Evolution of neutronic parameters vs burnup for the Moroccan TRIGA RR

Bilal El Bakkari

Experiments Manager & Reactor Manager Assistant

Reactor Operating Unit – CNESTEN

(bakkari@cnesten.org.ma)

Contents

1. Introduction

The changes in the proprieties of a nuclear research reactor over its life-time are determined by the changes in composition due to fuel burnup and the manner in which these are compensated.

The promotion of utilization of a research reactor is strongly affected by the efficiency of fuel utilization.

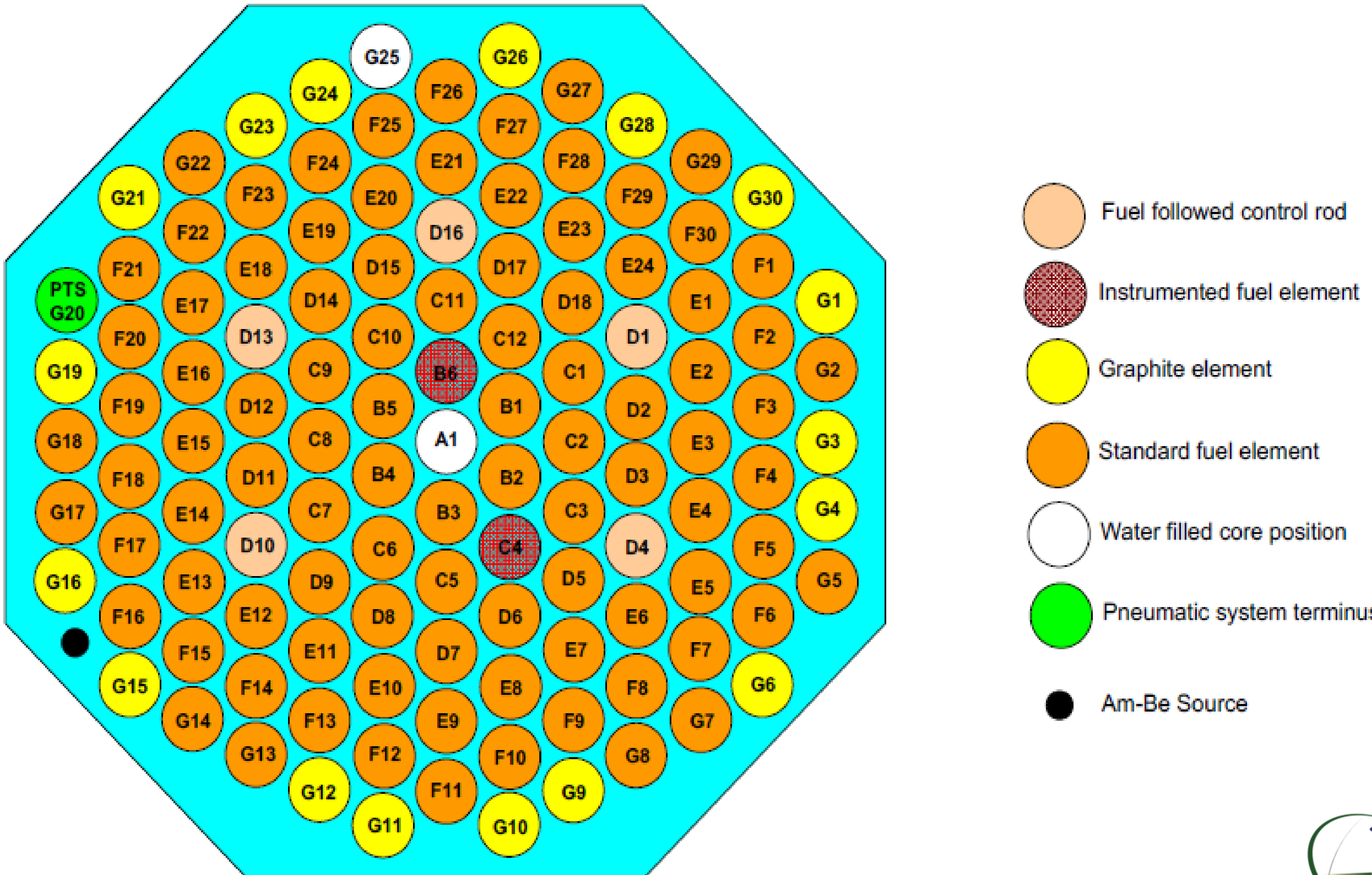
A burnup dependent neutronic analysis of the core parameters of the Moroccan TRIGA research reactor is performed.

2. Moroccan TRIGA RR

The TRIGA MARK II research reactor was commissioned at Centre d'Etudes Nucléaire de la Maâmora, Rabat, Morocco in 2006 and it went critical on *2 May 2007*.

The reactor is designated to implement the various fields of basic nuclear research, manpower training and production of radio-isotopes for their use in medecin, industry and agriculture.

The reactor is a light water cooled, graphite-reflected one, designated for operation at a steady-state power level of 2MW (thermal) under naturel convection mode.


G25 G26
G24 F26 G27
G23 F25 F27 G28
G22 F24 E21 F28 G29
G21 F23 E20 E22 F29 G30
F22 E19 D16 E23 F30
F21 E18 D15 D17 E24 F1
PTS G20 E17 D14 C11 D18 E1 G1
F20 D13 C10 C12 D1 F2
G19 E16 C9 B6 C1 E2 G2
F19 D12 B5 B1 D2 F3
G18 E15 C8 A1 C2 E3 G3
F18 D11 B4 B2 D3 F4
G17 E14 C7 B3 C3 E4 G4
F17 D10 C6 C4 D4 F5
G16 E13 D9 C5 D5 E5 G5
F16 E12 D8 D6 E6 F6
F15 E11 D7 E7 F7
G15 F14 E10 E8 F8 G6
G14 F13 E9 F9 G7
G13 F12 F10 G8
G12 F11 G9
G11 G10
Fuel followed control rod
Instrumented fuel element
Graphite element
Standard fuel element
Water filled core position
Pneumatic system terminus
Am-Be Source

TRIGA Fuel

Uranium-Zirconium Hydride U-Zr $H_{1.6}$

Part of U in Fuel [wt%] : 8.5

Enrichment in U235 [wt%] : 20

2 cylindrical graphite sections

Zirconium rod

Molybdenum disc

SS-304 cladding

Central thimble

The reactor is equipped with a central thimble for access to the point of maximum flux in the core.

The central thimble consists of an aluminium tube that fits through the centre hole of the top and bottom grid plates

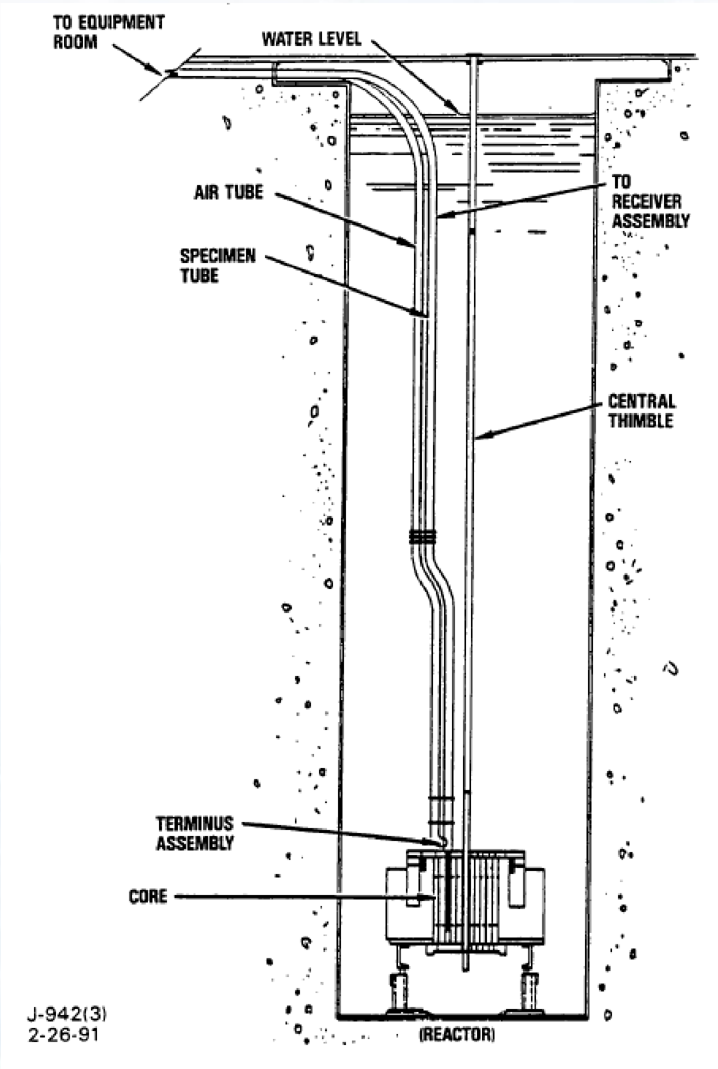
Experiments with the central thimble include irradiations of small samples and the exposure of materials to a collimated beam of neutrons or gamma rays.

Pneumatic transfer system in-core terminus

A pneumatic transfer system, permits irradiation of short-lived radioisotopes.

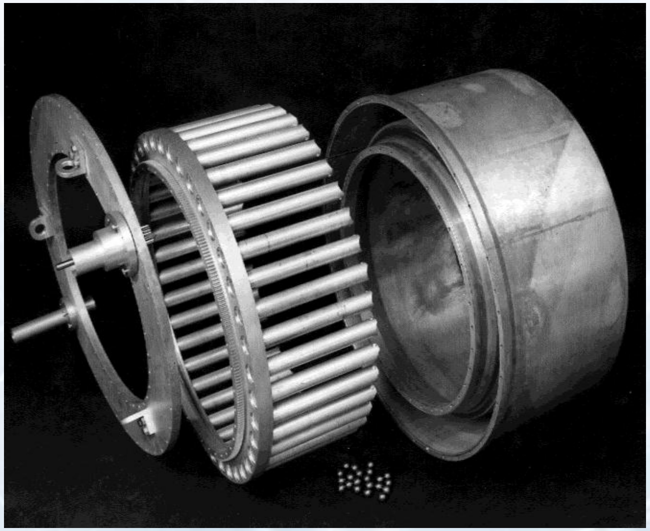
The in-core terminus of this system is normally located in the outer ring of fuel element positions, a region of high neutron flux.

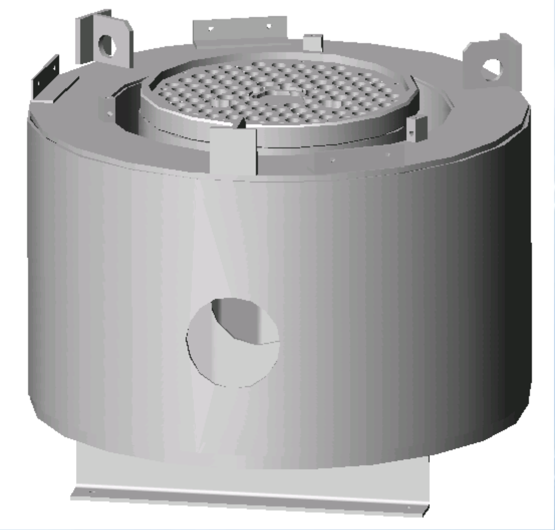
The sample capsule (rabbit) is conveyed to a receiver-sender station via 3.18 cm o.d. aluminium tubing. Effective space in the specimen transfer capsules is 1.7 cm diameter by 11.4 cm height.


TO EQUIPMENT ROOM
WATER LEVEL
AIR TUBE
SPECIMEN TUBE
TO RECEIVER ASSEMBLY
CENTRAL THIMBLE
TERMINUS ASSEMBLY
CORE
J-942(3)
2-26-91
(REACTOR)

Rotary specimen rack

A rotary, multiple-position (40) specimen rack located in a well in the top of the graphite reflector provides for the large-scale production of radioisotopes and for the activation and irradiation of multiple samples.





Neutron beam ports

Four beam ports (NB1, NB2, NB3 and NB4) are provided around the circumference of the reactor.

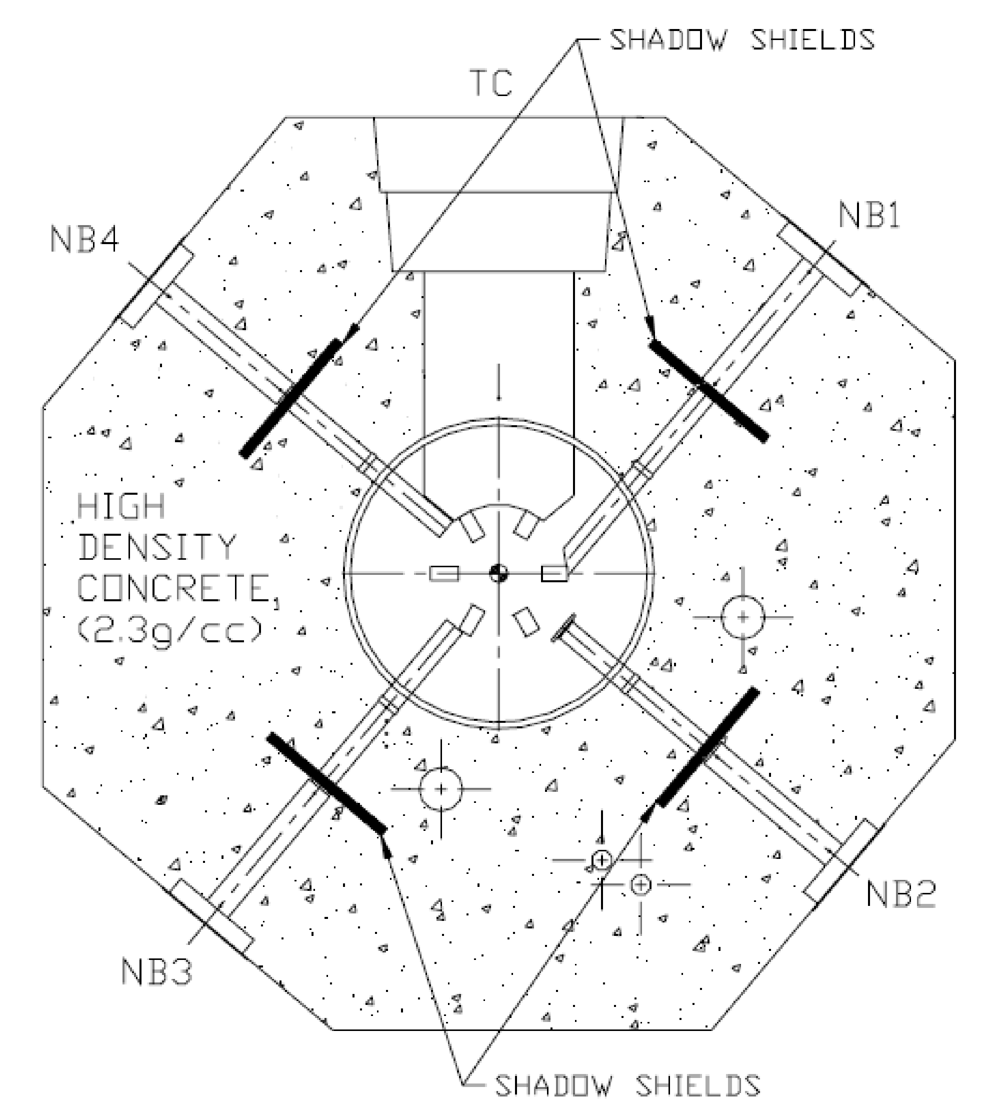
Beam port NB1 is located tangential to the reactor core edge, while the remaining beam ports NB2, NB3 and NB4 are positioned radial to the core centreline.

The beam ports are spaced approximately 90 degrees apart, with their axes slightly below the reactor core centre line elevation.

Thermal Column

The thermal column, is a 1219-mm square assembly, located in the side of the reactor shield structure, which facilitates irradiation of large experimental specimens. It is located between beam ports NB1 and NB4.

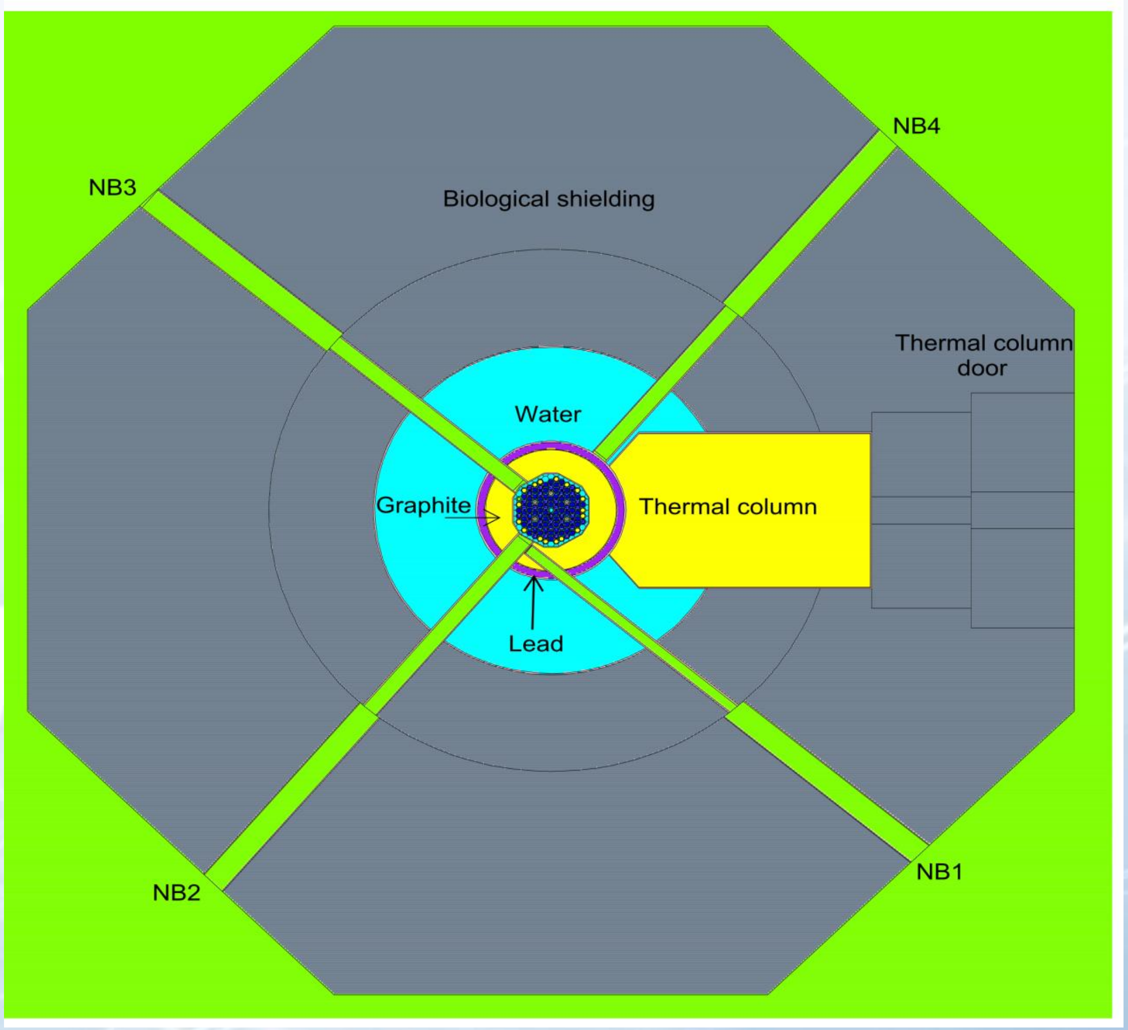
The thermal column consists of three sections. The inner section, which is an integral part of the reactor tank, and a middle section and door enclosure, which are integral to the reactor shield structure. All three sections are aligned on a common axis with the centre of the reactor core.

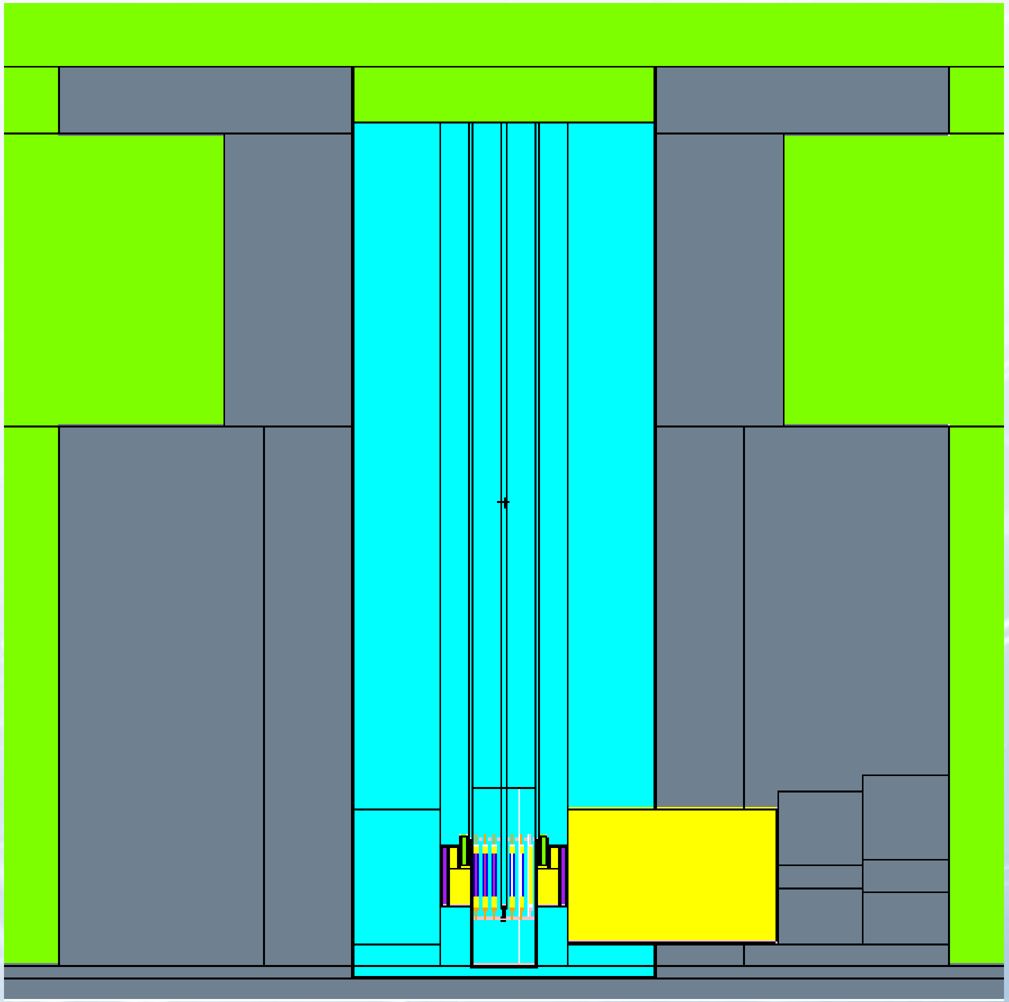

SHADOW SHIELDS
TC
NB1
NB4
HIGH
DENSITY
CONCRETE
(2.3g/cc)
NB2
NB3
SHADOW SHIELDS

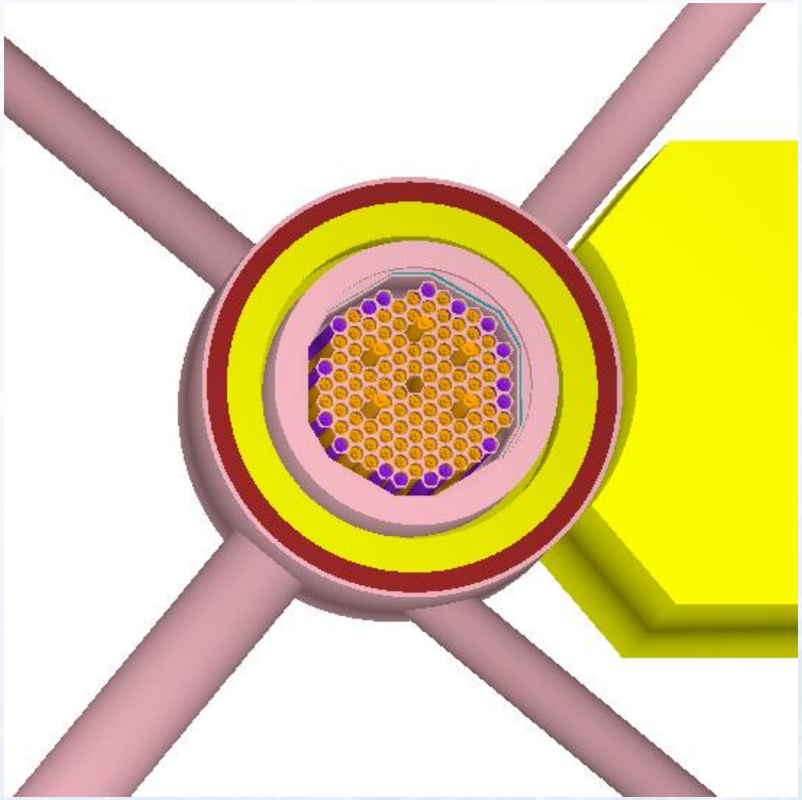
3. Modeling of TRIGA reactor

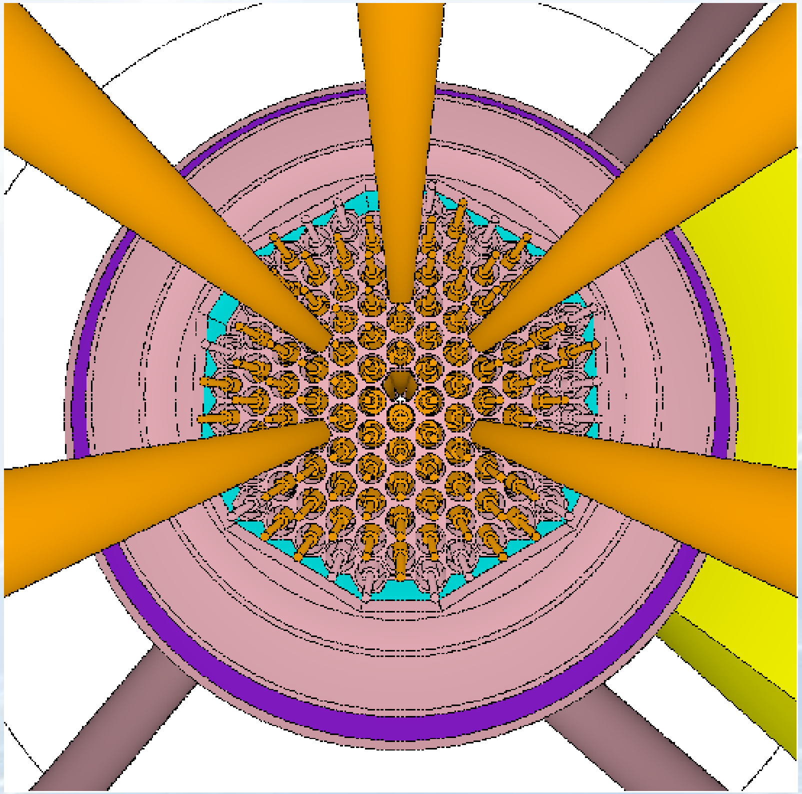
3-D continuous energy Monte Carlo code MCNP5 was used to develop a versatile and accurate full-model of the TRIGA reactor.

The model represents all components of the reactor using the maximum data allowed by GA.


NB4
NB3
Biological shielding
Thermal column
door
Water
Graphite
Thermal column
Lead
NB2
NB1





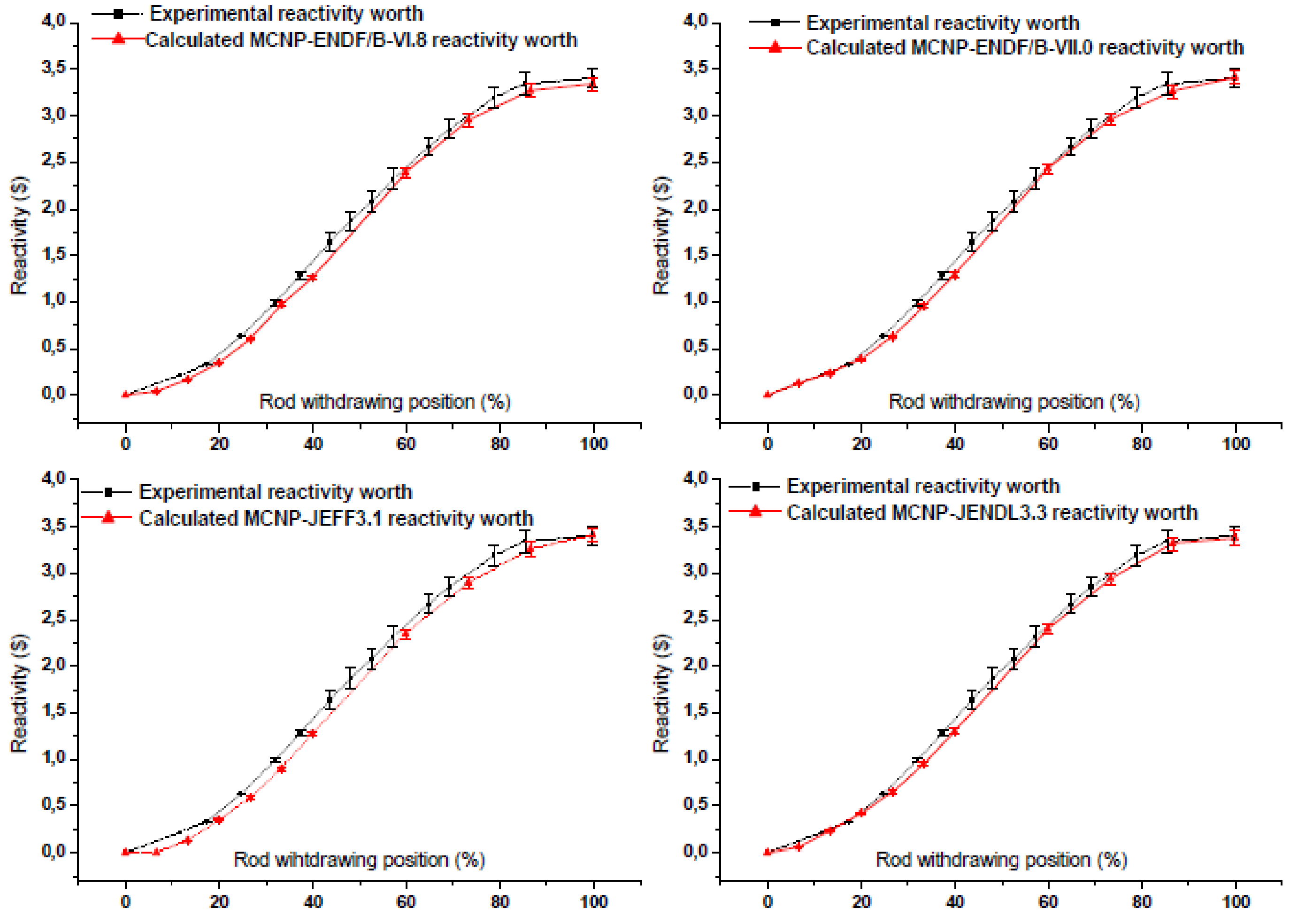


The developed MCNP model of the CENM TRIGA RR was validated by the caculation of several neutronics parameters.

	Core excess reactivity ($)	C/E
Measurement	**10.27**	–
ENDF/B-VII	**10.55**	**1.027**
ENDF/B-VI.8	**10.14**	**0.987**
JEFF-3.1	**10.03**	**0.977**
JENDL-3.3	**10.02**	**0.975**

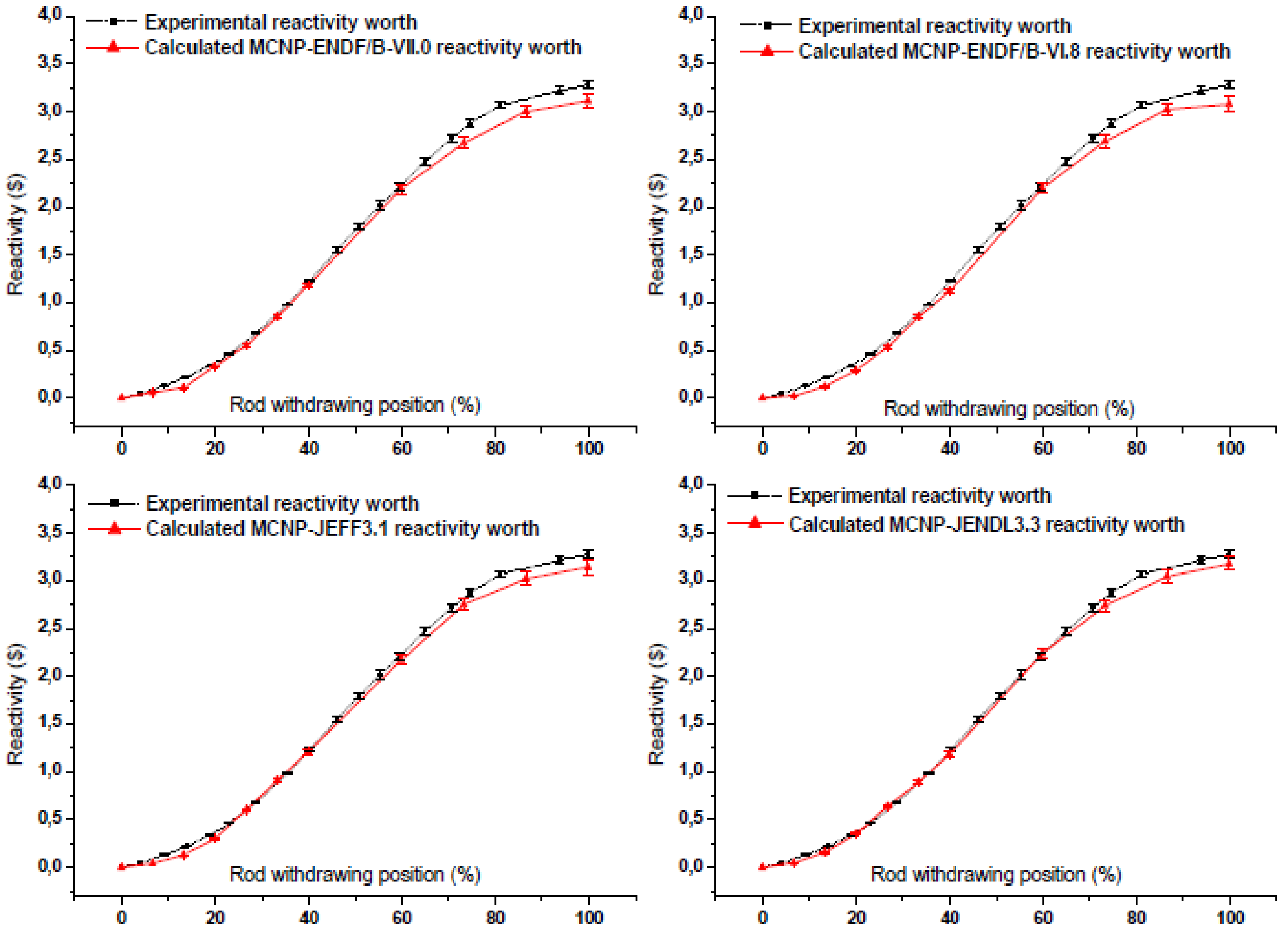
	C/E			
Control rods	ENDF/B-VII	ENDF/B-VI.8	JEFF-3.1	JENDL-3.3
Shim I	1.00	0.98	1.00	0.99
Shim II	0.93	0.92	0.94	0.95
Shim III	1.02	1.02	1.04	1.02
Shim IV	0.94	0.94	0.93	0.92
Regulating	0.96	0.95	0.94	0.97

Shim I



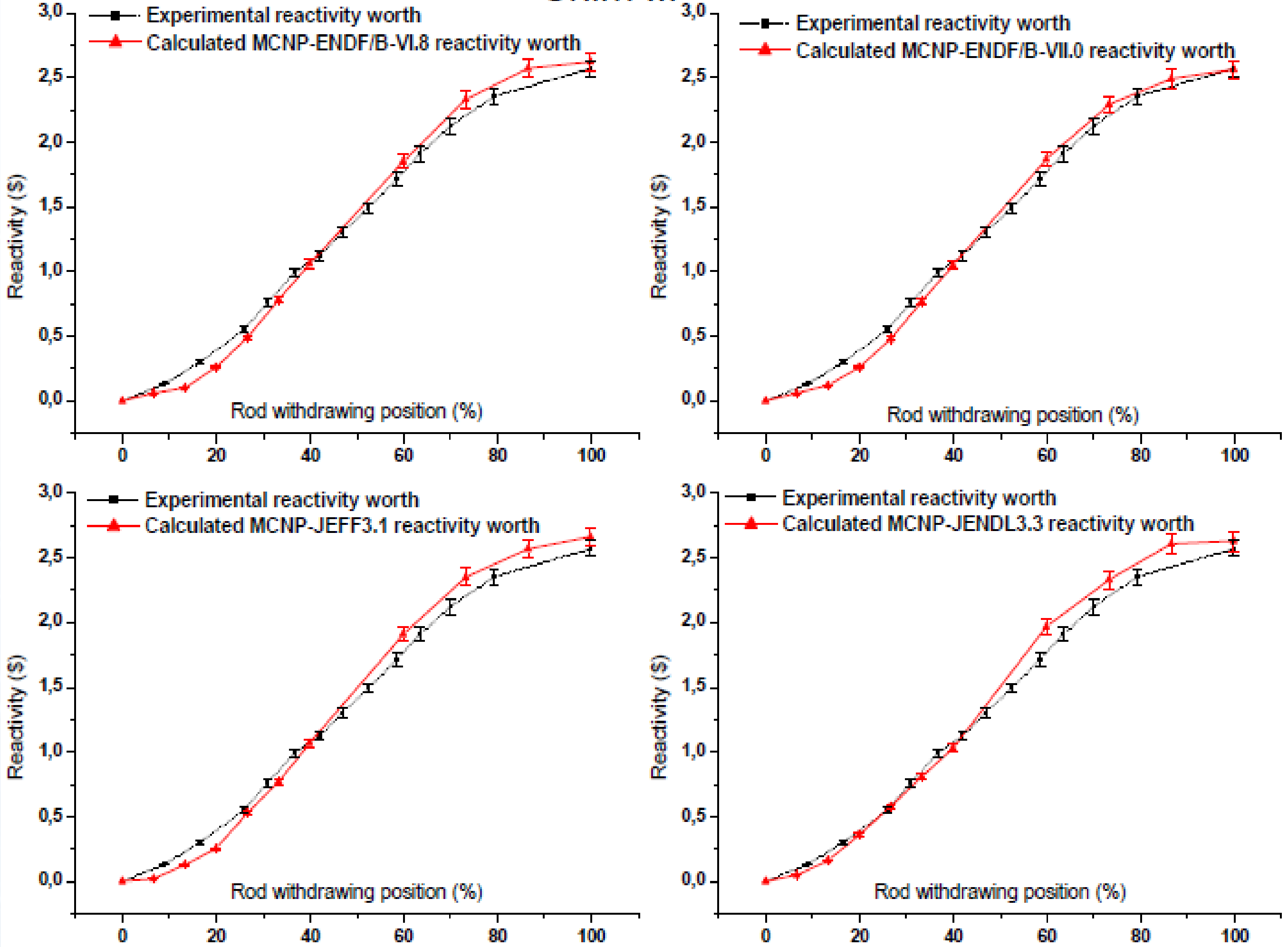
Experimental reactivity worth
Calculated MCNP-ENDF/B-VI.8 reactivity worth
Experimental reactivity worth
Calculated MCNP-ENDF/B-VII.0 reactivity worth
Experimental reactivity worth
Calculated MCNP-JEFF3.1 reactivity worth
Experimental reactivity worth
Calculated MCNP-JENDL3.3 reactivity worth
Reactivity ($)
Rod withdrawing position (%)
Rod wihtdrawing position (%)

Shim II

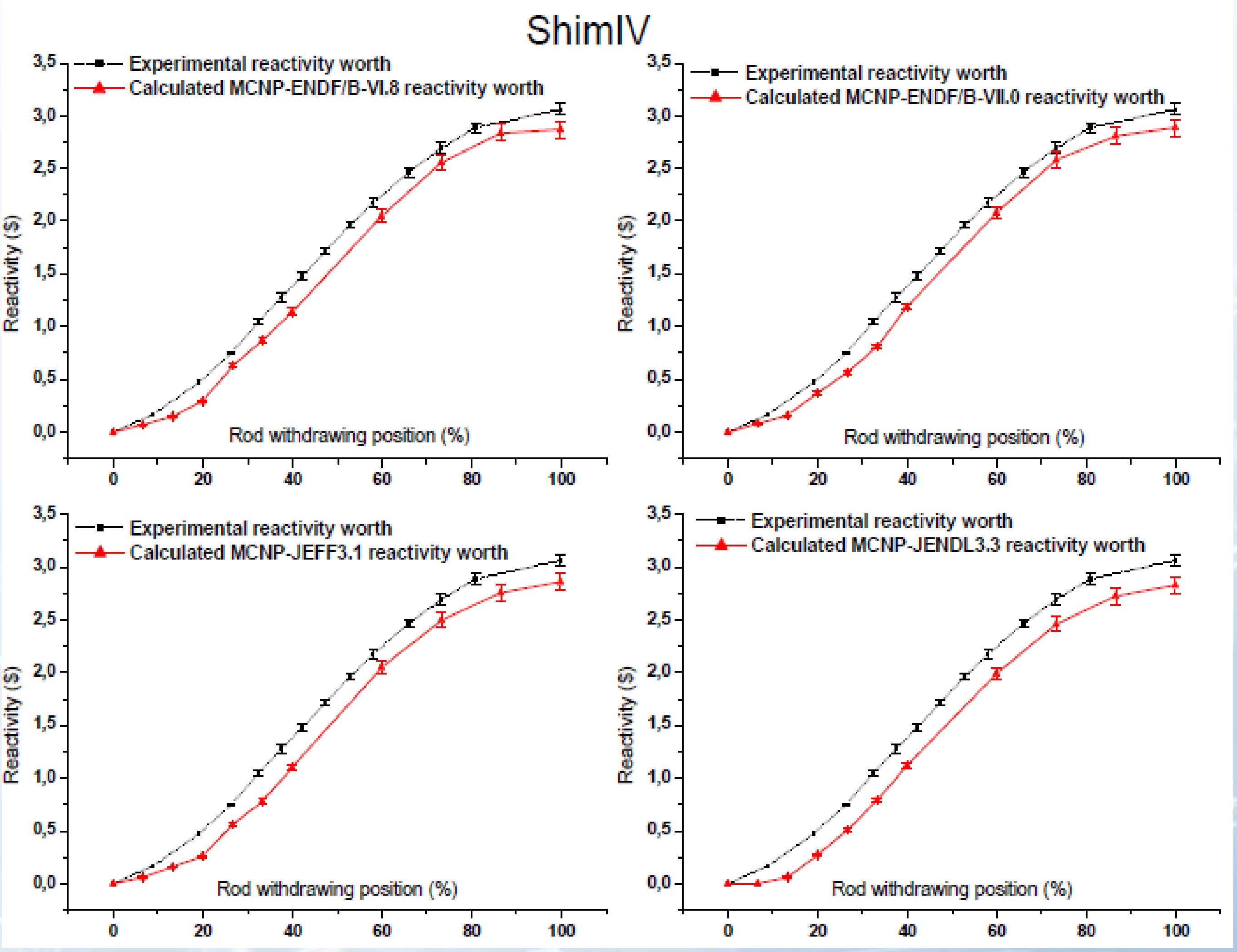


Experimental reactivity worth
Calculated MCNP-ENDF/B-VII.0 reactivity worth
Experimental reactivity worth
Calculated MCNP-ENDF/B-VI.8 reactivity worth
Experimental reactivity worth
Calculated MCNP-JEFF3.1 reactivity worth
Experimental reactivity worth
Calculated MCNP-JENDL3.3 reactivity worth
Reactivity ($)
Rod withdrawing position (%)

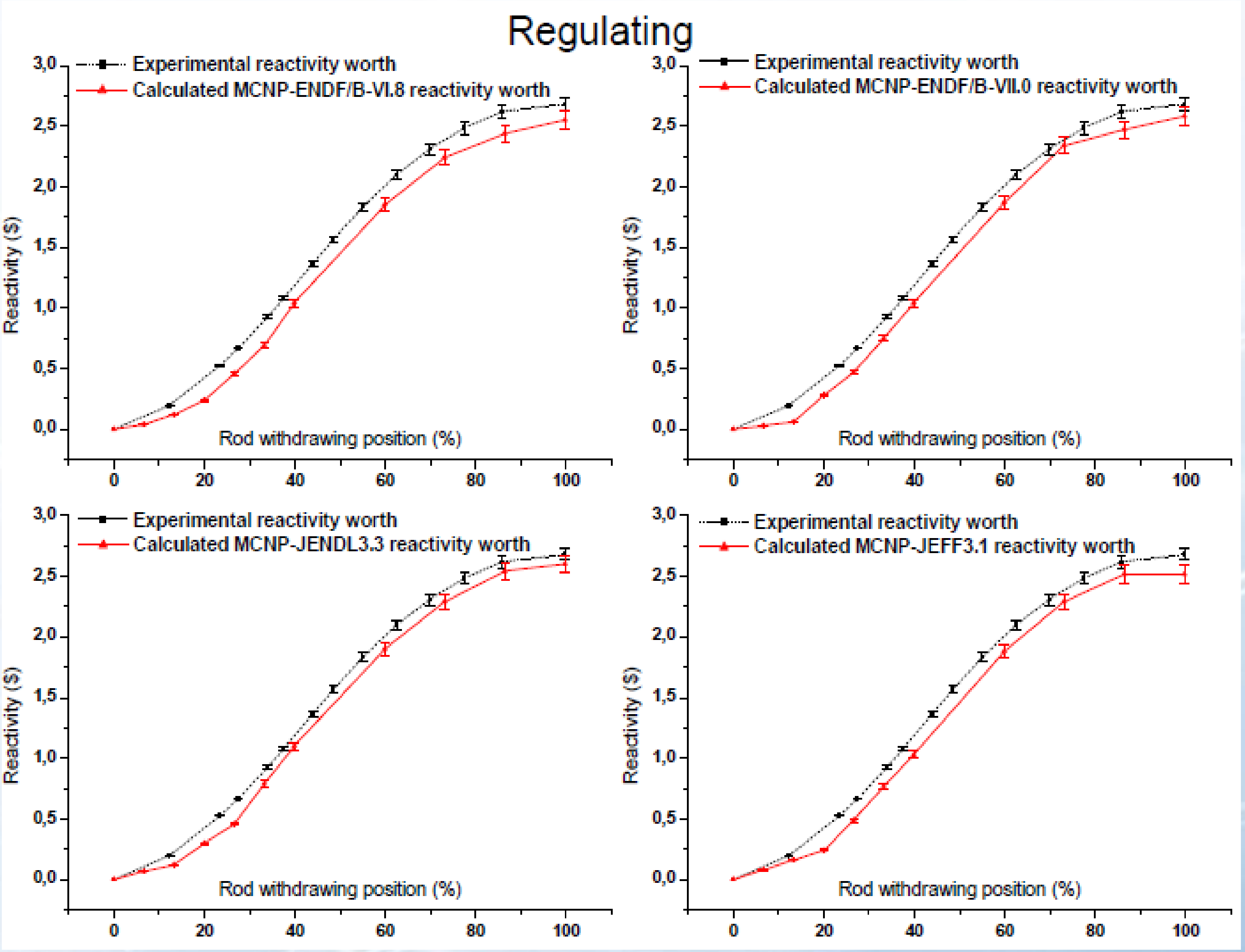
Shim III



Experimental reactivity worth
Calculated MCNP-ENDF/B-VI.8 reactivity worth
Experimental reactivity worth
Calculated MCNP-ENDF/B-VII.0 reactivity worth
Experimental reactivity worth
Calculated MCNP-JEFF3.1 reactivity worth
Experimental reactivity worth
Calculated MCNP-JENDL3.3 reactivity worth
Reactivity ($)
Rod withdrawing position (%)
3,0
2,5
2,0
1,5
1,0
0,5
0,0
0
20
40
60
80
100



ShimIV
Experimental reactivity worth
Calculated MCNP-ENDF/B-VI.8 reactivity worth
Calculated MCNP-ENDF/B-VII.0 reactivity worth
Calculated MCNP-JEFF3.1 reactivity worth
Calculated MCNP-JENDL3.3 reactivity worth
Reactivity ($)
Rod withdrawing position (%)


Regulating
Experimental reactivity worth
Calculated MCNP-ENDF/B-VI.8 reactivity worth
Calculated MCNP-ENDF/B-VII.0 reactivity worth
Calculated MCNP-JENDL3.3 reactivity worth
Calculated MCNP-JEFF3.1 reactivity worth
Reactivity ($)
Rod withdrawing position (%)
3,0
2,5
2,0
1,5
1,0
0,5
0,0
0
20
40
60
80
100

4. Burnup utility

BUCAL1(*B. El Bakkari et al., 2009*) is a FORTRAN computer code elaborated by "*Equipe de Radiation et Systèmes Nucléaires (ERSN)*" at *University ABDELMALIK ESSAÄDI*, *Morocco*.

The code is designed to aid in analysis, prediction, and optimization of fuel burnup performance in a nuclear reactor. It was developed to incorporate the neutron tally/reaction information generated directly by MCNP5 code in the calculation of fissioned or neutron-transmuted isotopes for multi-fueled regions.

$$\frac{dN_i}{dt} = \sum_j \gamma_{ji}\sigma_{f,i}N_j\phi + \sum_k \sigma_{c,k\rightarrow i}N_k\phi + \sum_l \lambda_{l\rightarrow i}N_l - (\underbrace{\sigma_{f,i}N_i\phi + \sigma_{a,i}N_i\phi} + \lambda_i N_i)$$

1 2 3 4 5 6

where

1: Change rate in concentration of isotope i,

2: Production rate per unit volume of isotope i from fission of all fissionable nuclides,

3: Production rate per unit volume of isotope i from neutron transmutation of all isotopes including (n, γ), (n, 2n), etc…,

4: Production rate per unit volume of isotope i from decay of all isotopes including β-, β+, α, γ decay, etc…,

5: Removal rate per unit volume of isotope i by neutron absorption

6: Removal rate per unit of isotope i by decay.

Predictor-corrector depletion approach::

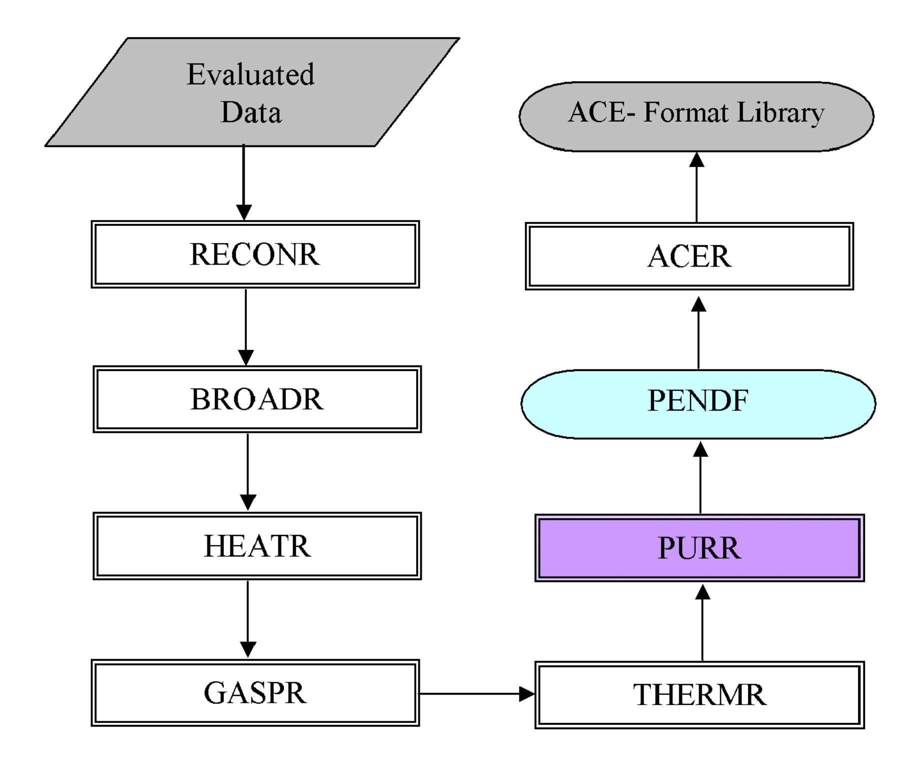
- A burnup calculation is completed in BUCAL1 to the final time step [ti →tf]. (Predictor step)
- Fluxes and reaction rates are calculated in a steady- state MCNP calculation at the final time step tf.
- Then the calculated fluxes and reaction rates are used to burn over the full time step [ti →tf]. (Corrector step)
- The average atom densities from these two calculations are taken as the end-of-time step material compositions.

BUCAL1 Charac.

- Standard burnup calculations
- Multi-fuel cycles calculations
- Shuffling of FE during calculation
- Loading of FE during calculation
- Stop & Restart options

5. Neutron data preparation

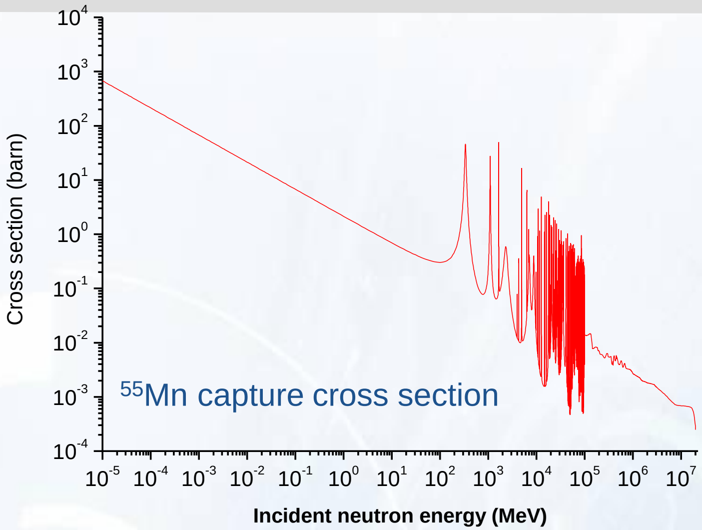
ENDF/B-VII was used in this study. The process to construct the continuous cross section data libraries from the data source files is typically performed by the NJOY system code with its recent update file “up304”

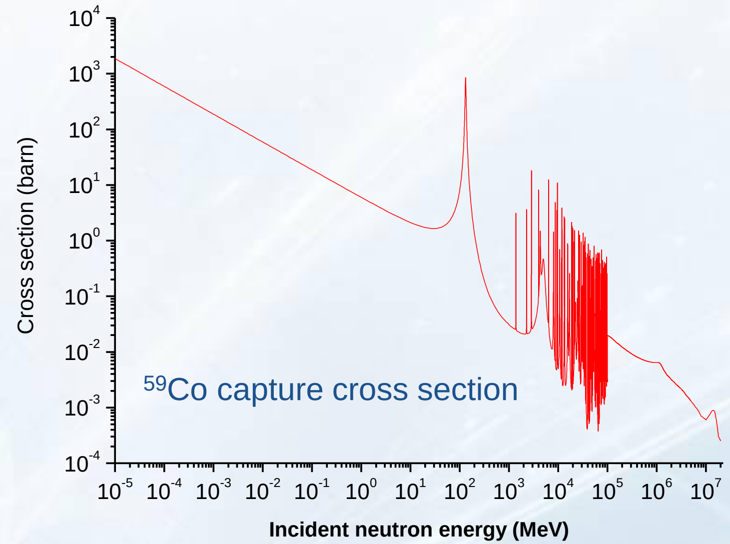


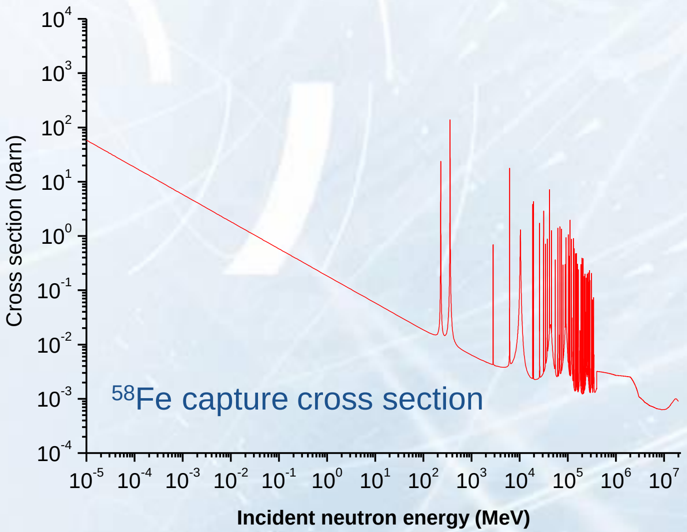
The qualification of the processed nuclear data library ENDF/B-VII was realized through the analysis of the integral parameters of Godiva, Flattop-25 and Bigten benchmarks from the *International Handbook of Evaluated Criticality Safety Benchmark Experiments* - ICSBE *Handbook*.

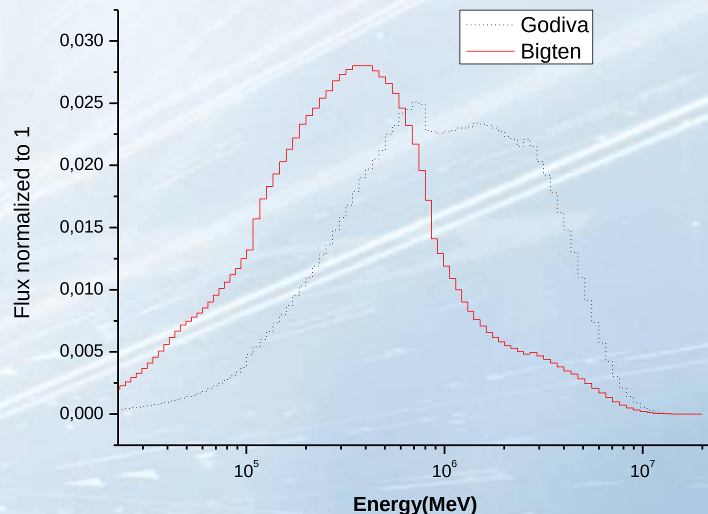
Benchmark	Integral parameters	Experiments (NEA, 2007)	ENDF/B-VII	(C-E/E)%
Godiva	k_{eff}	1.000 (± 1.0e-3)	0.99974 (±1.9e-4)	-0.026
HEU-MET-FAST-001	$\sigma_f\,^{238}U/\sigma_f\,^{235}U$	0.1643 (± 1.8e-3)	0.16055 (± 3.2e-5)	-2.282
	$\sigma_f\,^{233}U/\sigma_f\,^{235}U$	1.59 (± 3.0e-2)	1.56931 (± 3.1e-4)	-1.301
	$\sigma_f\,^{237}Np/\sigma_f\,^{235}U$	0.8516 (± 1.2e-2)	0.84258 (±1.7e-4)	-1.059
	$\sigma_f\,^{239}Pu/\sigma_f\,^{235}U$	1.4152 (± 1.4e-2)	1.38755 (±2.8e-4)	-1.954
	$\sigma_c\,^{55}Mn/\sigma_f\,^{235}U$	0.0027 (± 2.0e-4)	0.00338 (±6.8e-7)	25.11
	$\sigma_c\,^{59}Co/\sigma_f\,^{235}U$	0.038 (± 3.0e-3)	0.00549 (±1.1e-6)	-85.56
	$\sigma_c\,^{93}Nb/\sigma_f\,^{235}U$	0.03 (± 3.0e-3)	0.03384 (±6.8e-6)	12.80
	$\sigma_c\,^{197}Au/\sigma_f\,^{235}U$	0.1 (± 2.0e-3)	0.09362 (±1.9e-5)	-6.38

Benchmark	Integral parameters	Experiments (NEA, 2007)	ENDF/B-VII	(C-E/E)%
Flattop-25	k_{eff}	1.000 (± 1.0^{e-3})	1.00284 (±2.0^{e-4})	0.284
HEU-MET-FAST-028	$\sigma_f\,^{233}U/\sigma_f\,^{235}U$	1.608 (± 3.0^{e-3})	1.56620 (±3.1^{e-4})	-2.600
	$\sigma_f\,^{238}U/\sigma_f\,^{235}U$	0.1492 (± 1.6^{e-3})	0.12765 (±2.6^{e-5})	-14.446
	$\sigma_f\,^{237}Np/\sigma_f\,^{235}U$	0.7804 (± 1.0^{e-2})	0.70543 (±1.4^{e-4})	-9.606
	$\sigma_f\,^{239}Pu/\sigma_f\,^{235}U$	1.3847 (± 1.2^{e-2})	1.33321 (±2.7^{e-4})	-3.719
Bigten	k_{eff}	0.996 (± 2.0^{e-3})	0.99453 (±1.6^{e-4})	-0.148
IEU-MET-FAST-007	$\sigma_f\,^{238}U/\sigma_f\,^{235}U$	0.03739 (± 3.4^{e-4})	0.03364 (±6.7^{e-6})	-10.026
	$\sigma_f\,^{237}Np/\sigma_f\,^{235}U$	0.3223 (± 3.0^{e-3})	0.29768 (±6.0^{e-5})	-7.638
	$\sigma_f\,^{239}Pu/\sigma_f\,^{235}U$	1.1936 (± 8.4^{e-3})	1.15210 (±2.3^{e-4})	-3.476
	$\sigma_f\,^{233}U/\sigma_f\,^{235}U$	1.58 (± 3.0^{e-2})	1.53954 (±3.1^{e-4})	-2.561
	$\sigma_c\,^{238}U/\sigma_f\,^{235}U$	0.11 (± 3.0^{e-3})	0.10837 (±2.2^{e-5})	-1.478
	$\sigma_{(n,\alpha)}\,^{10}B/\sigma_f\,^{235}U$	1.011 (± 1.4^{e-2})	0.975104 (±2.0^{e-4})	-3.55
	$\sigma_{(n,\alpha)}\,^{27}Al/\sigma_f\,^{235}U$	0.000078 (± 2.0^{e-6})	0.000079 (±1.6^{e-8})	0.70
	$\sigma_{(n,p)}\,^{46}Ti/\sigma_f\,^{235}U$	0.0013 (± 3.0^{e-5})	0.001372 (±2.7^{e-7})	5.51
	$\sigma_{(n,p)}\,^{47}Ti/\sigma_f\,^{235}U$	0.00215 (± 9.0^{e-5})	0.001873 (±3.7^{e-7})	-12.90
	$\sigma_{(n,p)}\,^{48}Ti/\sigma_f\,^{235}U$	0.000036 (± 1.0^{e-6})	0.000031 (±6.2^{e-9})	-13.20
	$\sigma_{(n,p)}\,^{54}Fe/\sigma_f\,^{235}U$	0.009 (± 3.0^{e-4})	0.008308 (±1.7^{e-6})	-7.69
	$\sigma_{(n,p)}\,^{58}Ni/\sigma_f\,^{235}U$	0.0123 (± 2.0^{e-4})	0.011088 (±2.2^{e-6})	-9.85
	$\sigma_c\,^{58}Fe/\sigma_f\,^{235}U$	0.0031 (± 1.0^{e-4})	0.004867 (±9.7^{e-7})	57.01
	$\sigma_c\,^{59}Co/\sigma_f\,^{235}U$	0.0095 (± 2.0^{e-4})	0.009034 (±1.8^{e-6})	-4.91
	$\sigma_c\,^{63}Cu/\sigma_f\,^{235}U$	0.0164 (± 1.0^{e-3})	0.018385 (±3.7^{e-6})	12.11
	$\sigma_c\,^{197}Au/\sigma_f\,^{235}U$	0.167 (± 3.0^{e-3})	0.170243 (±3.4^{e-5})	1.94


55Mn capture cross section
Cross section (barn)
Incident neutron energy (MeV)


59Co capture cross section
Cross section (barn)
Incident neutron energy (MeV)

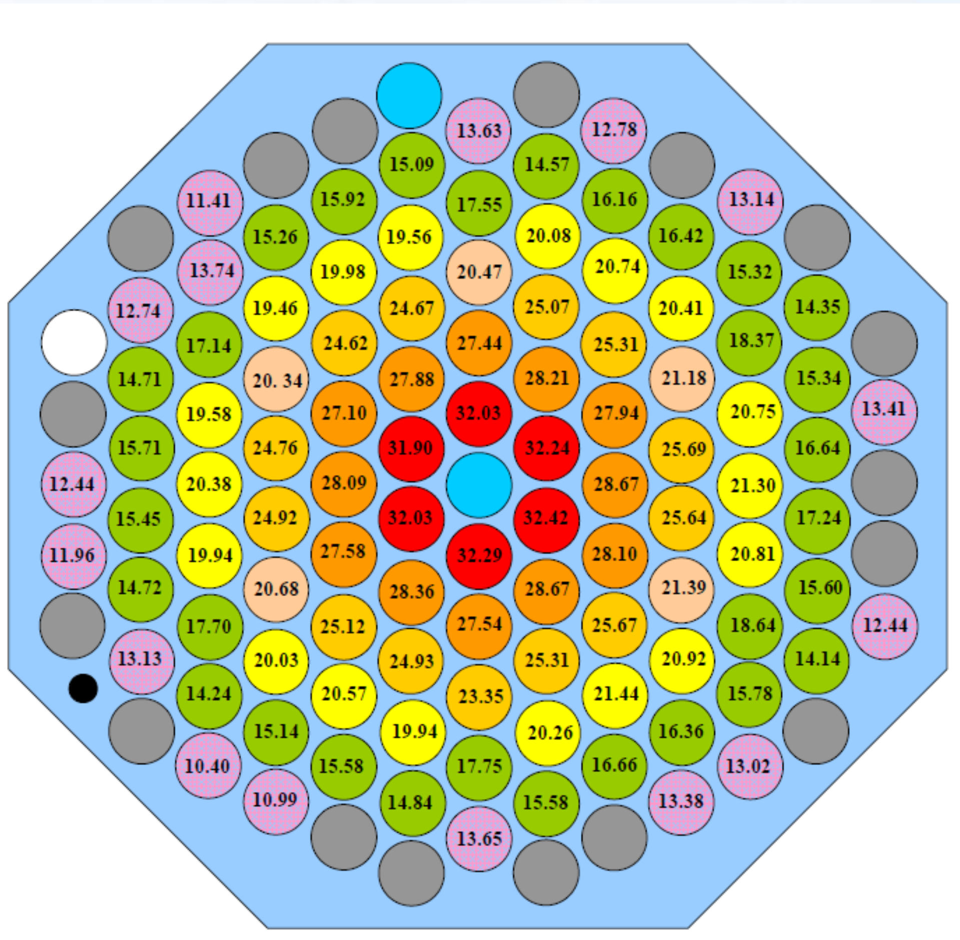

58Fe capture cross section
Cross section (barn)
Incident neutron energy (MeV)


Godiva
Bigten
Flux normalized to 1
Energy(MeV)

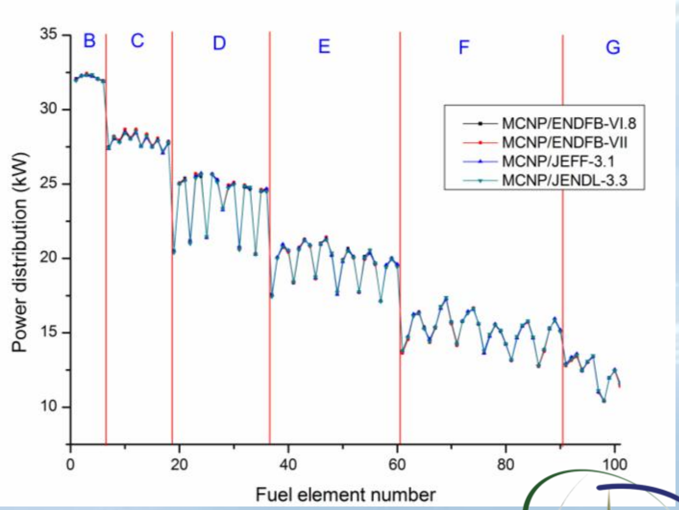
6. Results and discussions

The calculations were performed with 1500 cycles of iterations on a nominal source size of 5000 neutrons per cycle. The initial 100 cycles were skipped to insure homogeneous neutron source distribution.

The estimated statistical errors were reduced below 30 pcm *for keff* values and 5% for reaction rates calculations.

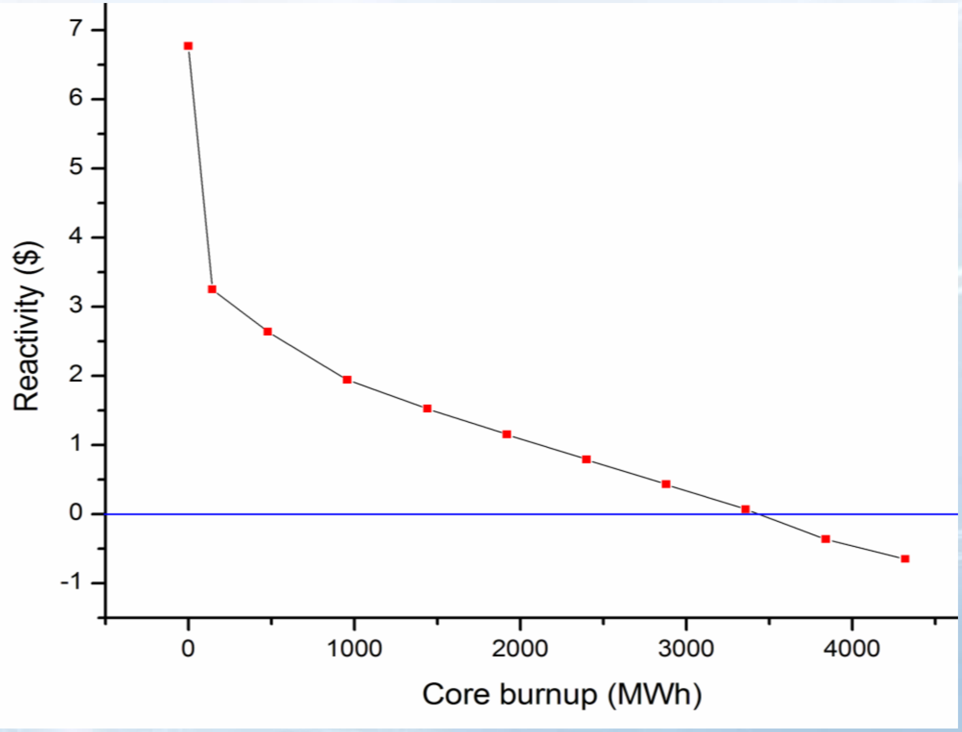


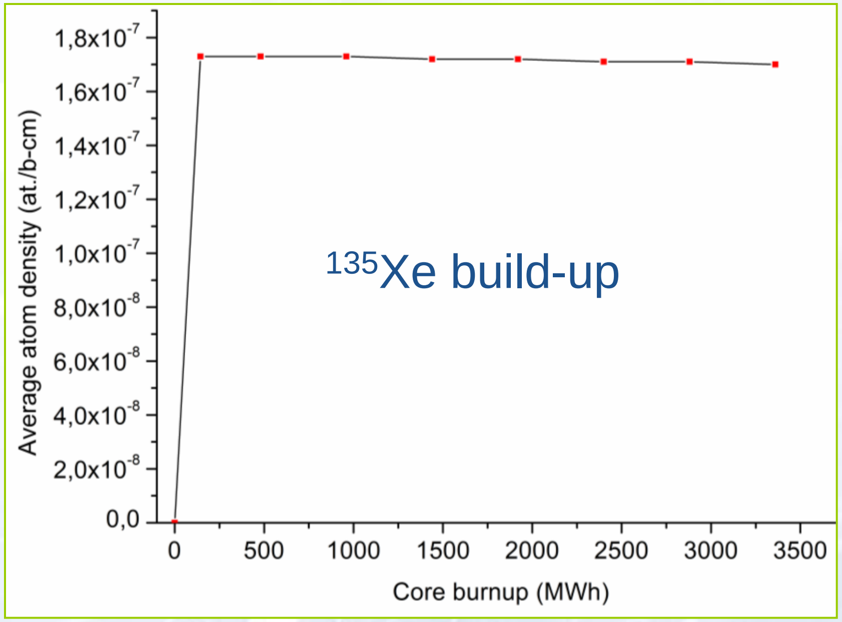
- Group K : (30.5 < P ≤ 33 kW)
- Group L : (26.5 < P ≤ 30.5 kW)
- Group M : (23 < P ≤ 26.5 kW)
- Group N [BC]: (20 < P ≤ 22 kW)
- Group O : (19 < P ≤ 23 kW)
- Group P : (14 < P ≤ 19 kW)
- Group Q : (P ≤ 14 kW)



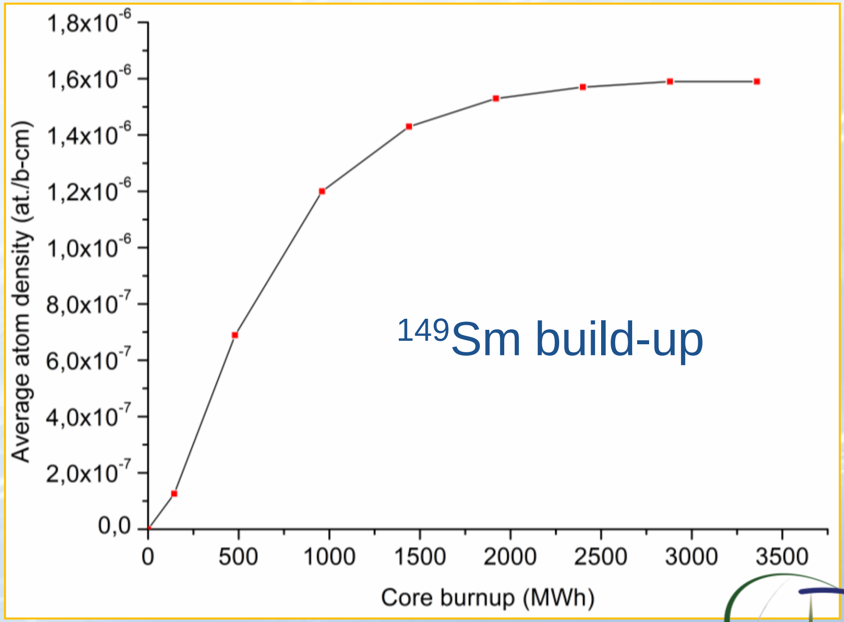
Reactor Life Time

The determination of *keff* and its relationship with burnup is of primary importance to determine the reactor core life time.



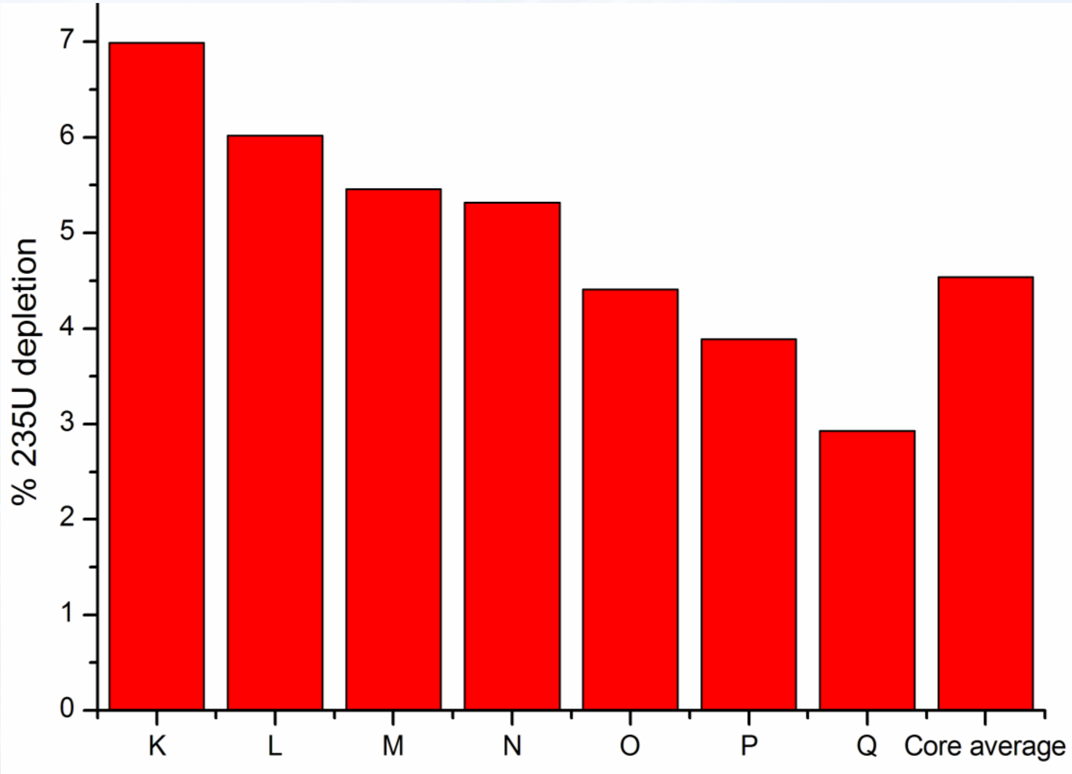


1,8x10-7
1,6x10-7
1,4x10-7
1,2x10-7
1,0x10-7
8,0x10-8
6,0x10-8
4,0x10-8
2,0x10-8
0,0
Average atom density (at./b-cm)
0
500
1000
1500
2000
2500
3000
3500
Core burnup (MWh)
135Xe build-up

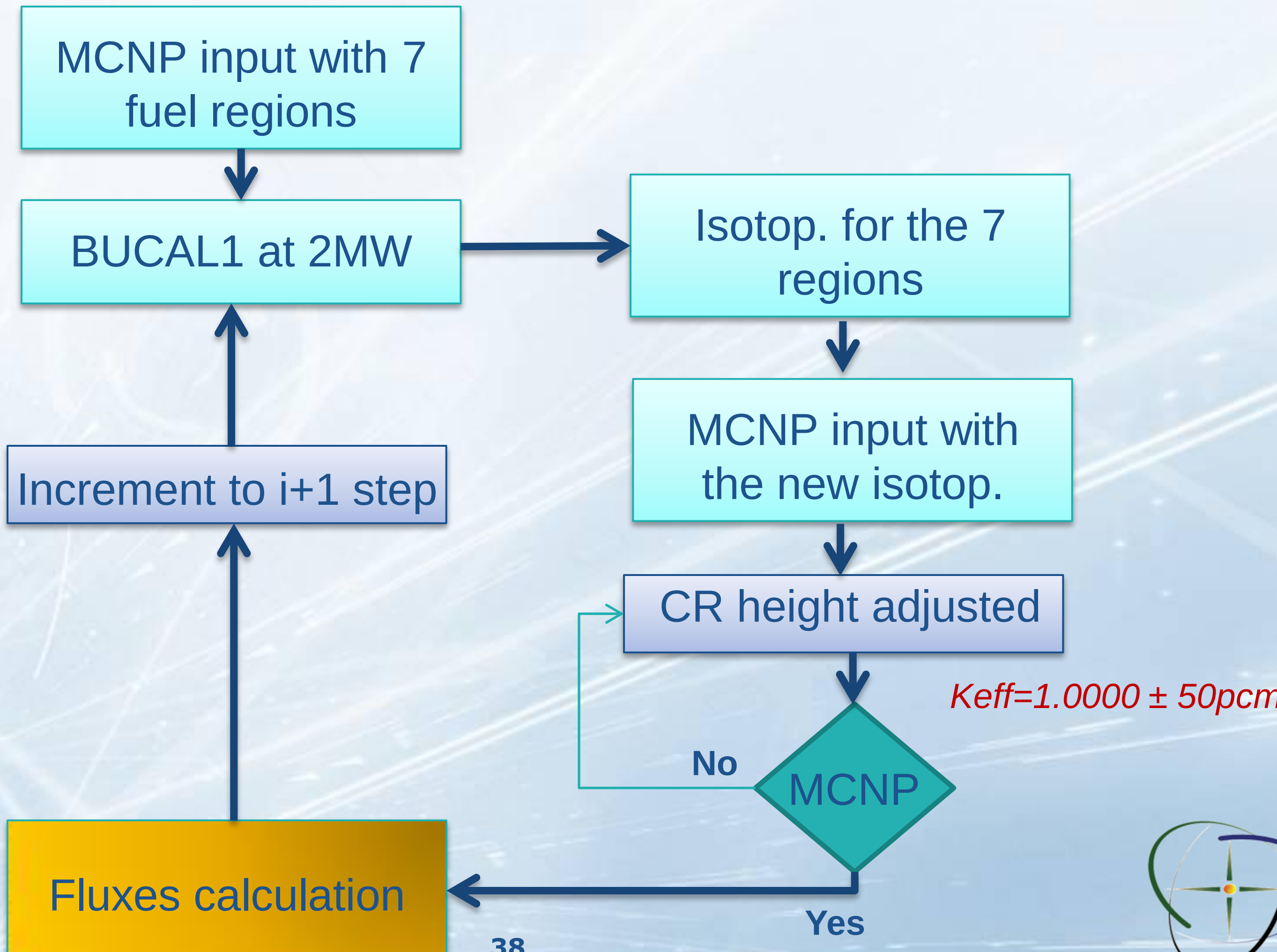


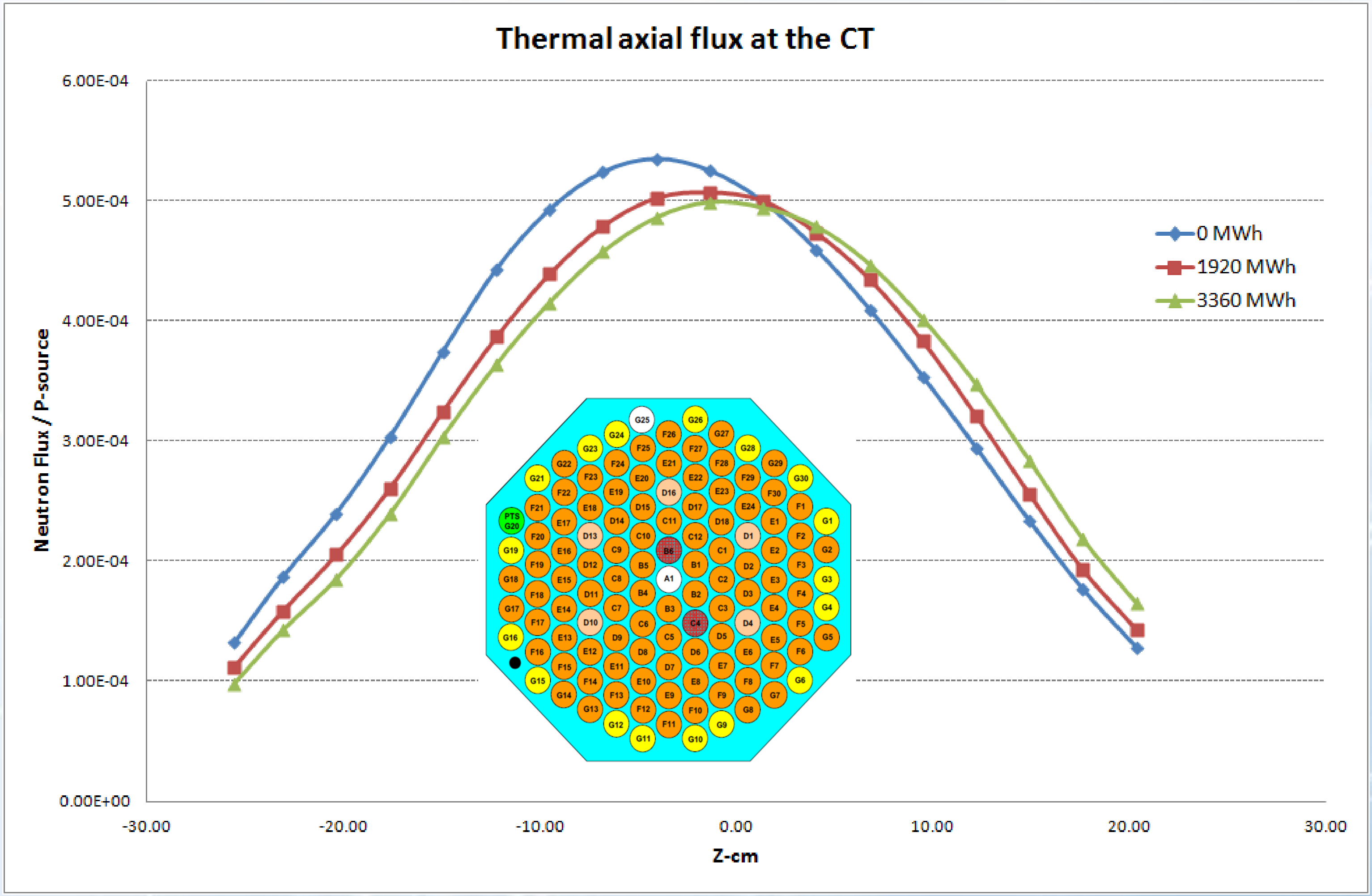
1,8x10-6
1,6x10-6
1,4x10-6
1,2x10-6
1,0x10-6
8,0x10-7
6,0x10-7
4,0x10-7
2,0x10-7
0,0
Average atom density (at./b-cm)
0
500
1000
1500
2000
2500
3000
3500
Core burnup (MWh)
149Sm build-up

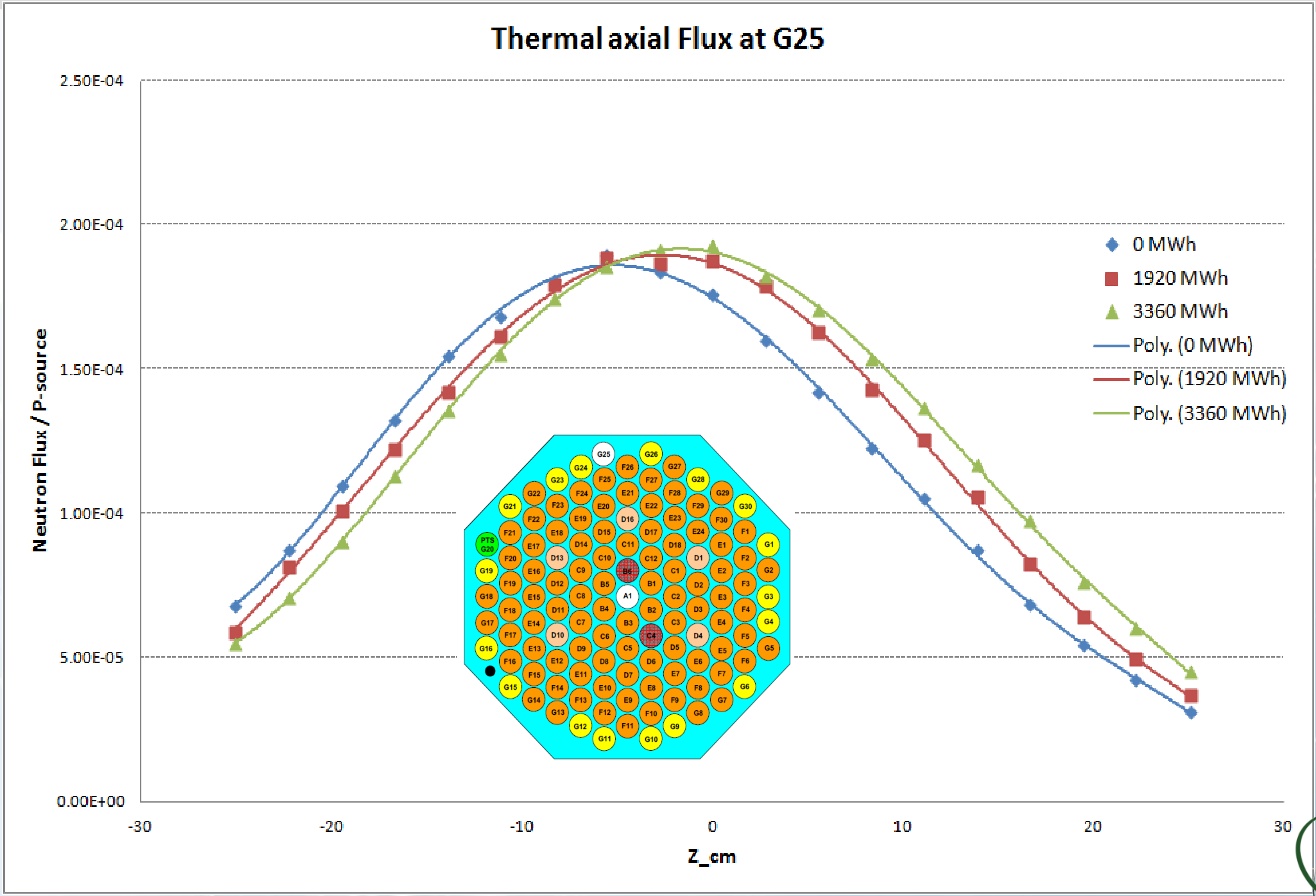
U235 depletion at EOC



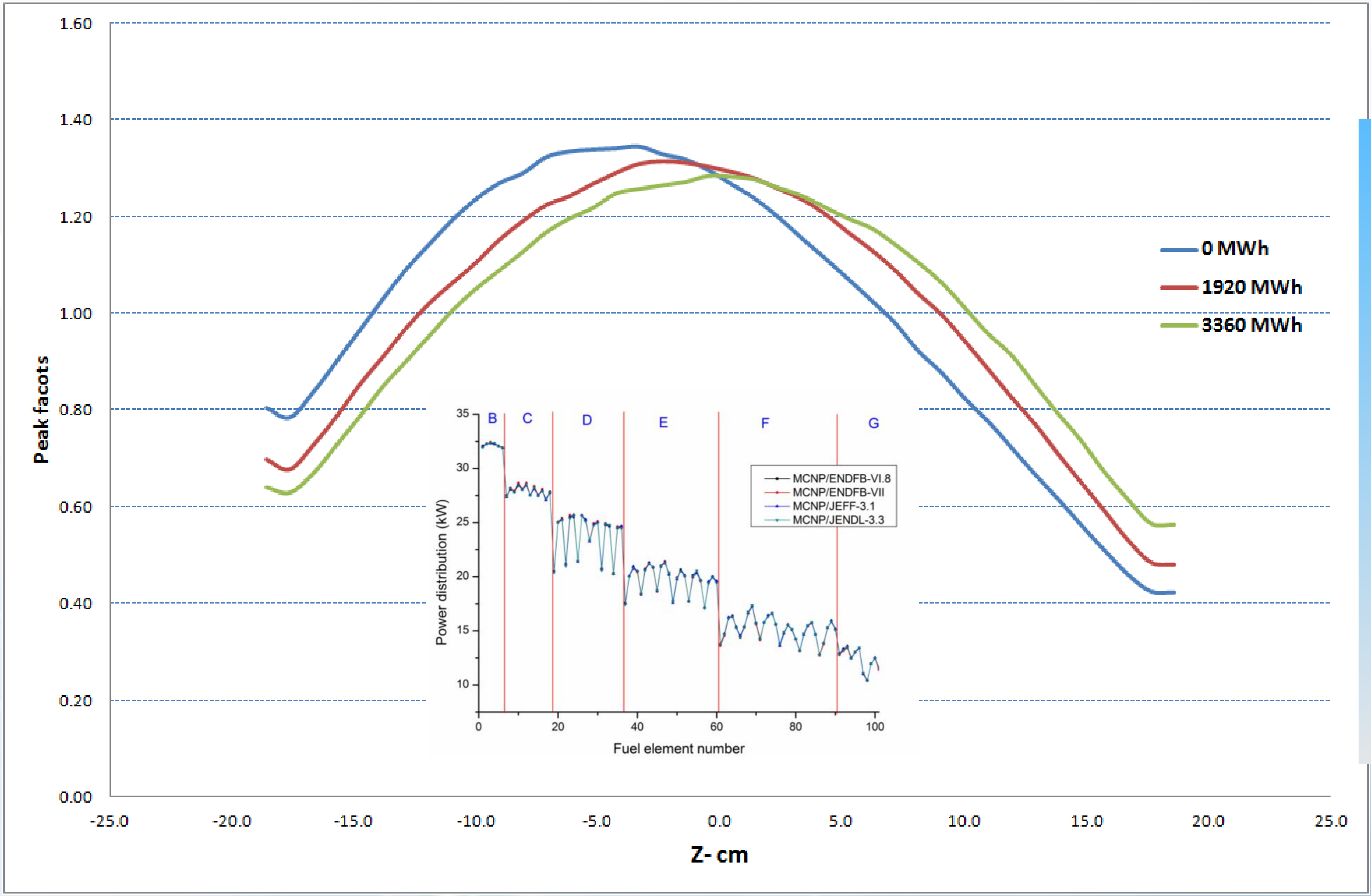
Fluxes versus burnup



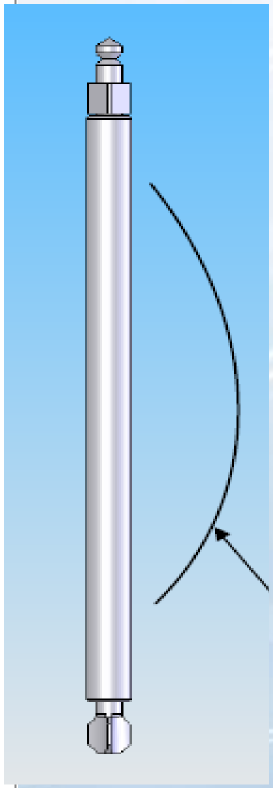

Thermal axial flux at the CT
Neutron Flux / P-source
Z-cm
6.00E-04
5.00E-04
4.00E-04
3.00E-04
2.00E-04
1.00E-04
0.00E+00
-30.00
-20.00
-10.00
0.00
10.00
20.00
30.00
0 MWh
1920 MWh
3360 MWh

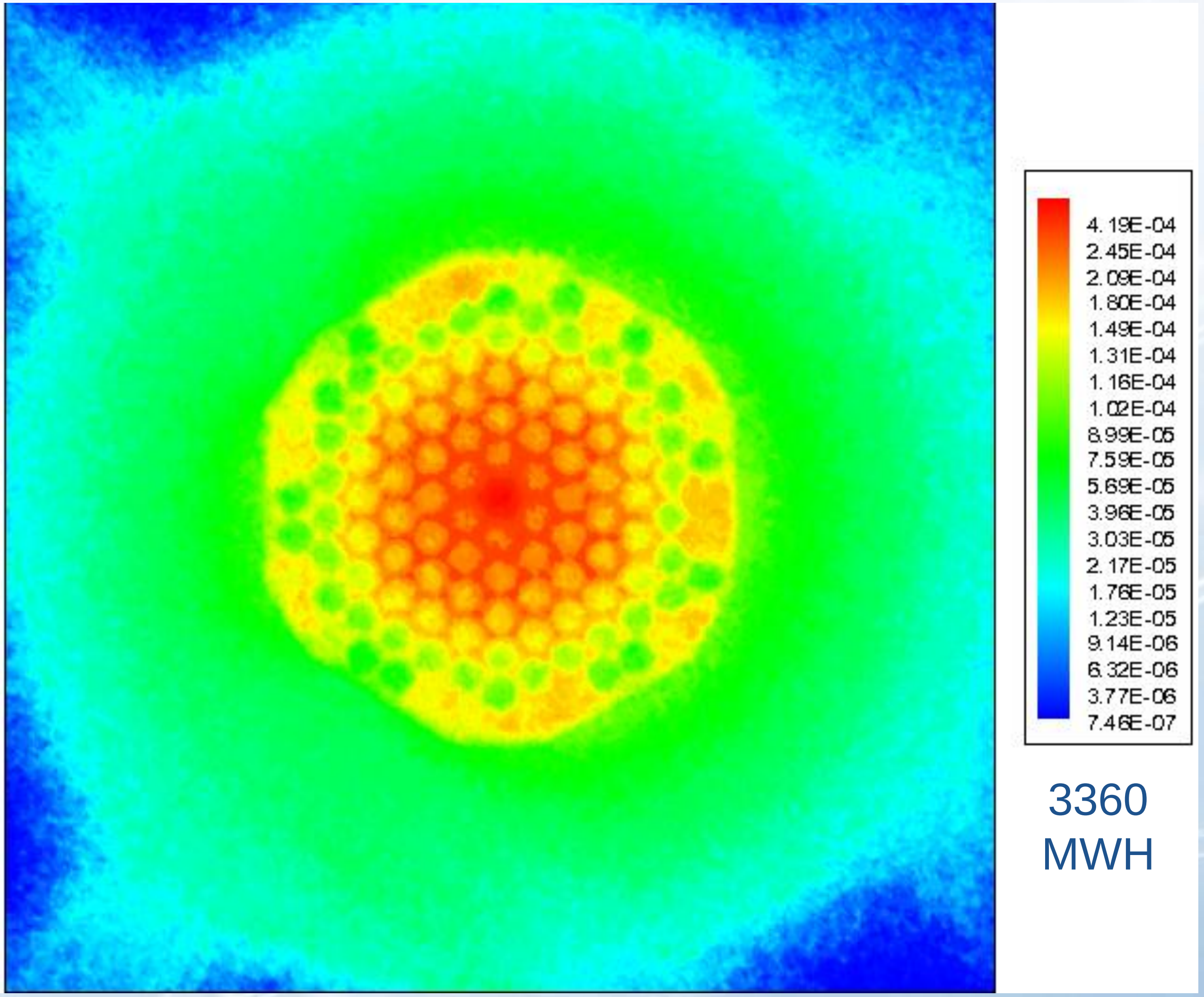


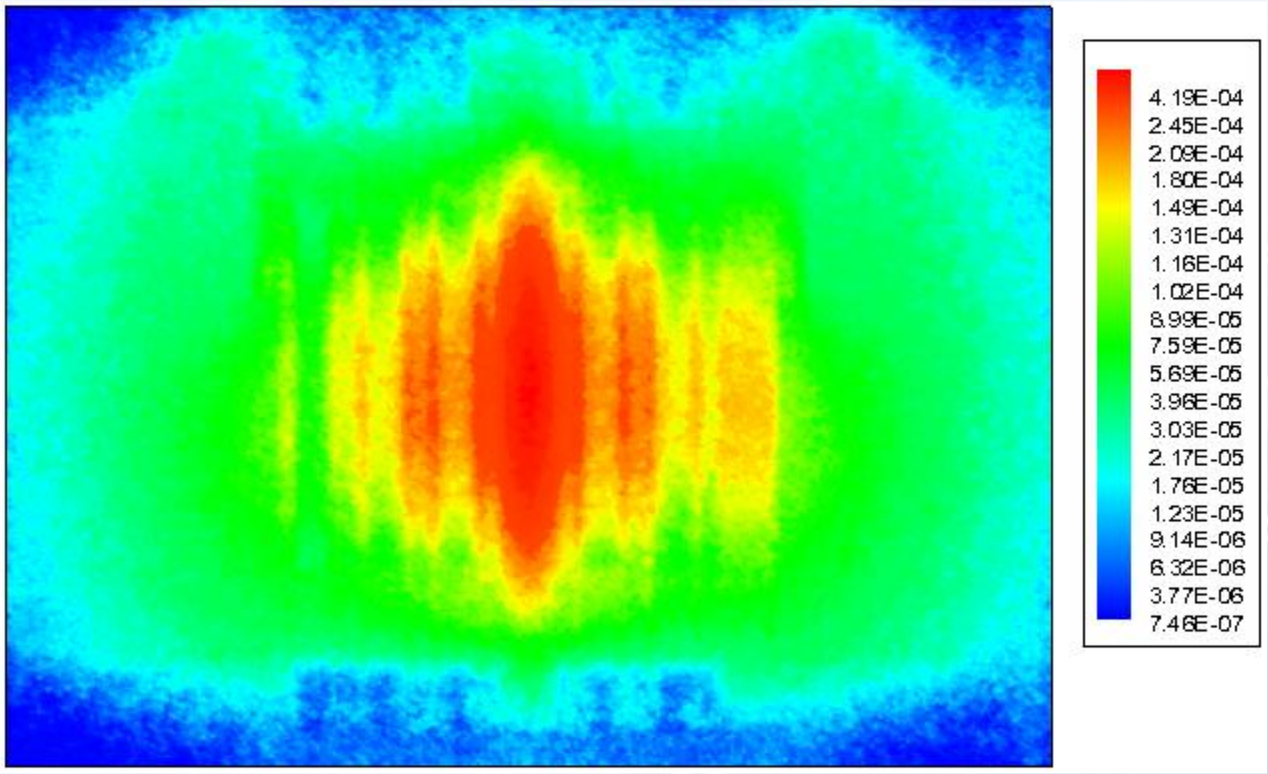
Thermal axial Flux at G25
Neutron Flux / P-source
Z_cm
0 MWh
1920 MWh
3360 MWh
Poly. (0 MWh)
Poly. (1920 MWh)
Poly. (3360 MWh)
2.50E-04
2.00E-04
1.50E-04
1.00E-04
5.00E-05
0.00E+00
-30
-20
-10
0
10
20
30



Peak facots
Z- cm
0 MWh
1920 MWh
3360 MWh
1.60
1.40
1.20
1.00
0.80
0.60
0.40
0.20
0.00
-25.0
-20.0
-15.0
-10.0
-5.0
0.0
5.0
10.0
15.0
20.0
25.0
Power distribution (kW)
Fuel element number
B
C
D
E
F
G
MCNP/ENDFB-VI.8
MCNP/ENDFB-VII
MCNP/JEFF-3.1
MCNP/JENDL-3.3




4. 19E-04
2.45E-04
2.09E-04
1.80E-04
1.49E-04
1.31E-04
1.16E-04
1.02E-04
8.99E-05
7.59E-05
5.69E-05
3.96E-05
3.03E-05
2.17E-05
1.76E-05
1.23E-05
9.14E-06
6.32E-06
3.77E-06
7.46E-07
3360
MWH



4.19E-04
2.45E-04
2.09E-04
1.80E-04
1.49E-04
1.31E-04
1.16E-04
1.02E-04
8.99E-05
7.59E-05
5.69E-05
3.96E-05
3.03E-05
2.17E-05
1.76E-05
1.23E-05
9.14E-06
6.32E-06
3.77E-06
7.46E-07

7. Conclusion

A detailed model of the Moroccan TRIGA MARK II research reactor was elaborated using the MCNP5 code.

Burnup calculations of the TRIGA reactor were established using BUCAL1 burnup code .

The reactor life time was found to be 3360MWH.

Radially, thermal fluxes decrease at the core center and increase slightly at the outer ring.

Axially, thermal fluxes peaks move up following the CR position

Thank You !

Bilal El Bakkari
Experiments Manager & Reactor Manager Assistant
Reactor Operating Unit – CNESTEN
(bakkari@cnesten.org.ma)