Propagation of $^{235,238}U$ and ^{239}Pu and STL (H in H_2O) Nuclear Data Uncertainties for PWR Core Analysis

An uncertainty propagation methodology based on Monte Carlo method is applied to PWR nuclear design analysis to assess the impact of nuclear data uncertainties in $^{235,238}U$, ^{239}Pu and Scattering Thermal Library for Hydrogen in water.

This uncertainty analysis is compared with the design and acceptance criteria to assure the adequacy of bounding estimates in safety margins.

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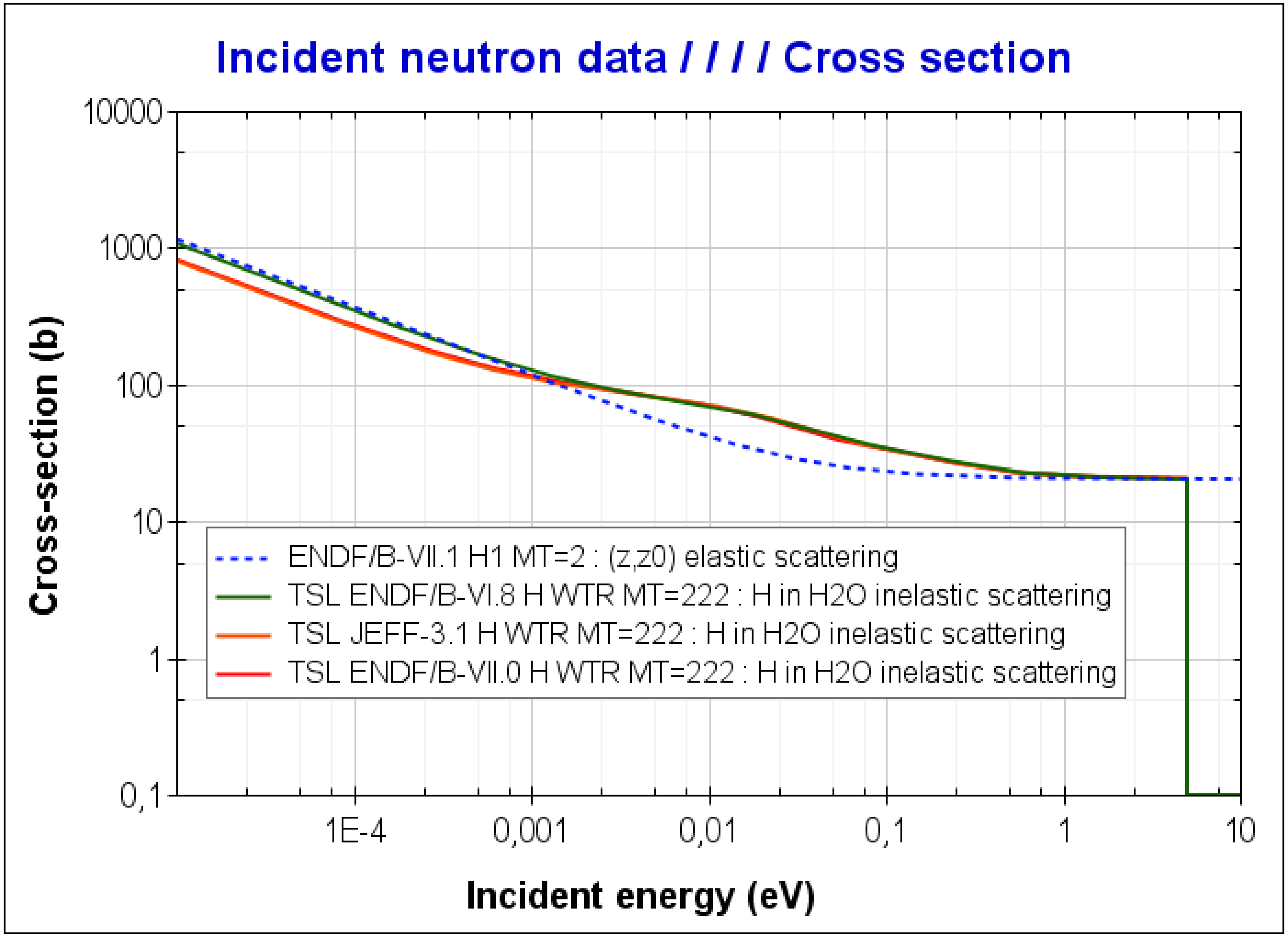
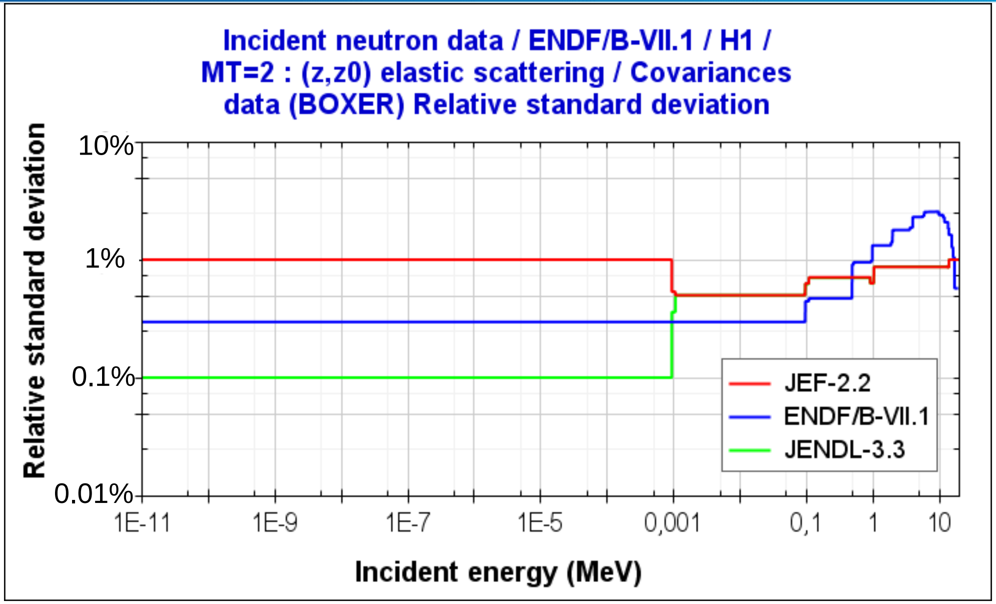
Summary and conclusions

I.1.1 Comparison of Elastic Cross Section: H (free gas model) versus H-in-H2O

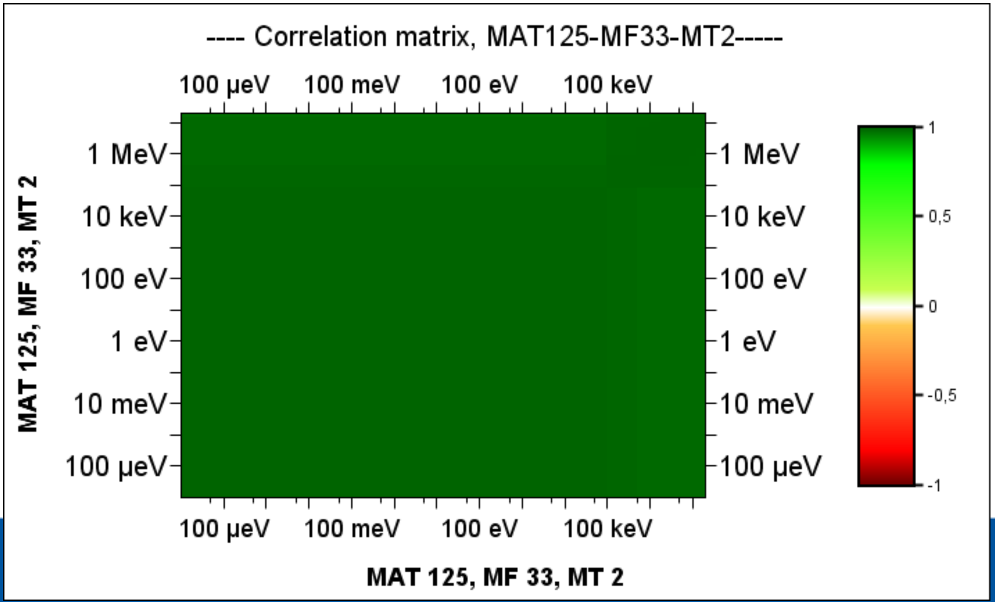


I.1.1 ENDF Uncertainties: Elastic Cross Section (MT2) for Hydrogen

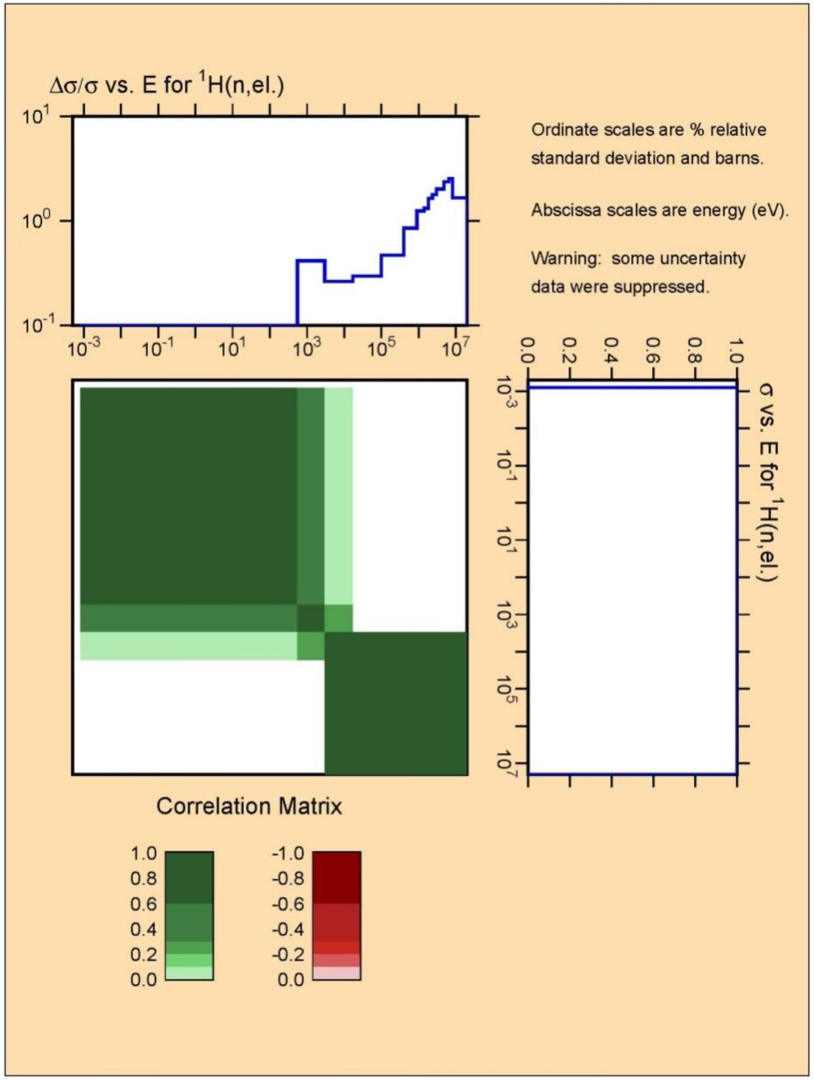


- **Comparison of relative standard deviation for different ENDFs sources**

- **Correlation matrix**



I.1.1 SCALE6.1/UN Correlation Matrix for Elastic Cross Section of Hydrogen (H)



Processing SCALE6.1/UN into ERRROR/BOXER format:

- ANGELO code to convert COVERX into ERRORR
- LAMDA to check covariance properties
- NJOY to process in BOXER format and to visualize with VIEWR

I.1.2 Random Inelastic Cross-Section: H in H2O

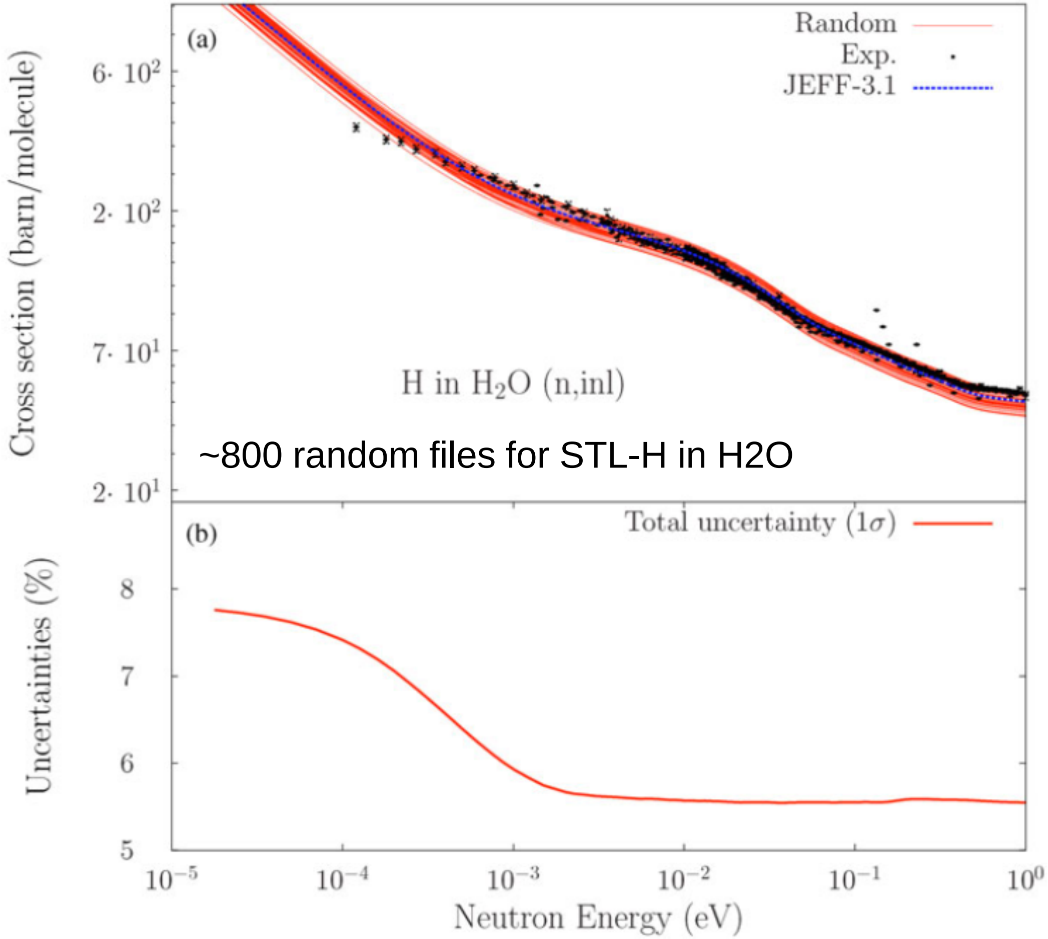


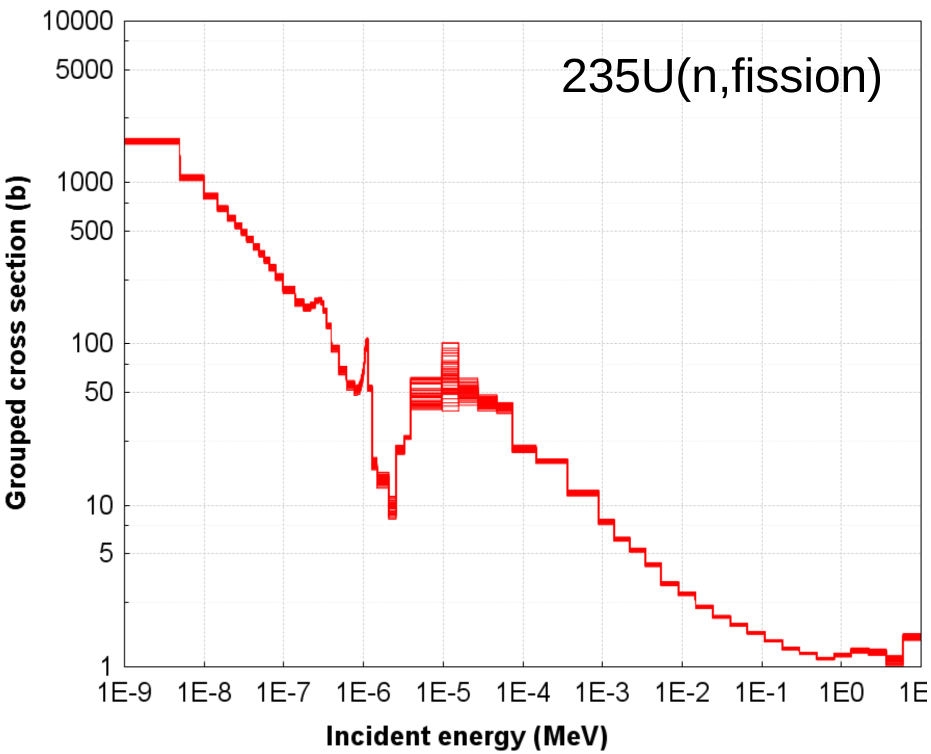
Figure: Incoherent random inelastic scattering cross section of H in H2O compared to experimental data and the inelastic cross section from the JEFF-3.1 library. Inelastic scattering is described by the scattering law *S*(α,β),

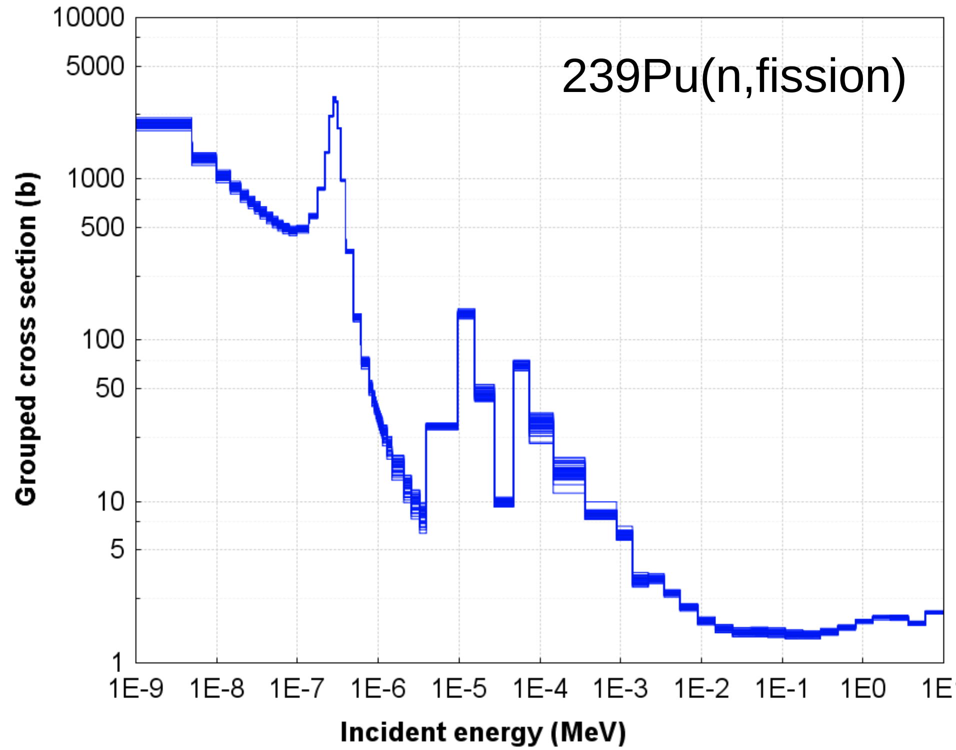
Figure: Uncertainties in the inelastic cross section calculated from 1330 random inelastic cross sections

- A NEARLY full correlated energy-energy correlation matrix for the incoherent inelastic scattering for H in H2O is found

Ref. "*Random Adjustment of the H in H2O Neutron Thermal Scattering Data*",
D. Rochman and A. J. Koning, NUCLEAR SCIENCE AND ENGINEERING: **172**, 287–299 (2012)

I.2 TENDL 2012 randoms: U235, U238 and Pu239

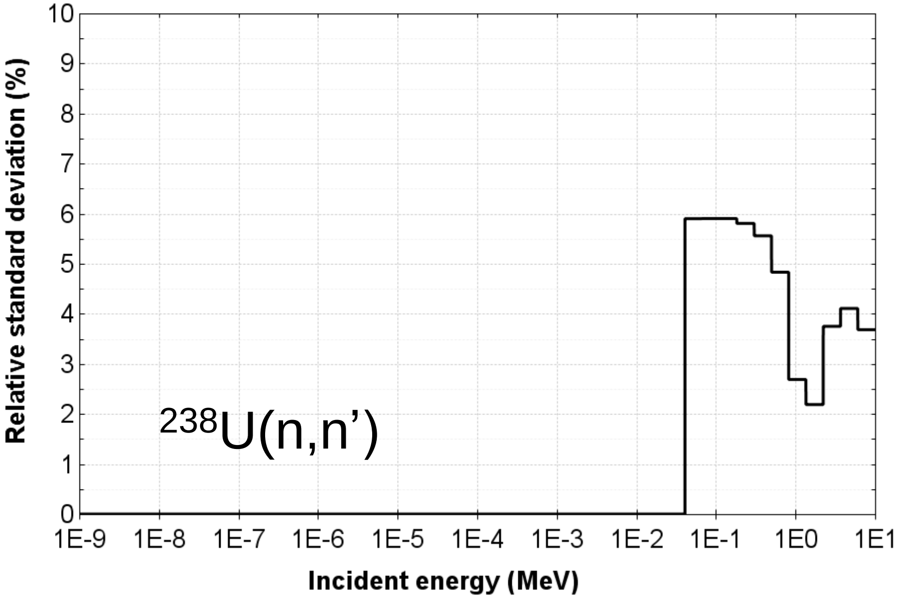


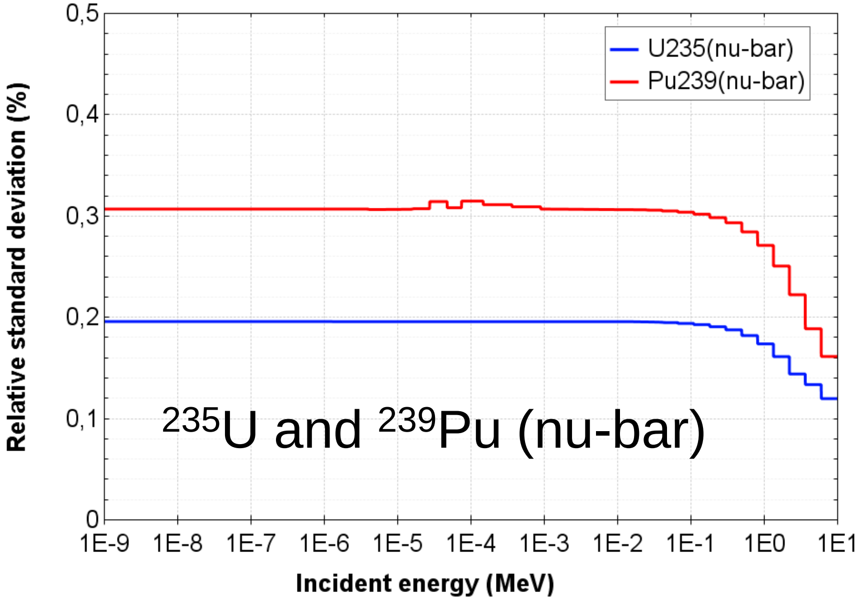


- First 50 TENDL2012 random files processed with NJOY/GROUPR in 69 energy groups at 293K with infinite dilution
- Random files: ~740 for ^{235}U, ~700 for ^{238}U and ~740 for ^{239}Pu

Relative standard deviation based on TENDL2012 random files

The most important contributors in criticality calculations: **^{239}Pu(nu-bar), ^{238}U(n,γ), ^{238}U(n,n'), ^{239}Pu(n,fission) and ^{235}U(nu-bar)** at 30 GWd/MTU

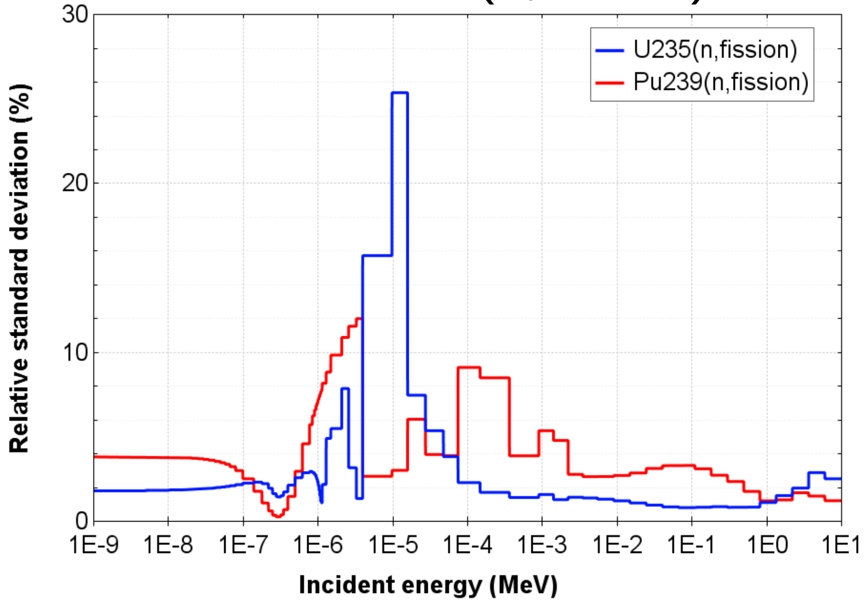




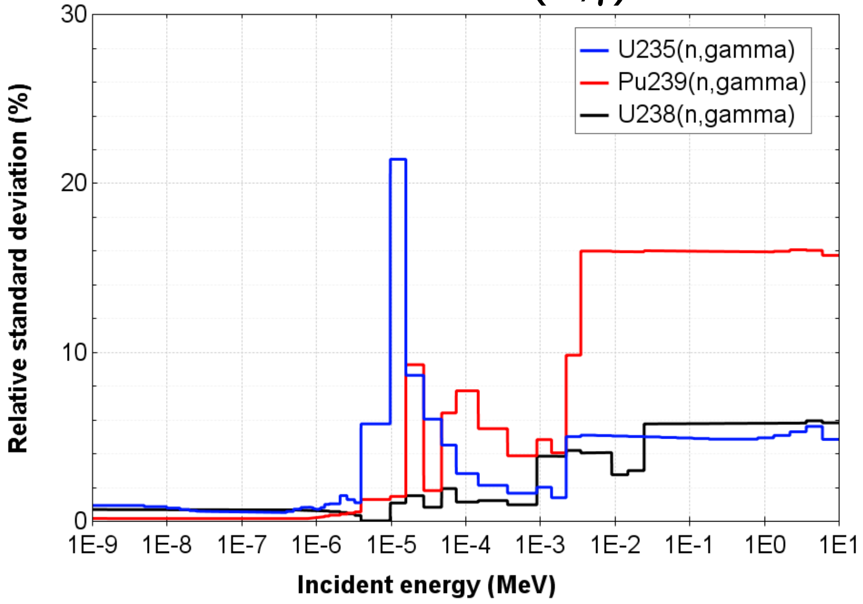
- ^{238}U(n,n'): < 6%
- ^{239}Pu(nu-bar) and ^{235}U(nu-bar): ~ 0.3-0.2%

Relative standard deviation based on TENDL2012 random files

^{235}U and ^{239}Pu (n,fission)



$^{235,238}U$ and ^{239}Pu (n,γ)



- Large uncertainty around 10 eV for ^{235}U(n,fission) and ^{235}U(n,γ)
- ^{239}Pu(n,g) at E> 5 keV the uncertainty remains high ~16%

However, large discrepancies of these uncertainties are found when comparing with current uncertainty nuclear data libraries, such as **SCALE6/UN**

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PART II. SEANAP System

II.1 Introduction to SEANAP

SEANAP = Sistema Español de Análisis de Núcleos de Agua a Presión

SEANAP system has been developed and applied for 3D PWR core analysis during near thirty years, as a close collaboration among the Polytechnical University of Madrid group of developers (J. M. Aragonés et al.) and the engineering groups of users at the several Spanish PWR units.

SEANAP is a mature, demonstrated, complete and integrated system of computer codes and procedures that provide full and independent PWR core analysis capabilities.

- **Integrated /coupled codes in SEANAP:**

  MARPIJ, COBAYA, DELFOS, SIMULA, SIMTRAN, COBRA, RELAP-5

- **SEANAP Applications:**

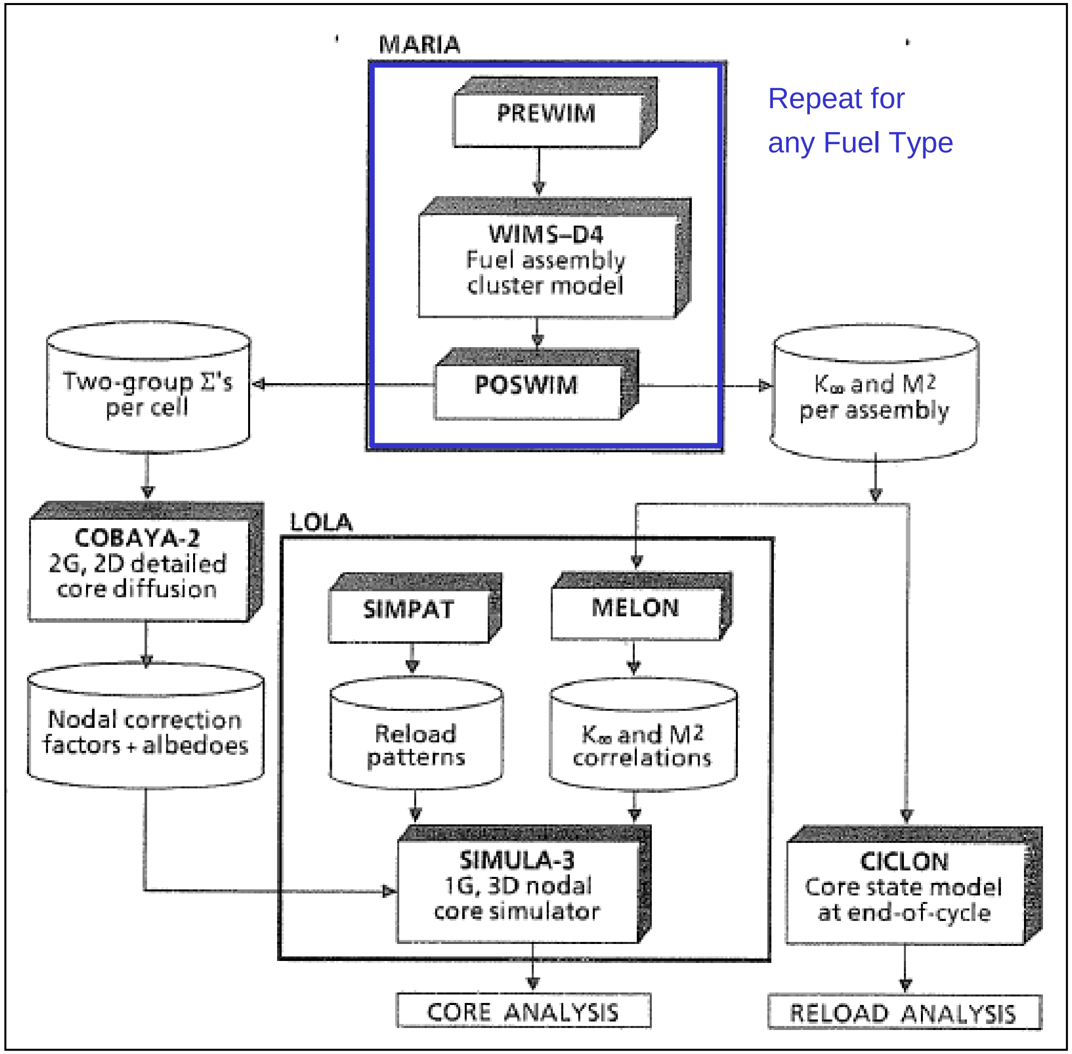
  - Fuel Loading Pattern Evaluations
  - Nuclear Design Analysis
  - On line 3D Simulations
  - Planning of Optimal Operational Maneuvers
  - Dynamic Core Analysis for Safety and Training Simulations

Reference: "*Continuous Validation and Development for Extended Applications of the SEANAP Integrated 3D PWR Core Analysis System*", C. Ahnert, J.M. Aragonés, O. Cabellos & N. García-Herranz, M&C99 (1999)

II.2 Scheme of the PWR Core Analysis SEANAP System

Figure 6. Scheme of the PWR Core Analysis System SEANAP-86

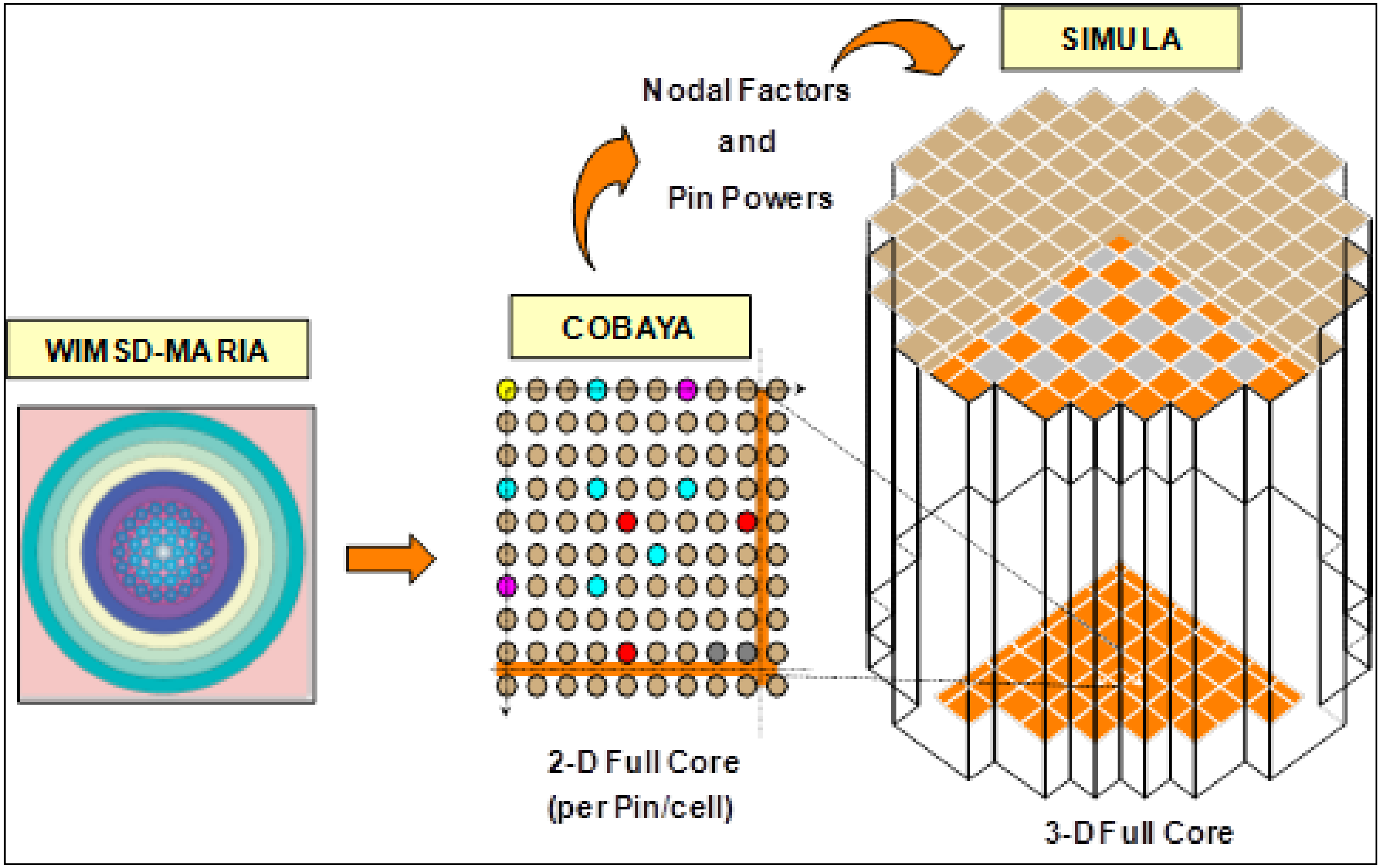
Ref.: *"Validation of PWR Core Analysis system SEANAP-86 with measurements in test and operation", C. Ahnert et al., M&C87*



SEANAP is integrated by 4 subsystems:

1. **MARIA** system for assembly calculations
2. **COBAYA** system for a detailed (pin-by-pin) core calculations at reference conditions
3. **SIMULA** system for 3D 1 group corrected-nodal core simulation
4. **CICLON** system for fuel management analysis of reload cycles

II.2 Scheme of the PWR Core Analysis SEANAP System



SEANAP: WIMS-D4+COBAYA+SIMULA

II.3 Validation of SEANAP

- SEANAP system has been applied in the last 30 years for 7 Spanish PWR units (Almaraz I and II, Ascó I and II, Trillo, Vandellós II and Zorita):
  - Fuel loading pattern optimization carried out for about **75 cycles** with very positive results
  - Full capability of the nuclear design for each cycle
    - **Start-up physic test at HZP**: critical end-point boron concentration , isothermal temperature coefficients, control bank worths, differential boron worth and power distribution
    - **Nominal operation**: boron concentration, in-core flux maps

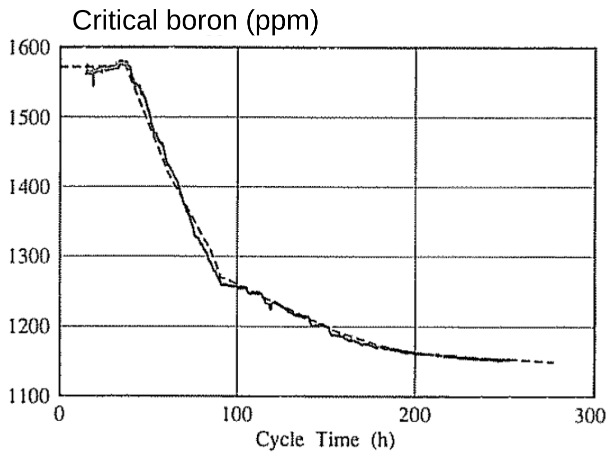


Figure: Critical boron (ppm) as a function of time (hours) from startup of cycle 8 in VandellosII: predictions (solid line) and chemical measurements (dashed line)

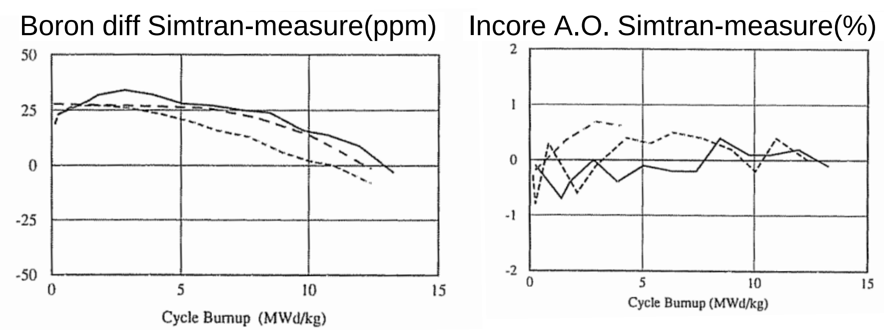


Figure: Diffreences Calculated_Measured in critical boron (ppm) and in-core axial offset (%) as a function of core burnup along cylces 6(dashed-line), 7 (solid line) and 8 (long dashed line) of Vandellos-II

II.3 Validation of SEANAP

- SEANAP system has been developed and implemented as an online simulator ~**20 cycles of three PWRs (Vandellós-II, Ascó-I and Ascó-II)**
  - **Every 5 minutes, continuos operational surveillance:** boron concentration, reaction rates at the excore detectors, A.O., fluid temperatures at the location of thermocouples, temperatures at the hot legs…
  - **Every month** incore flux maps: Incore/excore calibrations
  - **Planning of Optimal Maneuvers**, **Dynamic Core Analysis** for safety and training for plant engineers and operators.

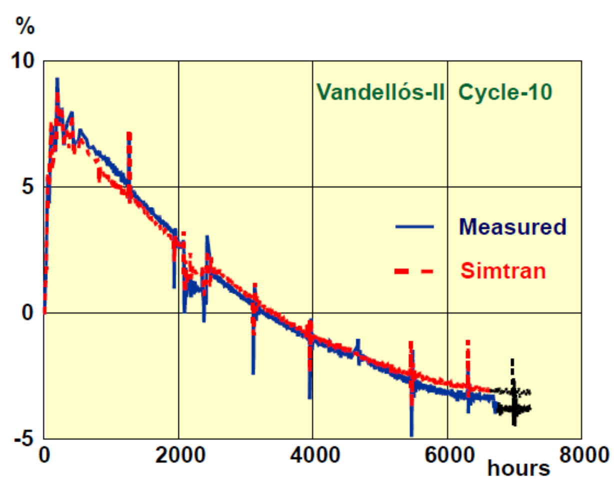


Figure: Delta-I of Incore Power as Measured and Calculated by SIMTRAN on-line

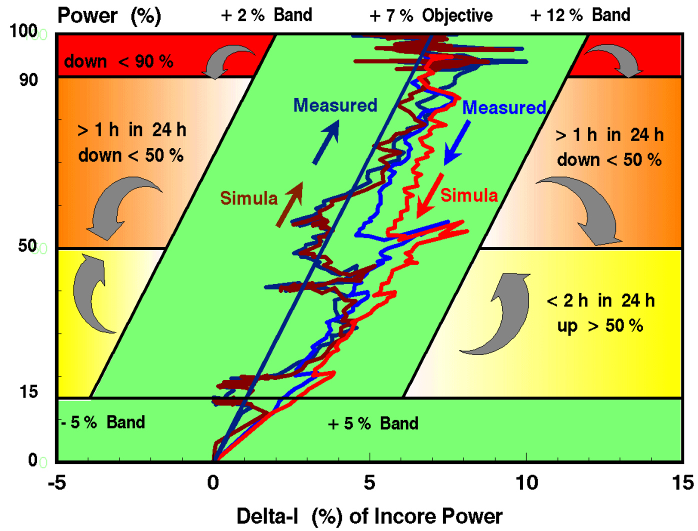


Figure: **M**easured and Simulated Power vs Delta-I in return to Power after a Short Shutdown

PART I. Uncertainty in Nuclear Data: TENDL2012 Random Files

- I.1 STLs
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  - I.1.2 Random Inelastic Cross-Section: H-H2O (“*Petten”* Methodology)
- I.2. ^{235,238}U and ^{239}Pu TENDL2012

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- III.2 PWR Core Analysis
- III.3 Uncertainty & Quantification

Summary and conclusions

III.1 PWR description

PWR (WESTINGHOUSE), 3 loops , 157 FA, power 2775. MWth

¼ CORE

1	13	4	21	6	21	16	14
13	11	15	2	16	6	20	7
4	15	3	21	8	22	19	
21	2	21	9	18	20	5	
6	16	8	18	12	17		
21	6	22	20	17			
16	20	19	10				
14	7						

AVE. BURNUP PER FUEL ASSEMBLY

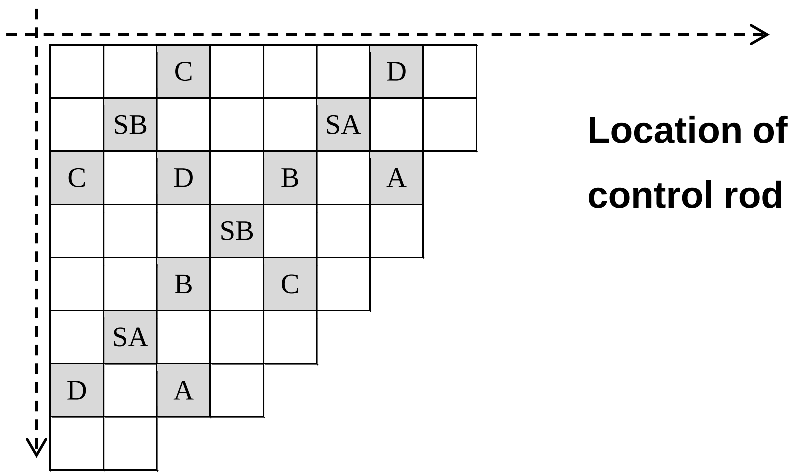
	1	2	3	4	5	6	7	8
1	18.137	11.662	27.397	**0.000**	30.867	**0.000**	14.984	11.662
2	11.662	16.188	13.130	28.902	12.155	28.866	**0.000**	30.191
3	27.397	13.130	27.572	**0.000**	22.778	**0.000**	**0.000**	
4	**0.000**	28.902	**0.000**	30.755	15.236	**0.000**	30.124	
5	30.867	12.155	22.778	15.236	13.123	14.882		
6	**0.000**	28.866	**0.000**	**0.000**	14.882			
7	14.984	**0.000**	**0.000**	30.503				
8	11.662	30.191						

FUEL	TYPE	w/o(%)	WABAS
1	OFA	2.10	0
2	OFA	3.10	0
3	OFA	3.24	0
4	OFA	3.24	0
5	OFA	3.24	0
6	OFA	3.24	0
7	OFA	3.24	0
8	OFA	3.24	0
9	OFA	3.24	0
10	OFA	3.24	0
11	OFA	3.24	0
12	AEF	3.60	0
13	AEF	3.60	0
14	AEF	3.60	0
15	AEF	3.60	0
16	AEF	3.60	0
17	AEF	3.60	0
18	AEF	3.60	0
19	AEF	3.60	0
20	AEF	3.60	4
21	AEF	3.60	8
22	AEF	3.60	12

III.1 PWR description

PWR (WESTINGHOUSE), 3 loops , 157 FA, power 2775. MWth

¼ CORE



Location of control rod banks

AVE. BURNUP PER FUEL ASSEMBLY

	1	2	3	4	5	6	7	8
1	18.137	11.662	27.397	**0.000**	30.867	**0.000**	14.984	11.662
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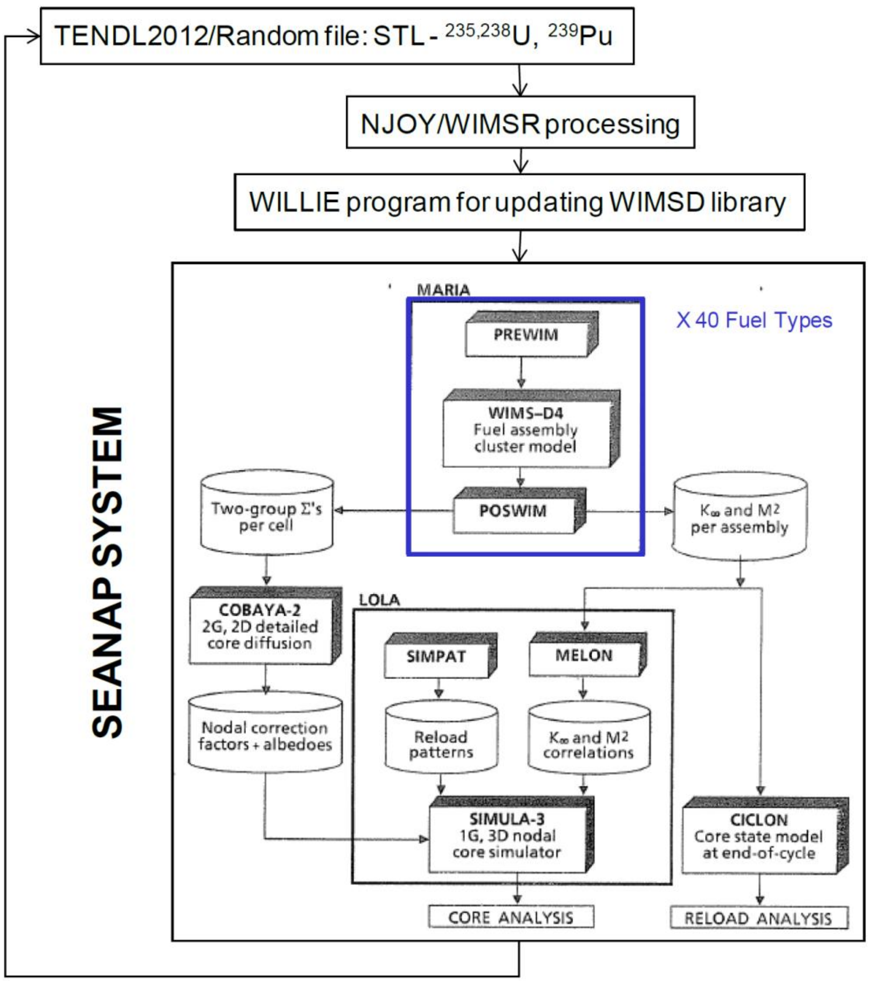
6	OFA	3.24	0
7	OFA	3.24	0
8	OFA	3.24	0
9	OFA	3.24	0
10	OFA	3.24	0
11	OFA	3.24	0
12	AEF	3.60	0
13	AEF	3.60	0
14	AEF	3.60	0
15	AEF	3.60	0
16	AEF	3.60	0
17	AEF	3.60	0
18	AEF	3.60	0
19	AEF	3.60	0
20	AEF	3.60	4
21	AEF	3.60	8
22	AEF	3.60	12

III.2 Methodology:
TMC with SEANAP System

Scheme of TMC for random nuclear data analysis:

- **TENDL2012** random file completely different from another one is processed with **NJOY**
- **WIMSR/NJOY** is needed to process this file in WIMSD format, and **WILLIE program** updates the WIMS-D4 library with the processed random file.
- This **random WIMSD library** includes a unique new material remaining the same for the rest of the library.
- **<u>MARIA sub-system is run for the 40 FA</u>** types generating the two-group cross-section and additional data for COBAYA and SIMULA codes.



III.3 U&Q of FAs: Reactivity at BOC, HFP and C_B=1348 ppm

#	Fuel Assembly	# Burnup absorbers/ Control Rods	Enrichment (w/o)	Burnup at BOC (GWD/TMU)	Average Kinf	Δk/k% due to:				
						STL	^{239}Pu	^{235}U	^{238}U	TOTAL
1	OFA	-	2.1	18147	0.91360	0.18	0.50	0.37	0.28	0.70
2	OFA	-	3.1	28827	0.89790	0.27	0.52	0.29	0.27	0.70
3	OFA	-	3.24	27054	0.93579	0.26	0.46	0.31	0.27	0.67
4	OFA	-	3.24	27519	0.96487	0.24	0.42	0.34	0.26	0.65
5	OFA	-	3.24	29887	0.96398	0.25	0.44	0.34	0.26	0.66
6	OFA	-	3.24	29340	0.96465	0.25	0.43	0.33	0.26	0.66
7	OFA	-	3.24	30232	0.96227	0.26	0.45	0.32	0.26	0.66
8	OFA	-	3.24	22908	0.98784	0.25	0.40	0.35	0.26	0.65
9	OFA	-	3.24	30456	0.94295	0.27	0.47	0.30	0.26	0.67
10	OFA	-	3.24	30053	0.96551	0.26	0.44	0.32	0.26	0.66
11	OFA	-	3.24	16273	1.09368	0.15	0.23	0.49	0.26	0.62
12	AEF	-	3.6	13050	1.13885	0.14	0.16	0.50	0.28	0.61
13	AEF	-	3.6	11577	1.14386	0.14	0.15	0.51	0.28	0.61
14	AEF	-	3.6	11695	1.12998	0.16	0.18	0.48	0.28	0.61
15	AEF	-	3.6	13263	1.13845	0.15	0.17	0.49	0.28	0.61
16	AEF	-	3.6	13285	1.13486	0.15	0.18	0.48	0.28	0.60
17	AEF	-	3.6	15024	1.11838	0.18	0.22	0.45	0.28	0.60
18	AEF	-	3.6	15233	1.12937	0.16	0.19	0.47	0.28	0.60
19	AEF	-	3.6	0	1.21566	0.09	0.00	0.63	0.27	0.69
20	AEF	4 WABAS	3.6	0	1.18455	0.12	0.00	0.64	0.27	0.71
21	AEF	8 WABAS	3.6	0	1.15216	0.15	0.00	0.65	0.27	0.72
22	AEF	12 WABAS	3.6	0	1.12014	0.18	0.00	0.66	0.27	0.73

➢ **Uncertainty due to ^{235}U decreases with Burnup**

III.3 U&Q of FAs:

Reactivity at BOC, HFP and C_B=1348 ppm

#	Fuel Assembly	# Burnup absorbers/ Control Rods	Enrichment (w/o)	Burnup at BOC (GWD/TMU)	Average Kinf	Δk/k% due to: STL	^{239}Pu	^{235}U	^{238}U	TOTAL
1	OFA	-	2.1	18147	0.91360	0.18	0.50	0.37	0.28	0.70
2	OFA	-	3.1	28827	0.89790	0.27	0.52	0.29	0.27	0.70
3	OFA	-	3.24	27054	0.93579	0.26	0.46	0.31	0.27	0.67
4	OFA	-	3.24	27519	0.96487	0.24	0.42	0.34	0.26	0.65
5	OFA	-	3.24	29887	0.96398	0.25	0.44	0.34	0.26	0.66
6	OFA	-	3.24	29340	0.96465	0.25	0.43	0.33	0.26	0.66
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9	OFA	-	3.24	30456	0.94295	0.27	0.47	0.30	0.26	0.67
10	OFA	-	3.24	30053	0.96551	0.26	0.44	0.32	0.26	0.66
11	OFA	-	3.24	16273	1.09368	0.15	0.23	0.49	0.26	0.62
12	AEF	-	3.6	13050	1.13885	0.14	0.16	0.50	0.28	0.61
13	AEF	-	3.6	11577	1.14386	0.14	0.15	0.51	0.28	0.61
14	AEF	-	3.6	11695	1.12998	0.16	0.18	0.48	0.28	0.61
15	AEF	-	3.6	13263	1.13845	0.15	0.17	0.49	0.28	0.61
16	AEF	-	3.6	13285	1.13486	0.15	0.18	0.48	0.28	0.60
17	AEF	-	3.6	15024	1.11838	0.18	0.22	0.45	0.28	0.60
18	AEF	-	3.6	15233	1.12937	0.16	0.19	0.47	0.28	0.60
19	AEF	-	3.6	0	1.21566	0.09	0.00	0.63	0.27	0.69
20	AEF	4 WABAS	3.6	0	1.18455	0.12	0.00	0.64	0.27	0.71
21	AEF	8 WABAS	3.6	0	1.15216	0.15	0.00	0.65	0.27	0.72
22	AEF	12 WABAS	3.6	0	1.12014	0.18	0.00	0.66	0.27	0.73

➢ **Uncertainty due to ^{239}Pu increases with Burnup**

III.3 U&Q of FAs: Reactivity at BOC, HFP and C_B=1348 ppm

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7	OFA	-	3.24	30232	0.96227	0.26	0.45	0.32	0.26	0.66
8	OFA	-	3.24	22908	0.98784	0.25	0.40	0.35	0.26	0.65
9	OFA	-	3.24	30456	0.94295	0.27	0.47	0.30	0.26	0.67
10	OFA	-	3.24	30053	0.96551	0.26	0.44	0.32	0.26	0.66
11	OFA	-	3.24	16273	1.09368	0.15	0.23	0.49	0.26	0.62
12	AEF	-	3.6	13050	1.13885	0.14	0.16	0.50	0.28	0.61
13	AEF	-	3.6	11577	1.14386	0.14	0.15	0.51	0.28	0.61
14	AEF	-	3.6	11695	1.12998	0.16	0.18	0.48	0.28	0.61
15	AEF	-	3.6	13263	1.13845	0.15	0.17	0.49	0.28	0.61
16	AEF	-	3.6	13285	1.13486	0.15	0.18	0.48	0.28	0.60
17	AEF	-	3.6	15024	1.11838	0.18	0.22	0.45	0.28	0.60
18	AEF	-	3.6	15233	1.12937	0.16	0.19	0.47	0.28	0.60
19	AEF	-	3.6	0	1.21566	0.09	0.00	0.63	0.27	0.69
20	AEF	4 WABAS	3.6	0	1.18455	0.12	0.00	0.64	0.27	0.71
21	AEF	8 WABAS	3.6	0	1.15216	0.15	0.00	0.65	0.27	0.72
22	AEF	12 WABAS	3.6	0	1.12014	0.18	0.00	0.66	0.27	0.73

➢ **Uncertainty due ^{238}U ~27% at different Burnup**

III.3 U&Q of FAs:

Reactivity at BOC, HFP and C_B=1348 ppm

#	Fuel Assembly	# Burnup absorbers/ Control Rods	Enrichment (w/o)	Burnup at BOC (GWD/TMU)	Average Kinf	Δk/k% due to:				
						STL	^{239}Pu	^{235}U	^{238}U	TOTAL
1	OFA	-	2.1	18147	0.91360	0.18	0.50	0.37	0.28	0.70
2	OFA	-	3.1	28827	0.89790	0.27	0.52	0.29	0.27	0.70
3	OFA	-	3.24	27054	0.93579	0.26	0.46	0.31	0.27	0.67
4	OFA	-	3.24	27519	0.96487	0.24	0.42	0.34	0.26	0.65
5	OFA	-	3.24	29887	0.96398	0.25	0.44	0.34	0.26	0.66
6	OFA	-	3.24	29340	0.96465	0.25	0.43	0.33	0.26	0.66
7	OFA	-	3.24	30232	0.96227	0.26	0.45	0.32	0.26	0.66
8	OFA	-	3.24	22908	0.98784	0.25	0.40	0.35	0.26	0.65
9	OFA	-	3.24	30456	0.94295	0.27	0.47	0.30	0.26	0.67
10	OFA	-	3.24	30053	0.96551	0.26	0.44	0.32	0.26	0.66
11	OFA	-	3.24	16273	1.09368	0.15	0.23	0.49	0.26	0.62
12	AEF	-	3.6	13050	1.13885	0.14	0.16	0.50	0.28	0.61
13	AEF	-	3.6	11577	1.14386	0.14	0.15	0.51	0.28	0.61
14	AEF	-	3.6	11695	1.12998	0.16	0.18	0.48	0.28	0.61
15	AEF	-	3.6	13263	1.13845	0.15	0.17	0.49	0.28	0.61
16	AEF	-	3.6	13285	1.13486	0.15	0.18	0.48	0.28	0.60
17	AEF	-	3.6	15024	1.11838	0.18	0.22	0.45	0.28	0.60
18	AEF	-	3.6	15233	1.12937	0.16	0.19	0.47	0.28	0.60
19	AEF	-	3.6	0	1.21566	0.09	0.00	0.63	0.27	0.69
20	AEF	4 WABAS	3.6	0	1.18455	0.12	0.00	0.64	0.27	0.71
21	AEF	8 WABAS	3.6	0	1.15216	0.15	0.00	0.65	0.27	0.72
22	AEF	12 WABAS	3.6	0	1.12014	0.18	0.00	0.66	0.27	0.73

➢ Unc. due STL depends on Burnup and Control

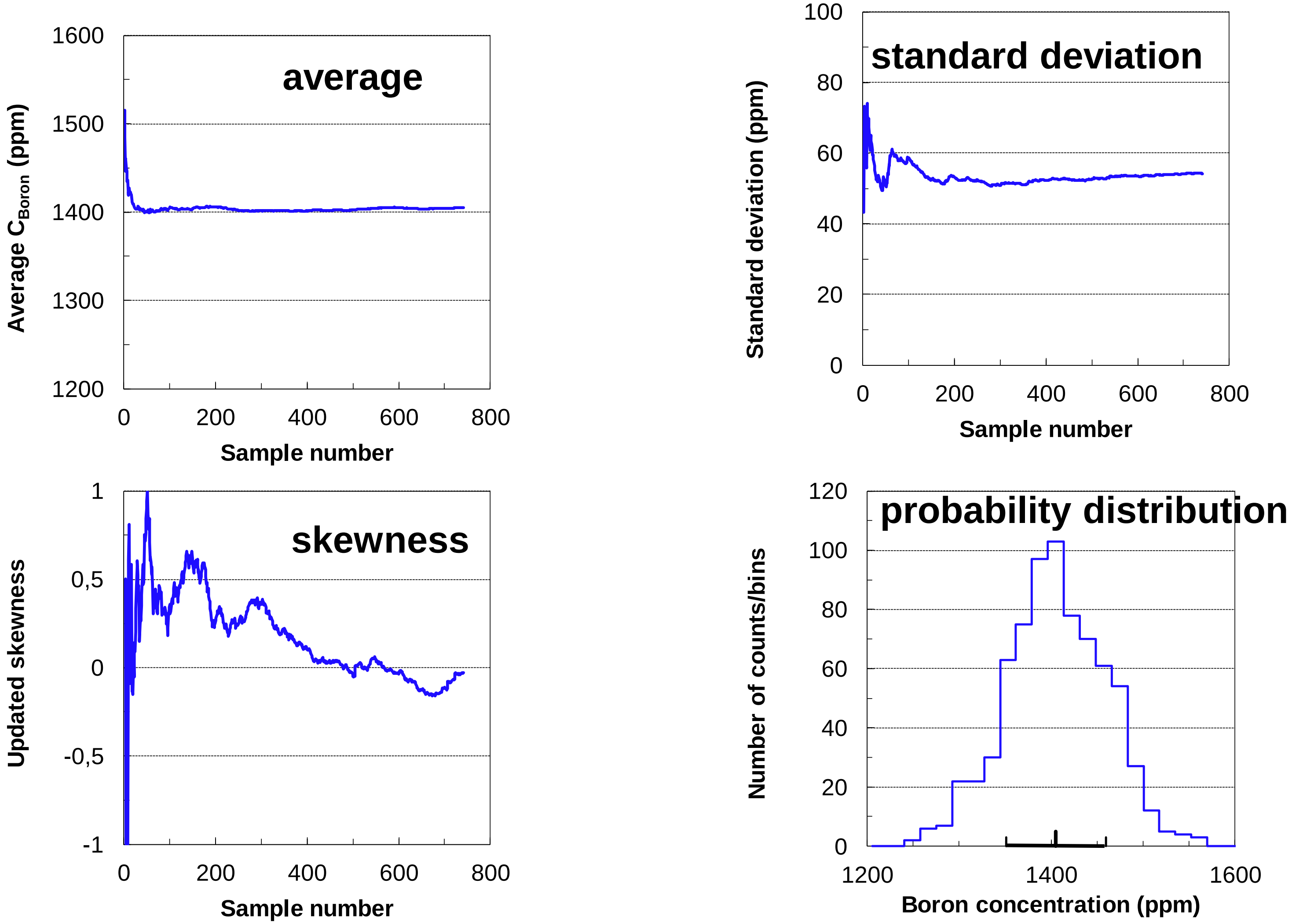
III.3 U&Q of FAs:

Reactivity at BOC, HFP and C_B=1348 ppm

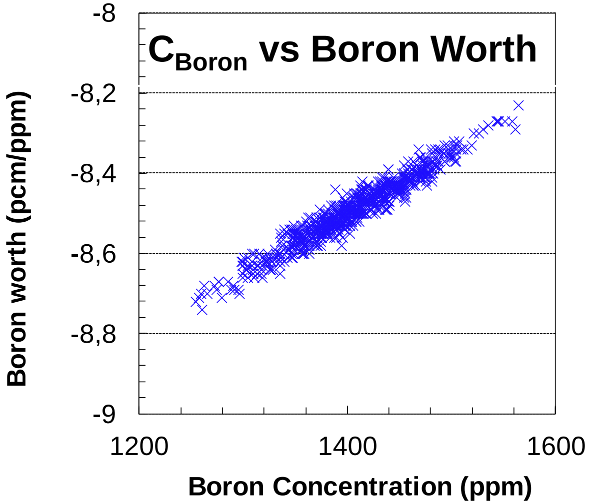
#	Fuel Assembly	# Burnup absorbers/ Control Rods	Enrichment (w/o)	Burnup at BOC (GWD/TMU)	Average Kinf	Δk/k% due to:				
						STL	^{239}Pu	^{235}U	^{238}U	TOTAL
23	OFA	CTL-Ag-In-Cd	3.24	27054	0.68538	0.63	0.48	0.30	0.27	0.90
24	OFA	CTL-Ag-In-Cd	3.24	27519	0.70843	0.62	0.45	0.33	0.27	0.88
25	OFA	CTL-Ag-In-Cd	3.24	29340	0.71012	0.62	0.45	0.32	0.27	0.88
26	OFA	CTL-Ag-In-Cd	3.24	22908	0.72835	0.62	0.43	0.34	0.27	0.87
27	OFA	CTL-Ag-In-Cd	3.24	30456	0.69436	0.64	0.49	0.29	0.27	0.89
28	OFA	CTL-Ag-In-Cd	3.24	16273	0.80665	0.56	0.25	0.48	0.26	0.82
29	AEF	CTL-Ag-In-Cd	3.6	13050	0.85273	0.53	0.18	0.49	0.28	0.80
30	AEF	CTL-Ag-In-Cd	3.6	13285	0.85183	0.54	0.19	0.47	0.28	0.79
31	AEF	CTL-Ag-In-Cd	3.6	0	0.90497	0.53	0.00	0.65	0.26	0.88
32	OFA	CTL-B4C	3.24	27054	0.62228	0.69	0.47	0.29	0.27	0.93
33	OFA	CTL-B4C	3.24	27519	0.64298	0.67	0.44	0.32	0.27	0.91
34	OFA	CTL-B4C	3.24	29340	0.64454	0.68	0.44	0.31	0.27	0.91
35	OFA	CTL-B4C	3.24	22908	0.66091	0.67	0.42	0.33	0.27	0.90
36	OFA	CTL-B4C	3.24	30456	0.63041	0.69	0.48	0.28	0.27	0.92
37	OFA	CTL-B4C	3.24	16273	0.73125	0.61	0.25	0.47	0.26	0.85
38	AEF	CTL-B4C	3.6	13050	0.77150	0.59	0.17	0.47	0.28	0.82
39	AEF	CTL-B4C	3.6	13285	0.77077	0.59	0.19	0.46	0.28	0.82
40	AEF	CTL-B4C	3.6	0	0.82126	0.59	0.00	0.63	0.26	0.90

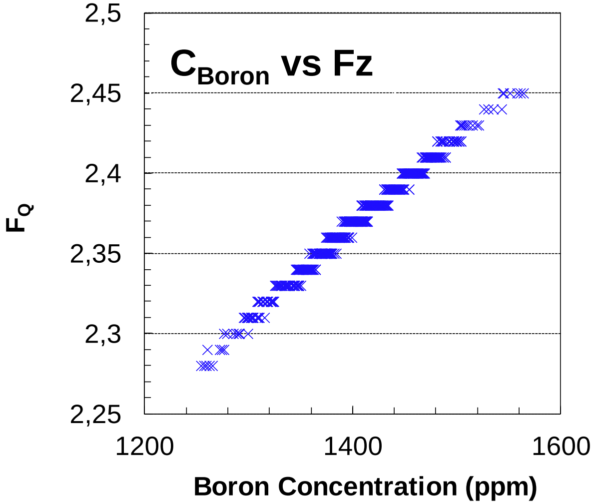
➢ **Large Unc. due STLs with Control Rods**

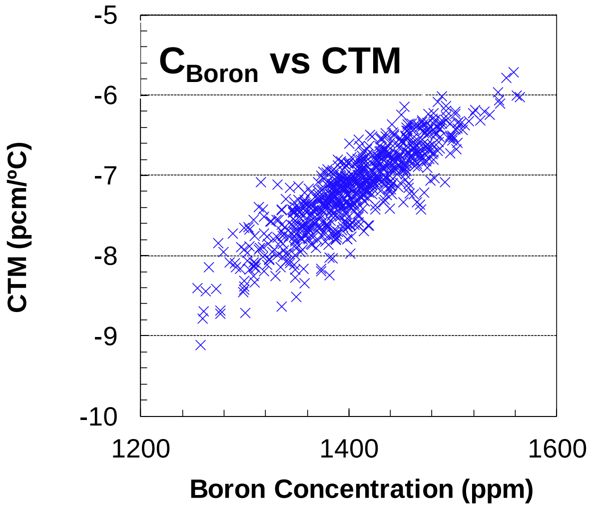
III.3 U&Q Core Analysis: Example of convergence for C_{Boron} (ppm) at BOC in the case of random ^{235}U nuclear data

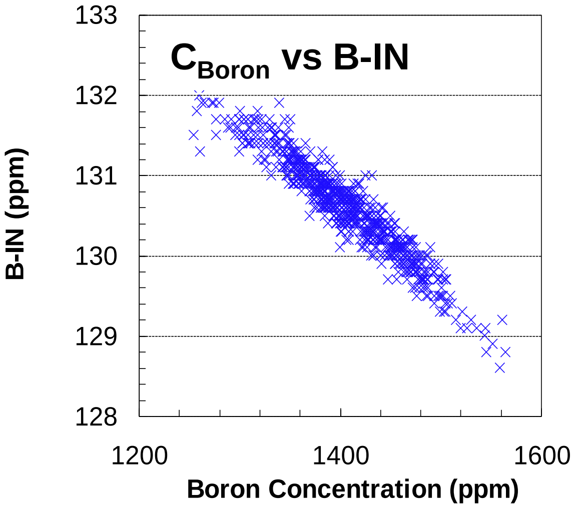


III.3 U&Q for Core Analysis: Example of correlations) at BOC, obtained by randomizing the ^{235}U transport nuclear data







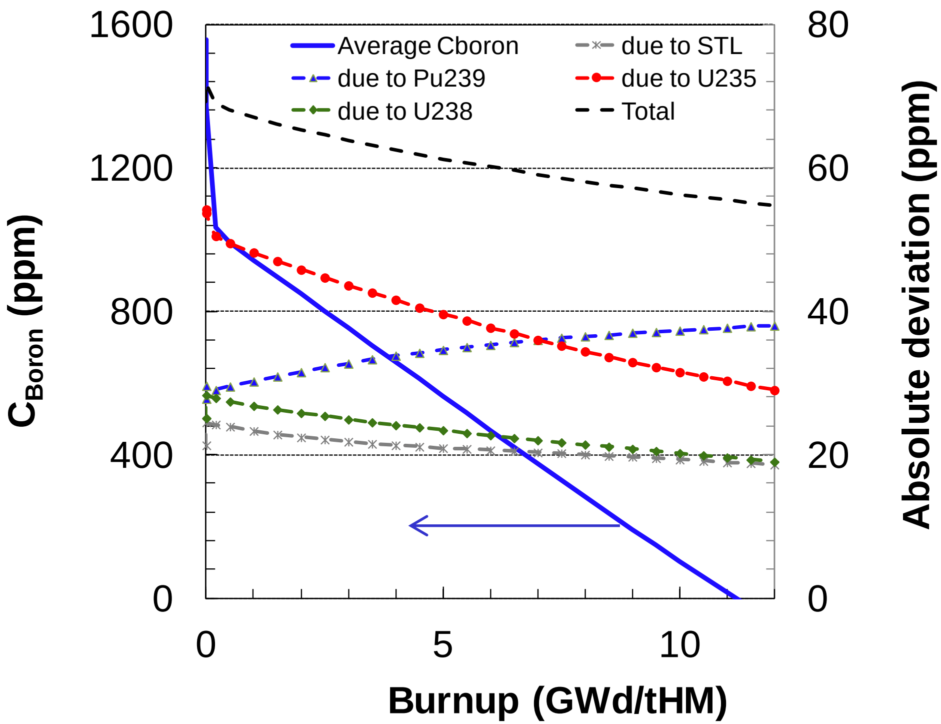


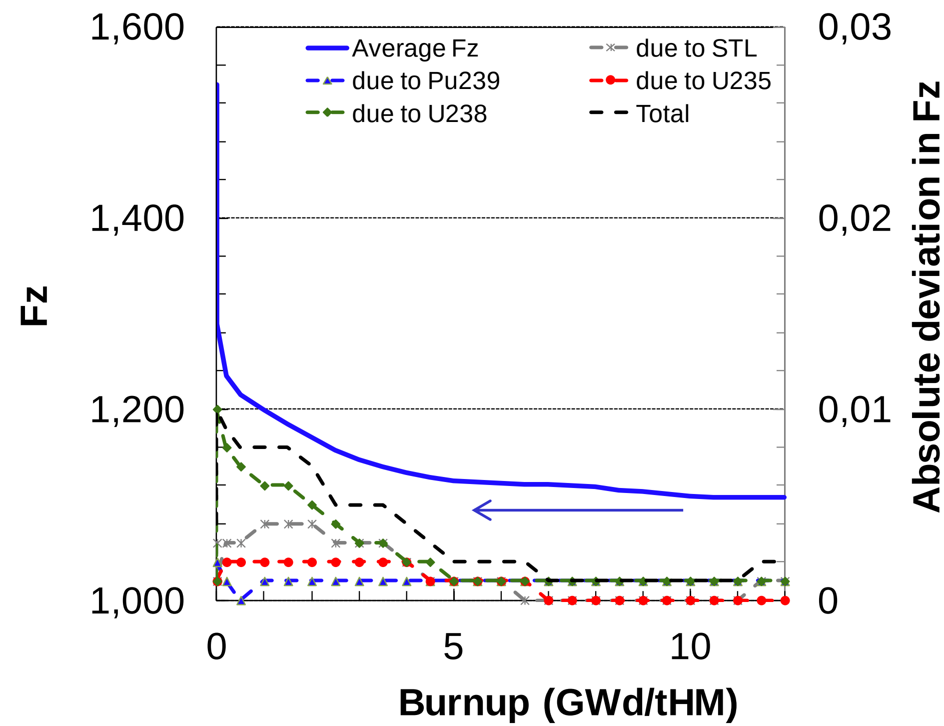
III.3 U&Q for Core Analysis: Average value and absolute deviation as a function of the burnup for control bank worth

	Average (ppm)	Absolute standard deviation (ppm)				
		STL	^{239}Pu	^{235}U	^{238}U	Total
D-IN	120.3	1.3	0.5	0.7	0.7	1.7
C-IN	92.2	0.8	2.6	1.4	1.1	3.2
B-IN	138.1	0.9	0.5	1.6	0.8	2.0
A-IN	92.3	0.5	3.5	3.9	0.7	5.3
SB-IN	88.9	1.1	3.3	2.4	1.4	4.5
SA-IN	120.3	0.8	2.3	3.2	0.6	4.0
D+C-IN	237.8	2.1	2.9	0.7	1.6	4.0
D+C+B-IN	419.2	3.5	4.5	1.0	2.4	6.2
D+C+B+A-IN	565.2	4.1	1.6	6.5	1.7	8.0
D+C+B+A+SB-IN	701.8	5.6	2.8	3.9	2.9	7.9
ARI	917.5	7.8	2.6	5.4	3.2	10.3

- Uncertainty for each of the nuclear data varied (STL-H in H2O, ^{235,238}U and ^{239}Pu) and the sum of the different contributions < ~**2%**

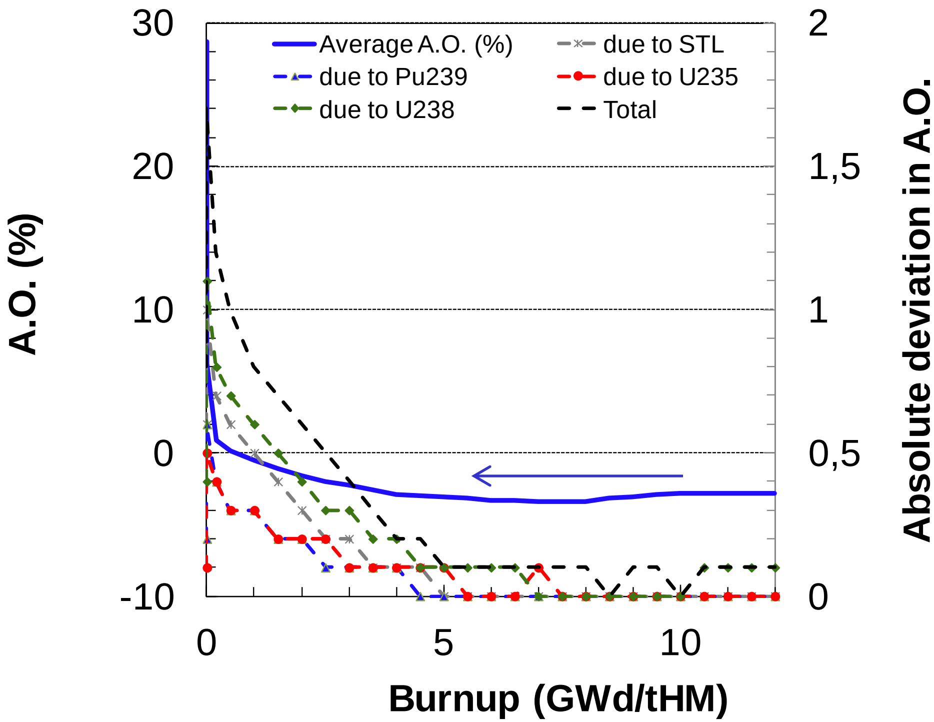
III.3 U&Q for Core Analysis: Average value and absolute deviation as a function of the burnup: C_{boron} and F_z

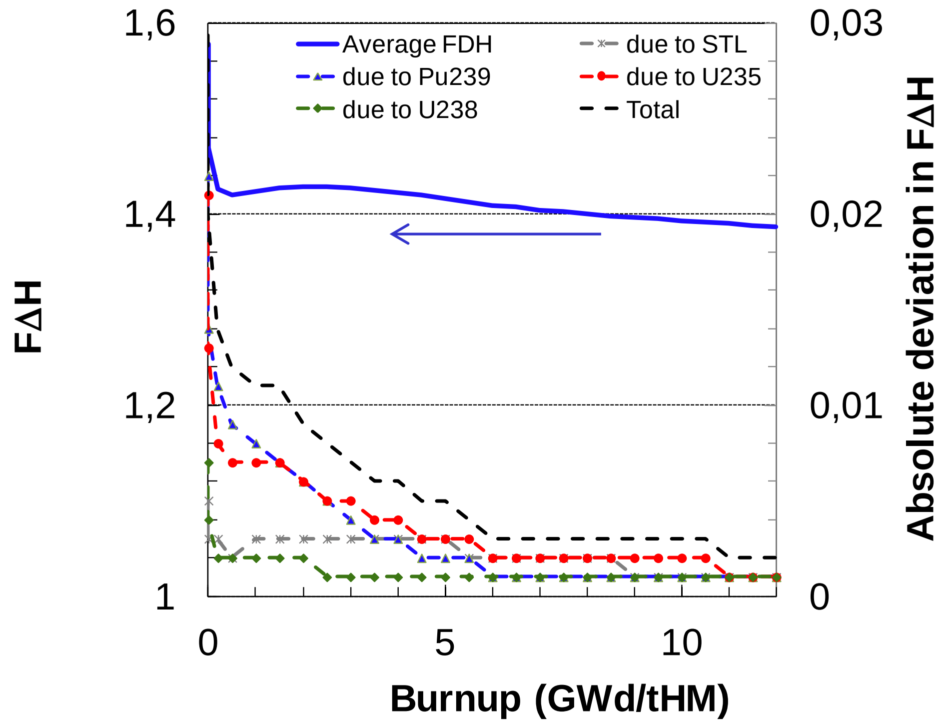




- C_{Boron} vs Burnup
  - Max unc.: 50-70 ppm
  - Most important contrib.: ^{235}U and ^{239}Pu
- F_z vs Burnup
  - Max unc.: 0.01-0.02 (~1-2%)
  - Most important contrib.: ^{238}U

III.3 U&Q for Core Analysis: Average value and absolute deviation as a function of the burnup: A.O. and $F_{\Delta H}$

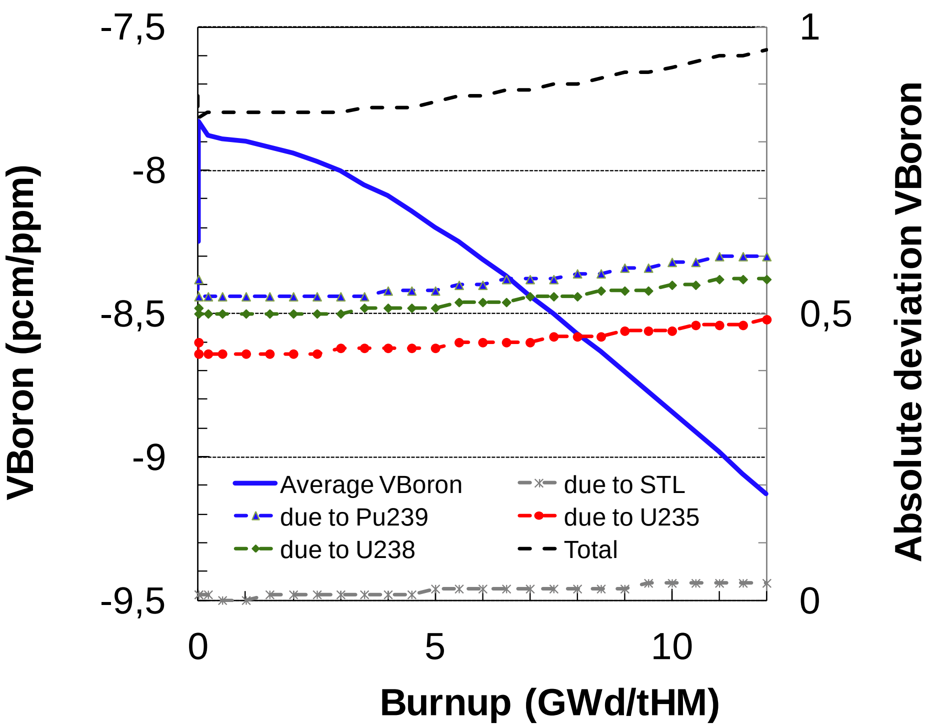


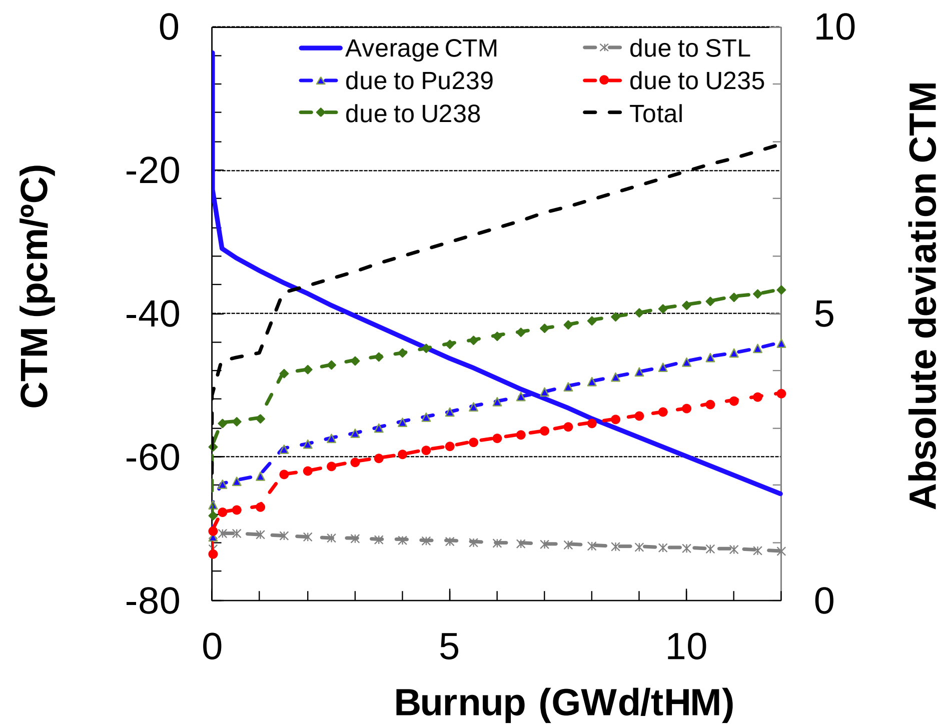


- **A.O. vs Burnup**
  - Max unc.: 1.5%
  - Most important contrib.: ^{238}U & STL

- **$F_{\Delta H}$ vs Burnup**
  - Max unc.: 0.01-0.02 (~1-2%)
  - Most important contrib.: ^{235}U & ^{239}Pu

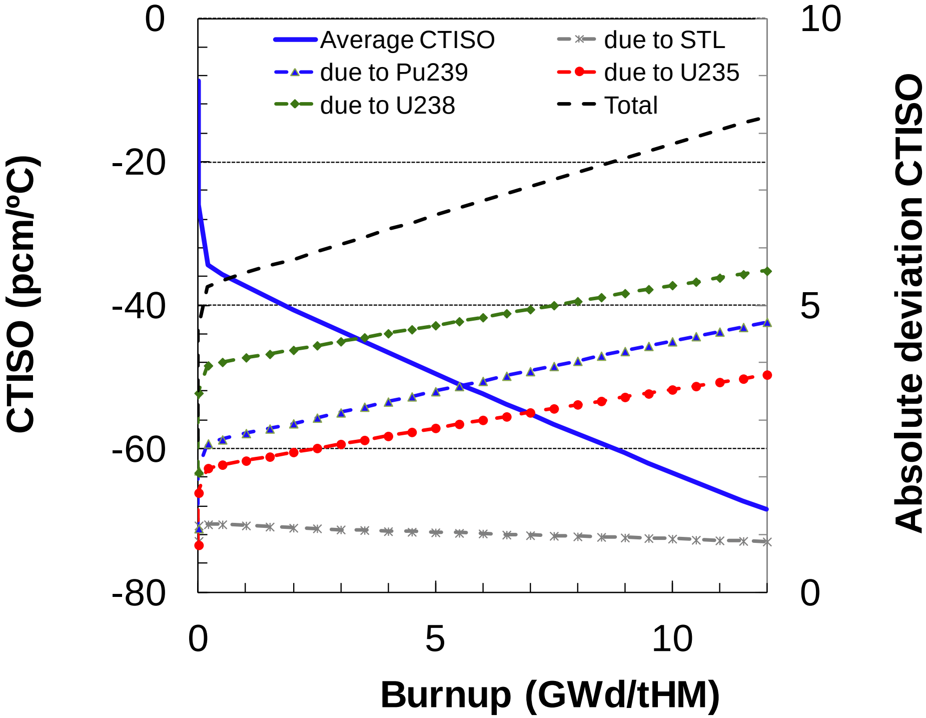
III.3 U&Q for Core Analysis: Average value and absolute deviation as a function of the burnup: V_{Boron} and CTM

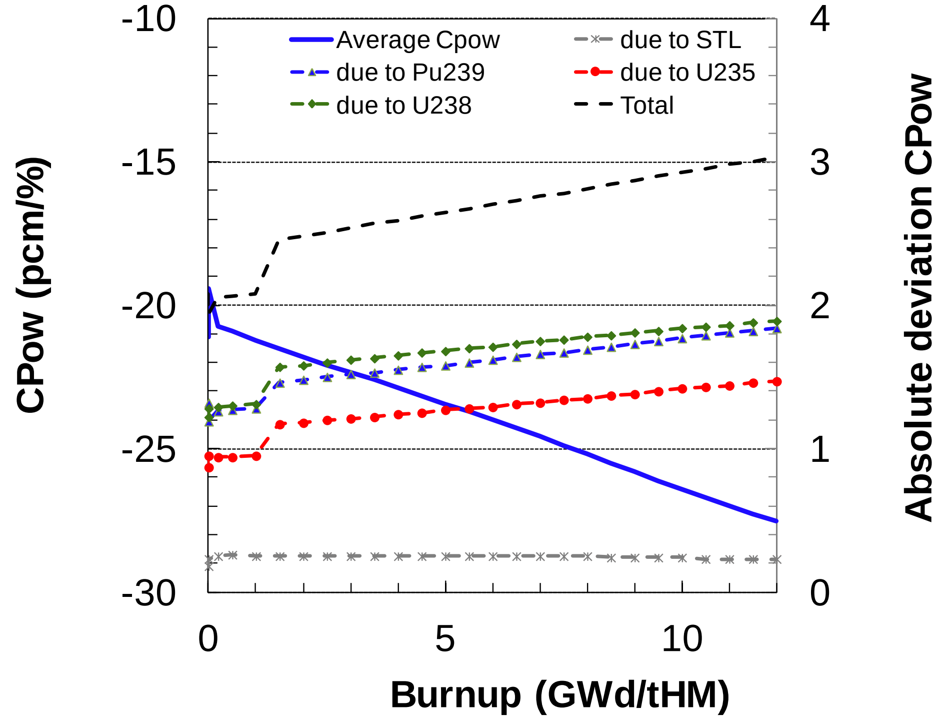




- **Boron Worth vs Burnup**
  - Max unc.: 1 pcm/ppm (~12%)
  - Most important contrib.: ^{235,238}U & $^{\mathbf{239}}$Pu
- **CTM vs Burnup**
  - Max unc.: 5-7pcm/ºC (~10%)
  - Most important contrib.: $^{235,\mathbf{238}}$U & ^{239}Pu

III.3 U&Q for Core Analysis: Average value and absolute deviation as a function of the burnup: CTISO and CPow





- **CTISO vs Burnup**
  - Max unc.: 5-8pcm/ºC (~10%)
  - Most important contrib.: $^{235,\mathbf{238}}U$ & ^{239}Pu

- **CPOW vs Burnup**
  - Max unc.: 2-3pcm/% (~10%)
  - Most important contrib.: $^{235,\mathbf{238}}U$ & ^{239}Pu

III.3 Design and acceptance criteria for start-up and operation: Calculated vs Measured

Core parameter	Design criteria	Acceptance criteria
Critical boron concentration ARO	$\|(C_B)^M_{ARO}-(C_B)^C_{ARO}\|$ <50 ppm	$\|\alpha C_B \times D(C_B)_{ARO}\|$< 1000 pcm
Isothermal temperature coefficient ARO at HZP	$\|(\alpha^{ISO}{}_T)^M_{ARO}-(\alpha^{ISO}{}_T)^C_{ARO}\|$ <3.6 pcm/ºC	$\|(\alpha^{ISO}{}_T)^M_{ARO}-(\alpha^{ISO}{}_T)^C_{ARO}\|$ <6.62 pcm/ºC
Moderator temperature coefficient ARO at HZP	$(\alpha^{CTM})^{HZP}_{ARO}$ <9 pcm/ºC	
Boron Worth Coefficient at HZP	$\| (\alpha C_B)^M-(\alpha C_B)^C \|$< 0.7 pcm/ppm	
Control banks worth for Reference Bank	$\| (I^{REF})^M-(I^{REF})^C \|$ <0.10x$(I^{REF})^C$	$\| (I^{REF})^M-(I^{REF})^C \|$ <0.15x$(I^{REF})^C$
Control Bank Worth value for other Banks using Rod Swap Technique	$\| (I^{CBW})^M-(I^{CBW})^C \|$ <0.15x$(I^{CBW})^C$ or 100 pcm	$\| (I^{CBW})^M-(I^{CBW})^C \|$ <0.30x$(I^{CBW})^C$ or 200 pcm
Total Control Bank Worth	1.10 x$(I^{TOT})^C$ > $(I^{TOT})^M$ >0.9x$(I^{TOT})^C$	$(I^{TOT})^M$ >0.9x $(I^{TOT})^C$
Axial Offset	$\| (AO)^M-(AO)^C \|$< 3%	
Max. Relative Assembly Power (P_A)	% $\| (P_A)^M-(P_A)^C \|/(P_A)^C$ { < 10% if P ≥90%; < 15% if P<90% }	

III.3 U&Q for Core Measurements: Measured boron concentrations (ppm) and calculated values versus cycle operation

	Burnup	Meas.	WIMS-D4		STL			Pu239		
Power (%)	(GWd/tHM)		C	M-C	C_Avg	UNC Abs. Dev.	M-C	C_Avg	UNC Abs. Dev.	M-C
50	0.015	1200	1141	59	1184	**22**	16	1149	**28**	51
75	0.031	1113	1062	51	1107	**23**	7	1070	**29**	43
100	0.134	985	990	-5	1035	**24**	-50	998	**29**	-13
100	1.34	870	883	-13	927	**24**	-57	892	**31**	-22
100	2.487	779	787	-8	830	**23**	-51	797	**33**	-18
100	2.842	755	758	-3	801	**23**	-46	768	**34**	-13
100	3.591	688	691	-3	732	**23**	-44	701	**35**	-13
100	4.441	604	617	-13	657	**23**	-53	627	**36**	-23
100	5.549	504	514	-10	552	**23**	-48	524	**38**	-20
100	6.692	412	405	7	443	**23**	-31	416	**40**	-4
100	7.716	319	305	14	341	**23**	-22	315	**41**	4
100	8.823	227	201	26	235	**23**	-8	211	**42**	16
100	10.284	101	57	44	90	**23**	11	68	**44**	33
100	11.351	4	-51	55	-19	**23**	23	-41	**45**	45

C= Calculated M= Measured

III.3 U&Q for Core Measurements: Measured Axial Offset (%) and calculated values versus cycle operation

	Burnup	Meas.	WIMS-D4		STL			Pu239		
Power (%)	(GWd/tHM)		C	M-C	C_Avg	UNC Abs. Dev.	M-C	C_Avg	UNC Abs. Dev.	M-C
50	0.015	7.7	4.8	2.9	5.7	**0.5**	2.0	4.9	**0.2**	2.8
75	0.031	3.8	2.7	1.1	3.8	**0.6**	0.0	2.9	**0.3**	0.9
100	0.134	-0.7	-0.7	0.0	0.7	**0.7**	-1.4	-0.5	**0.4**	-0.2
100	1.34	-1.6	-2.0	0.4	-1.2	**0.4**	-0.4	-1.9	**0.2**	0.3
100	2.487	-2.4	-3.0	0.6	-2.6	**0.2**	0.2	-3.0	**0.1**	0.6
100	2.842	-2.8	-3.0	0.2	-2.7	**0.2**	-0.1	-3.0	**0.1**	0.2
100	3.591	-3.8	-4.3	0.5	-4.1	**0.1**	0.3	-4.2	**0.1**	0.4
100	4.441	-3.2	-3.2	0.0	-3.1	**0.1**	-0.1	-3.2	**0.0**	0.0
100	5.549	-3.9	-3.7	-0.2	-3.7	**0.0**	-0.2	-3.7	**0.0**	-0.2
100	6.692	-4.2	-3.8	-0.4	-3.8	**0.0**	-0.4	-3.8	**0.0**	-0.4
100	7.716	-4.7	-4.3	-0.4	-4.4	**0.0**	-0.3	-4.3	**0.0**	-0.4
100	8.823	-3.6	-2.4	-1.2	-2.4	**0.0**	-1.2	-2.4	**0.0**	-1.2
100	10.284	-3.5	-1.6	-1.9	-1.6	**0.0**	-1.9	-1.6	**0.0**	-1.9
100	11.351	-3.4	-2.0	-1.4	-2.1	**0.0**	-1.3	-2.0	**0.0**	-1.4

C= Calculated M= Measured

III.3 U&Q for Core Measurments: Relative percentage assembly power calculated as: ((M-C) /C ·100)

- Relative percentage assembly power core distribution between measured (M) and calculated (C) **at BOC-HFP and Xenon equilibrium**
- Relative error in % for random cases (STL and ^{239}Pu) is provided
  - Low unc. due to STLs < 0.3%
  - Max. unc. due to ^{239} Pu < 1.8%

5.3 5.9±0.1 4.8±1.8	3.7 4.1±0.2 3.3±1.4	4.9 5.6±0.2 4.5±1.3	-0.6 0.0±0.2 -0.7±0.4	1.9 2.2±0.0 1.8±0.4	-1.8 -1.8±0.0 -1.6±0.6	-0.3 -0.9±0.1 -0.2±0.6	1.9 1.6±0.0 2.1±0.7
3.4 3.8±0.2 3.0±1.4	2.2 2.6±0.1 1.7±1.4	2.8 3.3±0.2 2.5±1.1	4.6 5.2±0.2 4.3±0.8	1.0 1.1±0.1 1.0±0.2	0.5 0.4±0.0 0.6±0.2	-1.3 -1.6±0.1 -0.9±1.1	3.4 3.1±0.0 3.4±0.6
4.0 4.7±0.2 3.6±1.3	2.2 2.6±0.2 1.8±1.1	4.6 5.2±0.1 4.4±1.0	-1.6 -1.2±0.1 -1.6±0.2	0.0 -0.1±0.1 -0.1±0.2	-1.6 -1.9±0.1 -1.3±0.8	-0.4 -0.7±0.1 0.0±1.3	
-1.7 -1.1±0.2 -1.8±0.4	3.5 4.1±0.2 3.2±0.8	-1.9 -1.6±0.1 -1.9±0.2	0.0 0.0±0.1 -0.2±0.5	-1.4 -1.9±0.2 -1.5±0.1	-2.0 -2.3±0.2 -1.7±0.8	2.5 2.0±0.0 2.5±0.7	
0.4 0.8±0.0 0.3±0.4	-0.4 -0.3±0.1 -0.5±0.2	-1.0 -1.0±0.1 -1.1±0.2	-2.3 -2.7±0.2 -2.4±0.1	-1.8 -2.4±0.3 -1.9±0.1	-0.7 -1.2±0.2 -0.7±0.2		
-3.1 -3.1±0.0 -2.9±0.6	-1.0 -1.2±0.0 -1.0±0.2	-2.9 -3.2±0.1 -2.6±0.8	-2.6 -2.9±0.2 -2.3±0.8	-1.0 -1.5±0.3 -1.0±0.2			
-0.8 -1.4±0.1 -0.7±0.6	-2.0 -2.2±0.1 -1.6±1.1	-1.5 -1.6±0.1 -1.0±1.3	0.5 0.2±0.0 0.7±0.7				
1.6 1.2±0.0 1.9±0.7	3.1 3.1±0.0 3.4±0.6						

WIMS-D4
Random STL
Random ^{239}Pu

Fresh Fuel

C= Calculated M= Measured

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Summary and conclusions

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Summary and conclusions

- The current methodology to predict the **calculational uncertainty** in a core analysis system is based on an extensive validation of the calculated results (using a computational code and its nuclear data) and the measured and design data at cycle start-up tests and nominal operation.
- However**, this calculational uncertainty ignores the term due to the uncertainty in NDs**:
  - We have analyzed the uncertainty of some safety core design parameters for a typical PWR in terms of the uncert. due to nuclear data in **^{235,238}U, ^{239}Pu and STL-H in H_2O**
  - To perform this uncertainty propagation study, a **Monte Carlo method** is applied, repeating similar calculations using a set of TENDL2012 random nuclear data files
  - Since **global uncertainties are within the design and acceptable criteria**, it can be concluded that calculation uncertainties due to nuclear data ensure bounding estimates in safety margins
  - However, **high uncertainties of 50-60 ppm in the boron concentration**, close to the design criteria, suggest that nuclear data uncertainties (for fission reactions of ^{235}U and ^{239}Pu) in TENDL2012 are very high compared with other uncertainty libraries, such as SCALE6/UN

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