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Review of JAEA's Monte Carlo codes for nuclear reactor core analysis

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Abstract. Japan Atomic Energy Agency (JAEA) has been developing a general-purpose continuous-energy Monte Carlo code MVP for nuclear reactor core analysis. Recently improvements to MVP have been focused on the development of an advanced neutronics/thermal-hydraulics coupling code. JAEA has also developed a new Monte Carlo solver Solomon for criticality safety analysis. Solomon aims to calculate the criticality of fuel debris. This paper provides an overview of the capabilities and reviews recent applications of MVP and Solomon.

1 Introduction

Japan Atomic Energy Agency (JAEA) has been developing a general-purpose continuous-energy Monte Carlo code MVP for neutron and photon transport calculations since the late 1980s. The MVP code is primarily designed for nuclear reactor applications such as reactor core design and analysis, criticality safety, and reactor shielding. The first version [1] of the code was released in 1994, and since then it has been utilized extensively in Japan. The second version [2] was released in 2005 with new capabilities. Modifications and enhancements were implemented using advanced Monte Carlo methodology for reactor physics applications. The third version (MVP3) was released in 2018 [3]. Recently improvements to the MVP code have been focused on the development of an advanced neutronics/thermal-hydraulics coupling code.

In 2015, JAEA initiated the development of a new Monte Carlo solver Solomon [4]. It is designed not only for usual criticality safety analysis but also for criticality calculations of fuel debris. Retrieving fuel debris generated from the severe accident at the Fukushima Daiichi Nuclear Power Station is planned in the near future; thus characterization of criticality for all the presumed conditions of fuel debris must be investigated to establish a principle of its criticality control. To address such systems including fuel debris, a randomized model and the delta-tracking capability were implemented into Solomon.

This paper reviews the capabilities of the MVP code and the Solomon solver and presents the recent applications with them. Sections 2 and 3 provide an overview of the capabilities and recent applications of the MVP code and the Solomon solver, respectively. Section 4 describes concluding remarks and the future perspective of the code and solver.

2 MVP code

The features and basic capabilities of the MVP code are briefly described in the following subsection. Useful capabilities for reactor analysis and recent applications are described in the subsequent subsections.

2.1 General description

MVP is designed to handle various types of problems, including eigenvalue and fixed-source calculations for neutron, photon, and neutron-photon coupled transport. Its capabilities are also extended to time-dependent problems, such as neutron noise simulation and pulsed neutron simulation; the implementation of a reactor kinetics simulation capability in the MVP code is currently under consideration. The code utilizes combinatorial geometry with multiple-lattice capability for precise calculation geometry description, allowing spatial regions to be defined using pre-defined bodies like spheres, cylinders, and parallelepipeds.

In terms of particle sources, MVP offers flexibility in specifying energy-, angle-, space-, and time-dependent distribution functions, with numerous sampling functions available. The code relies on dedicated nuclear data libraries generated from evaluated nuclear data using the LICEM code [5]. Notably, it employs the probability table method [6] for neutron cross-sections in the unresolved resonance region and can handle neutron cross-section data

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at any temperature specified in the input. MVP prepares the resolved-resonance cross-section data by using the SIGMA1 method in the LICEM code and the unresolved-resonance cross-section data and thermal scattering data with the temperature interpolation before it performs random walk calculations.

MVP incorporates various variance reduction techniques to enhance simulation efficiency, including Russian roulette, splitting, importance sampling, weight windowing, path stretching, and source biasing. For result estimation, it provides track length, collision, point, and surface crossing estimators. The eigenvalue (k_{eff}) is calculated using track length, collision, and analog estimators for neutron production and balance methods, with the most probable value and its variance determined by the maximum likelihood method.

The user-friendly features include the ability to use symbolic parameters as user-defined variables in the input data. MVP employs an event-based Monte Carlo algorithm known as the stack-driven zone-selection algorithm [7] for efficient computation on vector supercomputers. Extensive code validation has been performed using the ICSBEP [8] and IRPhEP [9] handbooks, primarily with the Japanese evaluated the nuclear data library JENDL-4.0 [10], with recent work [11] incorporating JENDL-5 [12].

2.2 Useful capabilities for reactor core analysis

2.2.1 Statistical geometry model

The MVP code employs the statistical geometry model [13] to address randomly distributed spheres, such as coated fuel particles utilized in high-temperature gas-cooled reactors (HTGRs). In this model, the location of the spheres is sampled stochastically along a neutron flight path. Upon the neutron's entry into the region where the coated fuel particles are filled, the position of a coated fuel particle is determined by sampling from the nearest neighbor distribution (NND). A neutron that enters the coated fuel particle repeats the normal random walk. If the neutron exits the coated fuel particle, the position of the coated fuel particle is determined again from the NND. The above process is repeated until the neutrons are outside the area where the coated fuel particles are located. Although the position of the coated fuel particle located in this way will be different for each neutron history, the randomly located coated fuel particle is stochastically represented from the perspective of the neutron population.

2.2.2 Burnup calculation

The MVP code package includes a burnup calculation module that enables the analysis of burnup for a variety of reactor types. Figure 1 depicts the flow of a burnup calculation with MVP-BURN [14]. Initially, cross-section data at the temperatures of interest must be prepared using the MVP-ART module. Subsequently, an eigenvalue calculation is conducted with MVP to obtain the distributions of the microscopic reaction rates at a burnup step. The depletion equation is solved with the calculated distributions to obtain the composition data in sub-steps up to one burnup step forward. MVP-BURN uses the Bateman method to solve the depletion equation. MVP-BURN performs a burnup calculation by repeating such a sequence of eigenvalue MVP and depletion calculations.

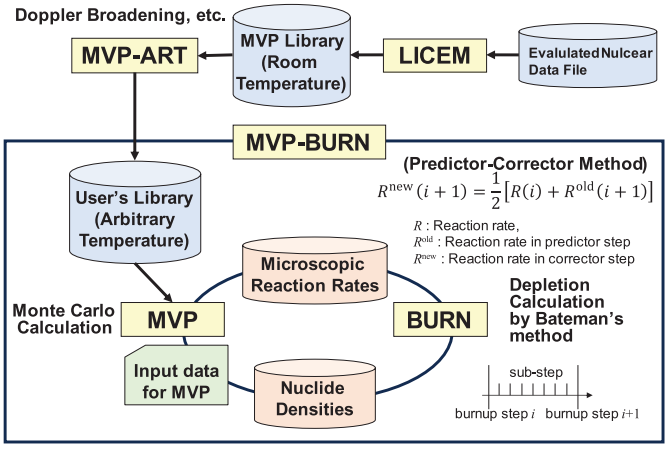


Fig. 1. Burnup calculation flow in MVP-BURN.

MVP-BURN has the capability of the predictor-corrector (PC) method [15] where a relatively long burnup step can be assumed. An average value is calculated for the microscopic reaction rates at the start and end points of a burnup step and then the depletion calculation for the step is redone from the start point with the average value. Since it is necessary to perform transport and depletion calculations twice for the same burnup step, the calculation time in the PC method is doubled compared to the calculation without the PC method. This method is effective for the burnup calculation where Gd_2O_3 is used as the burnable poison.

2.2.3 Perturbation for k_{eff} value

MVP can calculate the change in the k_{eff} value due to a perturbation in atomic densities [16]. This capability is useful for calculating a reactivity worth. The perturbation effect for the k_{eff} value can be estimated with the correlated sampling and differential operator sampling methods [17]. The differential coefficients can be estimated up to the eighth order. The impact on k_{eff} due to perturbed fission-source distribution can be also taken into account.

2.2.4 Group constant generation

MVP can generate effective microscopic and macroscopic scattering cross-sections with arbitrary energy bin structures [18]. The scattering matrix is calculated from group-to-group scattering reaction rates. In addition, MVP enables the calculation of the scattering matrix moment tally for the cosine of the scattering angle. The code can generate the Legendre components of group-to-group scattering cross-sections with an approximation of total-flux weighting instead of flux-moment weighting. In particular, the first-order scattering matrix moment tally yields the average cosine of the scattering angle. The isotropic diffusion coefficient can be thus calculated with the average cosine.

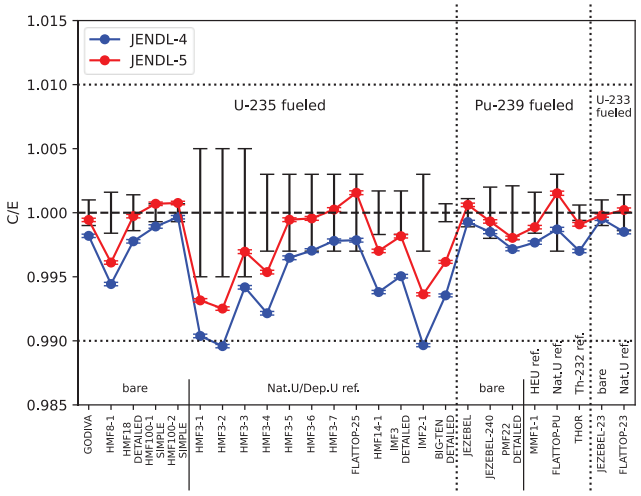


Fig. 2. C/E values of criticality for heavy-metal loaded systems. Error bars show the experimental uncertainties. Calculation uncertainties of 1 standard deviation are small enough to be hidden by markers.

2.2.5 Reactor kinetics parameters

MVP can calculate reactor kinetics parameters of the effective delayed neutron fraction β_{eff} and the generation time Λ. MVP calculates these parameters with the first-order differential operator sampling method [19,20].

2.3 Recent applications

2.3.1 Validation of JENDL

The MVP code was used for the validation of JENDL-5, which is the latest version of JENDL and was released in December 2021. Considerable efforts have been made for extensive integral tests of JENDL-5. Figure 2 shows the C/E values of criticality for heavy-metal-loaded systems as a part of the test results. Test problems were selected from the ICSBEP handbook [8]; the abbreviation such as HMF is commonly used in the handbook. Reflectors are natural or depleted uranium for the reflected systems except for MMF1-1 and THOR; their reflectors are high-enriched uranium and Th-232, respectively. In general, JENDL-5 gives better results than JENDL-4.0 for reactor core parameters. The MVP library for JENDL-5 can be downloaded at the website of [21].

2.3.2 BEAVRS benchmark

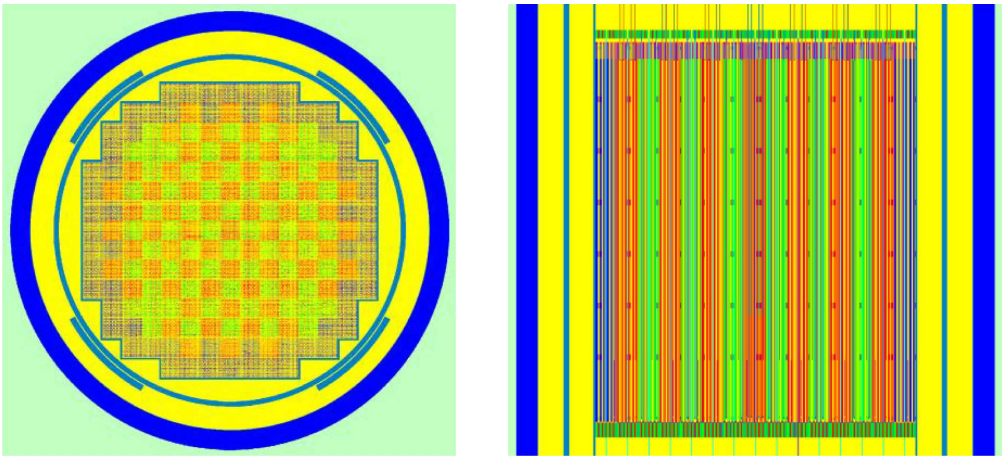
The BEAVRS benchmark problem [22] for the hot zero power (HZP) condition was solved for comprehensive validation of MVP3 although some previous work [23,24] has been done on MVP calculations. The benchmark specification of version 2.0.2 [25] was used to build the MVP calculation model; Figures 3a and 3b show the horizontal and vertical views of the model, respectively. The critical boron concentration and the control rod bank worth were calculated for 5 and 7 cases, respectively. The MVP calculations were performed for 50 million histories with a batch size of 100 000 histories with 200 inactive batches. The JENDL-4.0 evaluated nuclear data was used for all the cases.

Table 1 shows the comparison of the critical boron concentration between calculated and measured values. MVP with JENDL-4.0 can predict the critical boron concentration within 21 ppm. The critical boron concentration was calculated by interpolation of k_{eff} versus boron concentration curve. Table 2 shows the comparison of the control rod bank's worth between calculated and measured values. The maximum difference of -54 pcm can be found for case "SD with D, C, B, A, SE in". The results are comparable to those previously reported by Suzuki and Nauchi [23].

2.3.3 Neutronics/thermal-hydraulics coupled simulation

In recent years, JAEA has developed an advanced neutronics/thermal-hydraulics coupling code as a high-fidelity modeling and simulation tool for light-water reactors. To realize the high-fidelity simulation, a continuous-energy Monte Carlo code MVP and a multiphase CFD code JUPITER [26] are employed for the neutronics and thermal-hydraulics codes, respectively. The coupling code MVP/JUPITER aims to provide a reference solution for in-house reactor core design codes used in the industry.

A coupled simulation was performed for a BWR assembly with MVP/JUPITER on the multiphysics platform JAMPAN to accommodate coupled simulations [27,28]. Figures 4a and 4b show the calculation geometry for an 8×8 fuel assembly with a water rod positioned at the center. The pressure and the coolant temperature are 7.07 MPa and 549.15 K, respectively; water is injected at an inflow rate of 2.15 m/s from the bottom of the assembly. To obtain the void fraction distribution, a JUPITER calculation was initially conducted with an assumed power profile comprising a uniform horizontal distribution and a cosine-shaped axial distribution; the amount of heating was given as 71 840 W per single fuel rod. The outflow and non-slip boundary conditions were used for the top and side boundaries, respectively. JUPITER uses the Cartesian geometry; the cell size of the JUPITER was about 1 mm^3 ($130 \times 130 \times 3710$ meshes). Figure 5a shows the calculated instantaneous axial void fraction distributions in cells 1, 10, and 19; cell numbers are shown in Figure 4b. An MVP calculation was then performed with the averaged void fraction distribution. For material specification in the MVP calculation, each cell was divided into 4 sections horizontally as shown in Figure 4c, and 24 sections axially. The JUPITER meshes were mapped into the MVP material regions to average the moderator densities in each section. The MVP calculation was done for 1100 million histories with a batch size of 10 000 histories and 100 inactive batches. The calculated linear power densities for cells 1, 10, and 19 are shown in Figure 5b. It was confirmed that the a qualitatively reasonable result could be obtained for a test case simulation. We will perform steady-state neutronics/thermal-hydraulics coupled



(a) (b)

Fig. 3. MVP Calculation geometry of the BEAVRS benchmark problem for the HZP condition. (a) Horizontal view. (b) Vertical view.

Table 1. Comparison of critical boron concentration between calculated results and measured data.

Case	Measured critical boron concentration [ppm]	Calculated critical boron concentration [ppm]	Difference from measured value [ppm]
All rods out (ARO)	975	$984 \pm 0.4^*$	9
D in	902	923 ± 0.2	21
C, D in	810	826 ± 0.1	16
A, B, C, D in	686	688 ± 0.3	2
A, B, C, D, SE, SD, SC in	508	502 ± 0.2	−6

*The calculation uncertainty of 1 standard deviation is shown.

Table 2. Comparison of control rod bank worth between calculated results and measured data.

Case	Measured value [pcm]	Calculated value [pcm]	Difference from measured value [pcm]
D in	788	$766 \pm 16^*$	−22
C with D in	1203	1187 ± 18	−16
B with D, C in	1171	1203 ± 19	32
A with D, C, B in	548	552 ± 20	4
SE with D, C, B, A in	461	476 ± 21	15
SD with D, C, B, A, SE in	772	718 ± 20	−54
SC with D, C, B, A, SE, SD in	1099	1143 ± 20	44

*The calculation uncertainty of 1 standard deviation is shown.

simulations with multiple iterations of the JUPITER and MVP codes.

Another work was done for the design of an innovative reactor core called the resource-renewable BWR (RBWR) [29,30] with the MVP code. In this work, MVP including MVP-BURN was coupled with an in-house thermal-hydraulics code MOSRA-Hydro [31]. Coupled calculations were performed to investigate the calculation accuracy for the conventional core analysis method.

3 Solomon solver

3.1 General description

Solomon has been developed as a Monte Carlo solver written with the C++ language to build a code package for special applications to fuel debris systems. Solomon consists of several components such as geometry, tracking, physics model, material, nuclear data, source, tally, etc. The components are implemented as C++ classes and can be flexibly combined and replaced for special applications.

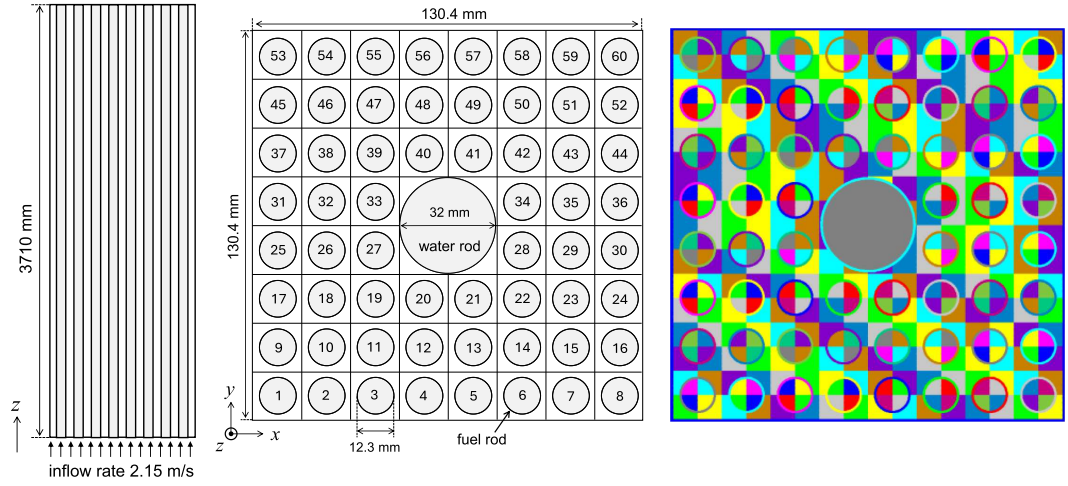


Fig. 4. Calculation geometry of the 8×8 fuel assembly for neutronics/thermal-hydraulics coupled simulation with MVP/JUPITER code. (a) Axial view. (b) Horizontal view. (c) MVP material/tally regions.

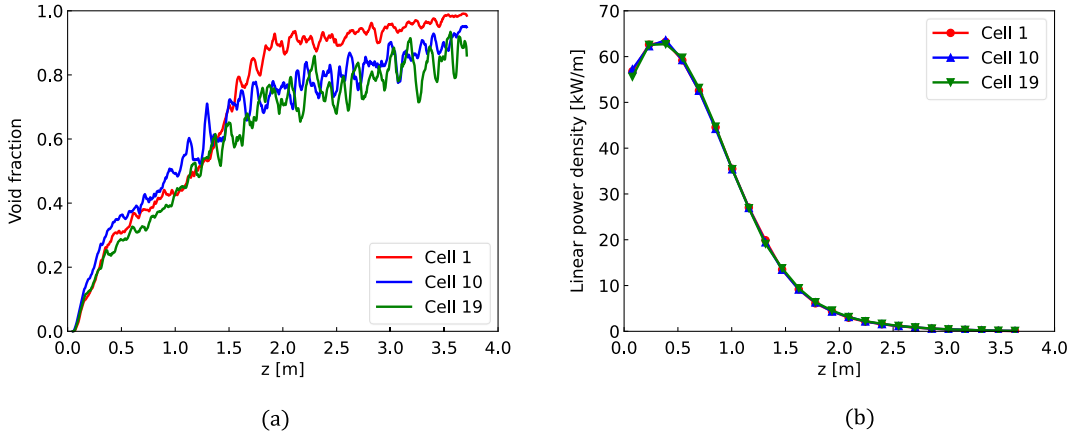


Fig. 5. Calculated results for the 8×8 fuel assembly with MVP/JUPITER code. (a) Void fraction along the z-direction. (b) Linear power density along the z-direction.

3.1.1 Physics

Solomon aims at criticality calculations with evaluated nuclear data files (ENDF); namely, only neutrons are transported with nuclear data files in the ACE format [32] and the k_{eff} value can be calculated. The ACE-formatted files can be generated with a nuclear data processing code such as FRENDY [33] or NJOY [34]. Solomon uses physics models of various LAWs defined in the ACE files. The probability table method is used in the unresolved resonance region to take the self-shielding effect into account. The free-gas model is used for neutron thermal scattering; target motion is taken into account by sampling the velocity of target nuclei from the Maxwell-Boltzmann distribution. Thermal scattering law $S(\alpha, \beta)$ tables are also available for thermal scattering.

3.1.2 Geometry

Calculation systems are modeled with surface-based geometry. Zones can be defined with pre-defined surfaces and set operations of intersection and union. The geometry definition in Solomon is similar to that in major Monte Carlo codes such as MCNP [35]. Repeated geometry is also available. Particle tracking is performed with conventional surface tracking. The delta-tracking method [36] can be also used for zones represented with the randomized Weierstrass function (RWF) model [37] or the RWF model superposed with voxel geometry.

3.2 Randomized geometry model

Solomon has a random geometry model in which the isotopic composition varies continuously using the randomized Weierstrass function (RWF). Users can generate a replica by inputting the average value and dispersion of the composition and the parameters as shown in Figure 6. Fuel debris is represented as a mixture of two materials, e.g., nuclear fuel and concrete with a ratio varying continuously in space. Since a replica is generated stochastically,

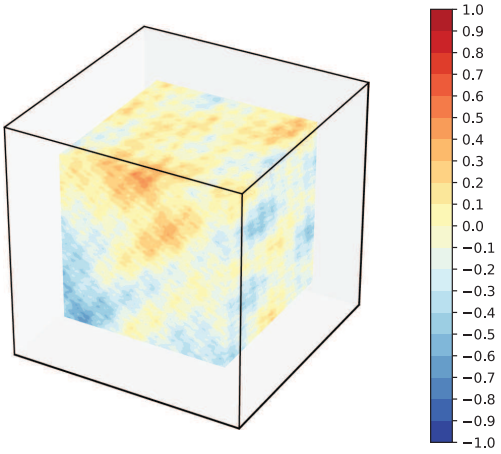


Fig. 6. A replica of the randomized Weierstrass function. The inner cube with a side of 100 cm is fuel debris, and the cube is surrounded with concrete of the thickness of 20 cm.

Table 3. Comparison of the k_{eff} values between lattice and random geometries.

Assembly type	k_{eff} (Lattice)	k_{eff} (RWF)	Ratio (RWF/Lattice)
UO_2	1.33285 (0.00022)*	1.27159 (0.00141)**	0.95404 (0.00106)***
MOX	1.16061 (0.00019)*	1.09294 (0.00660)**	0.94169 (0.00569)***

*One standard deviation for a Monte Carlo statistical uncertainty. **One standard deviation for 100 realizations of the RWF model. ***Only the standard deviation for the RWF model is taken into account.

each replica has a different isotopic composition distribution. Users can calculate the uncertainty due to the unknown distribution of the isotopic composition by calculating criticality for multiple systems and performing statistical processing.

Recently, benchmark exercises for criticality in random geometries [38] have been proposed to assess the impact of the random fragmentation of a fuel assembly on the key safety parameters for criticality calculations. Calculations were performed for exercise B with the RWF model of Solomon; the scaling factor S, which is one of the free parameters of the RWF model, is set to the assembly width of 21.45536 cm [39]. Table 3 shows the comparison of the k_{eff} values between lattice and RWF models; the lattice model is for the intact 17×17 PWR assembly. The second and third columns show the k_{eff} values for the lattice and RWF models, respectively; the fourth column shows the ratio of the k_{eff} values. One can observe that the RWF model gives the smaller k_{eff} value because of a less significant heterogeneity effect and yields a larger fluctuation in comparison with the Monte Carlo statistical uncertainty.

4 Concluding remarks

The capabilities of the MVP code were overviewed and recent applications for nuclear reactor core analysis were reviewed. MVP implements sufficient and useful capabilities for nuclear reactor core analysis. Recently MVP has been used for coupled simulations with thermal-hydraulics codes. MVP will be coupled with other physics codes such as a fuel performance code for multi-physics coupling simulations.

The capabilities of the Solomon solver were also reviewed. Solomon has a unique randomized geometry model for criticality safety analysis of nuclear systems including fuel debris. An application of the model to the benchmark exercises was reviewed. There have been some recent enhancements to the RWF model [40,41]. Future work for Solomon includes extensive verification and validation, user interface improvements, parallelization, etc. The Solomon solver is the leading candidate as a new Monte Carlo engine for MVP.

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Conflicts of interest

The author declares no conflict of interest.

Data availability statement

No data are associated with this article.

Author contribution statement

Y. Nagaya contributed the development of the Monte Carlo codes and overviewed the codes. Y. Nagaya also wrote and reviewed the manuscript.

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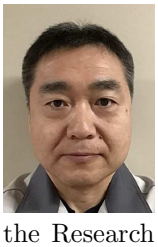

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