

# EFFECT OF THE UNIFORM FISSION SOURCE METHOD ON LOCAL POWER VARIANCE IN FULL CORE SERPENT CALCULATION

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## ABSTRACT

One of challenges of the Monte Carlo full core simulations is to obtain acceptable statistical variance of local parameters throughout the whole reactor core at a reasonable computation cost. The statistical variance tends to be larger in low-power regions. To tackle this problem, the Uniform-Fission-Site method was implemented in Monte Carlo code MC21 and its effectiveness was demonstrated on NEA Monte Carlo performance benchmark. The very similar method is also implemented in Monte Carlo code Serpent under the name Uniform Fission Source (UFS) method.

In this work the effect of UFS method implemented in Serpent is studied on the BEAVRS benchmark which is based on a real PWR core with relatively flat radial power distribution and also on 3x3 PWR mini-core simulated with thermo-hydraulic and thermo-mechanic feedbacks. It is shown that the application of the Uniform Fission Source method has no significant effect on radial power variance but equalizes axial distribution of variance of local power.

KEYWORDS: Serpent, Monte Carlo, Uniform Fission Source, variance reduction.

## 1. INTRODUCTION

The Monte Carlo codes, often coupled with depletion, thermal-hydraulic and thermal-mechanic solvers, are increasingly applied for full core reactor simulations. One of challenges of such simulations is to obtain acceptable statistical variance of local parameters throughout the whole reactor core at the same time keeping computation costs as low as possible. The statistical variance of a local tally such as local power is inversely proportional to the number of tally events and thus to the local power itself. The relative error thus tends to be larger in low-power regions as compared to high-power regions. To tackle this problem, the Uniform-Fission-Site method [1] was implemented in Monte Carlo code MC21 [2] and its effectiveness was demonstrated on NEA Monte Carlo performance benchmark [3]. The very similar method is also implemented in Monte Carlo code Serpent [4] under the name Uniform Fission Source (UFS) method [5]. This work describes the implementation of UFS in combination with fission source convergence acceleration (SCA) method [7] in Serpent and investigates the effect of UFS method on the BEAVRS benchmark [6], based on realistic PWR core.

## 2. UFS IMPLEMENTATION IN SERPENT

The idea of the UFS method is to get more source points in regions where fission power is low, and eventually improve statistics in the outermost fuel pins in full-core calculations. In the original implementation, the core geometry is covered by a mesh in which the code collects the fission source distribution during the inactive neutron cycles. This distribution is then used to adjust the number of emitted fission neutrons during the active cycles. The fission number is increased or decreased by a factor proportional to the inverse of the local fission rate, and the statistical weights are adjusted accordingly to avoid biasing the transport simulation.

Recently a fission source convergence acceleration (SCA) routine based on the response matrix method was implemented in the Serpent code [7]. The method provides a spatial distribution in a super-imposed mesh that approximates the converged fission source. It was shown that the SCA method greatly improves the initial guess for the fission source, and reduces the number of inactive cycles needed for source convergence.

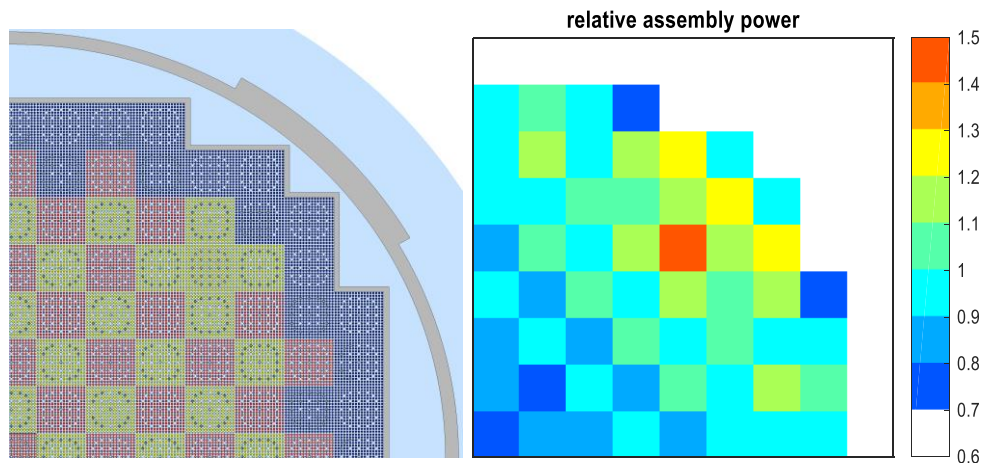
In this work, the combination of SCA and UFS methods is implemented and applied to obtain faster source convergence and improved statistics for the outermost fuel pins in a full-core calculation. This is accomplished by obtaining the spatial fission source distribution for the UFS directly from the response matrix solver. Since the approximated fission source distribution is also used to accelerate source convergence, the weighting factors for the UFS are obtained without additional computational cost.

## 3. TEST MODEL AND RESULTS

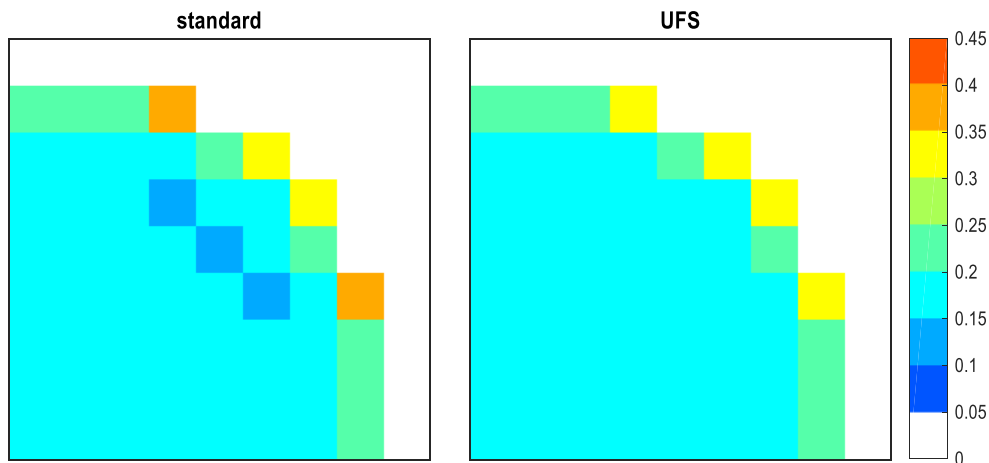
### 3.1. BEAVRS hot zero power test

The geometry and material specification for a test case could be found in the BEAVRS benchmark definition [6]. In this study the fresh reactor core is considered in hot zero power state, no burnup or xenon poisoning. The full core model was calculated without UFS (“standard” calculation) and with UFS. Each of standard and UFS calculations simulated  $20 \times 10^9$  neutron histories ( $4 \times 10^6$  neutrons in 5000 active cycles). Application of the UFS method did not influence simulation time.

The BEAVRS core layout and resulting assembly power distribution are shown in Figure 1. The calculation were performed in full core geometry but only quarter is shown in Figure 1 since results are symmetrical. The relative assembly powers vary between 0.73 and 1.43. The UFS effect on the maximum relative standard deviation of pin power in each fuel assembly is illustrated in Figure 2. The application of UFS has minor effect of a radial power uncertainty: the maximum pin power standard deviation in without UFS is 0.37% and with UFS is 0.34%, while average is 0.14% without UFS and 0.16% with UFS.



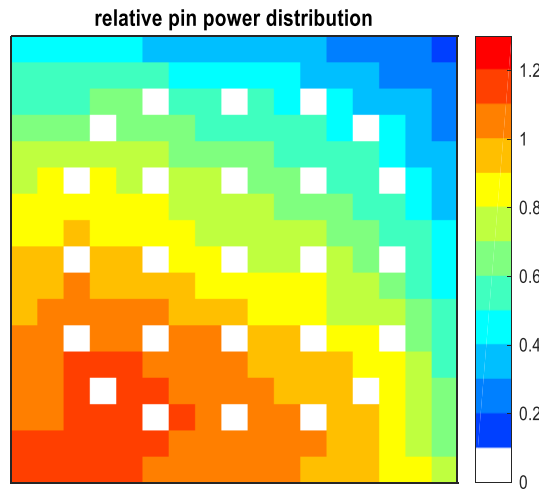
**Figure 1. Core power distribution.**



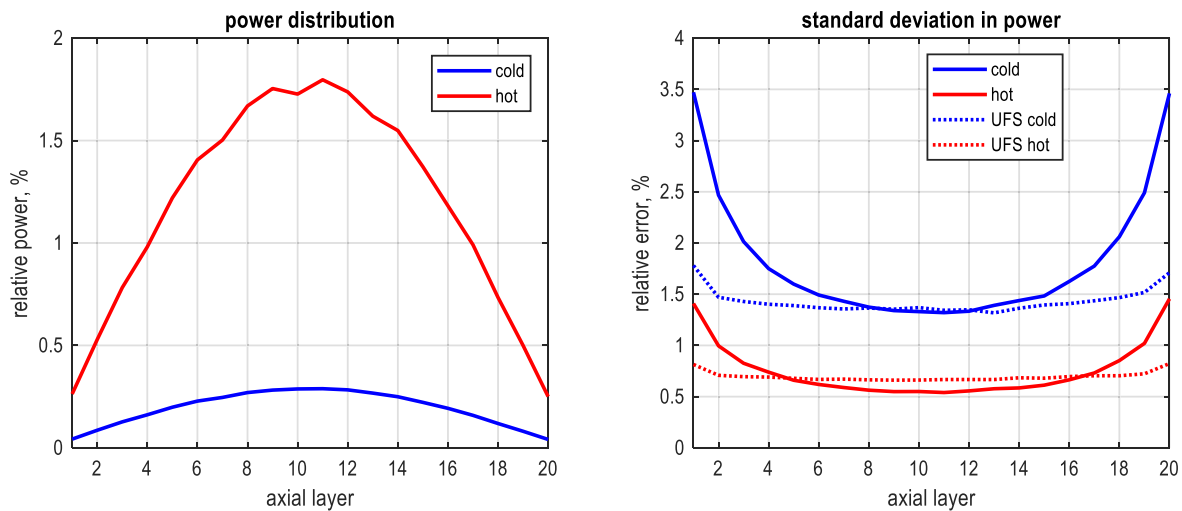
**Figure 2. Maximum relative standard deviation of pin power in each fuel assembly.**

The UFS effect on axial power uncertainty was studied on two fuel pins from a corner fuel assembly (right assembly in a top row in Figure 1). The relative pin power distribution is shown in Figure 3, pins in lower-left and top-right corners are selected as “hot” and “cold”, respectively. Axial power of selected pins and its relative standard deviations in two calculations are shown in Figure 4, with layer numbering from bottom to top. UFS effectively equalize statistical uncertainty in axial direction, improving statistics in periphery levels at the expense of statistics in a core center.

The maximum over the core standard deviation in a local power of a pin axial segment is reduced by UFS from 3.7% to 1.8%, while average value remain about the same – 0.71% in standard calculation and 0.66% in UFS.



**Figure 3. Pin power distribution in a corner assembly.**



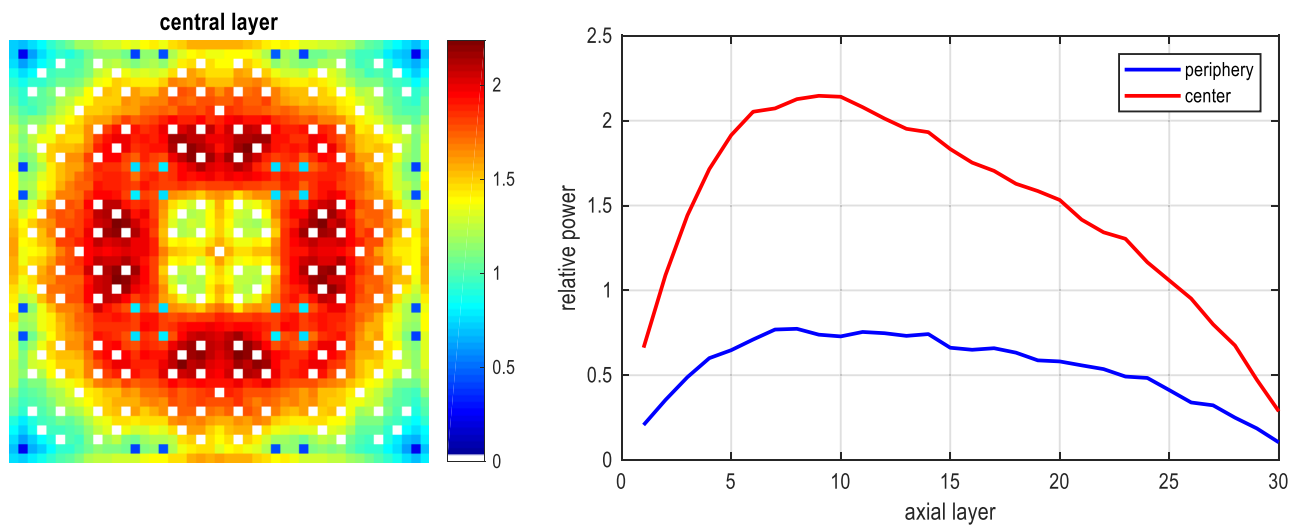
**Figure 4. Axial power its standard deviation distributions in selected “hot” and “cold” pins.**

It is also worth to mention that application of UFS does not affect resulting mean values of global multiplication factor and local power distributions, which indicates that fair game is insured.

### 3.2. PWR 3x3 mini-core with feedback

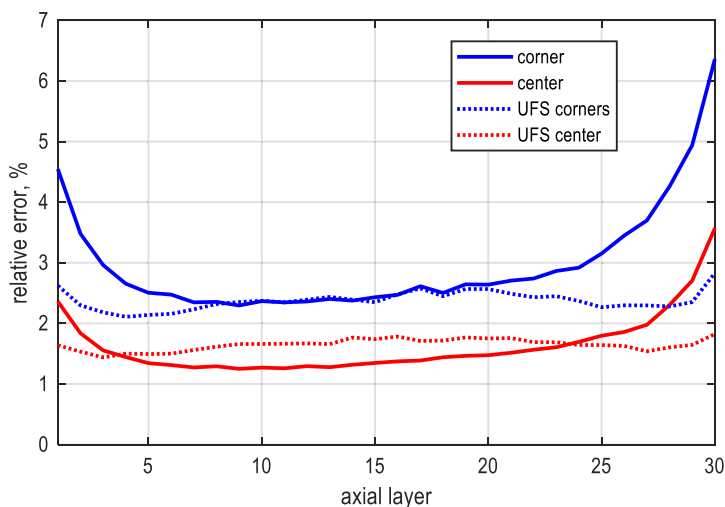
To evaluate effects of UFS method on a coupled calculation, the 3x3 PWR mini-core was simulated at the full power with thermo-hydraulic and thermo-mechanic feedbacks by the coupled system Serpent-Subchenflow-Transuranus. The coupling approach as well as test model are described in [8]. The mini-core model was chosen to reduce computation costs of a coupled calculation. The steady-state was simulated without burnup, with critical boron concentration search and equilibrium xenon concentration. For each coupled calculation 20 neutronics – thermo-hydraulic iterations were performed.

The model is axially subdivided into 30 layers. The radial power distribution in a central layer and axial distribution in a high-power “hot” central and low-power “cold” corner fuel pins are shown in Figure 5. Due to coolant density feedback the power is shifted down.



**Figure 5. Mini-core power distribution.**

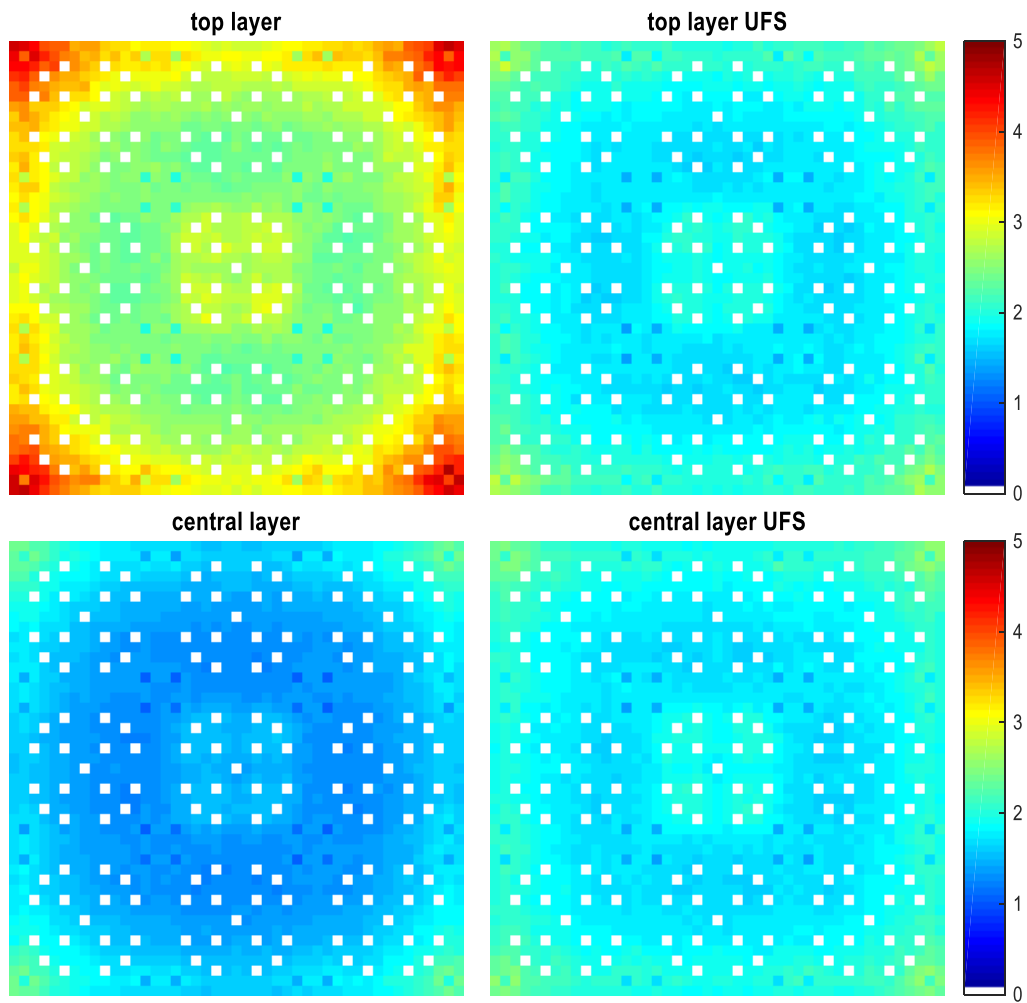
This section compares results obtained with and without UFS application. Figures 6 and 7 show relative standard deviation of the local power in percent, estimated by Serpent. The axial error distribution is illustrated in Figure 6. Just like in BEAVRS case, the UFS significantly reduces statistical errors in periphery layers at the same time slightly increasing error in a high-power middle layers.



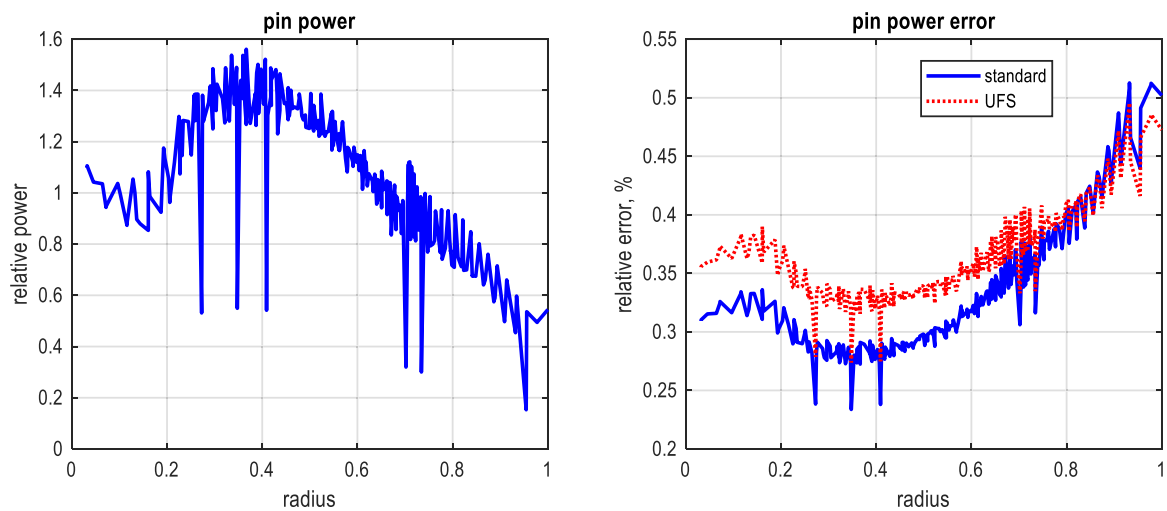
**Figure 6. Axial error distribution.**

The radial error distribution in top and central layers is shown in Figure 7. To illustrate the effect of UFS on radial error distribution, the relative pin power and its standard deviation are plotted in Figure 8 as a function of the distance from a core center. Application of UFS slightly increase statistical error in all pins except the most outer.

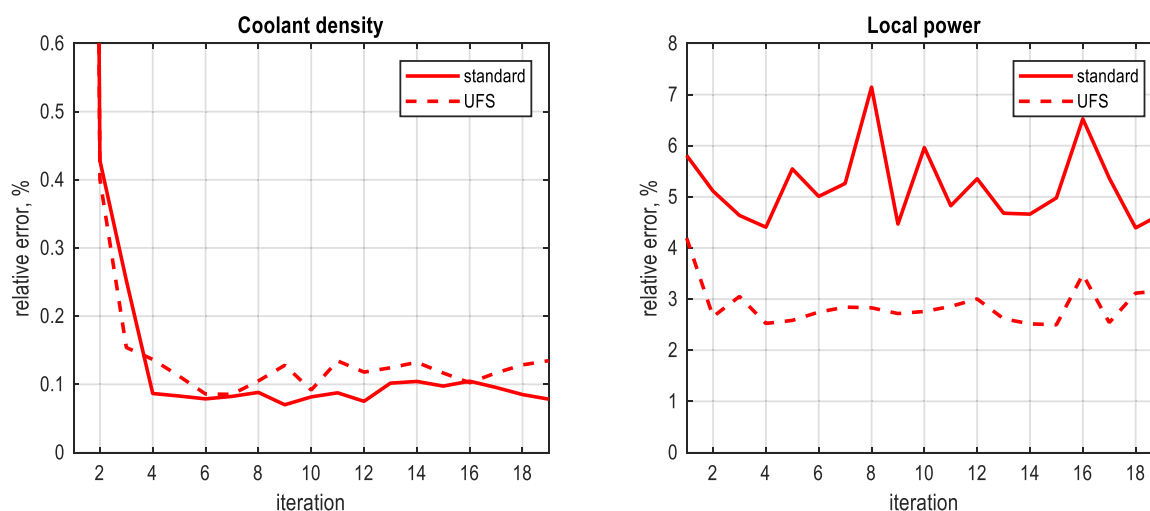
The maximum over the core standard deviation in a local power of a pin axial segment is reduced by UFS from 6.7% to 3.0%, while average value remain about the same – 1.8% in standard calculation and 1.7% in UFS.



**Figure 7. Radial error distribution in layers.**



**Figure 8. Axially averaged pin power error distribution.**



**Figure 9. Iteration convergence.**

The neutronic - thermo-hydraulic iterations convergence is illustrated in Figure 9, which shows maximal over the core relative deviation in chosen local parameters between two successive iterations. In this case certain level of convergence was reached in about 4 iterations and further convergence is limited by statistical error of Serpent calculation. Application of UFS did not influenced overall convergence, but it reduced maximal uncertainty in a local power, improving this convergence criterion.

#### 4. CONCLUSIONS

The UFS method implemented in Serpent redistributes fission source to equalize variance of the local tallies thought the reactor core. The implementation in Serpent allows to use UFS method together with source convergence acceleration without additional computational costs.

The effects of UFS are studied on BEAVRS full core case. In contrast to the NEA Monte Carlo performance benchmark which was used in [1], the BEAVRS benchmark is based on a real PWR core with relatively flat radial power distribution. The application of the Uniform Fission Source method has no significant effect on radial power variance. On the other hand, the axial power in a considered case has a typical cosine shape and variance of local power in top and bottom axial layers is few times higher than in a central layers. The application of the UFS method equalizes axial distribution of variance of local power, decreasing variance in periphery layers and increasing in central layers.

The application of UFS method to a coupled neutronic – thermo-hydraulic solution of a 3x3 mini-core test resulted in significant equalizing of an error in local power and thus the reduction of a maximal relative error, which occurs in a low-power regions on a core periphery. No significant influence on iteration convergence was found, other than improvement of a local power convergence.

In both considered cases, UFS application has reduced maximum statistical error in a local power by factor 2, keeping an average level and computational cost unchanged.

#### ACKNOWLEDGMENTS

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