Inventory Prediction and BUC Calculations Related to MEU/LEU IRT Fuels of LVR-15 Research Facility

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Outline

- Russian fuels for research reactors
- LVR-15 research facility of NRI at Rez: fuel conversion
- Calculation models of IRT fuels
- Criticality and depletion calculations
  - Reserve pool at LVR-15 reactor
  - VPVR transport cask
- Summary

Background

- In the past not only Russian NPPs (the most known is VVER type) were built in the former ‘Soviet block’ countries but also research reactors using highly enriched uranium (HEU) fuel

- In 2004, a bilateral agreement between the U.S. and Russian Federation governments concerning the repatriation of Russian-origin HEU research reactor fuel to Russia was signed

- Under the **‘Russian Research Reactor Fuel Return’ (RRRFR) program** participating countries agree to convert their research reactors from using HEU to LEU or shut them down

- HEU fuels have been (some part only will be) transported back to Russia from Belarus, Bulgaria, the Czech Republic, Germany, Hungary, Kazakhstan, Latvia, Libya, Poland, Romania, Serbia and Montenegro, the Ukraine, Uzbekistan and Vietnam.

- 16 **VPVR casks** made by the Czech high-tech company SKODA JS supported by the Nuclear Research Institute at Rez (NRI) as for design safety analyses are participating in transport under the RRRFR program

- The RRRFR program intends to move 2 tons of HEU and 2.5 tons of LEU spent fuel to Mayak, the Russian reprocessing plant, to 2012. The program supposes about 50 shipments of both fresh and spent fuel to remove all HEU fuel discharged before reactors are converted to LEU or shut down.

- 17 countries have Soviet-supplied research reactors, and there are 25 such reactors outside Russia, 15 of them still operational.

LVR-15 research reactor of Nuclear Research Institute at Rez

- The research facility of the Russian origin (1957) underwent several reconstructions, the last one in 1996 increased the reactor power to 15 MW (thermal)
- Light water pool reactor has rectangular lattice of FAs in Be reflected core; fuel length is 60 cm
- LVR-15 reactor is used for irradiation experiments (material testing etc.) and research related to Gen IV issues; core with experimental channels can be flexibly reshuffled

LVR-15 research reactor IRT FAs consist of tube fuel elements :

- The former fuel IRT-2M (HEU): 80 wt.% ^{235}U, 4 tube elements made of Al alloy cladded UAl alloy... all shipped to Mayak (2007)

- The current fuel IRT-2M (MEU): 36 wt.% ^{235}U, 4 tube elements made of Al alloy cladded UO_2 with dispersed Al…most shipped to Mayak (2007)

- The future fuel (since turn of 2009/2010) IRT-4M (LEU): 19.75 wt.% ^{235}U , 8 tube elements made of Al alloy cladded UO_2 with dispersed Al…partly supplied

Fuel density

IRT-2M HEU: 3.8 g/cm^3

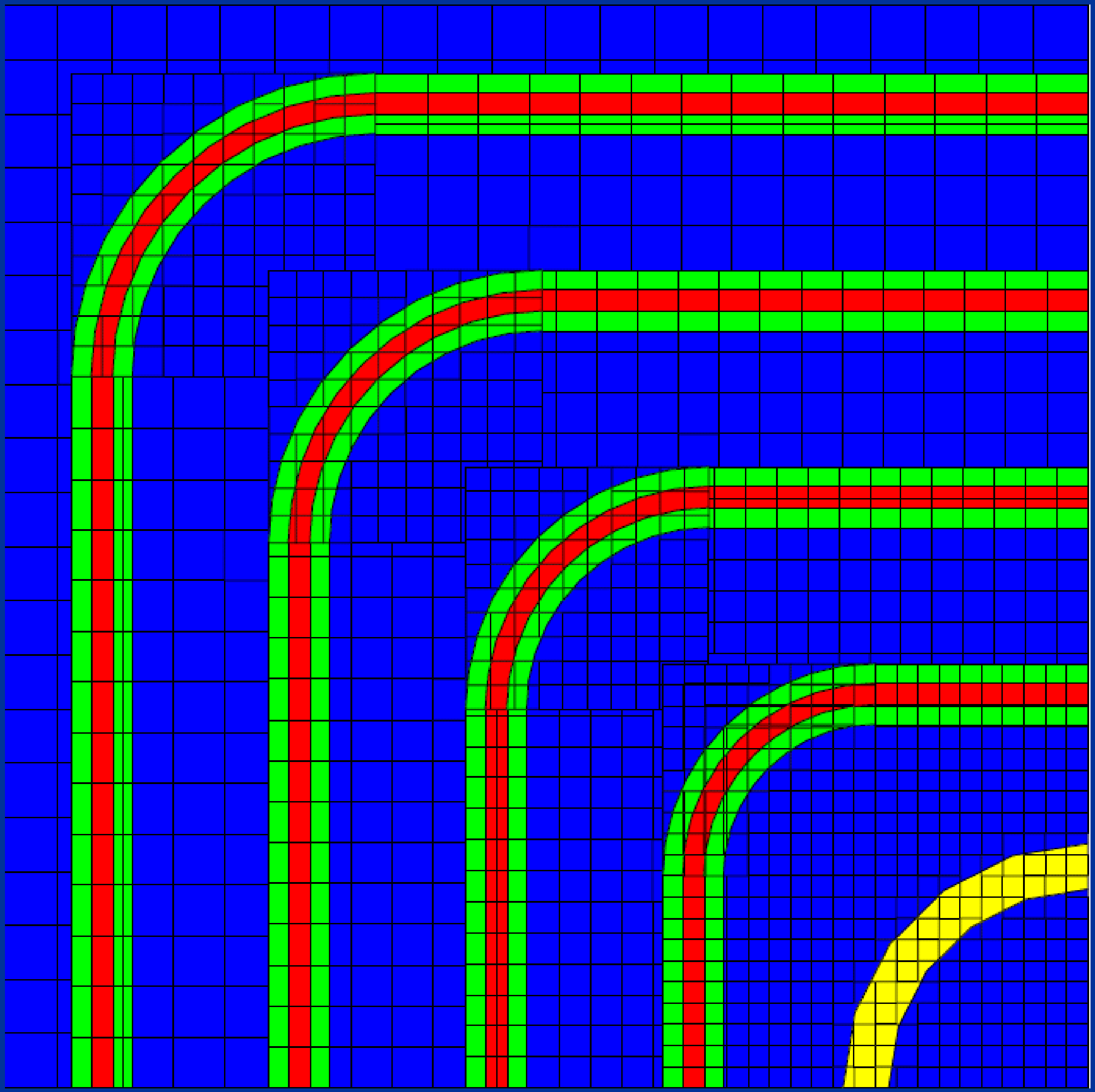
IRT-2M MEU: 4.3 g/cm^3

IRT-4M LEU: 7.4 g/cm^3

Fuel is fabricated in Novosibirsk chemical concentrates plant (http://www.nccp.ru) for TVEL Corporation (http://www.tvel.ru), the vendor of nuclear fuel for research reactors in Russia, Hungary, Kazakhstan, Uzbekistan, Poland, the Ukraine, the Czech Republic, Bulgaria, Vietnam and Libya

Quarter of horizontal section of IRT-2M four-tube fuel assembly

(TRITON output, real geometry)



The goals of calculation study

- To **build and verify models** of four- and eight-tube IRT fuel designs for new criticality assessments needed due to the change of IRT-2M MEU for IRT-4M LEU FAs of LVR-15 research facility

- To select independent and more advanced calculation methodology in comparison with those currently used for LVR-15 operation calculation

- To **check the models** (correctness, conservativeness, time consumption,..)

- To **perform initial scoping criticality, inventory and sensitivity calculations** related to the specific facilities connected with LVR-15 research reactor (fresh fuel storage, reserve pool and pool at the reactor, interim storage, spent fuel cask) using SCALE 5.1 codes for criticality and 2D depletion calculations

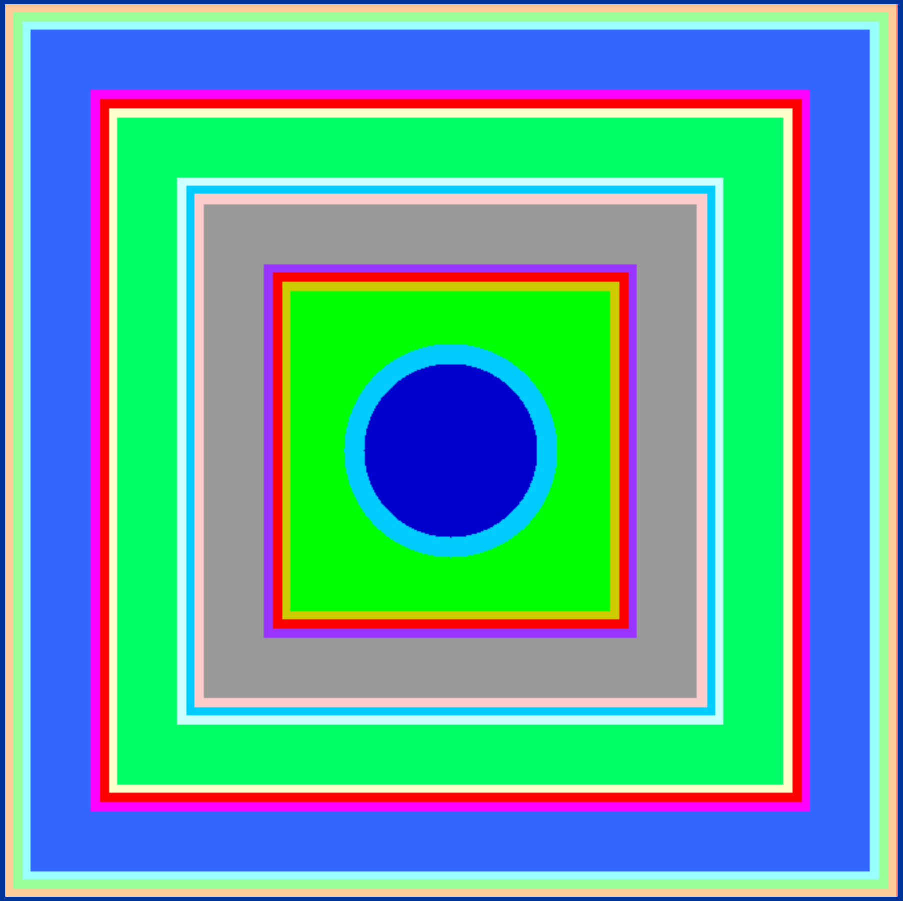
Calculation approach

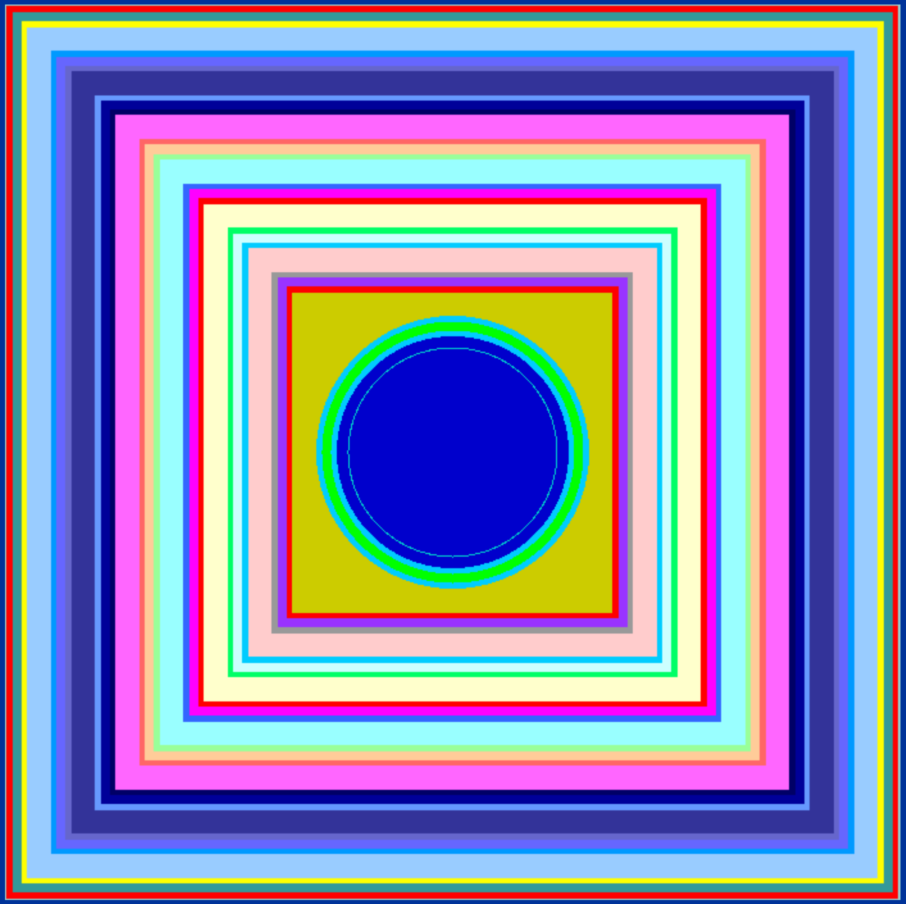
- **Rectangular model:** based on equivalent amount of materials in each tube element whose cross section is modeled squared
  - time consumption decrease for criticality calculations
  - not suitable for depletion calculation (more ‘materials’, modified number densities)
  - conservativeness must be proved

- **Real geometry model** describing each tube element with ‘round corners’
  - more correct

IRT-2M four-tube fuel assembly / IRT-4M eight-tube fuel assembly

(KENO Va output, modified geometry approach)



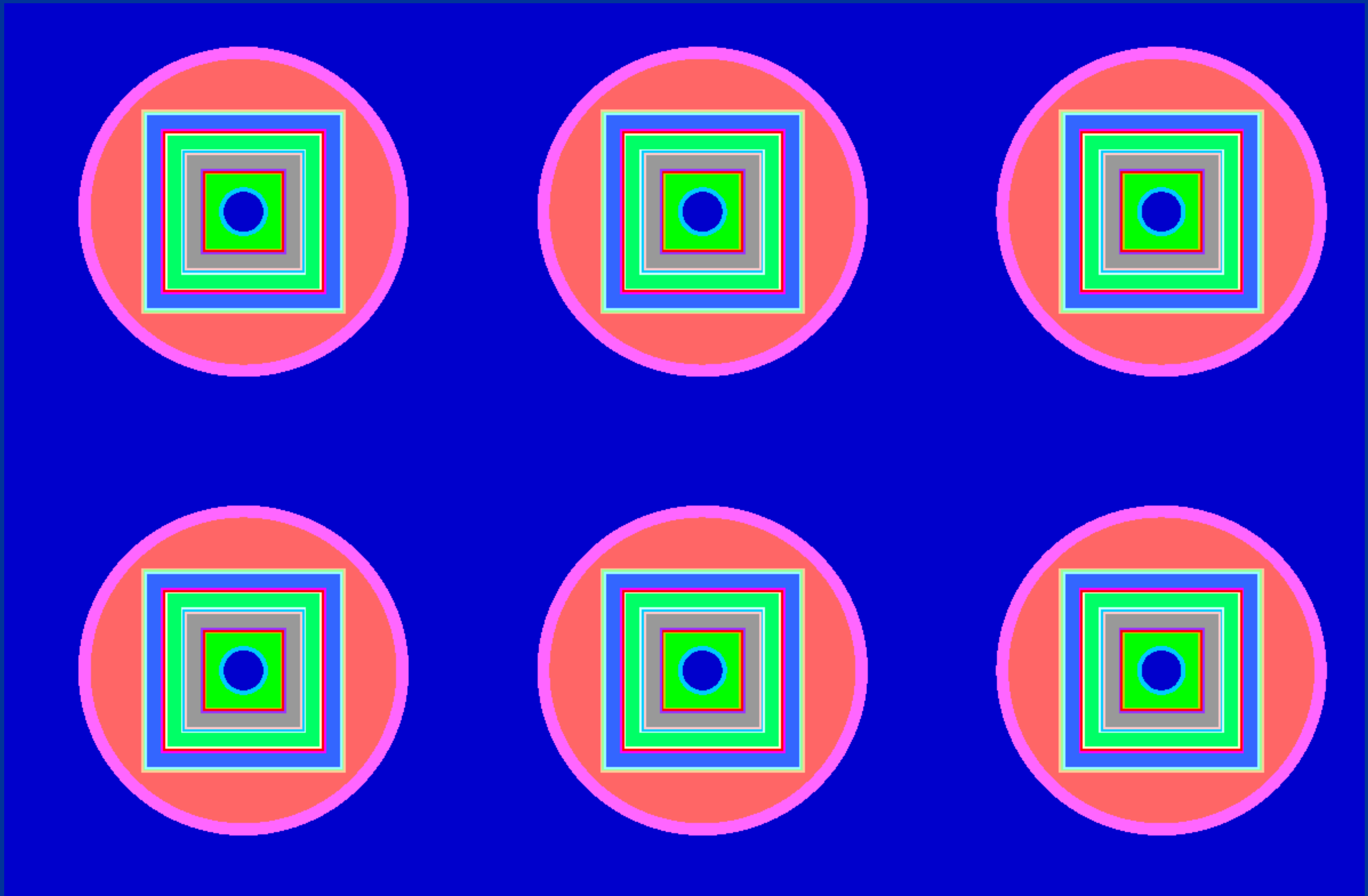


Model checking –
LVR-15 reserve pool application, fresh fuel approach

- **KENO Va / KENO VI criticality calculations for the pool fully loaded with current IRT-2M (MEU) fuel and future IRT-4M (LEU), both models used**
- **Result comparison**
  - KENO Va time consumption was smaller up to 6 times
  - Conservativeness of the ‘square model’ was proved (k_{eff} difference ≈ 1% to the real geometry ‘round’ model)
- **Calculation parameters tested**
  - use of 238-group neutron cross section library in comparison with 44-group library: **surprisingly minor impact** of ‰ order
  - data processing option - approach using either lattice cell or infinite homogeneous medium: **‘latticecell’** (asymmetric or symmetric array of slabs) **option recommended (**influence of % order)

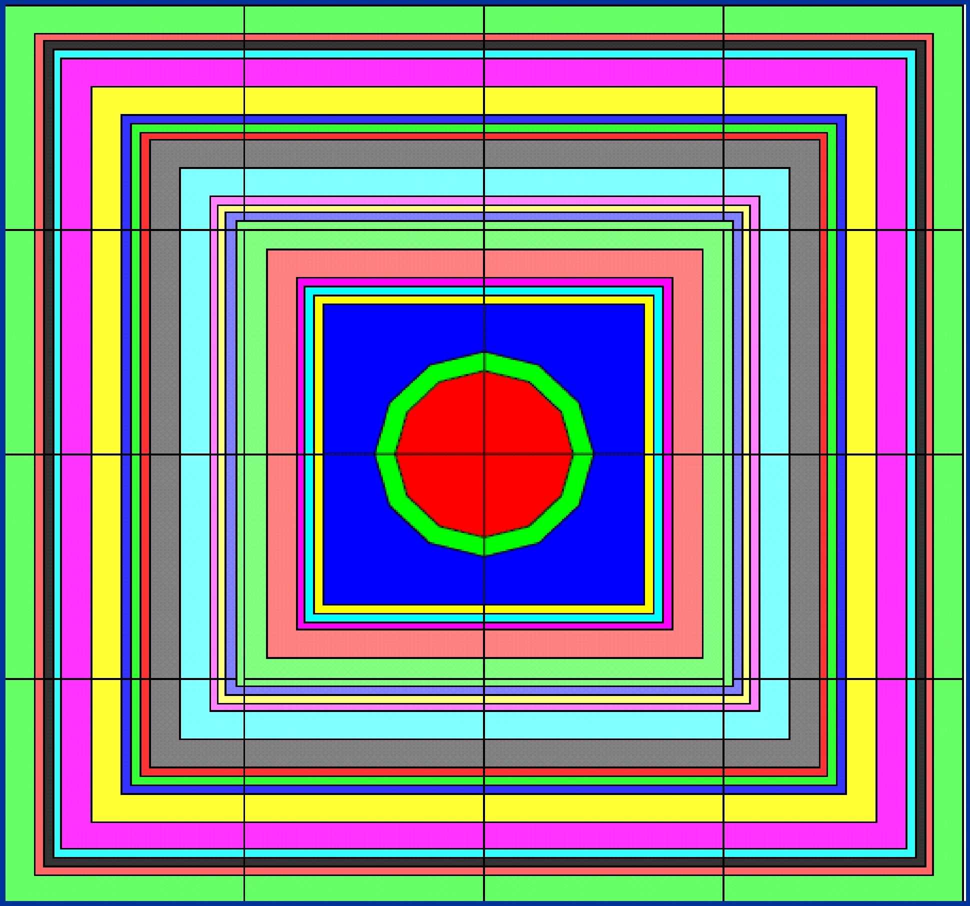
Array of IRT-2M MEU fuel assemblies inserted in Al storage tubes arranged at pitch of LVR-15 research reactor reserve pool

(KENO Va output, modified geometry approach)



IRT-2M MEU four-tube fuel assembly

(TRITON output, modified geometry approach, 'asymslabcell' lattice cell data processing option)



Model checking – inventory calculation

- TRITON depletion calculations (2D sequence with NEWT code) for current IRT-2M (MEU) fuel were made
- Typical irradiation history selected: 350 eff. days to reach burnup of about 195 MWd/kg_{HM} limiting fuel use in the core
- Both fuel models were checked
- Calculation parameters were tested
- Isotopics of 27 ‘BUC isotopes’ were used for follow-up criticality calculations of VPVR cask

Fixing the first and the last depletion intervals as very narrow, NEWT within TRITON provided 2D transport eigenvalue calculation for the given system so change of models or entering parameter values could be assessed for both ‘fresh’ and ‘spent’ fuel systems

- ‘Round’ model preferred: time consumption was smaller, unmodified resulting number densities allow more freedom in the use for criticality calculation

- 238-/44-group LWR cross-section libraries comparison: previous finding confirmed - small impact on the results

- lattice cell/infinite homogeneous medium data processing option: previous result confirmed – the lattice should be taken into consideration

- fuel zoning for the four-tube FA was checked (‘asymslabcell’ lattice cell data processing option): no significant effect for the given system

- NEWT calculation parameters tested:
  - grid coarseness selection: tested for each unit boundary of the model geometry, later experience is obtained, also compromise with time consumption taken
  - S_n and P_n orders: after finding small influence (less than ‰) S_4 and P_3/P_1 for moderator/otherwise were set up for the subsequent calculations

- depletion calculation parameters tested

NEWT 2D criticality calculations

(horizontally infinite array of fresh and spent IRT-2M FAs at reactor pitch)

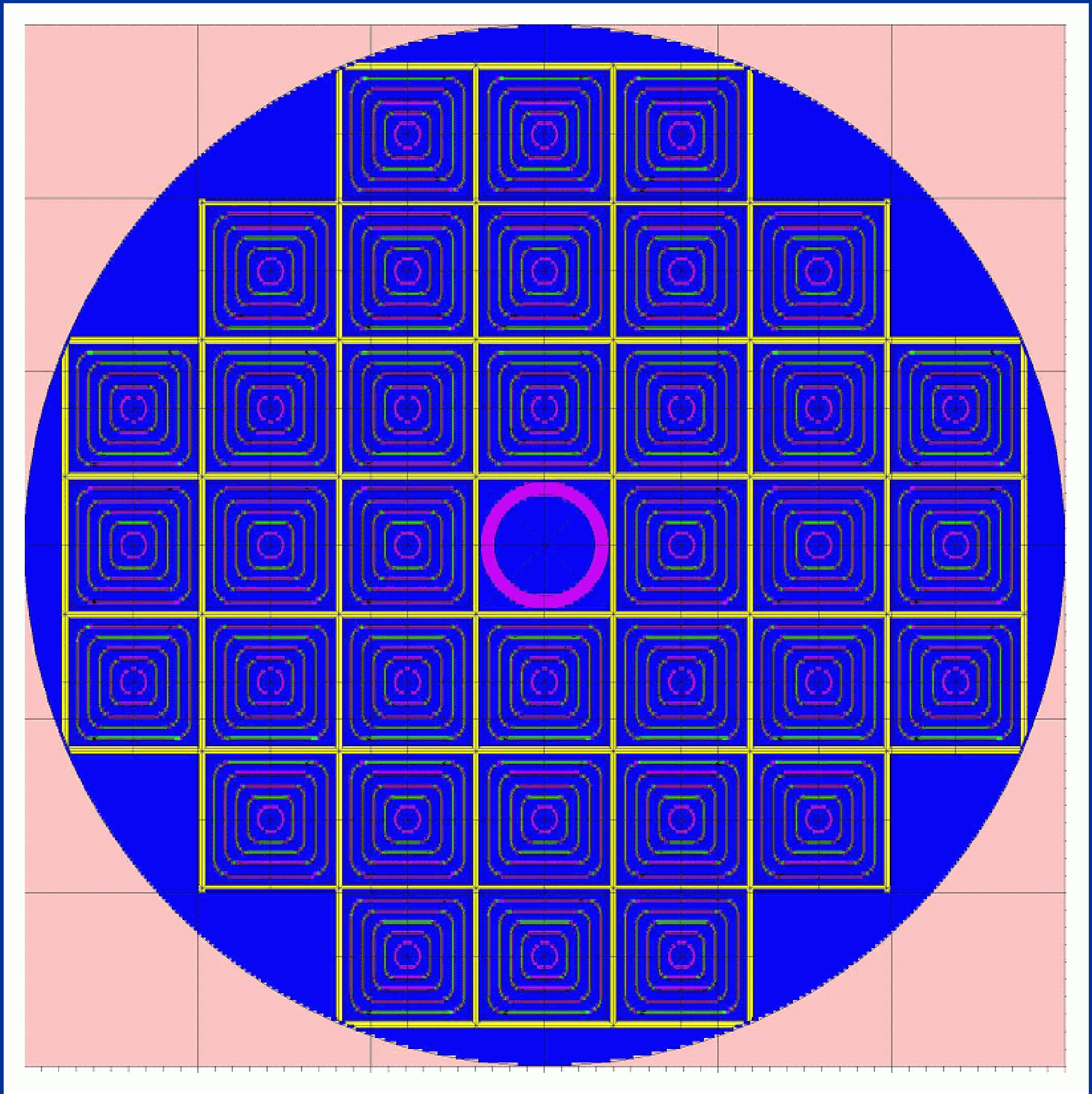
Type of the calculation: Model/grid/library	Data processing option	Cycle Splitting into	Number of ORIGEN libraries per cycle	Time [min] *)	k_{∞} (fresh)	k_{∞} (spent)
'Round' 20x20 44g	Latticecell	2	1	36	1.5058	1.0588
'Round' 20x20 44g	Latticecell	2	3	50	1.5058	1.0579
'Round' 20x20 238g	Latticecell	2	1	240	1.5061	1.0593
'Round' 20x20 44g	Latticecell	2	6	130	1.5058	1.0579
'Round' 20x20 44g	Latticecell	6	1	49	1.5058	1.0578
'Round' 20x20 44g	Inhommedium	6	1	50	1.5470	1.0672

*) value for intercomparison only

Cask criticality calculations

- A certain part of HEU/MEU spent fuel from the research reactors of the former Soviet origin (most cooled for long time) is still outside Russia, RRRFR program is going on
- Fresh HEU is still manufactured (e.g. IVV-10 design) and offered by TVEL, the Russian fuel vendor
- If cask used for shipments is well optimized (e.g. using BUC) it would enable increase of capacity and decrease of the number of shipments from some specific countries
- Criticality analysis of the VPVR cask specifically for IRT-2M MEU fuel was made as a calculation case

Horizontal section of VPVR cask basket loaded with IRT-2M fuel assembly of 36 wt.% ^{235}U enrichment (MCNP output)



VPVR cask criticality safety analyses for different research reactor fuels

(borated plates in basket, NRI design calculations due to licensing)

Fuel design	Enrichment wt.% ^{235}U	VPVR cask k_{eff} upper limit
IRT-2M	90	0.605
IRT-3M	90	0.733
S-36	36	0.556
VVR-M	36	0.522
VVR-M2	36	0.483
VVR-M5	36	0.570
VVR-M5	90	0.609
VVR-M7	90	0.609
EK-10	10	0.500
IRT-4M	19.75	0.676
IRT-2M	36	0.671
IRT-2M	80	0.603

(Re)calculation of the VPVR cask for IRT-2M MEU

Spent 1/Spent 2 – 'latticell'/'infhommedium' data processing option for TRITON depletion calculation, no axial burnup profile taken into account

IRT-2M 36 wt.% ^{235}U FA	VPVR basket intervening plates	Calculation time [h]	Cask k_{eff} (as calculated)	σ	Burnup credit BUC[%]
fresh	borated	53	0.5701	0.0001	
fresh	non borated	65	0.8920	0.0001	
Spent 1	borated	61	0.3076	0.0001	46
Spent 1	non borated	82	0.5261	0.0001	41
Spent 2	borated	62	0.2881	0.0001	49
Spent 2	non borated	84	0.4988	0.0001	44

Results of the VPVR cask criticality recalculations for IRT-2M MEU

- Basket intervening plates needn’t be borated
- Inventory computed using the infinite homogeneous medium option of the data processing (‘Spent 2’ data) resulted in highly non conservative k_{eff} of the cask system
- Optimizing VPVR cask taking BUC into consideration would be much beneficial (but, unfortunately, safe and correct BUC implementation was not possible due to lack of reliable data for most of the other spent fuels)

Summary

- **The use of the 44-group neutron cross section library is not strongly excluded**
- **Real ‘round’ model and the use of the lattice cell option for data processing are highly recommended**
- **As for the NEWT calculations within TRITON, S_4 and P_3/P_1 for moderator/otherwise are possible options; the grid coarseness should be tested for each specific case**
- **As for the TRITON depletion calculations, time-step length should be checked (possibility of splitting the cycle, change of number of libraries per cycle)**
- **Conservativeness of all simplifying options must be verified for each system/calculation case**

The fuel models and calculations presented were performed due to change of the LVR-15 research reactor fuel. They are considered initial scoping calculations using SCALE 5.1 codes for criticality and 2D depletion and as such can serve as a prototype for future modeling efforts (possibly also including BUC implementations)