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

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Abstract

The LVR-15 is the one of the main research facilities of NRI at Rez, the Czech Republic. As one of the older pool LWR research reactors of the Russian origin situated in many countries all over the world, the LVR-15 used IRT-2M HEU fuel enriched to 80 wt. % ^{235}U until recently. It was subsequently replaced by IRT-2M fuel of 36 wt. % ^{235}U (MEU) and finally by the IRT-4M fuel of 19.75 wt. % ^{235}U (LEU) being implemented at present under the Reduced Enrichment for Research and Test Reactors (RERTR) Program. The spent fuels are stored in the inner area of NRI - in pool at the reactor and then in the interim storage. In the course of the LVR-15 operation in NRI the original reactor was upgraded to give thermal power up to 15 MW and fuel is used up to quite a high burnup of about 60% of the ^{235}U initial amount. In the framework of the International RRRFR Project (Russian Research Reactor Fuel Return) Project of US DOE/GTRI (National Nuclear Security Administration's Global Threat Reduction Initiative) the spent fuel of the highest initial enrichment and most of fuel of the middle initial enrichment were transferred to the Mayak reprocessing plant (Russian Federation) in 2007. The rest of MEU spent fuel is being stored in NRI.

To try to optimize the storage capacity and assess burnup credit of the LVR-15 storage facilities the fuels were modeled for calculations by TRITON and KENO codes (SCALE 5.1). Models and results of preliminary calculations are performed and presented. The new tools for the depletion and criticality safety calculations should also support the LVR-15 operational calculations currently based on 1D depletion model (WIMSD4 code modified for IRT geometry by ANL under the RERTR program) and diffusion flux calculations (NODER, four-group 3D diffusion code) in connection with the requested fast and effective conversion to the LEU cores.

Introduction

The study described below was focused on preparation of IRT fuel models and making scoping calculations as a basis for the further use in several applications away of reactor as well as supporting research facility operation when results of 2D depletion calculations are needed.

IRT fuel is manufactured in Russia for the research reactors which the former USSR built in many countries all over the world. Until recently the LVR-15 research facility in NRI at Rez used the IRT-2M HEU fuel enriched to 80 wt. % ^{235}U. However, in 2004 the Russian-US government agreement on cooperation in removing Russian-made nuclear fuel from research reactors to Russia was signed /1/. The main goal of the Russian Research Reactor Fuel Return (RRRFR) program is to reduce global stockpiles of weapons-usable nuclear materials, to reduce the threat of international terrorism, and to prevent the weapon proliferation.
Under the RRRFR program, the eligible countries should convert their research reactors from using high-enriched (HEU) to low-enriched uranium (LEU) fuel upon availability, qualification, and licensing of suitable LEU fuel. Russia has agreed to take back spent and fresh nuclear fuel from research reactors whose operators agree to convert the reactors on LEU or shut down. 17 countries have Soviet-supplied research reactors, and there are 25 such reactors outside Russia, 15 of them still operational. The eligible countries receive financial and technical assistance from the United States and others to ship their fresh and spent research reactor fuel to Mayak, the Russia reprocessing plant. A total of approximately 838 kg of Russian-origin HEU (spent and fresh) have been returned from Bulgaria, the Czech Republic, Germany, Hungary, Kazakhstan, Latvia, Libya, Poland, Romania, Serbia, Uzbekistan and Vietnam. The RRRFR program intends to move 2 tons of HEU and 2.5 tons of MEU spent fuel to the Mayak plant to 2012./1/

For the time being the biggest shipment under RRRFR program was realized in Dec. 2007 from the Czech Republic. The shipment contained 252 IRT-2M spent fuel assemblies (37.3 kg) of 80 wt.% ^{235}U, 91 IRT-2M spent fuel assemblies (43.4 kg) of 36 wt.% ^{235}U and 252 EK-10 spent fuel assemblies of 10 wt.% ^{235}U (281.4 kg). The fuel assemblies (FA) spent in LVR-15 research facility of the Nuclear Research Institute at Rez near Prague (NRI) were transported through Slovakia and the Ukraine by train to Russia in 16 VPVR casks developed and made by the Czech high-tech company SKODA JS in collaboration with NRI. The specific VPVR casks were licensed for transport in the Czech Republic, Slovakia, the Ukraine, Russia, Hungary and Bulgaria and could be potentially used for similar shipments from around the globe during the next decade.

The cask safety analyses were performed in NRI for many specific designs of the Russian fuels used in the past as well as currently in the research reactors of the Russian origin. As the most reactive the IRT-3M eight-tube fuel design with high enrichment of 90 wt. % ^{235}U was evaluated. Unfortunately, many needed details of spent fuel irradiation history were unknown (not available) so neither any credits (e.g. burnup credit) nor cooling times (despite mostly very long cooling of spent fuel in the individual countries) could not be taken into cask design development and the developers had to accept quite a big data uncertainty. Finally, shielding and heat transfer did not become a crucial point of the analyses as the cask design was based on robustness. As for the criticality margin, it was much increased using ATABOR stainless steel plates containing 1.2 % of boron in the cask basket. Thus, the VPVR cask was very safe but somewhat heavy.

Mentioned below, the VPVR cask criticality was re-assessed specifically for the Czech spent fuel IRT-2M MEU as a calculation exercise. At the same time, new analyses and scoping criticality and shielding calculations are needed in connection with the change of LVR-15 research reactor operation from MEU to LEU under RRRFR program which is underway for the research facility. The new tools for the depletion and criticality safety calculations should support the LVR-15 operational calculations currently based on 1D depletion model (WIMSD4 code modified by ANL under the RERTR program) and diffusion flux calculations (NODER, four-group 3D diffusion code). The new calculation methodology using the realistic fuel models, multigroup approach and giving possibility to calculate fuel depletion in 2D geometry is thus considered as profitable for both the LVR-15 operation and the fuel management as well.

Calculation approach

Two approaches can be used for modeling the Russian IRT fuel of a specific shape for depletion and criticality calculations.

Generally, the IRT fuel assemblies and the other similar FA designs for research reactors made in Russia consist of several concentric fuel elements - cylindrical/cylinder-like tubes or tubes of square sections with round corners. The NRI LVR-15 research facility uses FAs consisting of concentric tubes of square sections with round corners (see the shape in Fig.1) building the rectangular core lattice. In the past, 80 wt.% ^{235}U enriched IRT- 2M fuel (HEU) was used and changed into IRT - 2M fuel with enrichment of 36 wt. % ^{235}U (MEU) in 1996-7. The latter fuel has been used up to now and its change for IRT-4M fuel of 19.75 wt. % ^{235}U (LEU) is planned for the near future under the RRRFR program. The IRT-4M design differs from IRT-2M consisted of four fuel elements creating FA not only in enrichment (LEU) but also in the number of fuel elements in FA (eight). The fuel element tubes for the fuel designs mentioned above are made of Al alloy cladded UAl alloy (HEU) or UO_2 with dispersed Al (MEU, LEU).

To incorporate the 'round corners' when modeling the fuel shape for the depletion and criticality analyses of the systems with the IRT fuel assemblies needs evaluation. The simplest way is to create an equivalent rectangular model which enables to use the criticality codes working quickly (e.g. KENO Va) in comparison with the codes working in a complex geometry (e.g. KENO VI or MCNP). On the other hand, in the core of the research reactor there are non-fuel experimental channels (by and among the fuel channels) where flux or spectra should be calculated as accurately as possible so modeling the adjacent fuel assembly in the real geometry can be important.

Therefore, two models of IRT FA were prepared for calculations - the realistic complex geometry model Fig.1 and an equivalent rectangular geometry model where the 'round corners' were neglected but the original amount of all the media were kept unchanged in the new volumes Fig.2.

As for the criticality calculations, the use of the realistic model could be expected for output of flux functionals, the rectangular model can be used when integral characteristics of the given system as multiplication factor are of main interest (if it gives a conservative result in comparison with the realistic model!) or when series of scoping calculations are made and time consumption is in view.

As for the depletion calculations, the equivalent rectangular fuel model using modified initial material data is not suitable for this case; the depletion calculations the results of which are mentioned in the paper were performed in the real geometry model.

Codes and Libraries Used for Calculations

The calculations described below were made with TRITON, KENO Va and KENO VI modules of SCALE 5.1 /2/ and with MCNPX code /3/.
Using TRITON sequence with NEWT code /2/ the 2D depletion calculations were performed, KENO modules were used for 3D criticality safety analyses of the reserve pool at the LVR-15 reactor Fig.3 and MCNP criticality calculations resulted in series of k_{eff} for VPVR cask loaded with fresh or spent ITR-2M fuel Fig.4.
As far as the LWR library for SCALE neutronics calculations is concerned, the use of 44-group SCALE library (44groupndf5) instead of 238-group library (238groupndf) had to be verified as - despite IRT being LWR fuel - FA shapes and enrichments are very specific. As for MCNP, DLC-205 and libraries of the previous RSICC MCNP packages up to 2005 were available for the cask criticality calculations.

Models and Parameters, Calculations Made

Criticality calculations of IRT fresh fuel system

The models of IRT fuel assemblies consisting of four and eight tube elements were prepared for TRITON calculations. Because KENO Va and KENO VI criticality codes use the same fuel geometry models as TRITON, the criticality calculations with these codes were made for fresh IRT-2M (four-tube FA) and IRT-4M (eight-tube FA) to check the built models at first. As for these calculations, the reserve pool (Fig.3) at LVR-15 reactor in NRI at Rez was selected. The application (described in /4/) contributed to series of the safety analyses needed to support licensing the planned change of IRT-2M (36 wt.% ^{235}U) for IRT-4M (19.75 wt.% ^{235}U) fuel. Besides the results themselves, calculation parameters and model features were checked to correct or optimize the fuel model.
Due to the Monte Carlo method of calculation the source convergence was carefully studied for both the finite reserve pool system and the reserve pool infinite lattice as well. A big attention was focused on modeling the IRT-2M and IRT-4M FAs themselves and on the assessment if it is possible (conservative?) and if it is worth (time consumption?) modeling the system in a rectangular geometry for using KENO Va. It was shown /4/ that the equivalent rectangular model gives fairly conservative results for the given system and therefore can be used for safety analyses proving subcriticality. The relative difference between the multiplication factors of the systems using either the real geometry ('round corners') or equivalent rectangular model ('square corners') models of the IRT tube fuel elements was **less than 1%**. The calculations for the finite pool system were well convergent. The use of 238-group neutron cross section library in comparison with the 44-group one within SCALE and the data processing option change (approach using either lattice cell or infinite homogeneous medium) brought about only a minor impact to the results, 'infhommedium' option was conservative as for the pool system (IRT fuel at pool pitch) but more correct approach is the 'latticecell' option, of course. The lattice unit was described in slab geometry as a symmetrical slab cell, effect of FA zoning was not checked up (but is planned to be evaluated). The time

consumption for this application when computing with KENO Va was found about 6 times smaller in comparison with KENO VI calculations using the real geometry model.

Inventory calculations

Further, depletion calculations with TRITON were made supposing that FA of IRT-2M fuel (36 wt. % ^{235}U) was working in the LVR-15 research facility core for a typical total of about 350 effective days to reach burnup of about 195 MWd/kg_{HM} limiting its use in the experimental cores.

The calculations were aimed at finding the number densities for the set of 'burnup credit' isotopes, actinides U-234, U-235, U-236, U-238, Pu-238, Pu-239, Pu-240, Pu-241, Pu-242, Am-241, Am-243, Np-237 and main fission products Mo-95, Tc-99, Ru-101, Rh-103, Ag-109, Cs-133, Nd-143, Nd-145, Sm-147, Sm-149, Sm-150, Sm-151, Sm-152, Eu-153 and Gd-155 present in spent fuel.

Using the TRITON sequence with NEWT code for 2D flux calculation the following parameters and their influence were checked for obtaining the correct results with sufficient accuracy for the given model:

- SCALE neutron cross section library (238-group, 44-group library)
- data processing option (infinite homogeneous medium, lattice cell)
- parameters connected with the discrete ordinates method used by NEWT (the space grid structure, the order of S_n quadrature, P_m scattering order)
- parameters connected with ORIGEN calculation (fuel cycle splitting, number of libraries per cycle)

As a tiny parts of the depletion cycle were divided at the beginning and at the end, the TRITON calculations provided also k_∞ multiplication factor for the horizontally infinite array of IRT-2M FA at the reactor core pitch as resulted from NEWT code eigenvalue calculation at the cycle beginning when the fuel is fresh and at the cycle end when the fuel is fully depleted. Thus, the system sensitivity to the checked parameters could be evaluated.

The lattice for data processing was described in a slab geometry using both symmetrical and asymmetrical slab cell. The former introduces a fuel model using only one fuel mixture in FA while the latter allows the FA zoning (for four-tube FA four fuel mixtures was generated, one for each fuel element the FA consists of). However, the results showed no significant influence on the depletion results (0.02% relative change of k_∞ for the given spent fuel system in comparison with the symmetrical slab option whose results were on the conservative side) while the time consumption increased about twice. The following result was obtained when checking the effect of the order of S_n quadrature. Fixing symmetrical slab option and varying n=4,6 and 8 the k_∞ of the spent fuel system differs 0.2 % and 0.07 % for S_4 and S_6 relative to S_8 approximation, respectively. Correctly, the k-sensitivity should be checked with a code based on a different method, nevertheless, the S_4 was fixed as sufficient for scoping calculations since the results were on the conservative side and time consumption for each higher order would increase about twice.

The other parameters, the influence of which was checked at first, were the number of energy group of the library selected within SCALE and the neutron cross section data processing option. Further, splitting the cycle of fuel depletion combined with changes of the number of libraries per cycle was made.

The resulting k_∞ are listed in the Table 1, below; there is visible that only the change of the data processing option - infinite homogeneous medium in comparison with the (symmetrical) lattice cell - shown an appreciable effect (of about 1%).

Table 1 NEWT k_∞ calculation results for horizontally infinite array of IRT-2M FAs, 36wt. % ^{235}U, at LVR-15 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	**Infhommedium**	6	1	50	**1.5470**	**1.0672**

Note:
S and P order options: S_4, P_3 - for moderator /P_1 - otherwise
*) value for intercomparison within the Table 1 only

Sensitivity calculations

Finally, using the results of the depletion calculations made with TRITON for IRT-2M MEU fuel assembly of four-tube fuel elements the following criticality calculations of the VPVR cask Fig.4 fully loaded with the IRT-2M FAs were performed with MCNP (/5/,/6/,/7/).
It should be noted that other IRT fuel - HEU IRT-3M eight-tube fuel design with enrichment of 90 wt. % ^{235}U - was evaluated as the most reactive fuel for the cask so the current calculations mentioned below for MEU IRT-2M show lower k_{eff}. The MEU IRT-2M fuel used (and still being used) in LVR-15 research facility of NRI at Rez was chosen for these recalculations because its irradiation history is best documented of all the other fuels for which the cask design was licensed for transport.
Two different sets of spent fuel data was used as resulted from the TRITON depletion calculation using either 'latticecell' or 'infhommedium' option for the cross section data processing. The IRT fuel was modeled in a realistic geometry ('round corners') Fig.1 in all the calculated cases. The resulting k_{eff} values both assessing the spent fuel input data processing option and evaluating a basket structural material influence are listed in the Table 2 below:

Table 2 MCNP k_{eff} calculation results for VPVR cask loaded with IRT-2M, 36wt.%^{235}U

IRT-2M 36 wt.% FA fuel	VPVR basket intervening plates	Number of neutrons per generation	Number of generations	Number of neutrons skipped	Time consumption *) [h]	VPVR cask k_{eff} (FAs at pitch of 7.4 cm)	Estimated StD	Burnup credit BUC [%]
fresh	borated	100000	550	50	53	0.5701	0.0001	
fresh	non borated	100000	550	50	65	0.8920	0.0001	
Spent 1	borated	100000	550	50	61	0.3076	0.0001	**46**
Spent 1	non borated	100000	550	50	82	0.5261	0.0001	**41**
Spent 2	borated	100000	550	50	62	0.2881	0.0001	**49**
Spent 2	non borated	100000	550	50	84	0.4988	0.0001	**44**

Note:

Spent 1 data resulted from the TRITON calculations using:

'latticecell' option for neutron cross section data processing, the same fuel mixture for all the tube elements, 1 burnup cycle split into 6 calculation cycles plus very short (the first and the last one) cycles due to k-calculation of the horizontally infinite array of four-tube FAs at the core pitch

Spent 2 data resulted from the TRITON calculations using:

'infhommedium' option for cross section data processing, the same fuel mixture for all the tube elements, 1 burnup cycle split into 6 calculation cycles plus very short (the first and the last one) due to k-calculation of the horizontally infinite array of four-tube FAs at the core pitch

BUC: (k_{eff} (fresh)-k_{eff} (spent))/k_{eff} (fresh)*100

*) MCNP criticality calculations ran on PC with Intel Core i7 965, 3.20 GHz

In addition to the results listed in the Table 2 above, the BUC value for horizontally infinite array of FAs arranged at the LVR-15 research reactor pitch of 7.15 cm can be mentioned to add knowledge on the MEU IRT-2M fuel system behavior:

Table 3 BUC for the system of the horizontally infinite array of IRT-2M FAs, 36 wt. % ^{235}U, at LVR-15 reactor pitch

IRT-2M MEU fuel	Fresh fuel k_∞	Spent fuel k_∞	BUC [%]
Spent 1	1.5058	1.0578	**30**
Spent 2	1.5470	1.0672	**31**

Taking into account the results shown in Table 2 and Table 3 the influence of the system structure on BUC as well as the influence of the data processing (Spent 1, Spent 2) on multiplication factor and therefore on BUC in both systems are visible. While in the case of the VPVR cask the options 'infhommedium' (Spent 2) and 'latticecell' (Spent 1) resulted in about 6 % (plates with boron in the basket) or 5 % (no boron in the basket plates) BUC difference, the horizontally infinite array of the same FAs at the reactor pitch shows the difference less than 1% only. Finally, it is worth mentioning that using 'infhommedium' option the results would not be conservative for the VPVR system (see Table 2) contrary to the results for the horizontally infinite array.

Conclusions

IRT models able to be used for safety analyses of fresh and spent fuel facilities connected with the LVR-15 research reactor (fresh fuel storage, reserve pool and pool at the reactor, interim storage, and spent fuel cask) or for management of other similar designs of the Russian-origin fuel for the research reactors were developed and checked up. The analyses presented in this paper 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 that integrate both specific data and modeling to add technical defensibility to future performance predictions (possibly also including BUC implementations).

The series of scoping calculations described above resulted in several findings:

- the use of the 44 library is not strongly excluded if low time consumption is needed when making scope calculations (despite being generated for different LWR spent fuel environment: the fine-group 238GROUPNDF5 cross sections were collapsed into the broad-group 44GROUPNDF5 structure using a fuel cell spectrum calculated based on a 17 × 17 Westinghouse PWR assembly, for more details see /2/)
- the use of the lattice cell ('latticecell') option for data processing is highly recommended (the FA tube element shape is taken into account in some way, the cross sections are corrected for both geometric and resonance self-shielding, for more details see /2/)
- as for the NEWT /2/ 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 /2/ depletion calculations, time-step length (there is a possibility to split a cycle into several successive shorter time intervals) and number of the libraries per cycle (problem-dependent cross-section processing) should be checked
- the conservativeness of all the simplifying options must be verified for each system/calculation case

Both IRT fuel shape models ('round' and 'square' tube elements) can be used for calculations. As for criticality calculations, the rectangular model can be recommended if it is conservative for the given system and requested output quantities would not be significantly affected by this approach. The advantage is KENO V time consumption which was proved to be about 6 times less than when modeling the IRT real fine shape in KENO VI. As for inventory calculations, results of the calculations for the 'round' model of the IRT fuel were shown and recommended. Running for a shorter time due to fewer materials involved in the depletion calculations (in comparison with the adequate 'square' model), the real geometry model uses non-modified data thus giving freedom for the following use of non-modified results.

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/7/ S. Flibor, 'MCNP criticality calculations of VPVR cask with ATABOR basket intervening shielding of 4 mm, Reports NRI 1269-Z and 1172-Z with Amendment, 2004 (proprietary)

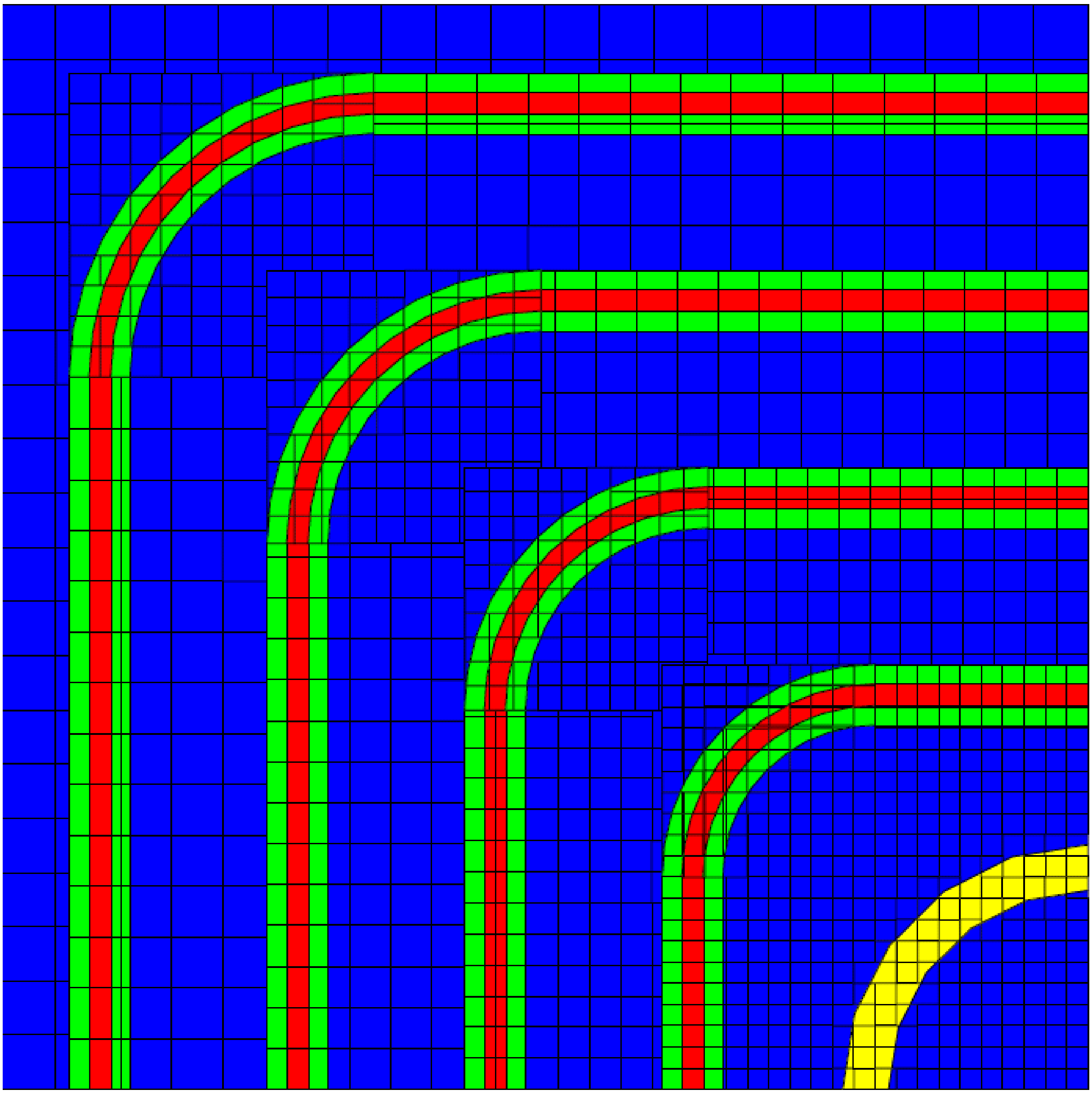

Fig.1. Quarter of horizontal section of IRT-2M four-tube fuel assembly (TRITON output)

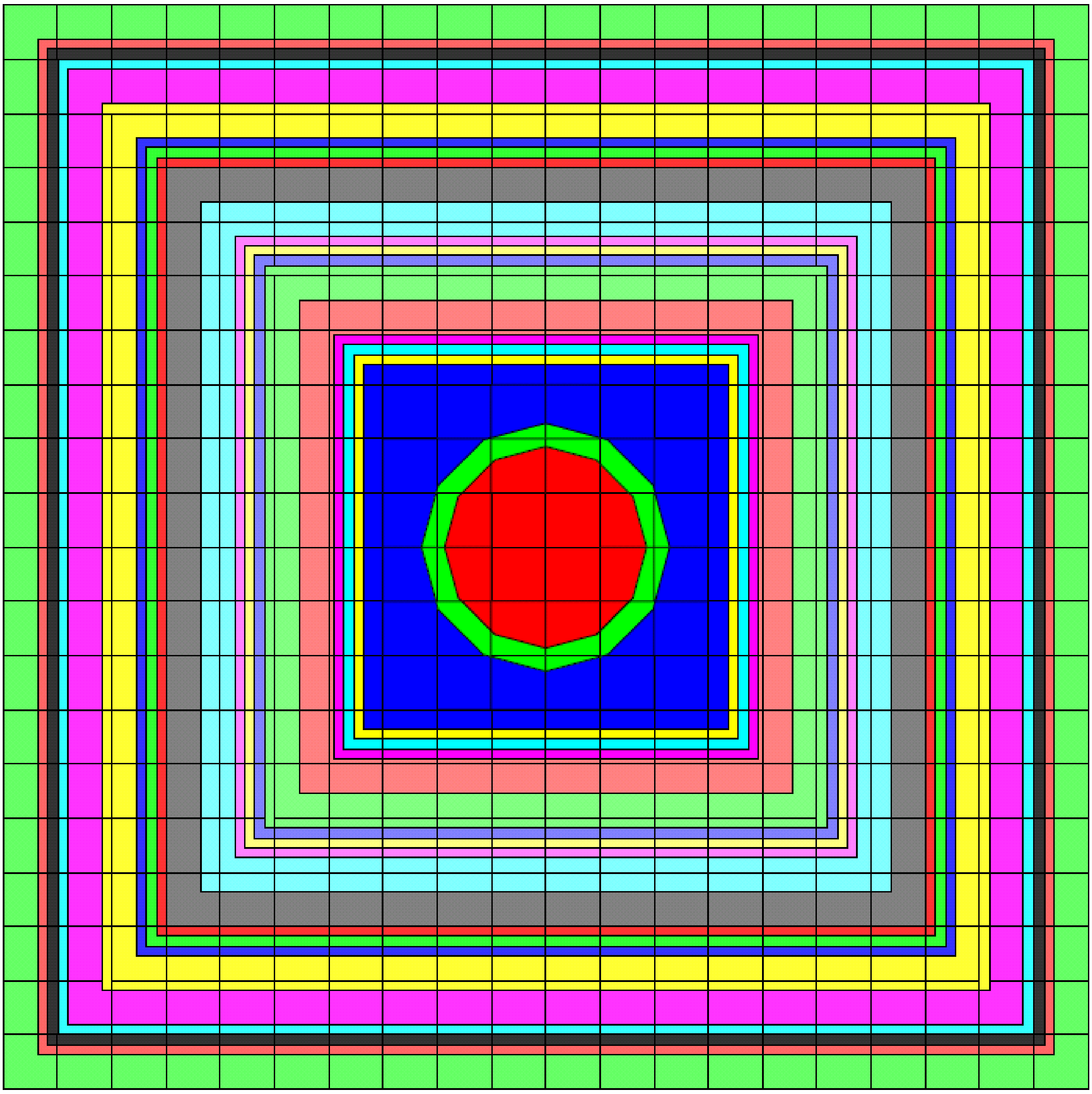

Fig.2. IRT-2M four-tube fuel assembly (TRITON output), modified geometry approach, model for 'asymmetric slab' as 'lattice cell' data processing option

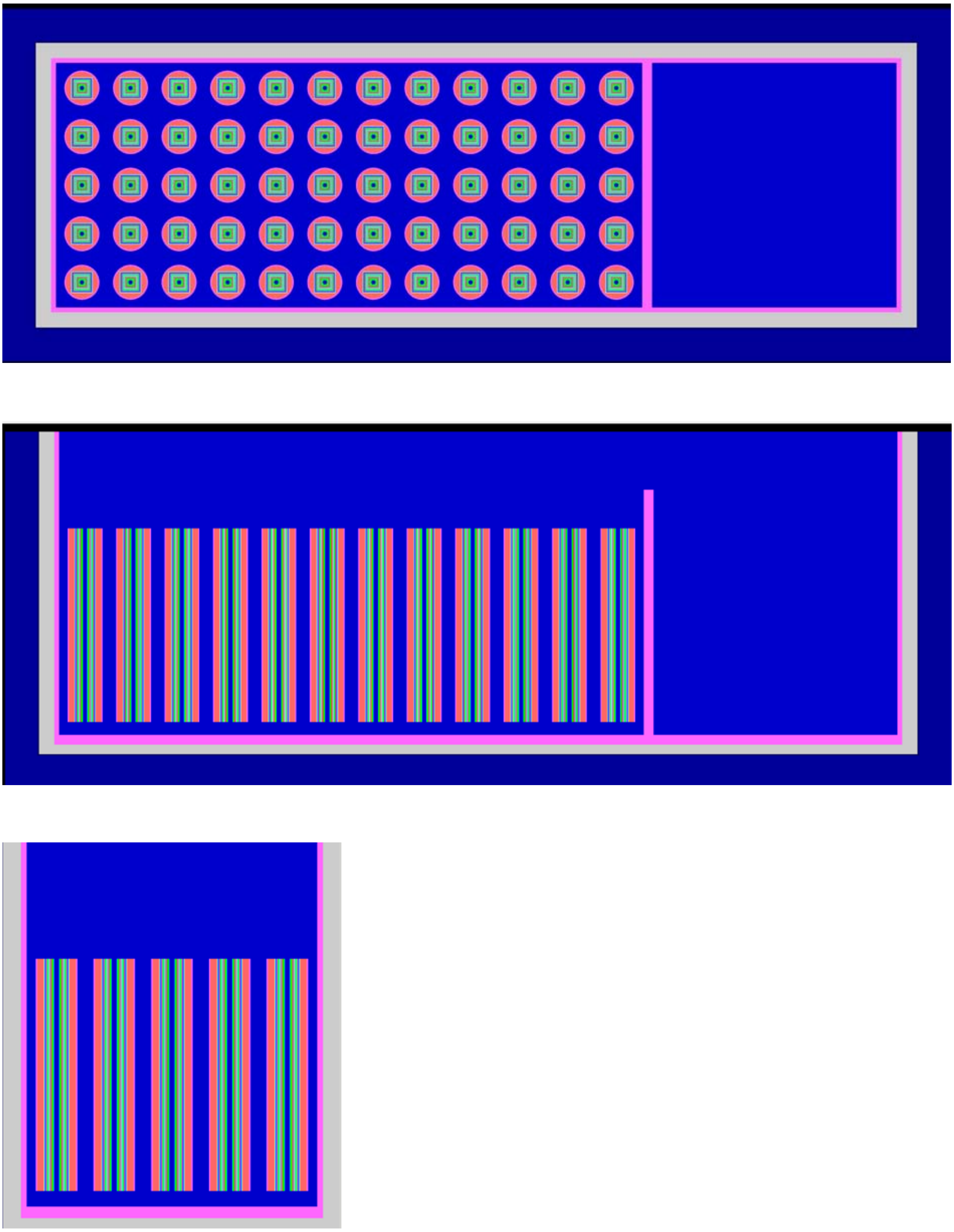

Fig.3. Reserve pool at LVR-15 research facility for IRT-2M or IRT-4M fuel assembly (KENOVa output), modified geometry approach, Al storage tubes

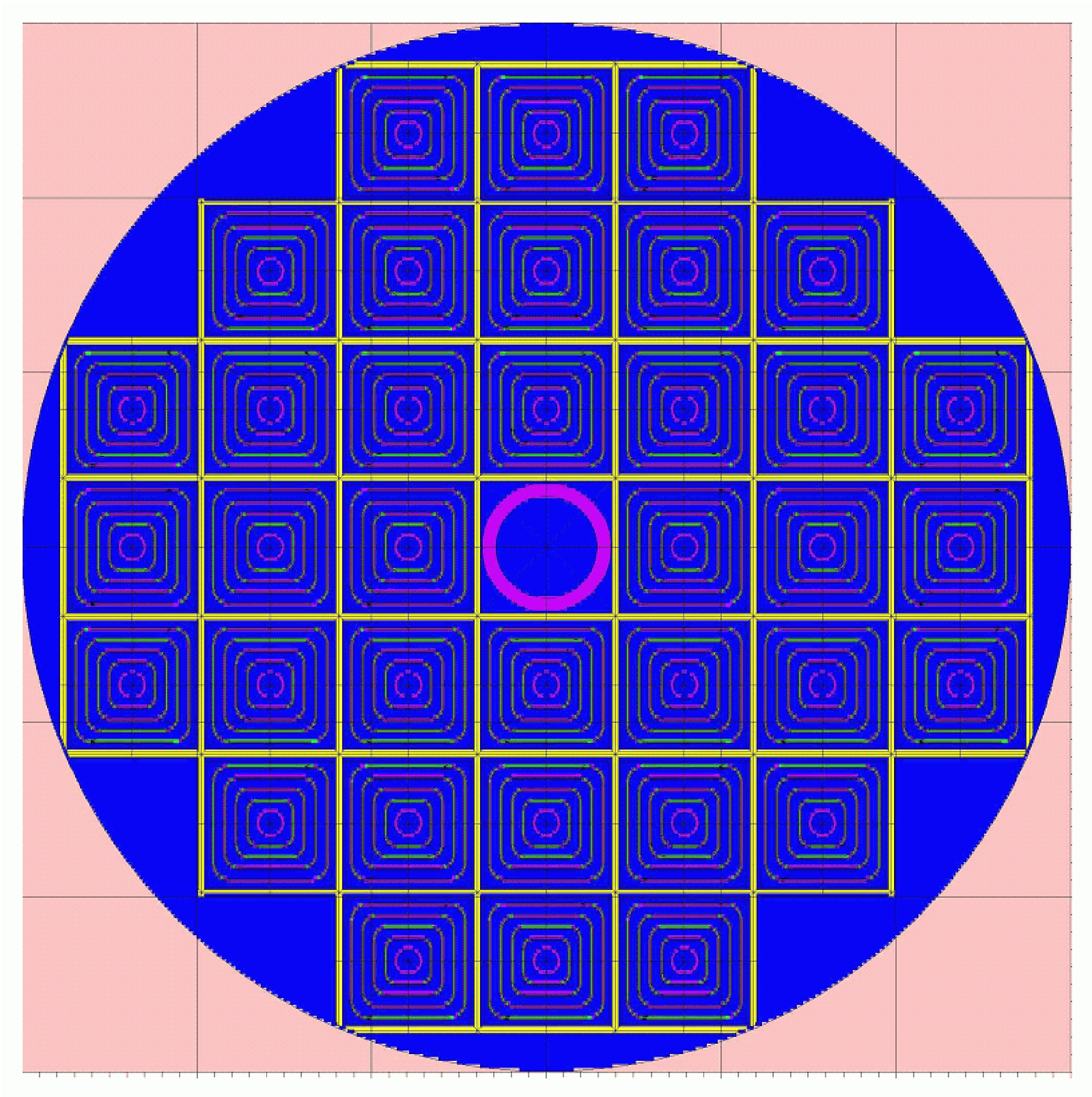

Fig.4. Horizontal section of VPVR cask basket loaded with IRT-2M fuel assemblies with initial enrichment of 36 wt.% ^{235}U , ATABOR intervening shielding plates (MCNP output)