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Research Article

Simulating a lead-cooled reactor campaign using the EUCLID/V1 code*

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Abstract

The paper presents the results of the development of the EUCLID/V1 integrated dynamic code designed to analyze and justify the safety of fast neutron reactor facilities with a liquid-metal coolant, in terms of simulating the reactor campaign. The relevance of this study lies in the need to simulate the behavior of the core at any time during the campaign. It lets us to obtain a full dataset for subsequent simulations of the reactor dynamic conditions (including transient states or accidents). The authors have developed a fuel archive to store calculated data in HDF5 format, created a computational model editor to generate input data in the fuel archive format, and also provided an example of computing the campaign of a lead-cooled fast reactor for three core design models shown in this paper. The main array of fuel assemblies was simulated as a single unit in the first model, as three units in the second model, and in the third every single assembly was unique. In addition, the authors have shown changes in the total masses of actinides in the core, revealed that the different core models have an insignificant effect on the evolution of the total masses of actinides, and given the fuel assembly burnup values for the three core models. For the third model, the largest difference between the minimum and maximum burnup values was obtained with an almost identical average over the fuel assemblies. The reactivity margin over time for the three core models was presented. It was shown that the values and behavior of the reactivity margin during the three micro-campaigns are almost equal. From the fourth to the sixth cycle, the reactivity margin value for the third core model was lower than for the first and the second ones. Finally, the authors conclude that it is desirable to evaluate the behavior of the reactivity margin for lead-cooled fast reactor campaigns based on the detailed model of the core.

Keywords

Fast-neutron reactor, reactor campaign, EUCLID/V1, fuel assembly, burnup, BPSD, neutronic parameters

Introduction

The need for a computational justification of the safety of liquid metal cooled reactor facilities stimulates the development of the modern computational tools that make it possible to simulate various operating modes of a reactor facility throughout its entire life cycle. One of such tools is the integrated code EUCLID/V1 developed at IBRAE. In this paper, we consider the possibilities of using the EUCLID/V1 code for computational simulations of reactor campaigns.

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Description of the EUCLID/V1 integrated code

The integrated dynamic universal computer code used for analysis and justification the safety of liquid metal cooled fast neutron reactors, EUCLID/V1, has been developed at the Nuclear Safety Institute (IBRAE) of the Russian Academy of Sciences since 2012, within the framework of the private project on the "New generation codes" of the "PRORYV" project (Alipchenkov et al. 2018, Mosunova 2018). The code is intended for simulating reactor facilities based on coupled neutron, fuel rod and thermal hydraulic computations. The application area of the EUCLID/V1 code includes both the conditions of normal operation (stationary states at permitted power levels, normal transients) and accidents. In 2019, the EUCLID/V1.2 computer program was certified for lead- and sodium-cooled reactor facilities computations (Attestation Certificate 2019).

The EUCLID/V1 code has a modular structure including three main modules: thermal hydraulic, fuel rod and neutron ones. Neutron calculations in the EUCLID/V1 code are carried out on the base of the DN3D module, which includes diffusion (single-point, seven-point, and hybrid computation schemes) and kinetic options (based on the discrete ordinate method). The diffusion and kinetic options are based on the use of the DOLCE VITA (Seleznyov et al. 2018b) and CORNER (Bereznev et al. 2015) codes, respectively, the stand-alone versions of which were certified in 2021.

Micro-campaign computations in the EUCLID/V1 code

In 2019, work began on modifying the EUCLID/V1 code to enable computations of reactor operating modes, taking into account core burnup and fuel assembly loading, as part of micro-campaigns based on the current version of the BPSD program (Seleznyov et al. 2018a) (the code was certified in 2021 (Attestation Certificate 2021). For this aim, the following modifications were made in the code:

- a fuel archive was created to organize the storage of calculated data in the HDF5 format (Electronic resource 2021);
- a computational model editor was created to arrange the input data in the format of the fuel archive;
- interfaces for data exchange with the fuel archive and the BPSD module were implemented in the DN3D neutronic module;
- nuclear data preparation was organized using the fuel archive for neutron calculations in the DN3D module.

Computations of modes, including burnup accounting, are associated with the processing and storage of large amounts of data, which necessitates the use of the fuel archive.

The fuel archive includes data on the parameters of the computational model, fuel assemblies, control and safety devices, material composition of the core, control parameters of the computation, etc.

The computational model editor has user interfaces for easy setting and adjustment of reactor plant parameters (Fig. 1).

To take into account changes in the nuclide composition during the computation of a micro-campaign, group cross-sections are prepared after each burnup step using the CONSYST program (Manturov et al. 2000). The prepared nuclear data are used in a stationary computation; the computed neutron flux density distribution is used in the BPSD module to determine new material compositions. After that, the archive is recorded and the nuclear data are prepared. To calculate the equilibrium fields of temperatures and densities at each burnup step, the equilibrating process

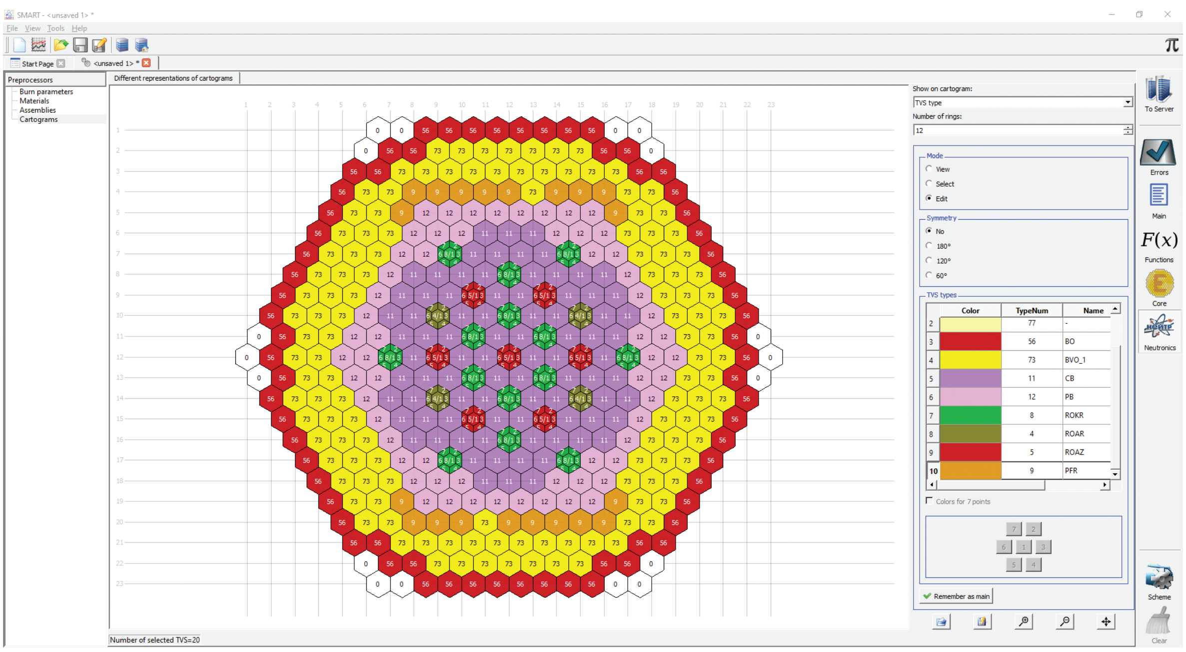

Figure 1. View of the workspace for defining a core loading pattern using the core constructor.

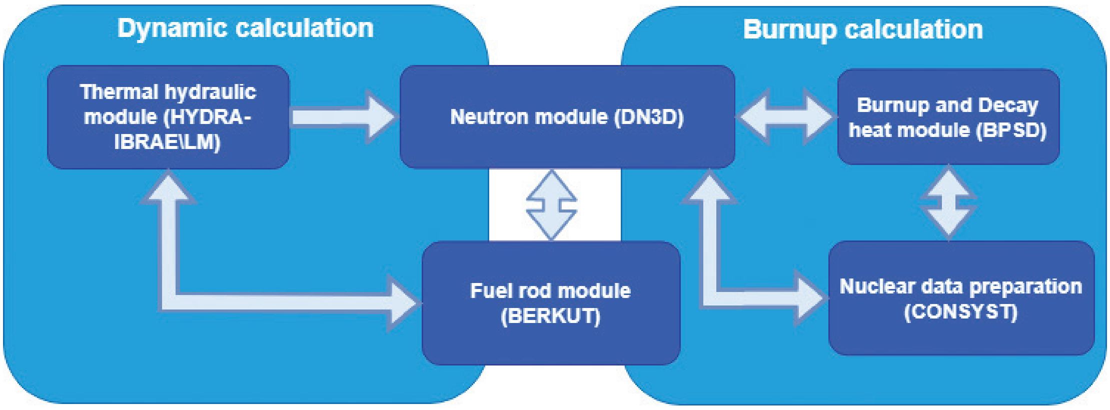

Figure 2. Block diagram of the subroutine for simulating campaigns in the EUCLID/V1 code.

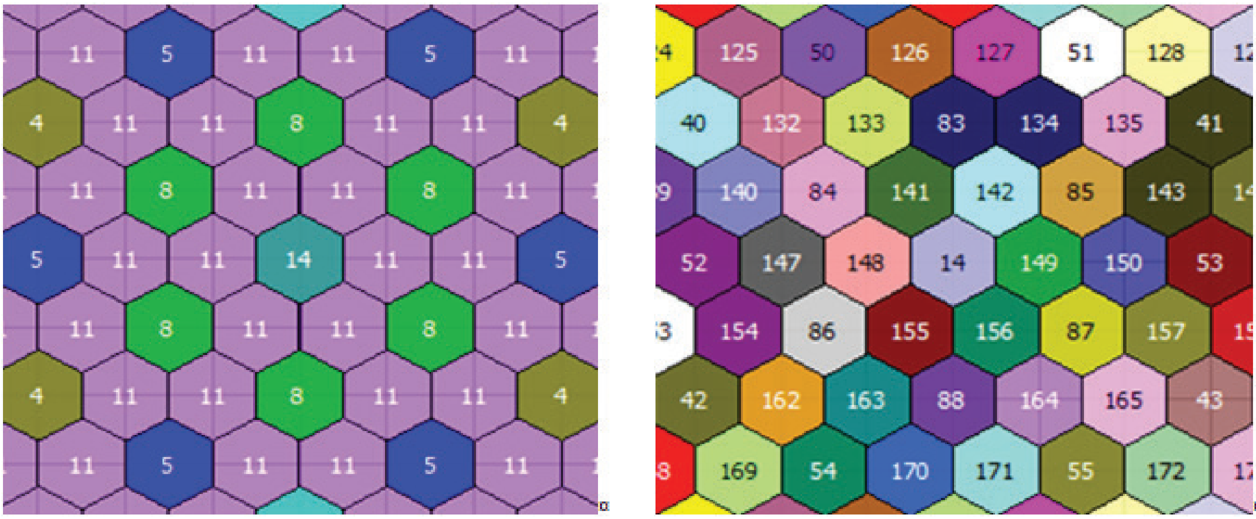

Figure 3. Two different partitioning options: on the left, with a combination according to the structural features of the assemblies; on the right, with a unique type for each fuel assembly.

is calculated using the thermal hydraulic and fuel rod modules. The block diagram of the subroutine for simulating campaigns in the EUCLID/V1 code is shown in Fig. 2.

To reduce computing costs, the combination of assemblies similar in neutronic and thermohydraulic characteristics in the form of single representatives is used. These representatives are used for further simulation of the core, taking into account sets of averaged parameters. An example of the core loading pattern with representatives of fuel assemblies is shown in Fig. 3; it displays two different partitioning options, taking into account the structural features of the assemblies: (a) all the fuel assemblies are of type 11, the assemblies with CPS elements are of type 4, 5, 8, 14 depending on the CPS type; (b) all the fuel assemblies are unique.

Description of the computational model of a lead-cooled reactor facility

As an example, let us consider a simulated campaign of a lead-cooled reactor plant lasting 900 days. The campaign is divided into 6 micro-campaigns of 150 effective days, each with stops for 30 days.

The first core loading pattern with oblique numbering of fuel assemblies is shown in Fig. 4 (Balovnev et al. 2019).

During the campaign, some permanent reactivity compensators and fuel assemblies are rearranged. At the same time, the temperature distributions of the materials throughout the entire computation are set constant.

Three computations of the reactor campaign with different partitions were carried out for fuel assemblies in the central and peripheral zones without the CPS actuators (CZ FAs and PZ FAs) (Table 1). For the fuel assemblies with the CPS actuators, a separate partition is specified, taking into account the type of absorber (boron or dysprosium).

Table 1. Various Partitioning Options

	Option 1	Option 2	Option 3
Partitioning	1 representative for CZ FAs and 1 representative for PZ FAs	3 representatives for CZ FAs (by concentric layers) and 1 representative for PZ FAs	Each FA is unique

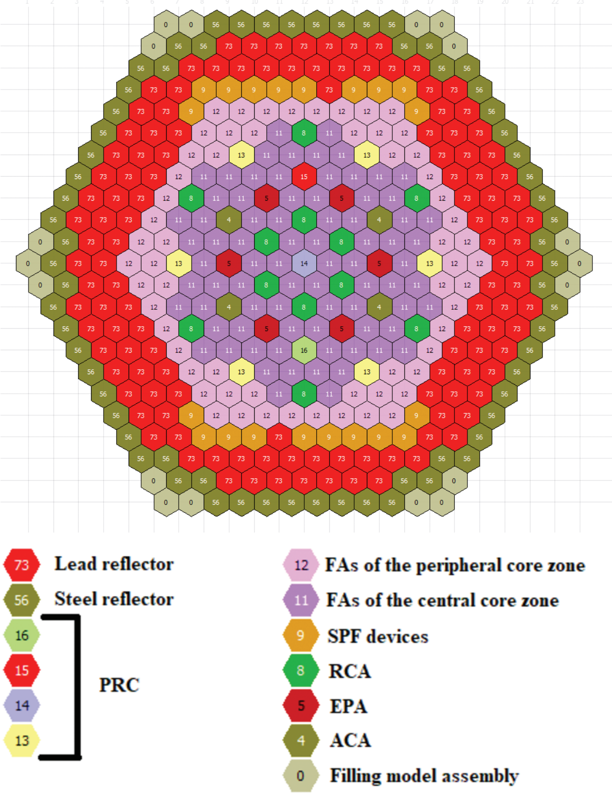

Figure 4. Computational model loading pattern.

Computations of the simulated campaign

For the CZ FAs and PZ FAs, burnup values were obtained at the end of the sixth micro-campaign (Table 2).

Table 2. Burnup Depth, % ha

Assembly	Option 1	Option 2	Option 3
CZ FAs			
Average	4.023	4.023	4.024
Minimum	–	3.5324	3.0837
Maximum	–	4.7501	4.7622
PZ FAs			
Average	2.4749	2.4748	2.4745
Minimum	–	–	1.7965
Maximum	–	–	3.1941

For ^{235}U, ^{238}U, ^{238}**Pu**, ^{239}**Pu**, ^{240}**Pu**, ^{241}**Pu**, ^{242}**Pu**, and ^{241}**Am** the time dependences of changes in the total masses of nuclides in the core were obtained. Figs 5 and 6 show the relative change in mass, normalized to the mass of a given nuclide in the initial load.

The mass changes over the entire core are practically independent of the type of partitioning. It is explained by the fact that the total power of the reactor is constant throughout the entire computation for all the options, that is, the total energy released in the core during the interaction of neutrons with nuclei does not depend on the type of model partitioning. At the same time, the burnup distributions for the fuel assemblies in the different partitioning options differ (Fig. 7). By way of illustration, the values for cells were taken in the one direction from the center.

The time-reactivity margin curve for the three options is also presented (Fig. 8).

The results presented in Fig. 8 show that the type of partitioning has an impact on the reactivity margin values.

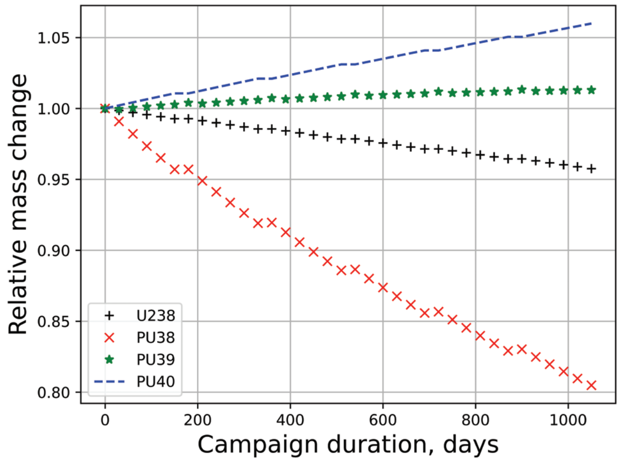

Figure 5. Mass changes of nuclides ^{238}U, ^{238}Pu, ^{239}Pu, and ^{240}Pu.

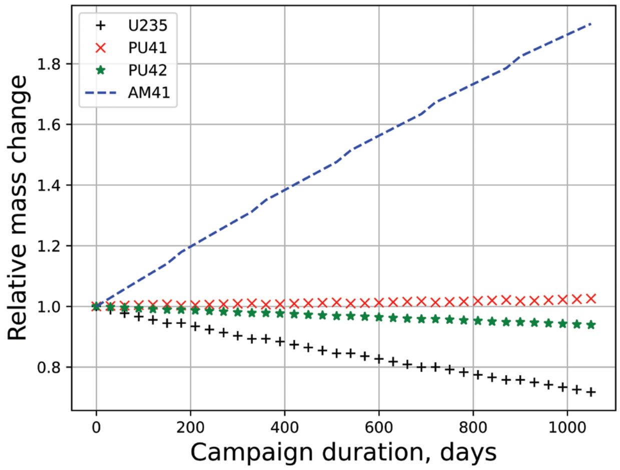

Figure 6. Mass changes of nuclides ^{235}U, ^{241}Pu, ^{242}Pu, and ^{241}Am.

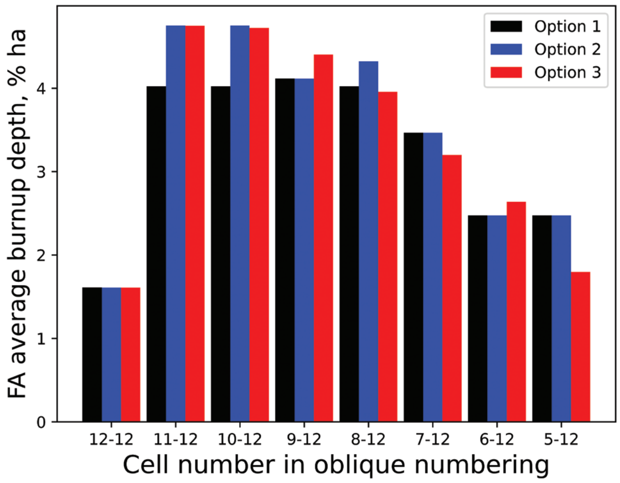

Figure 7. Average assembly burnup radial distribution for three options.

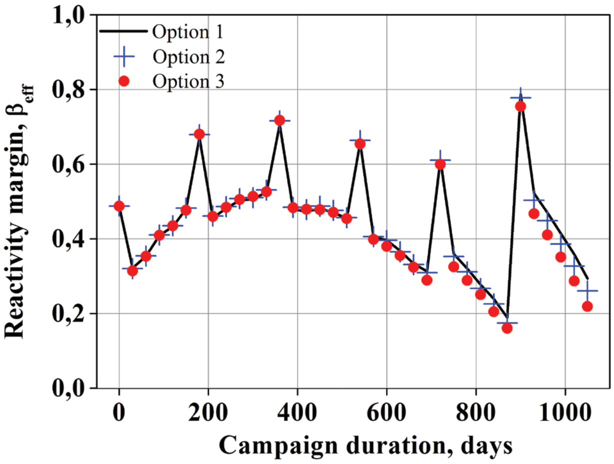

Figure 8. Time-reactivity margin curve.

This impact is described by the differences in the distributions of the neutron flux density over the reactor core caused by different burnup distributions in computations with different types of representation. To obtain the correct value of the change in the reactivity margin depending on the reactor core burnup, it is necessary to specify the core model in more detail. In turn, if the main interest in the computation is the change in the nuclide composition of the core, it is possible to use a simplified model, which, in this case, is computed in times faster due to the smaller number of the unique computational cells.

Conclusion

The EUCLID/V1 code implements the technology for computing simulated reactor plant campaigns using the nuclide kinetics code BPSD, including a core constructor and support for an archive of states in HDF5 format with a library of access functions.

A series of computations of a lead-cooled reactor plant simulated campaign was carried out. The results obtained were used as part of the work to improve the computing method, as well as to refine the functionality of the code.

Further work is aimed at improving the preprocessing tools, computational algorithms, and the core state archive format. In subsequent computations, it is planned to use the thermal hydraulic and fuel rod modules of the EUCLID/V1 code to take into account the effect of temperature-density feedbacks on the neutronic characteristics during the entire campaign.

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