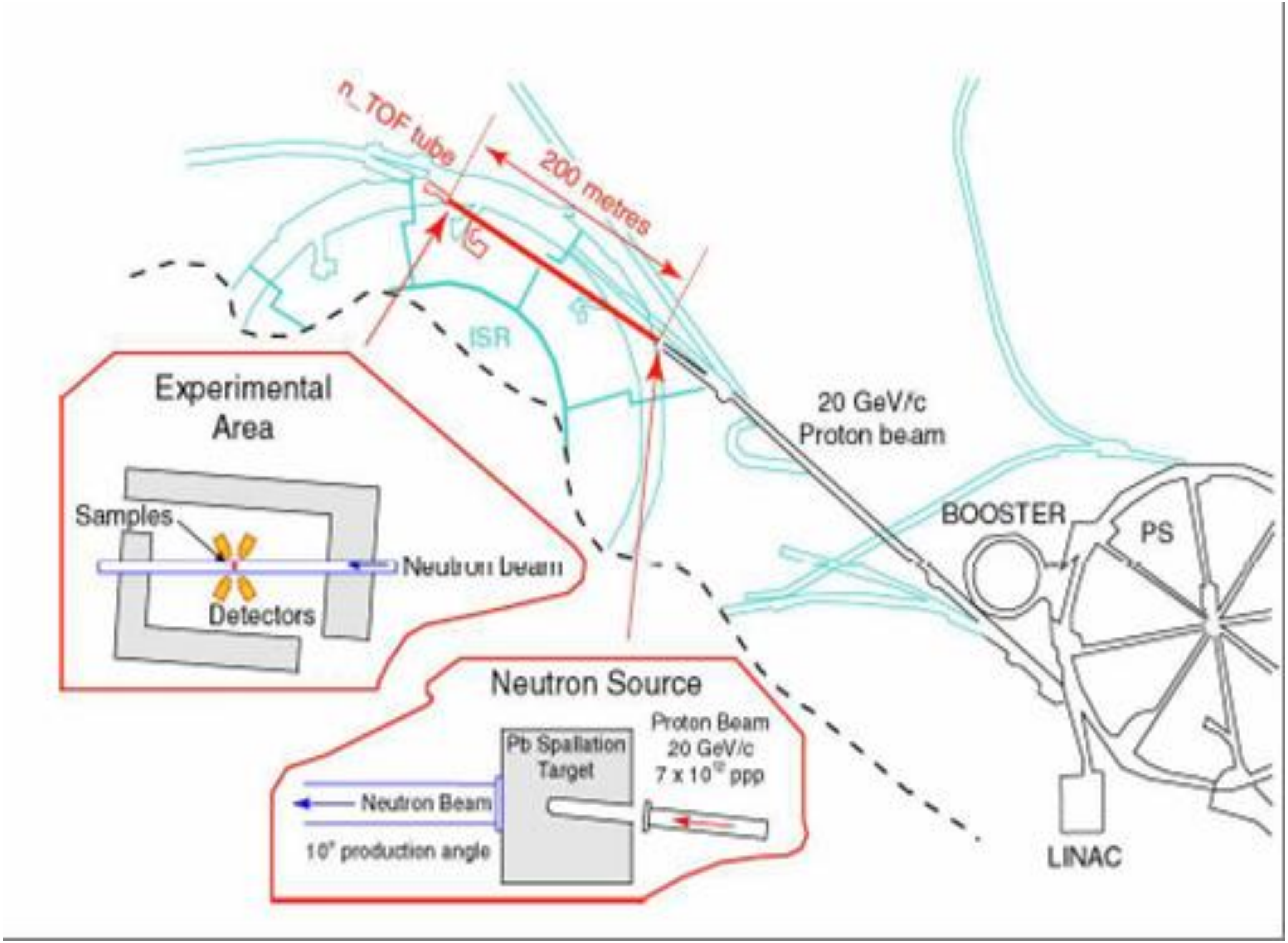
Typical thermal and fast reactor neutron energy spectra and neutron fission cross sections

From EXFOR to Data for Reactor Design Applications

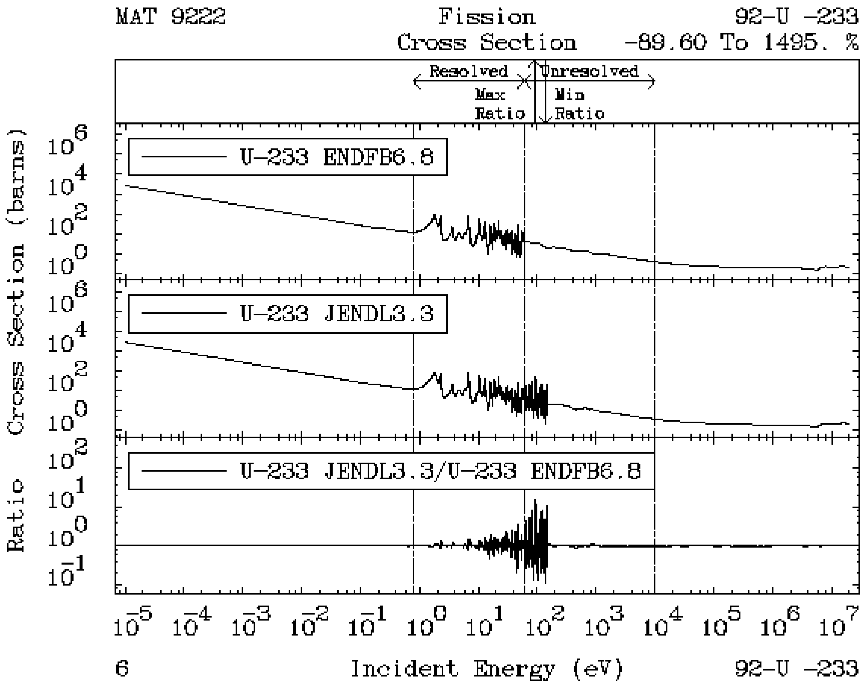
- **Experiments at all energies**
- **Basic nuclear physics experiments for all reactions**
- **Also for all nuclides of importance to**
- **Cover the entire range of energies (from a hew meV to MeV)**
- **Uncertainities**
    - Measurements
    - Evaluation
    - Processing
- **Obtaining pure samples**
- **Cutting edge facilities for measurements**

- Assessment of safety parameters
- Quantifying <u>design margins</u>
- Design decisions …

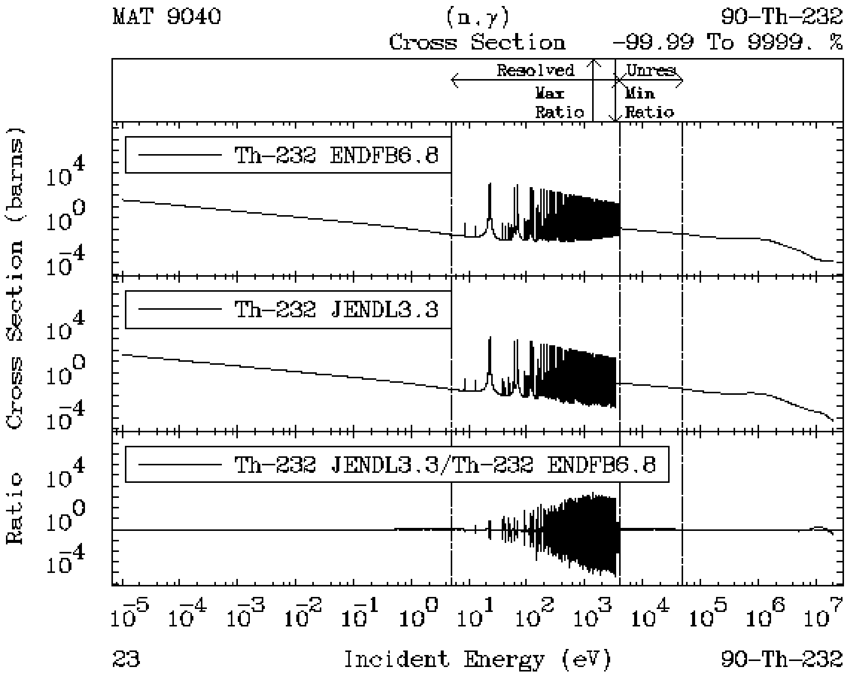
The n_TOF facility at CERN



Status of evaluated nuclear data : Thorium fuel cycle an example



Ratio plot of fission cross section of ^{233}U in JENDL-3.3 and ENDF/B-VI.8



Ratio plot of capture cross section of ^{232}Th in JENDL-3.3 and ENDF/B-VI.8

The comparisons and the typical analysis for the thorium fuel cycle shows that there are a large discrepancies in the basic cross section data used in lattice analysis.

The isotopes of the thorium fuel cycle need to be well qualified for use in reactor applications.

It is also imperative that new experiments also are performed to have better and verified data.

Recipe to include ^{231}Pa and ^{232}U in the WIMS library

1. **^{232}Th (n,2n) ^{231}Th (β) ^{231}Pa (n,γ) ^{232}Pa (β) ^{232}U**
2. **^{232}Th (n,γ) ^{233}Th (β) ^{233}Pa (β) ^{233}U (n,2n) ^{232}U**
3. **^{232}Th (n,γ) ^{233}Th (β) ^{233}Pa (n,2n) ^{232}Pa (β) ^{232}U**
4. **^{230}Th (n,γ) ^{231}Th (β) ^{231}Pa (n,γ) ^{232}Pa (β) ^{232}U**
5. **^{233}U (n,2n) ^{232}U**
6. **^{230}Th (n,γ) ^{231}Th (β) ^{231}Pa (n,γ) ^{232}Pa (n,γ)^{233}Pa (β) ^{233}U (n,2n) ^{232}U**
7. **^{234}U(n,3n) ^{232}U**
8. **^{234}U(n,2n) ^{233}U (n,2n) ^{232}U**

$$\gamma(^{231}Pa) = \frac{<\sigma_{(n,2n)}(^{232}Th)>}{<\sigma_f(^{232}Th)>} \qquad \gamma(^{232}U) = \frac{<\sigma_{(n,2n)}(^{233}U)>}{<\sigma_f(^{233}U)>}$$

Isotope	**Description**	**Reaction rate ratios**		
		ENDF/B-VI.8	**JENDL-3.2**	**JEFF-2.2**
^{232}Th	WIMS group structure with EPRI spectrum (NJOY /GROUPR)	0.1715	-	-
	WIMS group structure with HWR spectrum (NJOY /GROUPR)	0.166	0.	-
	VITAMIN-J group structure with HWR spectrum (NJOY /GROUPR)	0.200	0.1854	0.224
	One group average with HWR spectrum (PREPRO /GROUPIE)	0.230	0.204	0.244
	One group average with (1/E + Maxwellian and fission spectrum) (PREPRO /GROUPIE)	0.275	0.237	0.292
^{233}U	VITAMIN-J group structure with HWR spectrum (NJOY/GROUPR)	8.09E-6	7.51E-6	1.4E-5

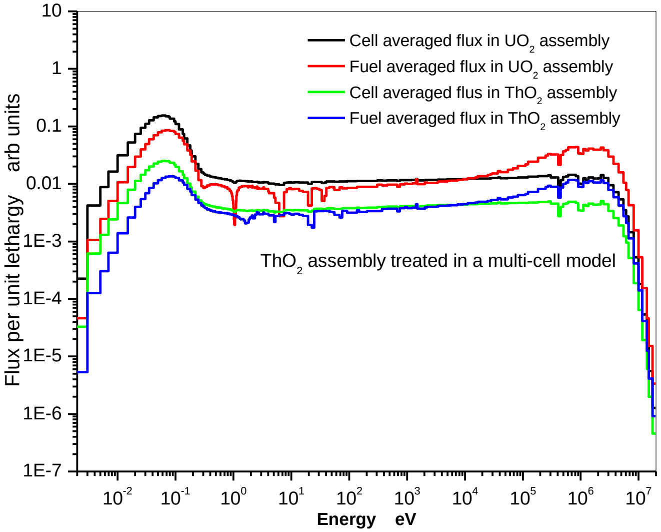


Figure 3.4a Typical neutron spectrum for ThO$_2$ and UO$_2$ fuel assemblies in a PHWR at 500 FPD

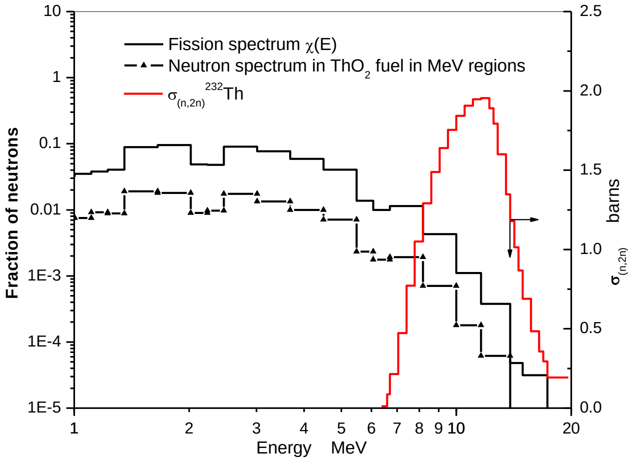
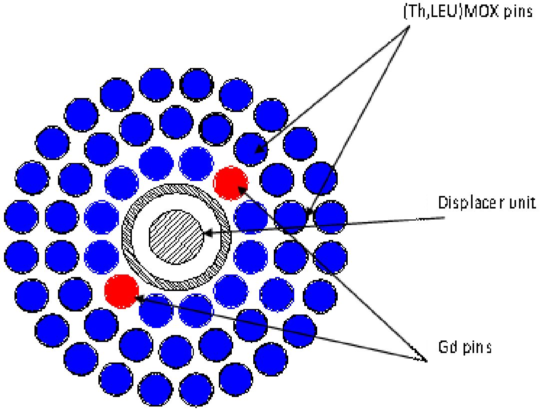


Figure 3.4b $\sigma_{(n,2n)}$ cross section of ^{232}Th in relation with fission spectrum and fuel neutron spectrum in ThO$_2$ lattice

Uncertainty analysis for AHWR fuel



AHWR (Th,LEU) fuel cluster

Uncertainty and sensitivity analysis is important for estimating the

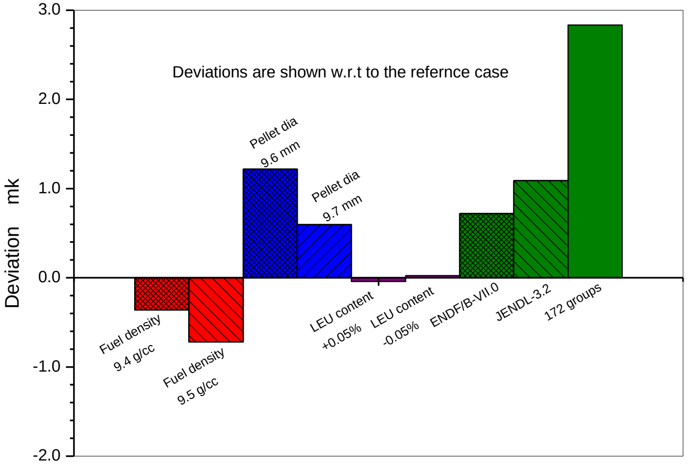
AHWR fuelled with (Th,LEU) taken up for this study

- *Reference fuel refers to average properties*

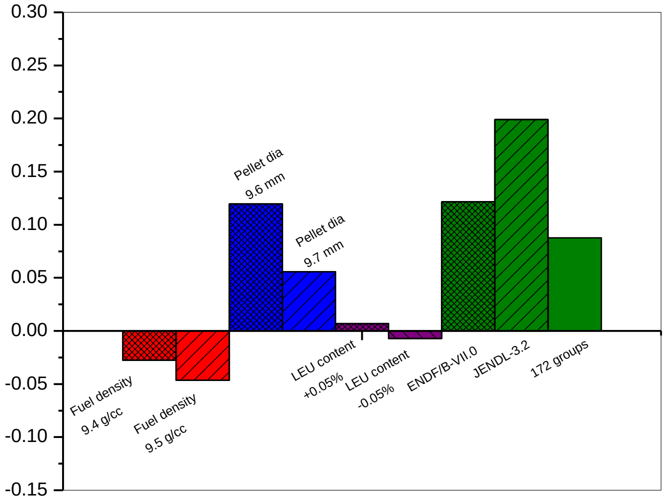
The uncertainties accounted for are

- *sensitivity to nuclear data (~10% in calculated CVR, 4% in FTC),*
- *sensitivity to modelling (172 group simulations gives 25% deviations in CVR),*
- *uncertainties in fuel pellet properties like pellet diameter (~11%,)*
- *fuel density and enrichments. (~7%)*

AHWR: Low Enriched Uranium Core



Uncertainty in coolant void reactiivty coeffiicent



Uncertainty in fuel temperature reactivity

Umasankari Kannan, Anek Kumar, Neelima Prasad Pushpam, P.D.Krishnani and R.K.Sinha, "Uncertainty analysis for safety parameters in thorium-LEU fuelled Advanced Heavy Water Reactor" CANDU Fuel Conference

Sensitivity of CVR in AHWR to nuclear data

Figure 16a Deviation group wise in contribution to CVR from ^{232}Th between ENDF/B-VII.0 and ENDF/B-VI.8 for AHWR at BOC

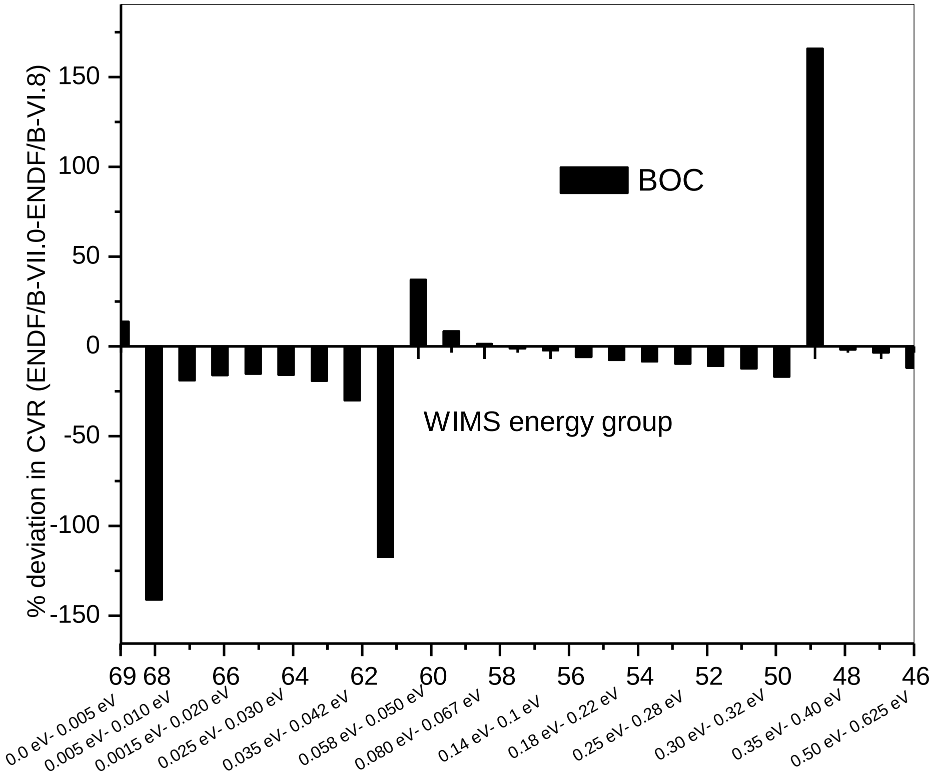
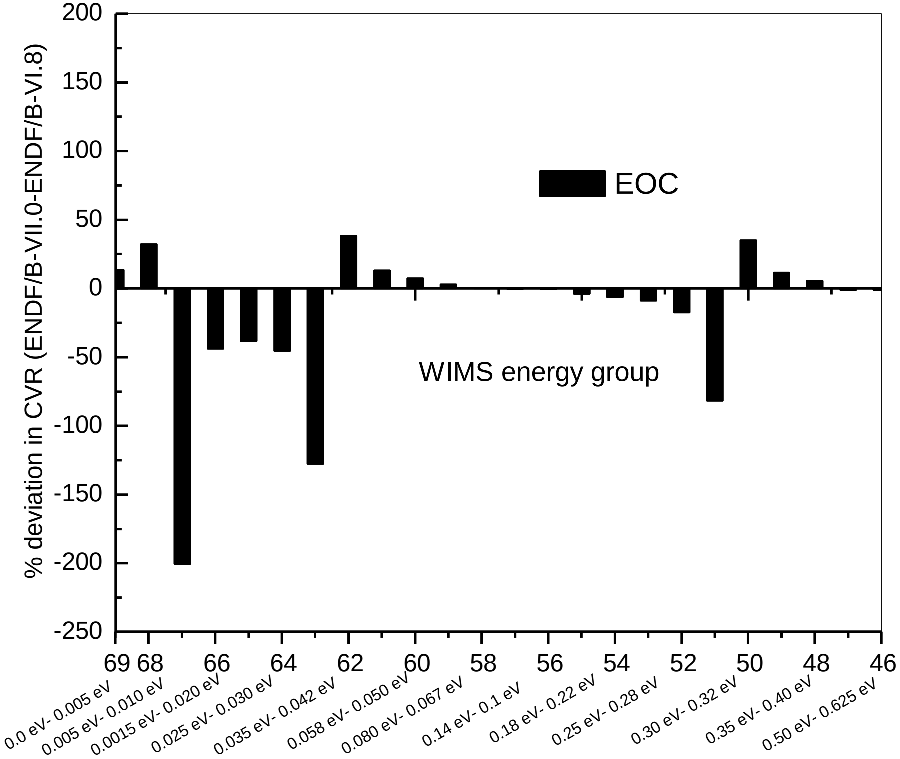


Figure 16b Deviation in group wise contribution to CVR from ^{232}Th between ENDF/B-VII.0 and ENDF/B-VI.8 for AHWR at EOC



Relative absorptions have been quantified by normalising to one fission neutron .

A scatter / deviation of 10% in CVR exists . Experiments are planned in the CF to estimate CVR in similar spectrum

Umasankari Kannan and S. Ganesan, "Analysis of coolant void reactivity of Advanced Heavy Water Reactor", *Nuclear Science and Engineering, Journal of the American Nuclear Society,* Vol. **167**, Number 2, Feb 2011 pp 105-121 (2011).

Sensitivity of CTR to nuclear cross sections

It can be seen that the contribution form ^{232}Th is less negative with the ENDF/B-VII.0 data set by 0.88 mk. This is due to the reduction in capture cross section of ^{232}Th in the ENDF/B-VII.0.

The sensitivity to ^{233}U data has been estimated as -0.3 mk at BOC and -0.412 mk at EOC.

This is attributed to the difference in the η of ^{233}U between the two data sets. The difference in η of ^{233}U between ENDF/-VI.8 and ENDF/-VII.0

η is lower by 2% in thermal energies and a maximum of about 11.8% in the resonance energies in the 69 group ENDF/B-VII.0 data set (Umasankari and Ganesan, 2007). The scattering cross section in the ENDF/B-VII.0 data set is lower by 6% from 0.01 eV to 0.1 eV as compared to the ENDF/B-VI.8 data set.

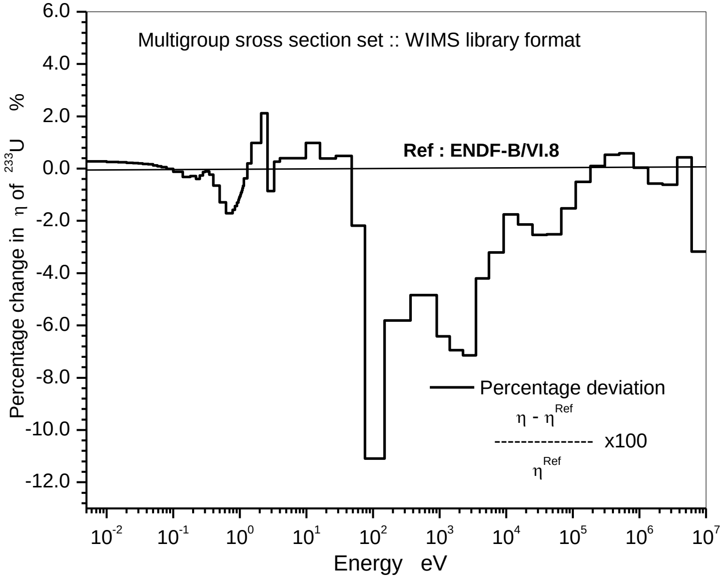


Figure 7.5 Comparison of η of ^{233}U in ENDF/B-VI.8 and ENDF/B-VII.0

The isotopic breakdown of the CTR shows that the major contributors to the CTR are H and ^{233}U which have positive components and ^{232}Th, ^{239}Pu and ^{240}Pu which have negative components.

The increase in the CTR with burnup and subsequent decrease is due to the change in η or more precisely $d\eta/\eta$.

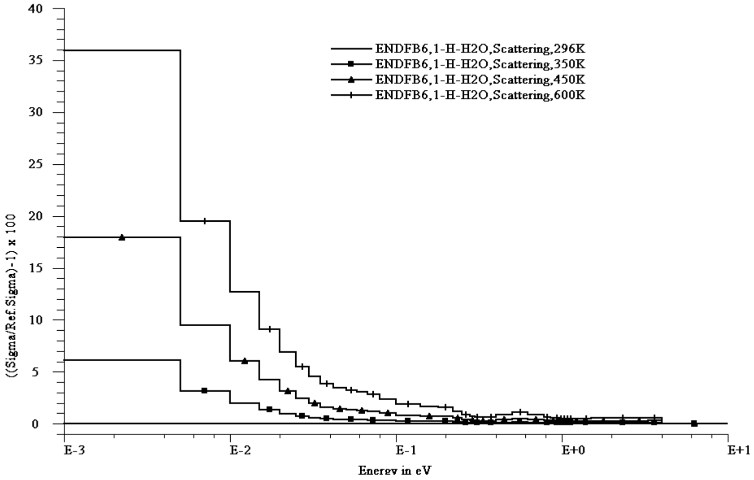
Umasankari Kannan and S. Ganesan, “Effect of newly available ENDF-B/VII.0 cross-section set on AHWR design studies”, CDROM Proceedings of National Symposium on radiation Physics, NSRP-17, Kolkatta, 14-16, Nov. 2007, (2007).

Scattering in Hydrogen : Temperature dependence

- Hydrogen shows a positive contribution which increases with burnup.
- Total scattering cross section increases by 35% at 600K as compared to that at 293 K.

$$\sigma(T) = \sigma(T_0)\sqrt{\frac{T_0}{T}}$$

- The 1/v behaviour (1/√E) of H absorption dominates over the density change, in this temperature range.
- The reduced absorptions in the coolant make its contribution to CTR positive
- The reduction in the density results in a reduction of the hydrogen atoms which again introduces a positive component. Or in other words, the water absorption reduces and fuel absorption increases thereby contributing to an increase in k_∞ and hence a positive component.



Analysis of Fuel Temperature Coefficient (FTC) of reactivity in PHWR fuel cluster

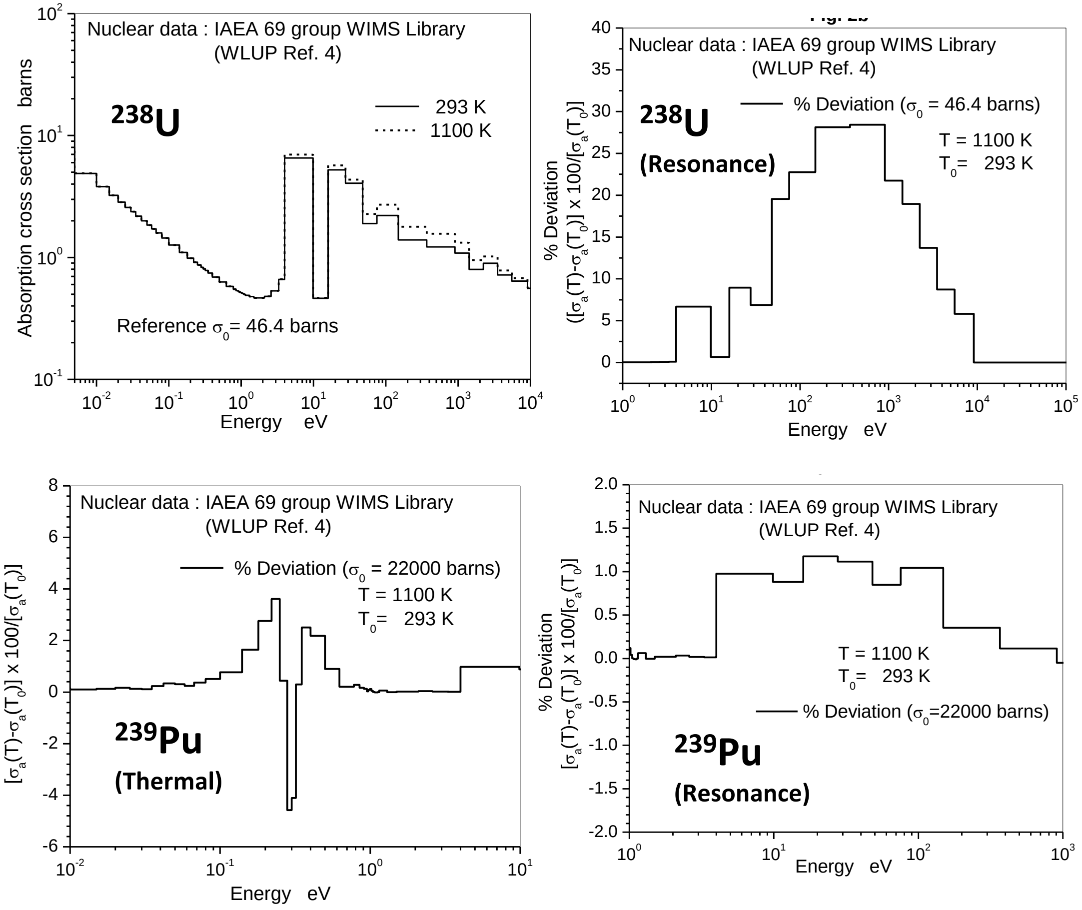
19 element UO_2 fuel cluster

This study was motivated by a specific over-power reactivity transient incident in the existing 220 MWe operating PHWR (KAPS-1) power reactor.

The isotopic and energy dependence of major contributors of the FTC for the PHWR 220 MW(e) reactor lattice has been calculated

A different approach was used where the temperature dependence of cross sections were switched off manually in the different energy regions namely

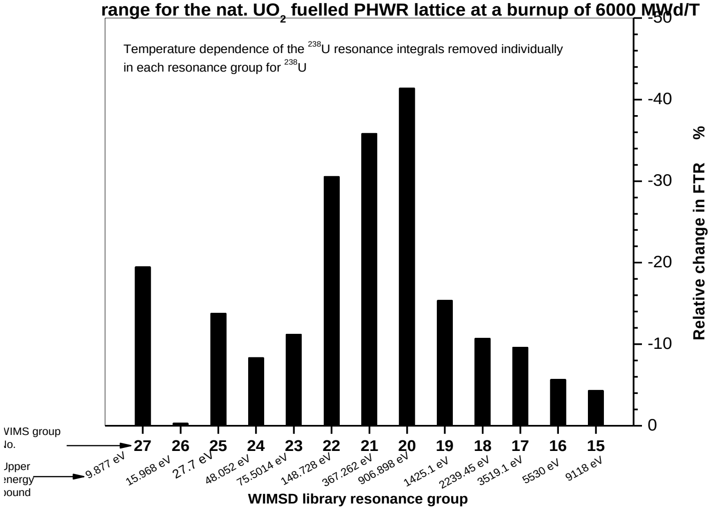
Resonance (9.118 keV to 4.0 eV) and Thermal (4.0 eV to 0.0 eV)



Examples of temperature dependence of ^{238}U in resonance and ^{239}Pu in thermal energy ranges

Quantifying the resonance contribution of ^{238}U to FTC

Fig. 6a Contribution to the FTR due to ^{238}U resonances in the resonance energy range for the nat. UO_2 fuelled PHWR lattice at a burnup of 6000 MWd/T



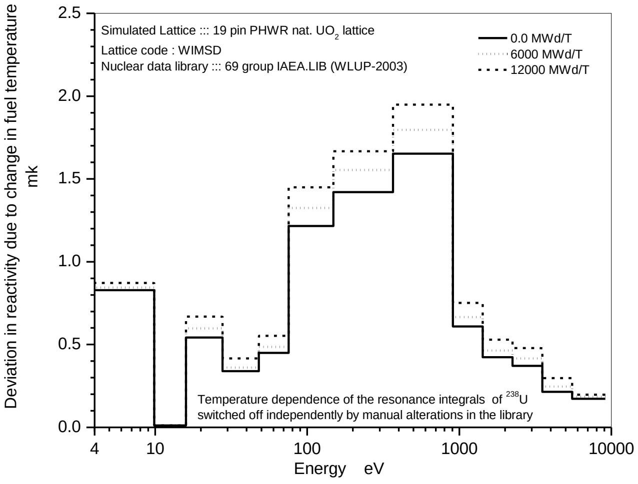
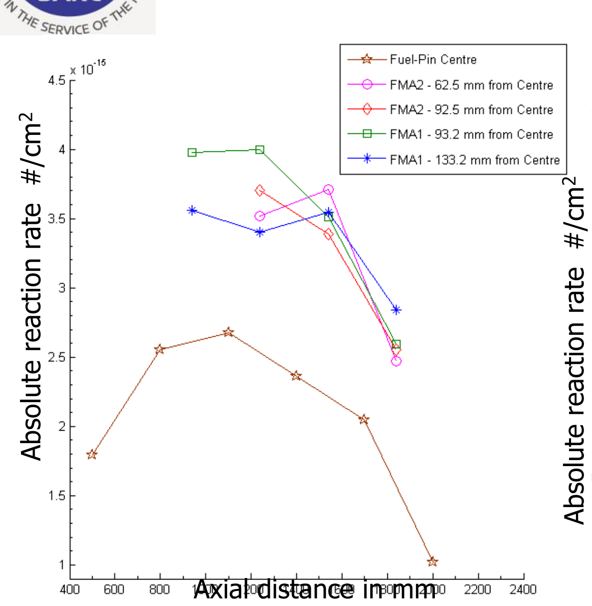


Figure 8.6b Sensitivity of ^{238}U to Fuel Temperature Reactivity (FTR) in the resonance energy region

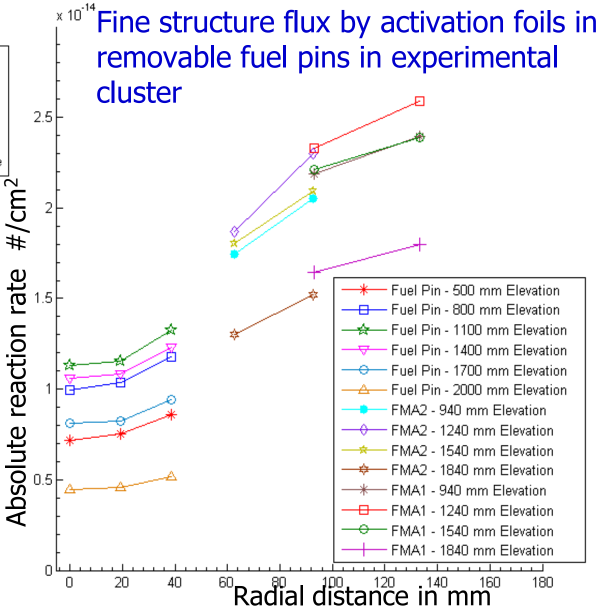
- The total contribution due to ^{238}U captures at equilibrium burnup of 6000 MWd/T is +8.972 mk
- The individual contribution from each resonance group has been estimated
- This amounts to 206.775% with a minimum of –0.277 % to a maximum -41.4 %
- The Doppler reactivity due to ^{238}U resonances is actually dominated by the resonances in the 75 eV to 1000 eV range. The maximum contribution of -41.4 % comes from the WIMS energy group 20, which ranges from 906.898 eV to 367.262 eV
- The 6.7 eV capture resonance alone contributes nearly -19 % of the Doppler effect of 238U from all its resonances up to 9.118 keV.

Umasankari, K., and Ganesan, S., 2007. Isotopic and Energy Groupwise Dependence of Fuel Temperature Coefficient of Reactivity in Natural Uranium Fuelled PHWRs. *Nuclear Science and Engineering*, Volume **156,** Number 2 · June 2007 · Pages 267-279.

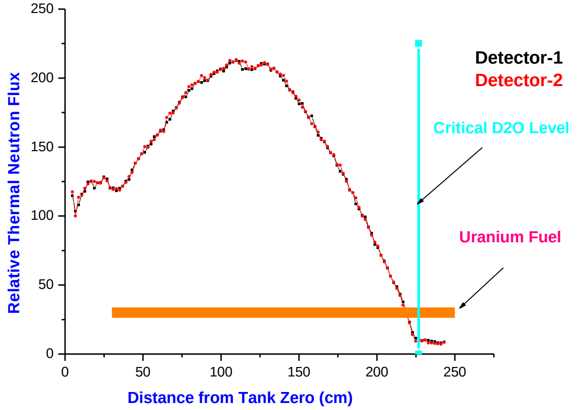
A few results from AHWR-CF



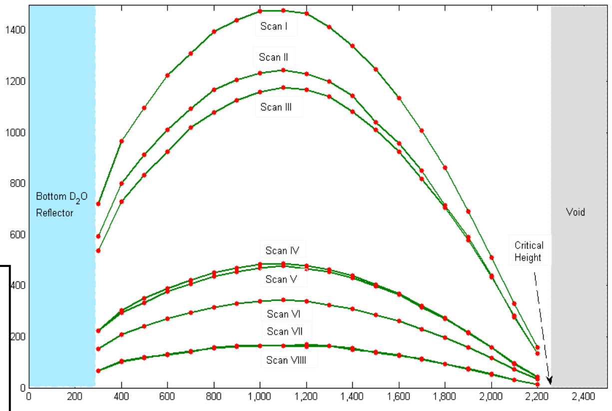
Axial Neutron Flux Distribution Inside and Around E5 Central Cluster Using Cadmium Covered Gold Foils


Fine structure flux by activation foils in removable fuel pins in experimental cluster

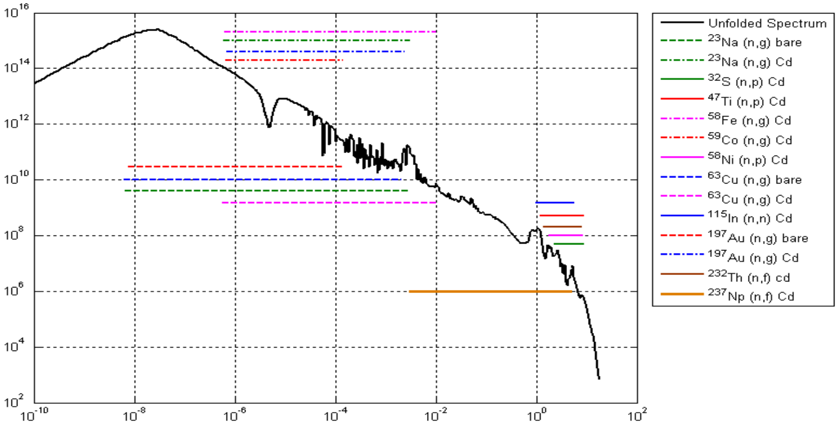
Radial Neutron Flux Distribution



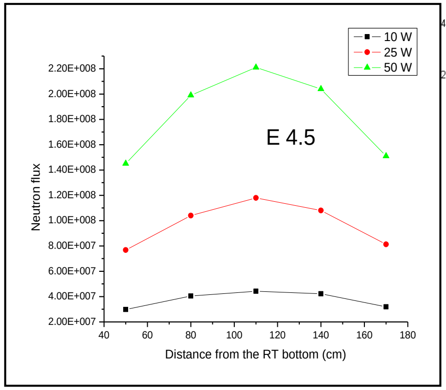
Measured axial flux distribution (Cu wire)



Gamma scanning of Fuel pin irradiated in AHWR-CF



Neutron Spectrum Measurement



Measured reaction rates used to calibrate flux mapping system (FCs)

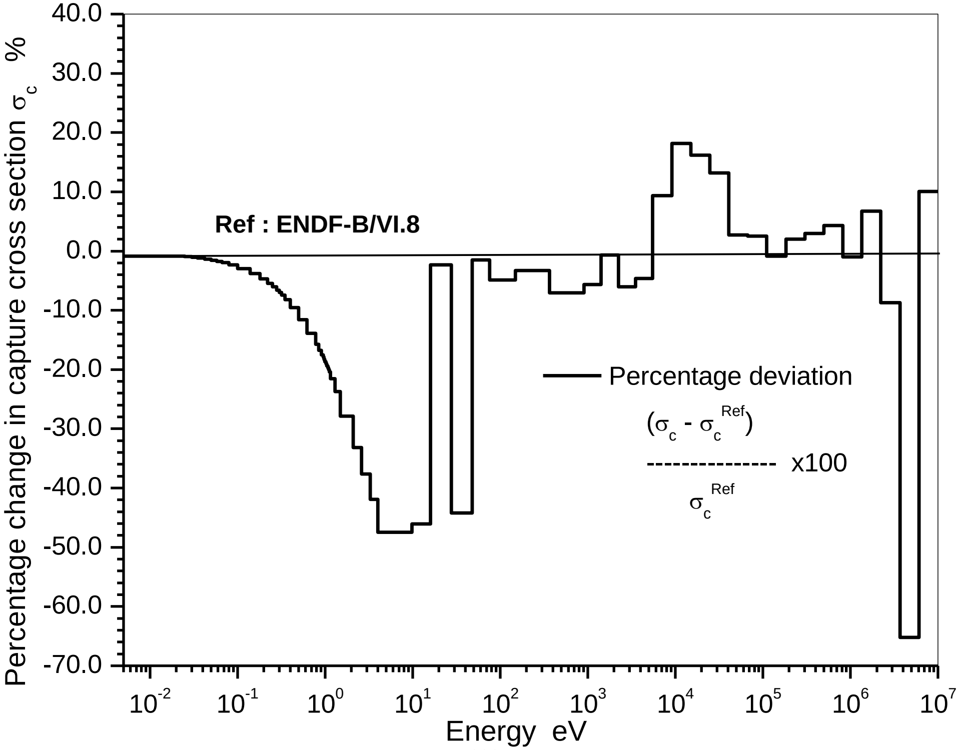


Figure 6.7 Comparison of ^{232}Th capture cross section in ENDF/B-VI.8 and ENDF/B-VII.0

Can we bridge this gap ????

Thank You