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Further Development of RAPID code extension for TRIGA reactor 3D burnup calculations

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ABSTRACT

Accurate and efficient calculations of nuclear fuel isotopic inventory and burnup are vital for fuel management, its storage safety and safeguards. The Virginia Tech Transport Theory Group (VT^{3}G) has been working on advanced computation tools for real-time simulations of nuclear systems. One of which is a novel burnup methodology bRAPID, which utilized the RAPID code system. As a part of collaboration between "Jožef Stefan" Institute and Virginia Tech we plan to couple RAPID with its bRAPID algorithm and use it on JSI TRIGA reactor for on-line calculations of fuel isotope inventory and burnup. Here we show an analysis of changes in Fission Matrix (FM) coefficients due to fuel burnup at different irradiation times and in different position in the reactor core. Our results demonstrate that primary contribution for changes of FM coefficients is the burnup of the destination region. In addition we evaluate the burnup effect of a region between the source and the destination. This method of calculating TRIGA burnup using the bRAPID algorithm will be extended in the future and finally validated using the irradiated fuel gamma spectrometry measurements.

1 INTRODUCTION

Determination of accurate 3D pin-wise fuel burnup in nuclear reactors is vital from the standpoint of fuel management, spent fuel storage safety and safeguards. In addition, the need for accurate and efficient burnup calculations has become more urgent for the simulation of advanced reactors and monitoring of spent fuel pools. To accomplish this, the Virginia Tech Transport Theory Group (VT3G) has been working on advanced computational tools for accurate modelling and simulation of nuclear systems in real-time [1]. One such capability is a novel methodology for performing 3D fuel burnup calculations, bRAPID [2], which utilizes the RAPID Code System [3,4,5]. RAPID is based on the Multi-stage Response-function Transport (MRT) methodology, that decouples a problem into independent stages that are then coupled in real-time via transfer functions/coefficients.

Recently, we initiated activities to benchmark the bRAPID methodology using the well characterized Jozef Stefan Institute's TRIGA Mark-II research reactor [6]. Thus far, we have created a database including full operational history that allows for burnup validation possibilities in the form of measured excess reactivity [7]. Additionally, for further confirmation, we are planning to perform burnup measurements using the fuel gamma spectrometry.

In this paper, extension of the bRAPID algorithm for its application to the TRIGA research reactor is presented. The paper focuses on bRAPID's database pre-calculation procedure and its automation. We isolated the changes of the Fission Matrix (FM) coefficients for burnups of different fuel elements and their effect on the coefficients of nearby elements.

2 THE RAPID CODE SYSTEM

The RAPID Code System [3,4,5] is based on the MRT methodology and it's capable of calculating 3-D steady-state (criticality) and time-dependent fission neutron source for prompt and delayed neutrons; criticality and prompt-criticality eigenvalues, k_{eff} and k_p; effective delayed neutron fraction, β_{eff}; sub-critical multiplication factor, M; kinetic parameters such as the effective neutron generation lifetime, λ_{eff}, and $Rossi-\alpha$; and detector responses. RAPID's MRT approach relies on the detailed pre-calculation of response functions and coefficients that are then coupled in real-time via linear system of equations. The generation and utilization of such response functions has been applied for patent [8]. RAPID has been computationally and experimentally benchmarked against the OECD/NEA nuclear reactor core benchmark [4], spent nuclear fuel casks [9,10] and pools [2,3,11], subcritical facilities [12], the Flattop criticality benchmark, and the JSI TRIGA benchmark included in the International Criticality Safety Benchmark Evaluation Project (ICSBEP) [9,13].

2.1 The Multi-Stage Response-Function Transport (MRT) Methodology

Based on the MRT methodology, RAPID decouples the analysis of a nuclear system into a series of independent stages, selected based on the physical features of the problem. For each of the stages, response functions and/or coefficients are calculated with best-estimate techniques (e.g., continuous-energy Monte Carlo) as a function of system-dependent parameters (e.g., burnup, cooling time, position of control rods, moderator temperature, etc.) within given physics-based ranges of validity. Once these coefficients are pre-calculated, the parameter dependent response functions/coefficients are compiled into a formatted RAPID database (RDB). The RDB is utilized in real-time via interpolation and combination of its entries within the RAPID code, for calculation of the desired distributions, parameters, and responses by solving linear systems of equations.

RAPID's MRT approach is schematized in a flowchart in Fig 1. In particular, the orange stages identify the pre-calculation of response functions and coefficients; the blue stage identifies the generation of the RDB; while the green stages are the real-time calculations.

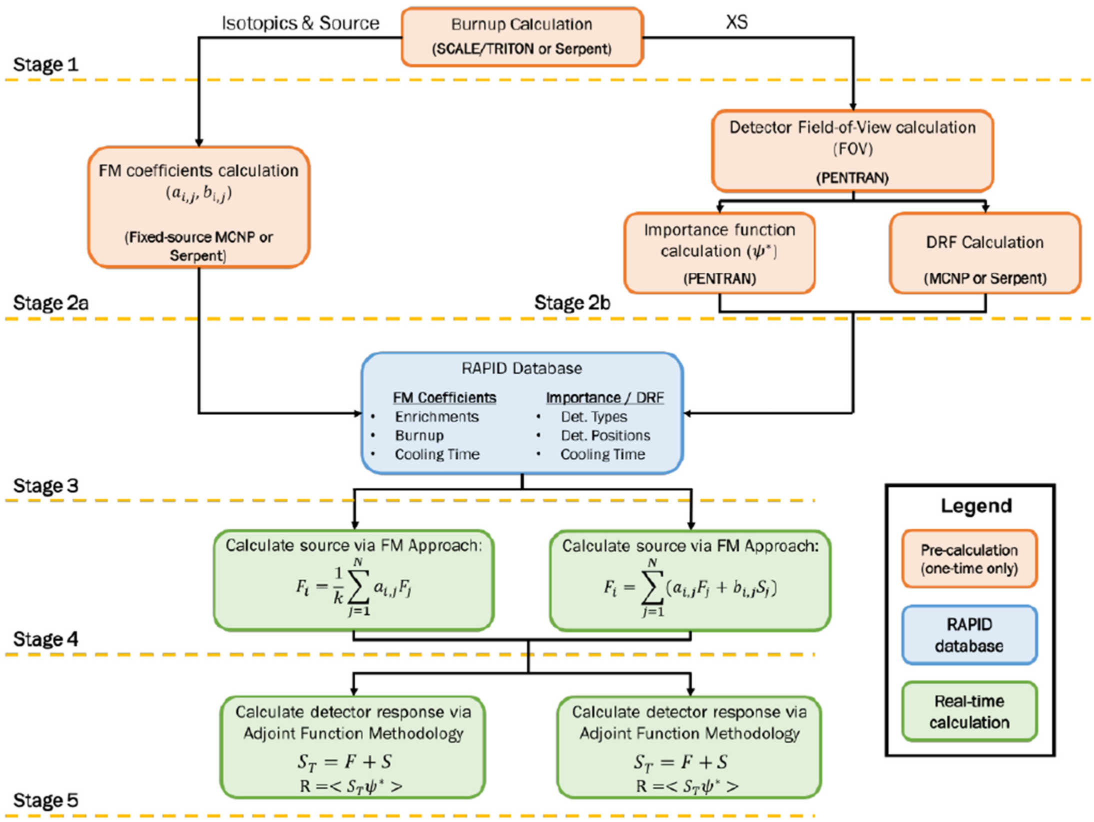

Figure 1: RAPID's MRT structure flowchart.

2.2 The Fission Matrix (FM) Method

The Fission Matrix (FM) approach to solve a neutron transport problem [14] consists of rewriting the Linear Boltzmann Equation into its matrix form. Its criticality form can be written in operator form as

$$H\Psi(\mathbf{r}, \mathrm{E}, \widehat{\Omega}) = \frac{1}{k_{eff}} F\Psi(\mathbf{r}, \mathrm{E}, \widehat{\Omega}), \tag{1}$$

Where $\Psi(\mathbf{r}, \mathrm{E}, \widehat{\Omega})$ is the angular neutron flux in phase-space $(\mathbf{r}, \mathrm{E}, \widehat{\Omega})$, H represents the transport operator[1] and F the fission operator[2]. The fission spectrum, $\chi(E)$, is extracted from the fission operator as $F = \chi(E)\tilde{F}$. By multiplying Eq. 1 with inverse transport operator times the fission operator $H^{-1}\tilde{F}$, we obtain

$$\tilde{F}\Psi(\mathbf{r}, \mathrm{E}, \widehat{\Omega}) = \frac{1}{k_{eff}} \left(\tilde{F}H^{-1}\chi(E)\right) \tilde{F}\Psi(\mathbf{r}, \mathrm{E}, \widehat{\Omega}), \tag{2}$$

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[1] $H = \widehat{\Omega} \cdot \nabla - \int_{4\pi} \mathrm{d}\Omega' \int_0^{\infty} \mathrm{d}E' \sigma_s \left(\boldsymbol{r}, E' \to E, \widehat{\Omega}' \longrightarrow \widehat{\Omega}\right)$

[2] $F = \int_{4\pi} \mathrm{d}\Omega' \int_0^{\infty} \mathrm{d}E' \frac{\chi(E)}{4\pi} \nu\sigma_f\left(\boldsymbol{r}, E'\right)$

where the left-hand side represents the fission neutron source. The right-hand side can be expressed with an operator $A = \tilde{F}H^{-1}\chi$ and by integrating Eq. 2 over the angle, energy and discretizing space into N_{cells} of fissionable regions, we obtain

$$\mathbf{S} = \frac{1}{k_{eff}}\underline{A}\mathbf{S} \quad \longrightarrow \quad S_i = \frac{1}{k_{eff}}\sum_{j=1}^{N_{cells}} a_{i,j}\, S_j, \qquad i \in [1, N_{cells}] \tag{3}$$

where $\mathbf{S}$ is the discretized total neutron fission source, and $\underline{A}$ is the discretized form of operator A and is referred to as the Fission Matrix (FM). Each element $a_{i,j}$ of its discretized form represents the number of fission neutrons generated in fission region i due to a neutron born in fission region j . FM coefficients are easily calculated using any neutron transport code. For our case Serpent-2 neutron transport and depletion code [15] was used. The process of their generation is described in detail in [14].

2.3 bRAPID Extension for Burnup Calculations

The novel bRAPID algorithm is based on the same fundamental methodology as RAPID [2], where a series of pre-calculations are performed to build a library of data (containing FM coefficients, material compositions and source strengths), which covers a range of scenarios (specific powers and exposure times) for the wanted system. A simulation is then performed by interpolating these data, to fit a specific model of the system, and solving a linear system of FM equations in real-time.

The bRAPID MRT methodology is partitioned into several stages as follows:

- PRE-CALCULATION:
  - Stage 1: Burnup calculation to determined fuel isotopic composition.
  - Stage 2: Calculation of FM coefficients using fixed-source Monte-Carlo calculations.
- REAL-TIME FUEL BURNUP:
  - Stage 3: Evaluation of the fission neutron source distribution via solution of the FM equations at a specific burnup step.
  - Stage 4: Determination of the cell-wise burnup and fuel isotopic composition.
  - Stage 5: Repeat Stage 3 and Stage 4 for each burnup step.

The general bRAPID algorithm is explained in greater detail in [2]. Only a brief description is given here. After obtaining the complete RAPID database real-time calculations are performed. Using RAPID fission source distribution is determined and used to determine the specific power distribution throughout the system. These specific powers along with the given exposure time are used to determine isotopics and also the FM coefficients for the next time step. The bRAPID algorithm for the JSI TRIGA research reactor is presented in a flowchart in Fig. 2. The grey coloured part of the chart represents the FM coefficient pre-calculations for different reactor powers and irradiation times. The blue part represents the real-time solving of the linear set of FM equations is performed from $a_{i,j}$ obtained from the interpolation based on the calculated neutron source.

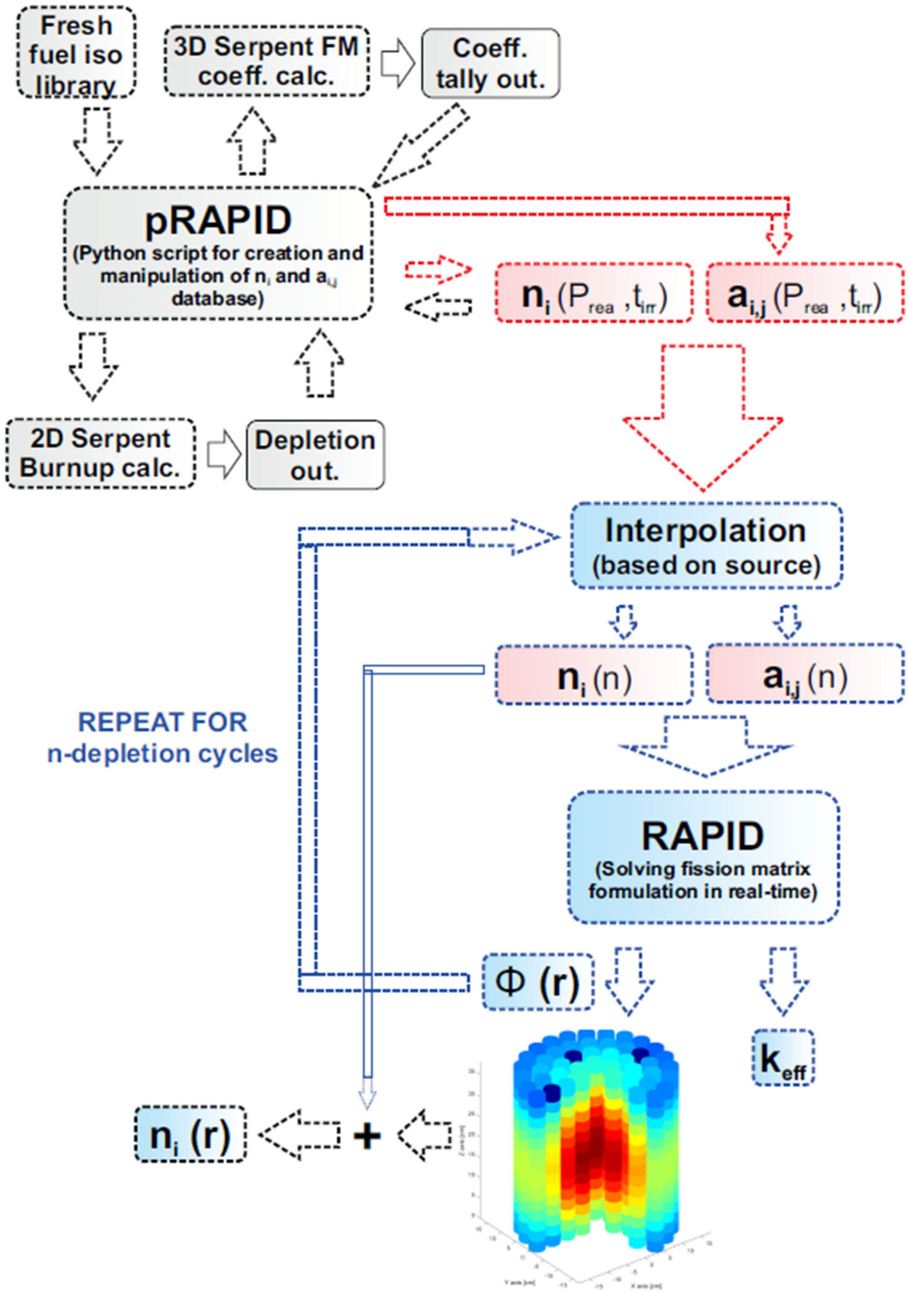


Figure 2: Schematic flowchart of the bRAPID algorithm, adapted from [2]. Pre-calculation is depicted in black, created isotope inventory and FM coefficient database in red and real-time calculation using RAPID in blue.

3 BRAPID FOR THE JSI TRIGA RESEARCH REACTOR

The JSI TRIGA Mark II research reactor is an open pool type reactor with maximum steady-state power of 250 kW. The reactor has a circular lattice with 91 fuel element locations arranged in 6 concentric rings (from A to F) as presented in Fig. 3. Fuel elements are made of a U-Zr-H mixture (12 wt. % of 19.75 % enriched uranium) with a central Zr rod. Core is equipped with four control rods. Three of them are equipped with a fuelled follower, while the "transient" (P) one is equipped with an air follower. Several irradiation channels are inserted during standard steady-state operation, such as a central channel in the centre, triangular channel (area of three fuel elements), and several channels in the outermost ring F. All the above-mentioned features make the core of the JSI TRIGA reactor asymmetrical.

Due to the asymmetry of the reactor core we have decided to calculate the isotope inventory database for different reactor powers and irradiation times $n_i = (P_{rea}, t_{irr})$ in 2D, using the Serpent-2 depletion code [15]. With this we are able to incorporate the asymmetry of the core loading pattern and the inserted irradiation channels. The next step is to calculate the FM coefficients database. For each set of (P_{rea}, t_{irr}) we calculate the $a_{i,j}$ for all the fuel elements, divided into 15 axial zones. For a standard core loading pattern the total number of source and destination locations is N_{cells}. Fission source distribution is used to determine the specific power distribution and along with the given exposure times it is used to determine isotopic inventory and also the FM coefficients for the next step. With such procedure axial burnup distribution is determined. Similar process could be applied for the radial distribution, where the initial burnup would be calculated on a fuel unit cell, however further analysis of the behaviour of FM coefficients in the TRIGA burned fuel have to be performed. Initial analysis is presented in this paper.

4 CHANGES OF FM COEFFICIENTS WITH BURNUP

Most important part of bRAPID methodology development is to determine how the FM coefficients $a_{i,j}$ change with fuel burnup. From the definition of the coefficients there are a couple of different processes that could affect their dependence on burnup, such as burnup of the source and destination region, burnup of region a neutron passes through and the difference of burnup under different neutron spectrum (for example near an irradiation channel). To test such effects, we analysed a hypothetical case where burnup was exaggerated and effect of different burnup regions isolated. We focused on four regions in the core; position B-05, C-09, D-13 and E-17 (presented on Fig. 3. and analysed the changes of FM coefficients by changing the burnup of just one fuel element.

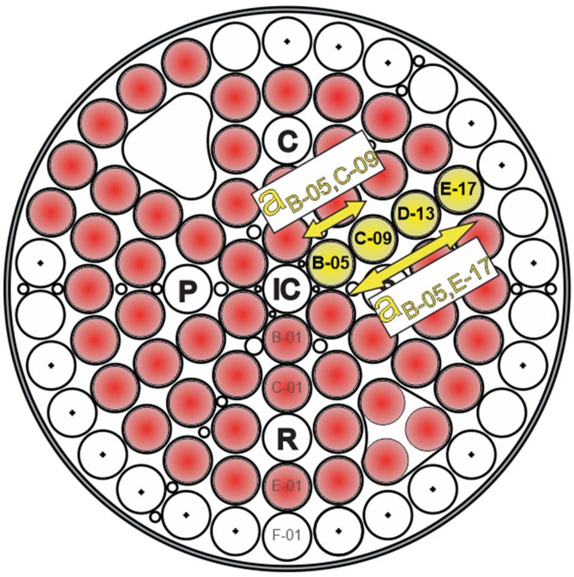
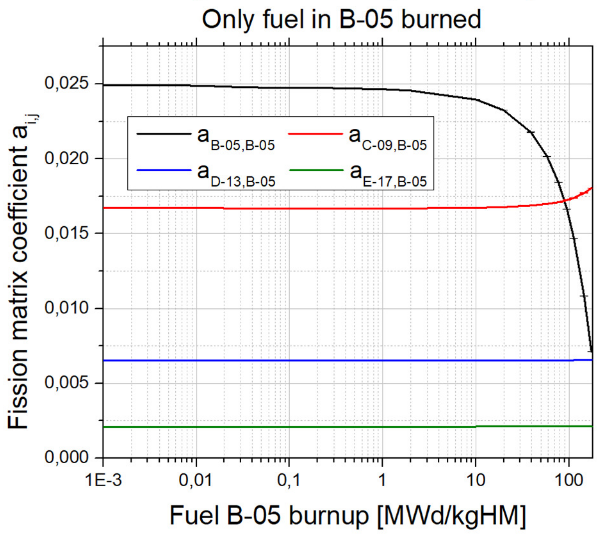


Figure 3: Schematic view of the JSI TRIGA reactor core with depicted control rods (P,R,C,S) and irradiation channel IC. With yellow colour four core positions that were used in the analysis are highlighted and their corresponding FM coefficients $a_{i,j}$ depicted.

For all the cases studied only one fuel element was burned under the same conditions up to 170 MWd/kgHM, which is an exaggeration, compared to current burnup of TRIGA fuel elements 20 MWd/kgHM [7]. We calculated $a_{i,j}$ using fixed source calculation with $10^{\wedge 8}$ neutrons for all four regions, depicted in Fig. 3. Source region *j* was in the middle of fuel element in position B-05. Two cases were studied, where fuel element in B-05 and C-09 were burned. It can be observed that $a_{i,j}$ decrease if the destination region *i* is burned and that $a_{i,j}$ increase slightly nearby regions. Another observation is that the coefficients change only in nearby regions, as in E-17 region no change is observed if elements in B-05 or C-09 are burned. The results are presented in Fig. 4 and Fig. 5.



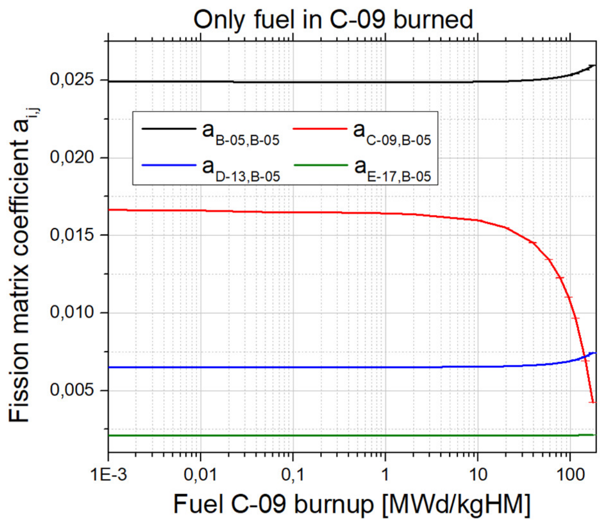


Figure 4: Changes of Fission Matrix coefficients presented in Fig. 3 as a function of burnup of two fuel elements in position B-05 and C-09.

In order to further evaluate the changes relative changes of $a_{i,j}$ were studied. Their dependence on fuel burnup are presented on Fig. 5. The changes of $a_{i,j}$ in the destination region *i* are directly connected with its burnup and are independent on the source region. This can be observed that even if source region *j* was at B-05, the relative change is the same in all destination regions *i*. Another visible effect is that $a_{i,j}$ changes are higher (up to 15 % at 170 MWd/kgHM) if the burned region is between the source and destination. This can be observed on top right and bottom left graph of Fig. 5. Burned region C-09 was in the first case between source *j = B-05* and destination *i = D-13* and burned region D-13 was in the second case between source *j = B-05* and destination *i = E-17*.

Knowledge obtained from this analysis will help us with the development of the method and finally decrease the total computational time, since knowing the extent of FM coefficient changes reduces pre-calculation time.

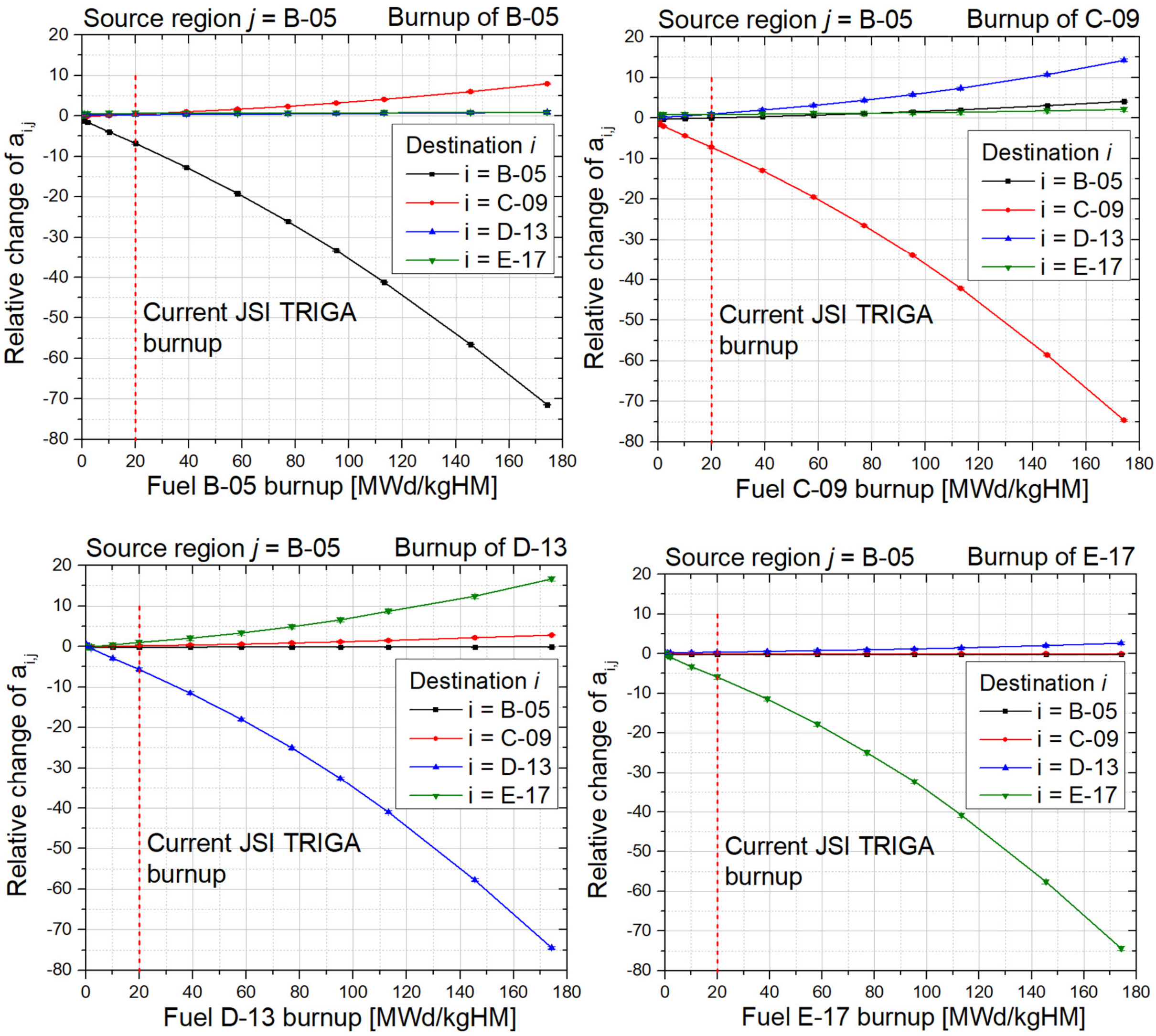


Figure 5: Relative changes of Fission Matrix coefficients $a_{i,j}$ as a function of different fuel element burnups presented on each graph. In all cases source region j = B-05 was chosen to eliminate the effect of fuel position in the core and focus on burnup.

5 CONCLUSIONS

This paper presents the initial idea of application of the bRAPID algorithm on the JSI TRIGA Mark II research reactor. It demonstrates how the RAPID code system can be utilized for a burnup calculations of research reactors with diverse operations. The utilization is dependant on knowing the changes of FM coefficients due to burnup. Hypothetical case was analysed which enabled the isolation of different effects, such as burnup of source and destination region. It was demonstrated that the primary change is due to burnup of the destination region (75 % change at 170 MWd/kgHM). In addition, effect of burnup of region between source and destination was analysed and changes up to 15 % at burnup of 170 MWd/kgHM were observed. Analysis presented in this paper serves as a basis of knowing FM coefficient changes and the interpolation process in the bRAPID methodology.

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