FUEL BURN-UP CALCULATION FOR THE DALAT NUCLEAR RESEARCH REACTOR BY USING SERPENT AND MCNP6 COMPUTER CODES

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- Serpent-MCNP6 computer codes
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Introduction

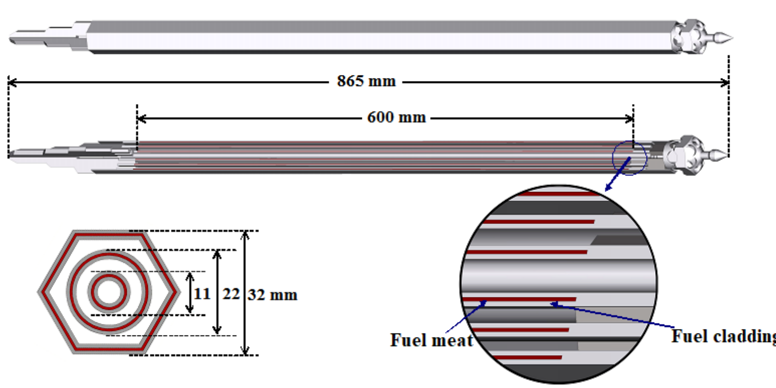
- Fuel burn-up effect to the safety of reactor
- Burn-up determination by: experiments, calculation
- Burn-up calculation using MC method: MCNP6+Serpent
- Why do apply MC method? Main reasons: obtain exact neutron flux, reaction rate in space and energy, excellent library (can be updated)

Introduction

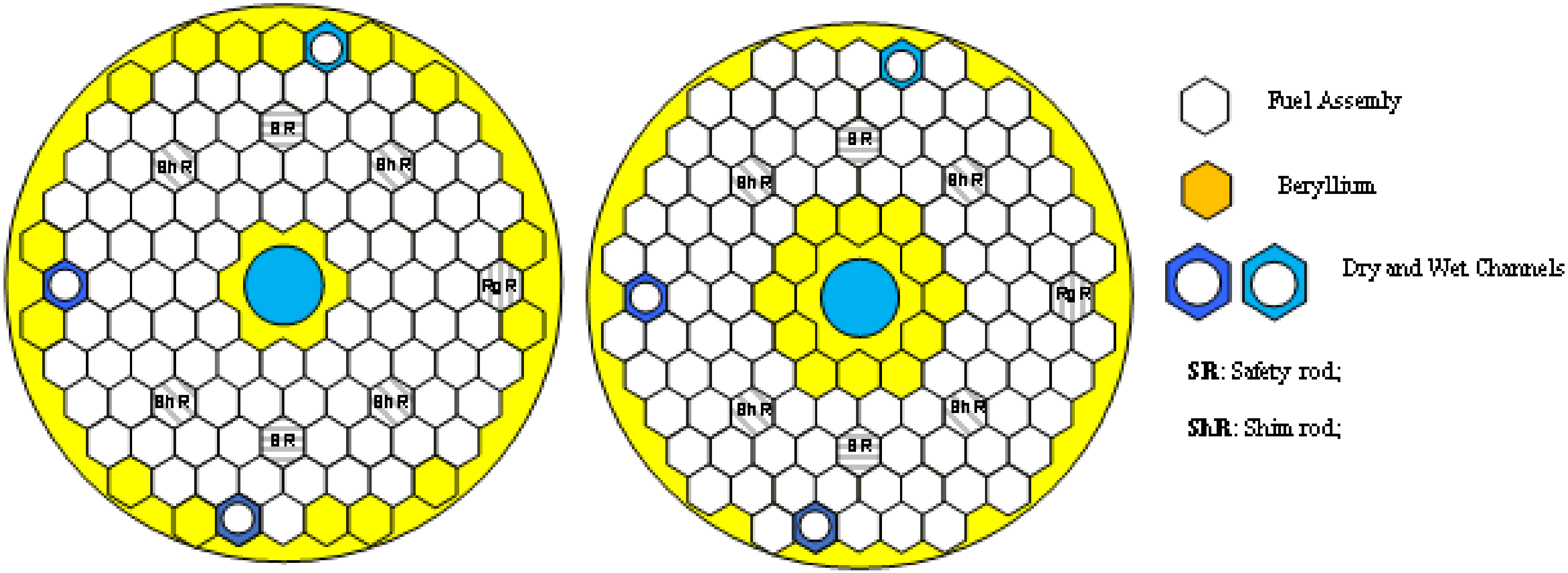
- Parameters for depletion calculation: Neutron flux and reaction rate in 1 group energy.
- Long burn-up step 50 to 100 days.
- Complex core configurations including many type of fuels with different enrichments and geometries
- Validation by apply to VVR-M2 HEU and LEU fuels and HEU core 89 FAs, LEU core 92 FAs of the DNRR.
- Obtained burn-up results compared between MCNP6 and Serpent codes.

Introduction

Parameter	HEU	LEU
Enrichment, %	36	19.75
Average mass of ^{235}U in FA, g	40.20	49.70
Fuel meat composition	U-Al Alloy	UO_2+Al
Uranium density of fuel meat, g/cm^3	1.40	2.50
Cladding material	Al alloy (SAV-1)	Al alloy (SAV-1)
Fuel element thickness (fuel meat and 2 cladding), mm	2.50	2.50
Fuel meat thickness, mm	0.70	0.94
Each cladding thickness, mm	0.90	0.78

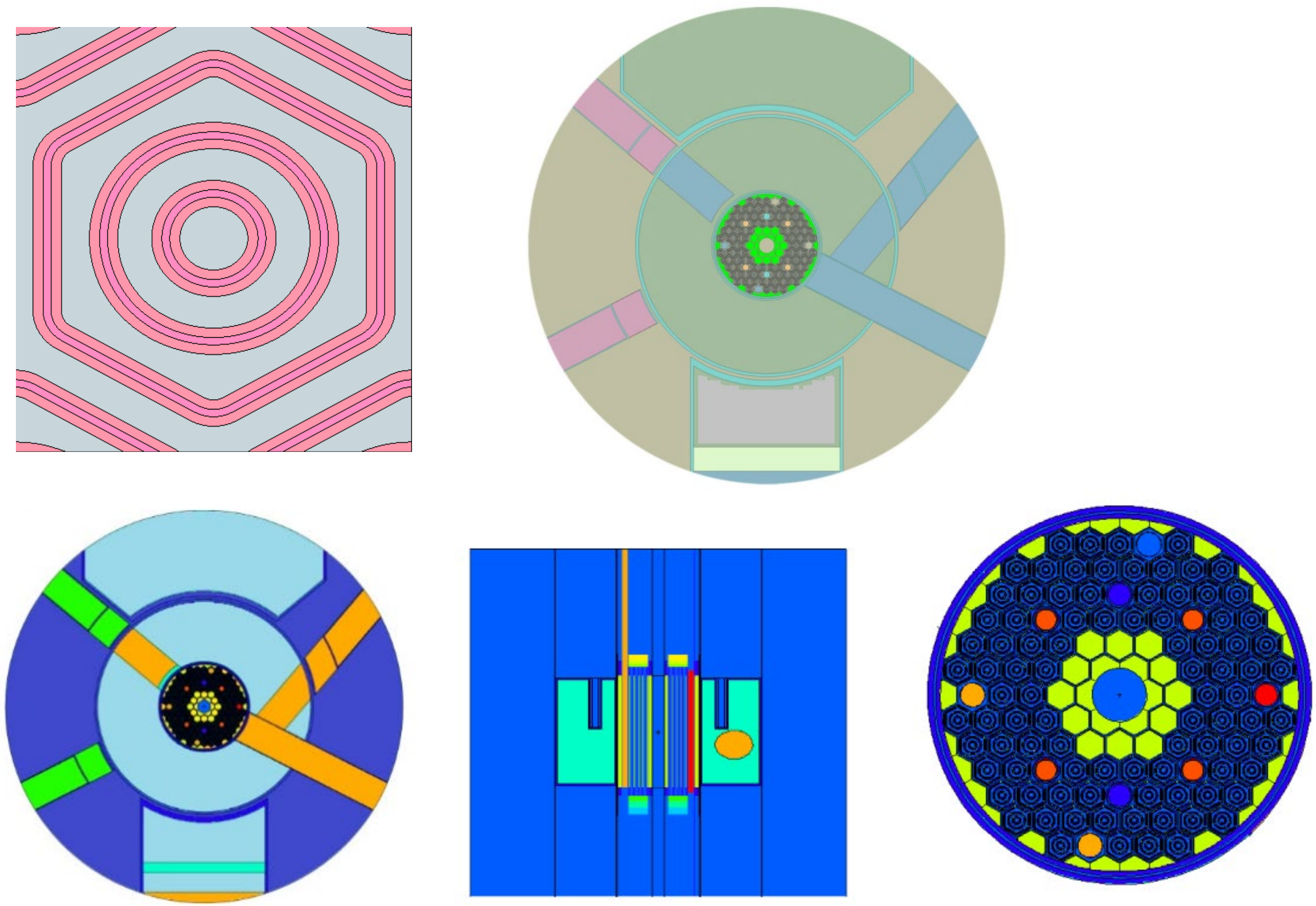


Introduction



HEU and LEU working cores at start-up time in 1984 and 2012

Serpent-MCNP6 computer codes

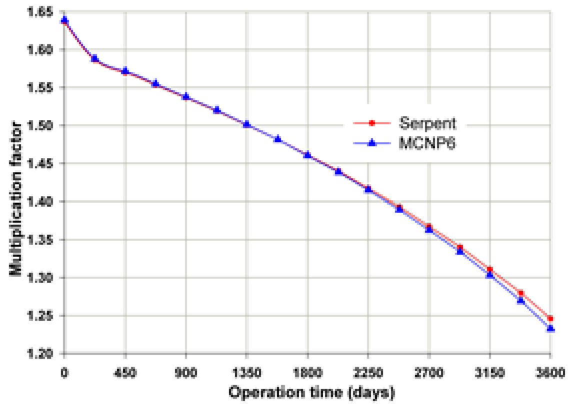


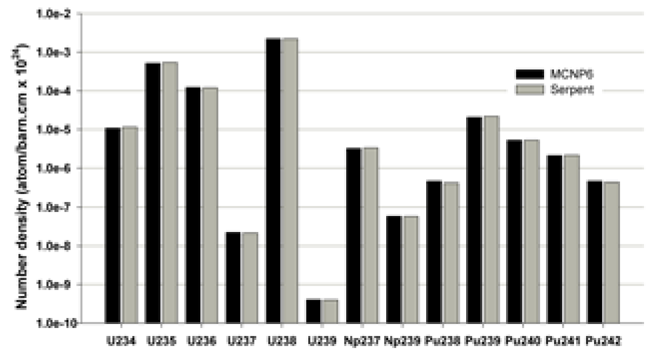
Calculation models of Serpent and MCNP6 codes

Serpent-MCNP6 computer codes

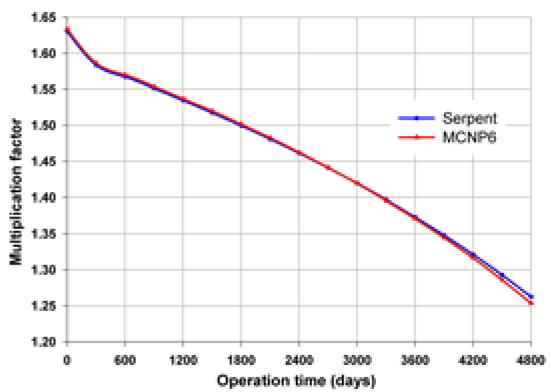
MCNP6	SERPENT
Monte Carlo Method (PL: Fortran - GFORTRAN) MPI and OpenMPI	Monte Carlo Method (PL: C - GCC) MPI or OpenMPI
CINDER90 – MCNPX2.7 Analysis method with PC	CRAM – TTA – PC
ENDF/B VII.0 (AEC files)	ENDF/B VII.0 (AEC files)
Limitation of number particle	~ 1 million
Running: 8 nodes ~ 64 CPUs	8 nodes ~ 8 CPUs
Big output file	Big output file
OMIT Isotope	Free for setting historical isotopes
No beryllium poisoning	No beryllium poisoning
All control rods are withdrawn	All control rods are withdrawn

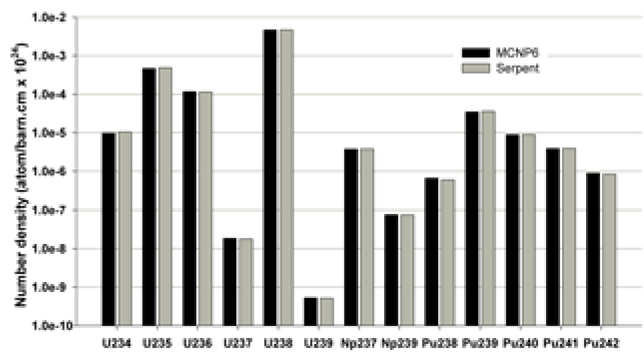
Calculation results of fuels and reactor cores of the DNRR





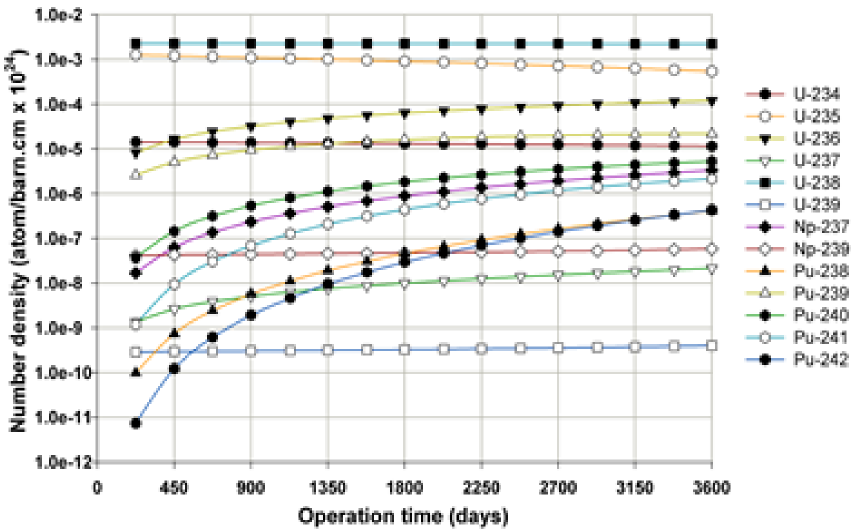
Infinite multiplication factor of HEU fuel depending on burn-up steps and atom density of actinide isotopes at the end of burn-up step (~ 60% burn up of U-235)

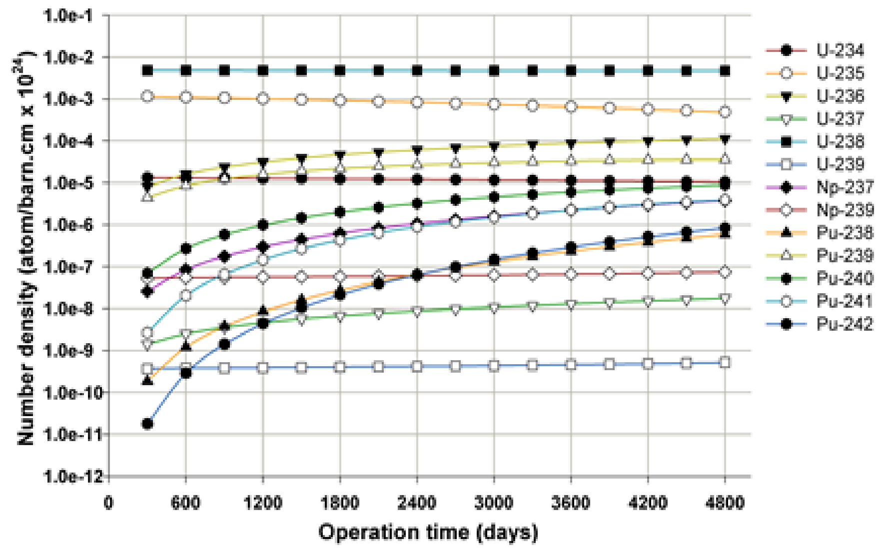




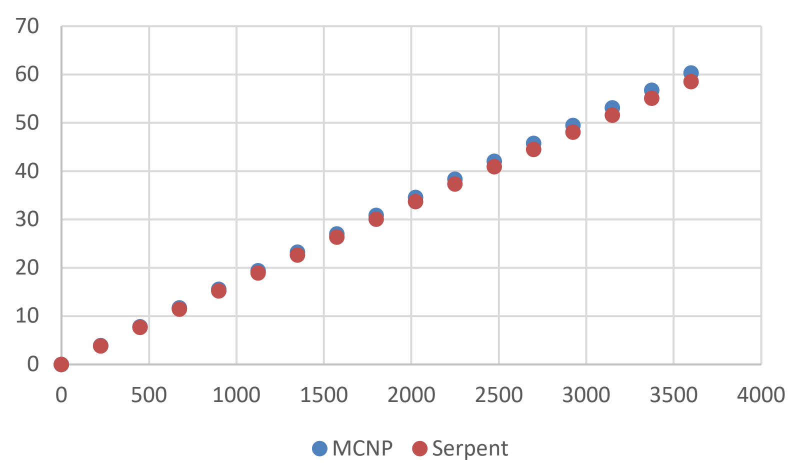
Infinite multiplication factor of HEU fuel depending on burn-up steps and atom density of actinide isotopes at the end of burn-up step (~ 60% burn up of U-235)

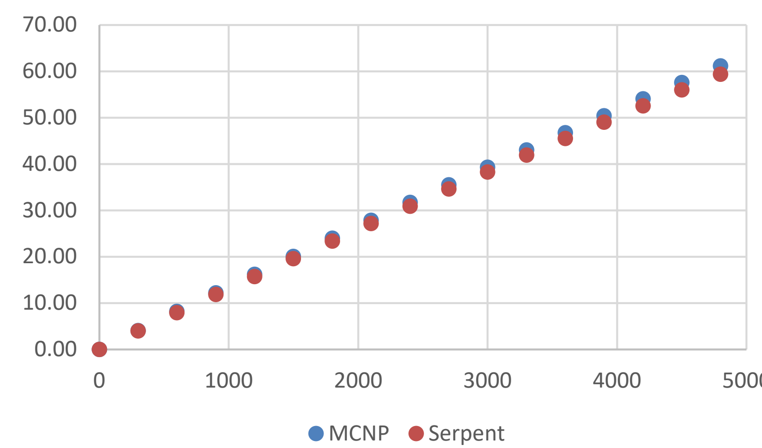
Calculation results of fuels and reactor cores of the DNRR





Number density of 13 actinide isotopes of HEU and LEU FA at the end of burn-up step (~ 60% burn up of U-235)





Calculation results of fuels and reactor cores of the DNRR

- Total burn-up of HEU and LEU fuel: about 60% U-235
- Error of neutron flux, reaction rate and multiplication under 0.1% and standard deviation of keff under 0.005%
- Difference of multiplication factor between MCNP6 code and SERPENT code is smaller than 800 pcm for both cases
- Maximum difference of atom densities of all isotopes is under 10% to actinide isotopes Pu-238.

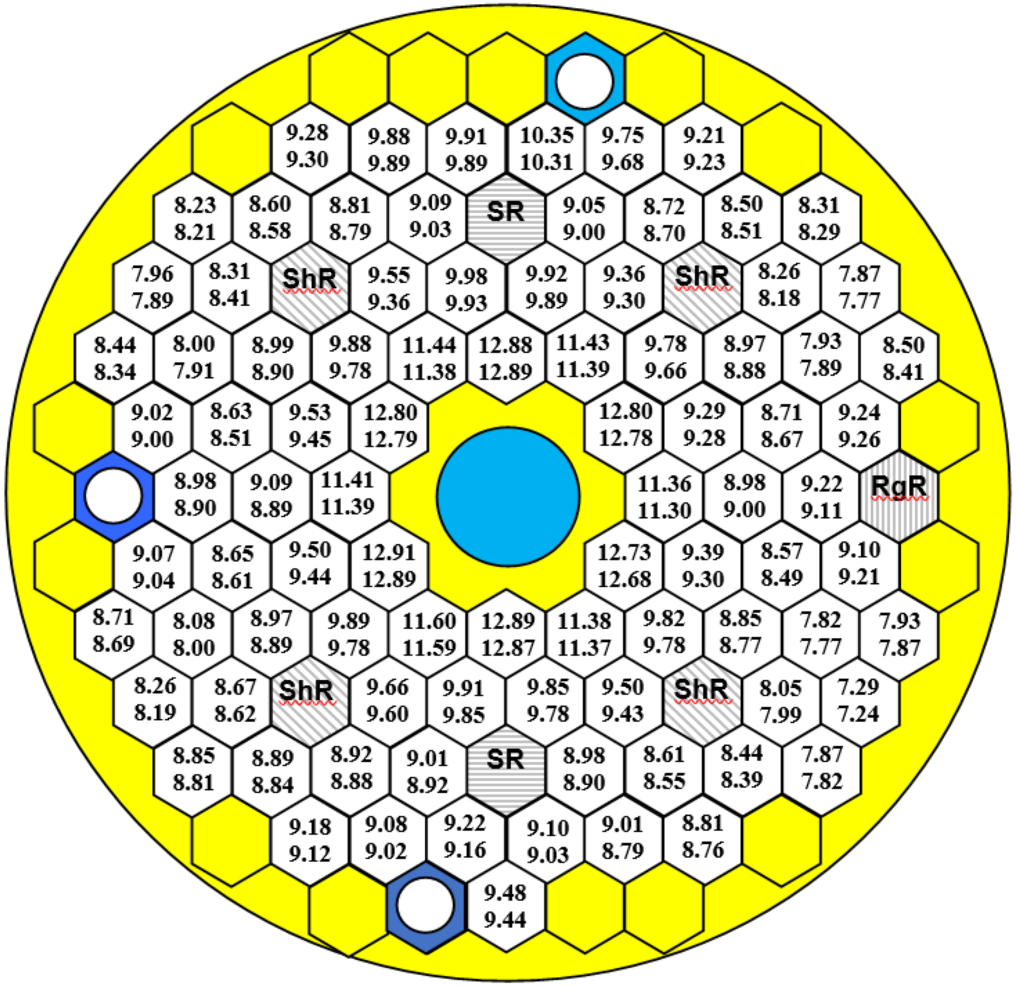
Calculation results of fuels and reactor cores of the DNRR

- Depletion in axial direction: 5 nodes same volumes.

-HEU core: 89 FAs, 15 fresh Be rods, 538 FPDs and 100 cooling day, the difference excess reactivity at the end of fuel cycle between two codes was about 120 pcm.

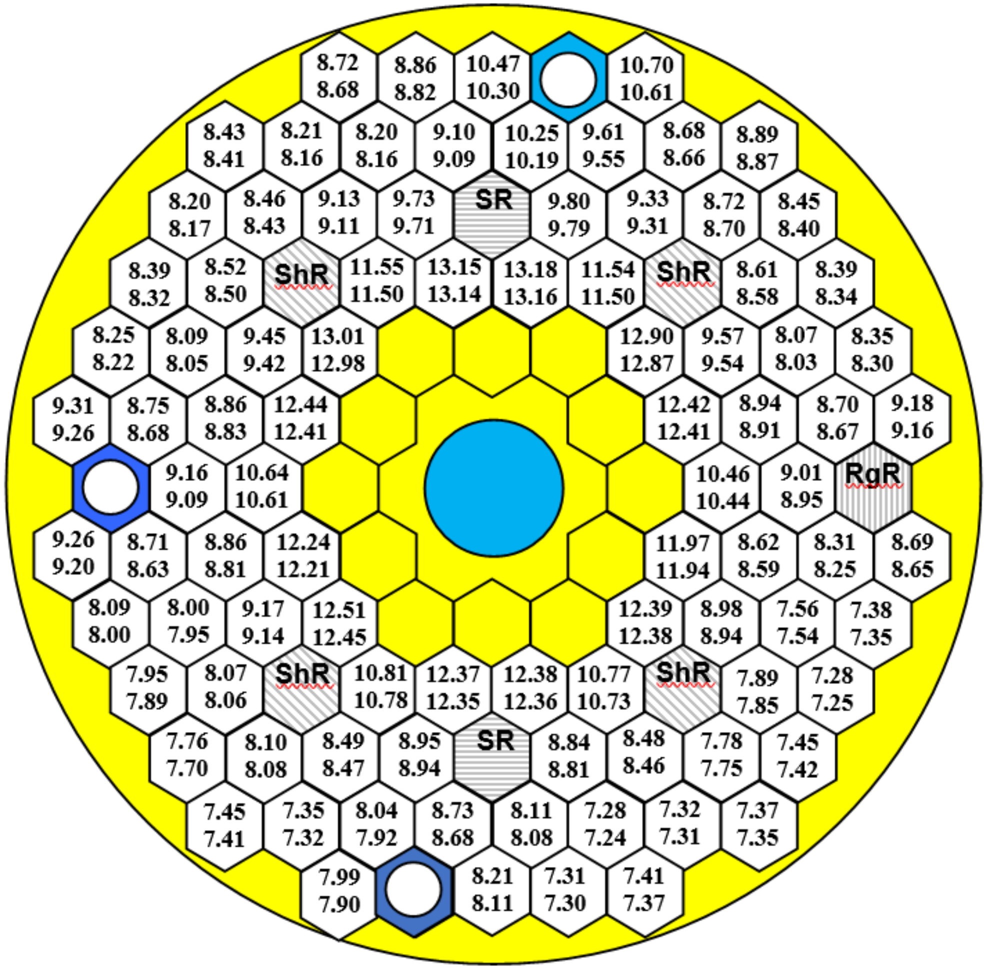
-LEU core: 92 FAs, 12 poisoned Be rods, 700 FPDs and 100 cooling day, different excess reactivity 100 pcm.

Calculation results of fuels and reactor cores of the DNRR



Burn-up (% U-235) distribution of fresh HEU core after 538 FPDs operation (MCNP6 code at upper values – Serpent code at lower values)

Calculation results of fuels and reactor cores of the DNRR



Burn-up (U-235%) distribution of fresh LEU core after 700 FPDs operation
(MCNP6 code at upper values – Serpent code at lower values)

Conclusions

- Both codes can be used for fuel and core management.
- Serpent code has more advantage than MCNP about running time, using number of CPUs.
- Need to consider about position of control rods.
- Beryllium poisoning need to be taken into account.
- Serpent code can be coupled with other computer codes as Kinetics, T-H, CFD

Thank you for your attention !