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Experimental and Computational Validation of Novel Depletion Algorithm in the RAPID Code System Using JSI TRIGA Reactor

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ABSTRACT

The Real-time Analysis for Particle-transport and In-situ Detection (RAPID) Code System, developed based on the Multi-stage Response-function Transport (MRT) methodology, enables real-time simulation of nuclear systems such as reactor cores, spent nuclear fuel pools and casks, and sub-critical facilities. This paper presents the experimental and computational validation of a novel fission matrix-based burnup methodology using the well-characterized JSI TRIGA Mark II research reactor. The validated methodology allows for calculation of nuclear fuel depletion by combination and interpolation of RAPID's burnup dependent fission matrix (FM) coefficients to take into account core changes due to burnup. The computational validated is conducted by comparison to the Serpent-2 Monte Carlo depletion calculations. The results show that the burnup methodology for RAPID (bRAPID) implemented into RAPID is capable of accurately calculating the keff burnup changes of the reactor core as the average discrepancies throughout the whole burnup interval are 37 pcm. Furthermore, capability of accurately describing 3D fission source distribution changes with burnup is demonstrated by having less than 1% relative discrepancies compared to Serpent-2. Good agreement is observed for axially and pin-wise dependent fuel burnup and nuclear fuel nuclide composition as a function of burnup. Experimental validation is performed using excess reactivity measurements, obtained from the JSI TRIGA operational history analysis. Comparison of measured and calculated reactivity gradients due to burnup for three mixed TRIGA cores are presented. The results show great agreement between calculations and measurements for both Serpent-2 and RAPID code system, validating the performed simulations.

INTRODUCTION

Determination of nuclear fuel burnup in nuclear fission reactors is important from standpoint of reactor safety, safeguards, operation and fuel management. As the fuel burnup measurements are impractical and special equipment is needed [1], calculations are commonly performed to determine the nuclide composition as a function of reactor operation. The modelling of the long-term changes in the nuclide composition of the fuel is very computationally demanding [2] because coupled system of Bateman's depletion equations and Boltzmann's transport equation has to be solved [3].

Traditional approach in solving depletion problems has been to use deterministic codes to solve the coupled system at the unit-cell level and use the calculated parameters to obtain solutions for the entire core using diffusion approximation (e.g., the CORD2 package [4]). For smaller reactor cores (e.g., research reactors) such approximations are not as accurate in regions of high neutron flux gradients (e.g., vicinity of strong absorbers) [2]. In such cases, Monte Carlo neutron transport should be use to obtain the neutron flux distribution in the reactor core and use it to solve Bateman's depletion equations [5]. Detailed Monte Carlo burnup simulations are computationally intensive, because for each depletion step neutron transport with changed nuclide composition has to be performed. Recently, a new method for calculating nuclear fuel burnup (bRAPID methodology [6]) using hybrid methods has been in development and applied to the JSI TRIGA reactor [7].

An important step in the process of developing new reactor simulation methods is their validation and verification. In this paper validation refers to comparison of simulations to experimental measurements, while verification refers to code-to-code comparison using Monte Carlo code results as a reference. The operational history of the JSI TRIGA research reactor is well-documented and analysed [2]. Hence, it was decided to perform burnup simulations of the operational history and compare calculated integral parameters of JSI TRIGA core to measurements performed during reactor operation. In addition, it was decided to perform extensive computational comparison of the RAPID [8] with Serpent-2 depletion code [5].

First part of the paper briefly introduces the RAPID code system with its bRAPID methodology and the Serpent-2 depletion code. The second part of the paper presents the validation and verification. Computational verification is conducted by determination of changes of k_{eff} and 3D fission source with burnup. Experimental validation is performed using excess reactivity measurements, obtained from the JSI TRIGA operational history analysis.

1 FUEL BURNUP CALCULATION METHODOLOGIES

1.1 Stochastic Serpent-2 Depletion Calculations

Serpent-2 [5] is a continuous-energy 3D Monte Carlo particle transport code in which the geometry can be defined in detail and which is capable of performing fixed-source and criticality calculations of nuclear systems. In addition to the steady-state Monte Carlo calculation, it has an automated built-in depletion algorithm capable of calculating detailed 3D fuel depletion [9]. This capability is possible due to introduction of a novel matrix exponential method CRAM (Chebyshev Rational Approximation Method) for solving the Bateman equations and it enables simultaneously solving an entire depletion system consisting of 1200-1700 nuclides. The simulation of JSI TRIGA operational history is conducted in such manner that for each cycle (operation on one core configuration at P = 250 kW) Monte Carlo neutron transport simulation is performed and calculated parameters (e.g. neutron flux in each depletion region) are used to solve Bateman's depletion equations for the given depletion time step. In such a way nuclide vector is tracked from the start of operation to the end. Use of Serpent-2 for JSI TRIGA depletion calculation and the validity of reactor operation simulation approximations can be found in [2].

1.2 Hybrid RAPID Depletion Calculations Using bRAPID Methodology

The RAPID code system [8] is developed based on the MRT methodology [10] and is used for 3D real-time simulation of nuclear systems by pre-calculating response functions/coefficients for a given problem using detailed Monte Carlo calculations. These coefficients are compiled in a database to solve various problems. For the simulation of TRIGA core configuration, the pre-calculation of the coefficients takes ~ 9 hours on the JSI cluster processor2 and the criticality calculation takes 20 seconds on PC. It should be noted that the coefficients for a given problem only need to be calculated once, making the methodology extremely fast overall.

The bRAPID methodology was first presented in [8] and applied to computational PWR benchmark and computationally verified on TRIGA reactor core [7], where more information regarding the methodology can be obtained. The methodology uses the same formulation as applied in the MRT methodology [10], where the analysis of the nuclear system is decoupled into multiple stages, coupled via response function/coefficients. In first stage the material composition of the nuclear fuel is calculated at reduced order (2D burnup calculations for JSI TRIGA) using Monte Carlo Serpent-2 depletion simulations. In the next stage, this material composition is used to generate detailed fission matrix (FM) coefficients $a_{i,j}$ as a function of problem relevant parameters. In the bRAPID methodology, the input parameters are the reactor power P, the irradiation time t_{irr}, the cooling time t_{cool}, the reactor core configuration, the number of nuclides to be calculated with and nuclear data library. The coefficients are calculated using fixed-source Monte Carlo transport (in our case Serpent-2 Monte Carlo code [5]) and then compiled into a database $\$a_{i,j}(P,t_{irr},t_{cool})$. The database enables the calculation of steady state criticality calculation and the calculation of burnup-dependent k_{eff}, 3D fission source distribution, and nuclide composition by linear combination and interpolation of the database entries. The schematic of the bRAPID methodology, adapted from [7], is presented in Figure 1.

To obtain the solution of the transport equation for each depletion step *n*, the FM formulation can be expressed as:

$$F_i^n = \frac{1}{k_{eff}} \sum_{j=1}^{N_{cells}} a_{i,j}(P, t_{irr})\, F_j^n, \qquad (1)$$

where F_i^n is the fission neutron source in region *i* at time t_{irr}, $a_{i,j}(P, t_{irr})$ is the power- and time-dependent FM coefficient. Since these FM coefficients contain information of the material properties of the system, they are directly dependent on nuclear fuel depletion. The discretization of the (P, t_{irr}) phase space is case-dependent. For the JSI TRIGA reactor discretization analysis of the phase space was conducted in [7] and it showed the minimum discretization for reactor power is $P \in [100\,W, 1\,kW, 100\,kW, 250\,kW, 400\,kW]$ and irradiation time $t_{irr} \in [0.1, 1, 3, 10, 30, 100, 300, 1000, 5000, 10000]\ days$. It was also shown that tracking 66 nuclides, that have a substantial effect (> 10 pcm) on core reactivity is sufficient for analysing TRIGA depletion calculations.

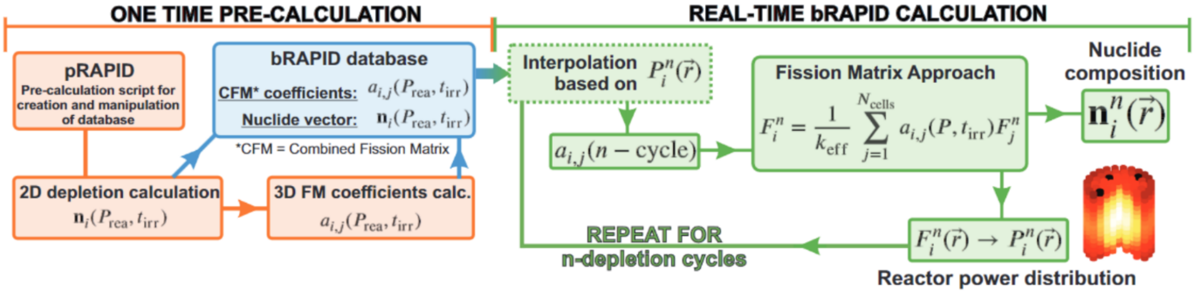

Figure 1: Schematic flowchart of the bRAPID methodology, employed in the RAPID code system. Pre-calculation is depicted in orange, creating nuclide inventory and combined fission matrix CFM database in blue and real-time calculations using RAPID's FM formulation in green. Figure is adapted from [7].

2 COMPUTATIONAL VERIFICATION

Computational verification of bRAPID methodology was conducted on three cases, differing in reactor power and irradiation times, devised in such a way to test the bRAPID interpolation algorithm. The defined reactor power $P_{reactor}$ was 250 kW, 142 kW and 13 kW. Cases 2 and 3 were chosen to be between the interpolation interval of the bRAPID database (e.g. $P_{reactor}$ = 13 kW is between database phase-space points 1 kW and 100 kW, and $P_{reactor}$ = 142 kW is between database phase-space points 100 kW and 250 kW). Irradiation time t_{irr} steps were chosen in the same way to test whether bRAPID calculations are accurate between phase-space points in bRAPID database. The verification is performed on multiple levels ranging from k_{eff} comparison to 3D fission source redistribution due to fuel burnup.

In both RAPID fixed source pre-calculation and Serpent-2 depletion simulation the same neutron data library ENDF/B-VII.1 [11] was used.

2.1 Core's Multiplication Factor Burnup Changes

To validate and test the accuracy of the bRAPID methodology, the core multiplication factor k_{eff} as a function of fuel burnup was analysed with Serpent and bRAPID for all previously mentioned cases. Multiplication factor as a function JSI TRIGA core No. 239 burnup and relative comparison to Serpent is presented in Figure 2.

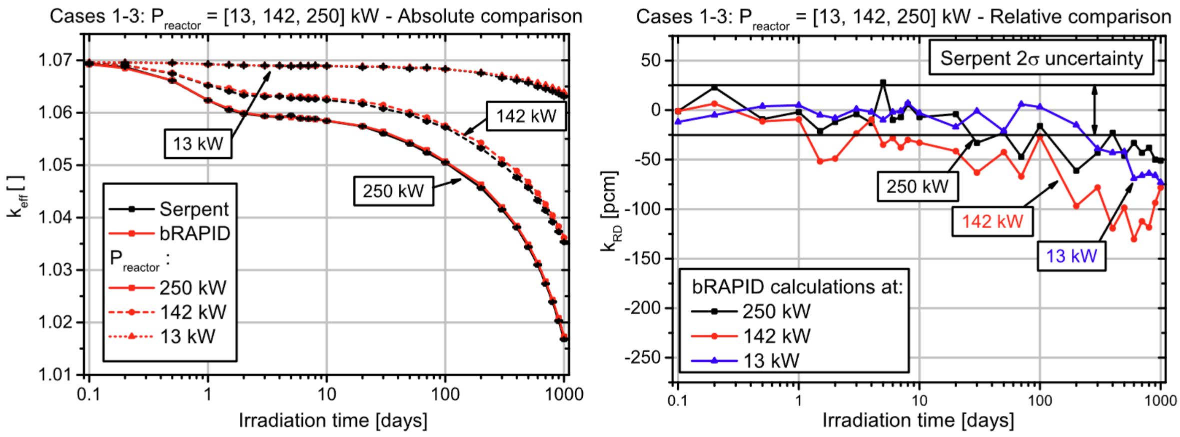

Figure 2: k_{eff} of the JSI TRIGA core as a function of reactor operation time operating on three different reactor powers. Relative difference k_{RD} is defined as $k_{RD} = (k_{eff}^{RAPID} - k_{eff}^{Serpent-2})/(k_{eff}^{Serpent-2})$

2.2 3D Fission Source Redistribution

The fission source redistribution due to the fuel burnup was studied for the three previously defined cases. The average uncertainties and average relative differences for the fission source distribution values, calculated using bRAPID and Serpent were analysed. The average relative difference of the fission source distribution between bRAPID and Serpent is within 0.6 % for all the cases, increasing to 6 % for case with larger burnup (7000 days on 250 kW). Figure 3 compares bRAPID and Serpent-2 calculated axially dependent pin-wise fission source distribution as a function of burnup. A clear redistribution of the fission source is observed, as reactor power inside of the core decreases by ~ 7 %, and increases in the periphery. This demonstrates that the bRAPID methodology accurately describes the fission source redistribution due to fuel depletion.

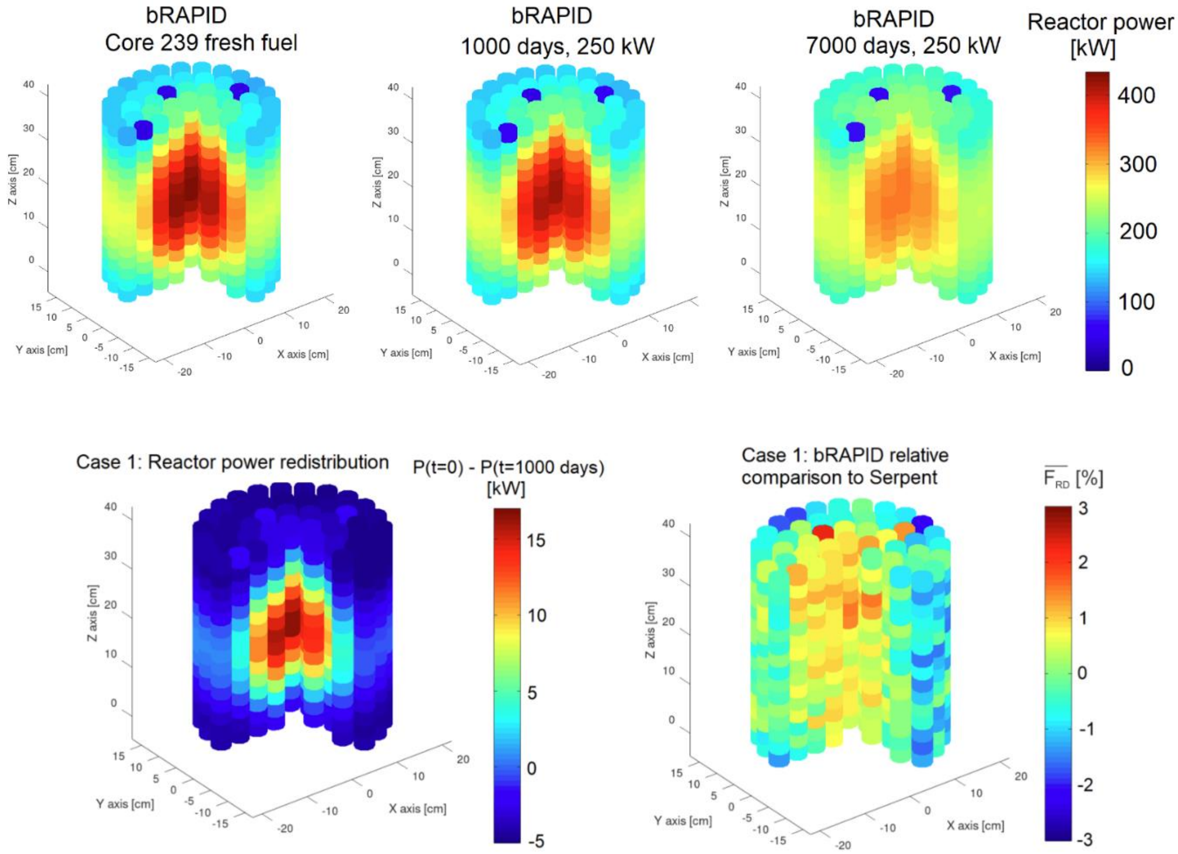

Figure 3: Comparison of 3D reactor power distribution calculated with RAPID between fresh fuel and burnup obtained by 1000 and 7000 days of operation at 250 kW, which resulted in average core burnups of 15.4 MWd/kg and 107.8 MWd/kg. Relative comparison of fission source distribution to Serpent-2 at 1000 days on 250 kW is presented on bottom right.

3 EXPERIMENTAL VALIDATION USING EXCESS REACTIVITY MEASUREMENTS

The weekly excess reactivity measurements conducted at the JSI TRIGA Mark II research reactor are used to validate RAPID's and Serpent-2 burnup and criticality calculations. From the complete operational history three core configurations were identified, which had the highest average core burnup build-up between reshuffling, ranging from 1.2 MWd/kg to 2 MWd/kg. Chosen cores are so-called mixed TRIGA cores as they consist of all four different types of fuel elements (SS 8.5 % and 12 %, AL 8.5 %, FLIP) that were in use in complete history of JSI TRIGA reactor operation. RAPID and Serpent-2 simulations and comparison with excess reactivity measurements and loading pattern schematics for three mixed TRIGA core configurations number 69, 129 and 216 is presented in Figure 4. Calculated burnup reactivity coefficients for all three core configurations are present in Table 1.

Main approximation used in this type of experimental validation is that reactivity change due to burnup is linear. It was showed in [12] that for burnup intervals of 3 MWd/kg linear approximation can be assumed if initial average TRIGA core burnup is above 4 MWd/kg. For the defined burnup intervals in which the linear approximation stands, burnup reactivity coefficient can be defined as

$$Burnup\ reactivity\ coefficient\ = \alpha_{BU} = \frac{\delta \rho_{excess}}{\delta Burnup}\left[\frac{pcm\ kg}{MWd}\right] \tag{2}$$

where $\delta\rho_{excess}$ represents the change in core's excess reactivity, and $\delta Burnup$ change in average core's burnup. Core's ρ_{excess} is defined as reactivity of the system if all control rods would be extracted from the reactor. It is measured by making the reactor critical at low powers and using control rod worth curves to calculate how much reactivity “worth” is left in the control rods.

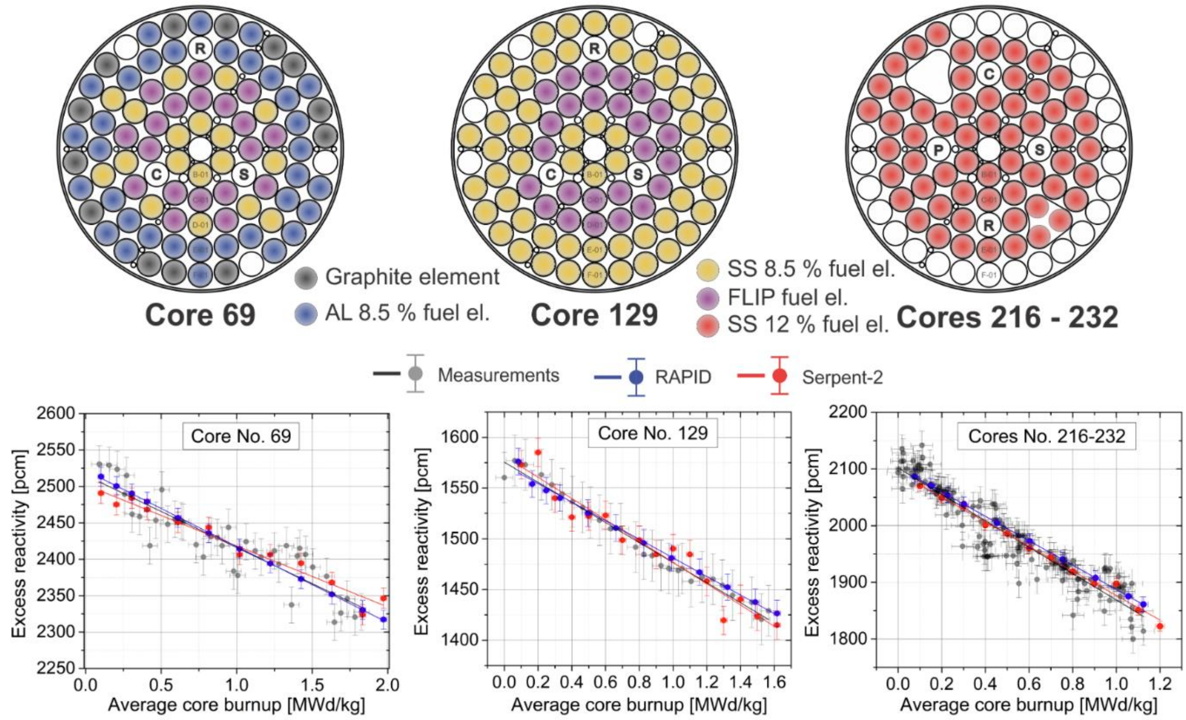

Figure 4: JSI TRIGA reactor core excess reactivity as a function of average core burnup for three mixed core configurations (from left to right: 69, 129, 216–232). For each core, measurements of ρ_{excess} are compared to calculated using burnup simulations with RAPID and Serpent-2 code. Core configurations 216–232 represent multiple identical loading patterns, between which, multiple core configurations were used on which negligible burnup was accumulated.

RAPID's experimental validation was conducted in such matter that bRAPID database was calculated for the three mixed TRIGA core configurations (Core 69, 129 and 216) and initial burnup calculations were performed to match the initial conditions at BOC. From that point, burnup simulation and criticality calculations were performed at multiple burnup intervals to describe the criticality changes due to burnup in the analysed burnup interval. For depicted core configuration 216, multiple fuel configuration changes were made in-between to create pulse-mode operational cores, which do not contribute to fuel burnup, due to low energy released [13].

The results presented in Figure 4 and Table 1 show great agreement between the measured and calculated reactivity burnup coefficient with both RAPID and Serpent-2. The predicted values using the RAPID code system with its bRAPID algorithm are within the 1σ measurements uncertainty for all three different mixed TRIGA cores. Furthermore, it can be observed that both codes accurately describe the change of the burnup reactivity coefficient when different fuel type is used. For cores configuration No. 69 and 129, FLIP type fuel elements were used, which were highly enriched (70 %) and also included burnable absorber erbium. It was showed in [2] that this burnable absorber highly effects the burnup reactivity coefficient, as the coefficient becomes positive for burnups higher than 10 MWd/kg. For fuel types in use today (SS 12 %) the burnup reactivity coefficient is higher as no burnable absorber is present in the fuel. Based on this analysis it can be concluded that both RAPID and Serpent-2 accurately describe the changes in core reactivity due to burnup in mixed TRIGA cores, where all types of TRIGA fuel are used. The results presented in this section experimentally validates RAPID's calculations of reactor core's burnup dependent integral parameters for mixed TRIGA cores with different fuel element types.

Table 1: Comparison of measured and calculated (Serpent-2 and RAPID) burnup reactivity coefficient for three mixed TRIGA cores, depicted in Figure 4. The reported 1σ uncertainty is the calculated uncertainty for standard linear regression for a set of data-points that each have its own uncertainty

	Burnup reactivity coefficient α_{BU} $\left[\frac{pcm\ kg}{MWd}\right]$		
	Core No. 69	**Core No. 129**	**Core No. 216**
Measurements	-99.1 ± 8.0	-98.9 ± 7.7	-232.8 ± 19.8
RAPID	-104.4 ± 0.5	-91.5 ± 2.3	-215.1 ± 1.3
Serpent-2	-83.8 ± 5.9	-104.9 ± 7.1	-216.4 ± 6.8
	$(\alpha_{BU,measured} - \alpha_{BU,calculated})/(1\sigma)$		
RAPID	-0.66	0.92	0.89
Serpent-2	1.54	-0.57	0.78

4 CONCLUSIONS

This paper describes the computational verification and experimental validation of the novel bRAPID methodology using the JSI TRIGA Mark II research reactor. The calculations are computationally verified by comparison to Serpent-2 Monte Carlo depletion calculations. The comparison between the two codes is conducted on three different cases, which vary in reactor power and irradiation time. The bRAPID-calculated k_{eff} as a function of burnup is within 2σ of Serpent-2 statistical uncertainty, indicating that bRAPID accurately describes core reactivity changes due to fuel burnup. Aditionally, the average fission source relative differences between bRAPID and Serpent are within 1 %, demonstrating that bRAPID accurately describes the fission source distribution changes due to fuel burnup. The experimental validation was conducted on excess reactivity measurements performed in the past on three different mixed TRIGA cores. Both codes were able to accurately calculate the changes in excess reactivity during reactor operation on the same core configuration. The validation was conducted using the *burnup reactivity coefficient* α_{BU} where linear changes in reactivity were assumed. For all three mixed TRIGA cores, calculated α_{BU} was within 1σ of the measurements, validating the burnup simulations.

It can be concluded that the bRAPID methodology yields accurate results for different burnup related parameters despite the employed approximations and the different approach. Main advantage of the RAPID code system with its bRAPID methodology is the computational time with the speedup factor of 674 in comparison to Serpent-2 for a typical TRIGA burnup simulation. Computational times and memory requirements are presented in Table 2.

Table 2: Timing and memory requirements for RAPID and Serpent-2 burnup calculations. Case with 43 burnup steps (1000 days at 250 kW) was chosen for both codes.

	# of comp. cores	*Memory usage*	*Pre-calc. time*	*Wall-clock time*	*Wall-clock speedup*
Serpent	40	230 Gb	/	59 h 34 min	/
RAPID	1	2.5 Gb	88 h 48 min	5 min 18 s	674

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