A New Monte-Carlo-Based Iterative Method for Neutron Noise Calculation

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1. Introduction

Power reactor neutron noise is the fluctuations or perturbations of neutron flux induced by small oscillations of various macro parameters (density, temperatures, mechanical vibrations, flow patterns, e.g.) in a steady-state reactor core that works at a high-power level. Neutron noise contains abundant information about the reactor core's operation, which has shown to be helpful in the diagnostic, surveillance and maintenance of a power reactor[1-3]. Neutron noise-based online reactor diagnostics and monitoring system has the main advantage of being non-invasive, which can use current ex-core and in-core detectors to detect possible abnormal phenomena[4,5].

Simulating the generation of neutron noise and its in-core transport process can provide fundamental knowledge for the operation and optimization of the reactor core and the development of the non-invasive reactor core online monitoring system. With the recent development of AI(Artificial Intelligence) and ML(Machine Learning) methods, it is very promising to classify and locate various noise sources[6,7].

In view of the importance of numerical simulation, researchers have developed several frequency-domain neutron noise simulators in recent years, especially under the European CORTEX project structure. Until now, several representative neutron noise simulators have been developed based on the Deterministic or Monte-Carlo Method[8-22]. They are summarized and listed in Table I. Looking at the overall history of the development of neutron noise simulators, it gradually evolves from the deterministic method in diffusion theory toward more accurate approximations of the neutron transport theory, and finally, located on the Monte-Carlo method with origin transport theory with increasing calculation cost.

Table I: Frequency-Domain Neutron Noise Simulators

Simulator	Theory and method
CORE SIM & CORE SIM+	Diffusion, Finite Difference
DYN-GFEM	Diffusion, GFEM
DYN-ACNEM & SD-HACNEM	Diffusion, Nodal like method
FEMFFUSION	Diffusion and SPn approximation, GFEM
APOLLO-3	Transport, Short characteristic method
NOISE-SN	Transport, Sn Method
FEMN	Transport, Finite element
CORCA-NOISE	SP3 approximation, Finite element
Yamamoto's in-house code	MC Transport, Weight-cancellation technique
MCNP4C(Yamamoto)	MC Transport
TRIPOLI-4	MC Transport, Pseudo cross-section
MGMC	MC Transport, Pseudo cross-section and Weight-cancellation technique with source particle sampling
NECP-MCX	MC Transport, Pseudo cross-section and source particle sampling

Neutron noise equations are complex-valued equations, but they could be solved by the Deterministic and Mont-Carlo methods. Generally, the Monte-Carlo method can provide more accurate results with fewer approximations but takes longer, as the Deterministic method can calculate faster with its iteration schema but needs to be more precise. On the Deterministic approach, initiated by Chalmers University with diffusion theory and finite difference method[8,9], the neutron noise equation has been solved by the nodal method and expanded to the GFEM method[10-14]. The transport noise equation has also been solved by the short characteristic method and the Sn method[15,16]. Moreover, recent research has shown more focus on searching for a proper approximation of transport equation, and SP3 approximation could reach a balance between speed accuracy[18,23].

On the Monte-Carlo approach, Yamamoto first introduced an algorithm similar to traditional power iteration methods with a weight cancellation technique[19] developed by the same author in 2013. This algorithm has shown satisfactory results but needs large-scale changes on the code, especially for the binning procedure in weight-cancellation, and the calculation time will be getting longer in the low-frequency region. Inspired by the calculation of alpha eigenvalue, Rouchon issued a new method by adding a term at both sides of the neutron noise equation and introduced pseudo-cross-sections[21], which makes the traditional fixed-source algorithm available. Rouchon's algorithm uses no weight cancellation technique and is easier to implement into the standard Monte-Carlo code. However, the problem of the explosion of the particle number outside the plateau region (<0.01Hz and

>1kHz) was observed but could be resolved by simply removing implicit capture and adjusting the η value.

Iteration methods are widely used in Deterministic Methods to help accelerate convergence. The same logic could also be brought into Monte-Carlo Method. Inspired by Qi's previous work on ADS (accelerator-driven subcritical) System[24] and the K-S iteration method, which was first issued by K.F. Raskach and V. V. Korobeinikov in 1998[25], a new Monte-Carlo method that does not need any weight cancellation technique, and able to calculate in all frequency region with a higher efficiency was developed. This new method borrows the idea of the iteration method in determinism and applies it to the Monte-Carlo method, and a significant increase in stability was observed.

The rest of the paper is organized as follows. Section 2 briefly introduces the general neutron noise equation and Rouchon's pseudo-cross-section method. Then the derivation of the new K-S iteration-based Monte-Carlo method is presented. Section 3 presents the NECP-MCX platform and the implementation of the code. Section 4 calculates a 2-D simplified UOX fuel assembly noise benchmark with both direct and iteration method and compare the two methods. Conclusions are finally drawn in section 5.

2. Methods

In this section, firstly, the essential formula derivation of neutron noise equations is briefly introduced. Secondly, we introduce Rouchon's pseudo-cross-section method, which is also applied in this work. Thirdly, we demonstrate the formula derivation of the new Monte-Carlo method.

2.1 Neutron Noise Equations

Various previous works have presented the basic neutron noise equations under the traditional theory [1,19,21,22]. Therefore only a brief introduction to their derivation is given here.

Consider time-dependent neutron transport equations as follow:

$$\begin{aligned}&\frac{1}{v(\boldsymbol{r},E)}\frac{\partial\Psi(\boldsymbol{r},\boldsymbol{\Omega},E,t)}{\partial t}=\\&-\boldsymbol{\Omega}\cdot\nabla\Psi(\boldsymbol{r},\boldsymbol{\Omega},E,t)-\Sigma_t(\boldsymbol{r},E,t)\Psi(\boldsymbol{r},\boldsymbol{\Omega},E,t)\\&+\int_0^{E_{\max}}\int_{4\pi}\Sigma_s(\boldsymbol{r},\boldsymbol{\Omega}'\to\boldsymbol{\Omega},E'\to E,t)\Psi(\boldsymbol{r},\boldsymbol{\Omega}',E',t)d\boldsymbol{\Omega}'dE'\\&+\frac{\chi_p(E)[1-\beta]}{4\pi k_{eff}}\int_0^{E_{\max}}\int_{4\pi}\upsilon\Sigma_f(\boldsymbol{r},\mathrm{E}',t)\Psi(\boldsymbol{r},\boldsymbol{\Omega}',E',t)d\boldsymbol{\Omega}'dE'\\&+\frac{1}{4\pi}\sum_{i_d}^{I_d}\chi_{d,i_d}(E)\lambda_{i_d}C_{i_d}(\boldsymbol{r},t)\end{aligned}\tag{1}$$

with the i_d -th group delayed neutron precursor nucleus:

$$\frac{\beta_{i_d}}{k_{eff}}\int_0^{E_{\max}}\int_{4\pi}\upsilon\Sigma_f(\boldsymbol{r},E',t)\Psi(\boldsymbol{r},\boldsymbol{\Omega}',E',t)d\boldsymbol{\Omega}'dE'-\lambda_{i_d}C_{i_d}(\boldsymbol{r},t)\tag{2}$$

All variables above are standard in nuclear engineering.

Perform a first-order perturbation on the equations, eliminate the steady state term, neglect the second-order term, and then take a Fourier transform from the time domain to the frequency domain. Finally, one obtains the original neutron noise equations written as follows:

$$\begin{aligned}&\frac{i\omega}{v(\boldsymbol{r},E)}\delta\Psi(\boldsymbol{r},\boldsymbol{\Omega},E,i\omega)=\\&-\boldsymbol{\Omega}\cdot\nabla\delta\Psi(\boldsymbol{r},\boldsymbol{\Omega},E,i\omega)-\Sigma_t(\boldsymbol{r},E)\delta\Psi(\boldsymbol{r},\boldsymbol{\Omega},E,i\omega)\\&+\int_0^{E_{\max}}\int_{4\pi}\Sigma_s(\boldsymbol{r},\boldsymbol{\Omega}'\to\boldsymbol{\Omega},E'\to E)\delta\Psi(\boldsymbol{r},\boldsymbol{\Omega}',E',i\omega)d\boldsymbol{\Omega}'dE'\\&+\frac{\chi_p(E)[1-\beta]}{4\pi k_{eff}}\int_0^{E_{\max}}\int_{4\pi}\upsilon\Sigma_f(\boldsymbol{r},\mathrm{E}')\delta\Psi(\boldsymbol{r},\boldsymbol{\Omega}',E',i\omega)d\boldsymbol{\Omega}'dE'\\&+\frac{1}{4\pi k_{eff}}\sum_{i_d=1}^{I_d}\frac{\chi_{d,i_d}(E)\beta_{i_d}\lambda_{i_d}}{\lambda_{i_d}+i\omega}\int_0^{E_{\max}}\int_{4\pi}\upsilon\Sigma_f(\boldsymbol{r},\mathrm{E}')\delta\Psi(\boldsymbol{r},\boldsymbol{\Omega}',E',i\omega)d\boldsymbol{\Omega}'dE'\\&+S(\boldsymbol{r},\boldsymbol{\Omega},E,i\omega)\end{aligned}\tag{3}$$

$$\begin{aligned}&S(\boldsymbol{r},\boldsymbol{\Omega},E,i\omega)=-\delta\Sigma_t(\boldsymbol{r},E,i\omega)\Psi_0(\boldsymbol{r},\boldsymbol{\Omega},E)\\&+\int_0^{E_{\max}}\int_{4\pi}\delta\Sigma_s(\boldsymbol{r},\boldsymbol{\Omega}'\to\boldsymbol{\Omega},E'\to E,i\omega)\Psi_0(\boldsymbol{r},\boldsymbol{\Omega}',E')d\boldsymbol{\Omega}'dE'\\&+\frac{\chi_p(E)[1-\beta]}{4\pi k_{eff}}\int_0^{E_{\max}}\int_{4\pi}\upsilon\delta\Sigma_f(\boldsymbol{r},\mathrm{E}',i\omega)\Psi_0(\boldsymbol{r},\boldsymbol{\Omega}',E')d\boldsymbol{\Omega}'dE'\\&+\frac{1}{4\pi k_{eff}}\sum_{i_d=1}^{I_d}\frac{\chi_{d,i_d}(E)\beta_{i_d}\lambda_{i_d}}{\lambda_{i_d}+i\omega}\int_0^{E_{\max}}\int_{4\pi}\upsilon\delta\Sigma_f(\boldsymbol{r},\mathrm{E}',i\omega)\Psi_0(\boldsymbol{r},\boldsymbol{\Omega}',E')d\boldsymbol{\Omega}'dE'\end{aligned}\tag{4}$$

where i is the imaginary unit and ω is the angular frequency, Ψ_0 represent steady state neutron flux, terms with δ represent perturbations.

2.2 Pseudo Cross-Section Method

Rouchon presented a new Monte-Carlo algorithm by adding a term at both sides of the equation. It makes the conventional fixed-source algorithm available and doesn't need more changes to the code. It also performs better than Yamamoto's method in typically interested frequency regions. The formula derivation is demonstrated as follows:

Add a term $\frac{(\eta-i)\omega}{v(\boldsymbol{r},E)}\delta\Psi(\boldsymbol{r},\boldsymbol{\Omega},E,i\omega)$ at both sides of the neutron noise equations, one obtains:

$$\begin{aligned}&\boldsymbol{\Omega}\cdot\nabla\delta\Psi(\boldsymbol{r},\boldsymbol{\Omega},E,i\omega)+\left[\Sigma_t(\boldsymbol{r},E)+\frac{\eta\omega}{v(\boldsymbol{r},E)}\right]\delta\Psi(\boldsymbol{r},\boldsymbol{\Omega},E,i\omega)\\&=\frac{(\eta-i)}{\eta}\eta\frac{\omega}{v(\boldsymbol{r},E)}\delta\Psi(\boldsymbol{r},\boldsymbol{\Omega},E,i\omega)\\&+\int_0^{E_{\max}}\int_{4\pi}\Sigma_s(\boldsymbol{r},\boldsymbol{\Omega}'\to\boldsymbol{\Omega},E'\to E)\delta\Psi(\boldsymbol{r},\boldsymbol{\Omega}',E',i\omega)d\boldsymbol{\Omega}'dE'\\&+\frac{\chi_p(E)[1-\beta]}{4\pi k_{eff}}\int_0^{E_{\max}}\int_{4\pi}\upsilon\Sigma_f(\boldsymbol{r},\mathrm{E}')\delta\Psi(\boldsymbol{r},\boldsymbol{\Omega}',E',i\omega)d\boldsymbol{\Omega}'dE'\\&+\frac{1}{4\pi k_{eff}}\sum_{i_d=1}^{I_d}\frac{\chi_{d,i_d}(E)\beta_{i_d}\lambda_{i_d}}{\lambda_{i_d}+i\omega}\int_0^{E_{\max}}\int_{4\pi}\upsilon\Sigma_f(\boldsymbol{r},\mathrm{E}')\delta\Psi(\boldsymbol{r},\boldsymbol{\Omega}',E',i\omega)d\boldsymbol{\Omega}'dE'\\&+S(\boldsymbol{r},\boldsymbol{\Omega},E,i\omega)\end{aligned}\tag{5}$$

where η is a real constant having the same sign as ω, and normally set to $\eta=1$.

The total cross-section in the second term of the left side of the equation is replaced by

$\Sigma_t^{pseudo}(\boldsymbol{r},E)=\Sigma_t(\boldsymbol{r},E)+\frac{\eta\omega}{v(\boldsymbol{r},E)}>0$, which we call it pseudo total cross-section. Similarly, the first term at the right side of the equation is considered as pseudo fission term, which has a pseudo fission cross-section of $\eta\frac{\omega}{v(\boldsymbol{r},E)}$. Thus, a conventional fixed-source algorithm could solve the neutron noise equation.

2.3 K-S iteration with Monte-Carlo Method

K-S iteration was first posted by K.F. Raskach and V. V. Korobeinikov [25] in 1998 as an effective determinist algorithm for calculating subcritical reactor systems with an external source. It was introduced into the Monte-Carlo method later for the calculation of time-dependent ADS subcritical problems [24].

It should be remarked that the neutron noise problem in the frequency domain is inherently subcritical,

Then the neutron noise equation with the pseudo-cross-section method could be written in the operator form as follow:

$$L\Psi = P\Psi + D\Psi + Ps\Psi + S \tag{6}$$

Where $L\Psi$ represent the transport operator, $P\Psi$ represent the prompt fission neutron operator, $D\Psi$ represent the delayed fission neutron operator, $Ps\Psi$ represent the pseudo fission neutron operator, S represent the external source operator.

Traditional fixed-source algorithm solves the neutron noise equation in direct form, which is relatively unstable in the convergence and its calculation time.

$$\Psi = (L - P - D - Ps)^{-1} S \tag{7}$$

In order to use K-S iteration method, we define ks as the ratio between the prompt fission term and the prompt fission term plus the external source term:

$$ks = \frac{\langle P\Psi \rangle}{\langle P\Psi \rangle + \langle S \rangle} \tag{8}$$

And rewrite S in the form

$$S = \langle S \rangle S_0 \tag{9}$$

Where $\langle S_0 \rangle = 1$, then S_0 represent the shape of the external source. ($\langle \bullet \rangle$ donates integration over the phase volume of the system)

Using the equations above, we may rewrite equation (7) into equation (10):

$$(L - D - Ps)\Psi = P\Psi + \frac{1-ks}{ks}\langle P\Psi \rangle S_0 \tag{10}$$

Equation (7) could be solved by solving equation (10) firstly. We may use the iterative scheme:

$$ks_n = \frac{\langle P\Psi_{n-1} \rangle}{\langle P\Psi_{n-1} \rangle + \langle S_{n-1} \rangle} \tag{11}$$

$$(L - D - Ps)\Psi_n = P\Psi_{n-1} + \frac{1-ks_{n-1}}{ks_{n-1}}\langle P\Psi_{n-1} \rangle S_0 \tag{12}$$

Compared to the direct method, our new K-S iteration method borrows the iteration idea from the deterministic way, puts the prompt fission term at the right side of the equation, and avoids the direct simulation of the tremendous history of a single particle. Therefore, the external noise source is underestimated by a ratio related with ks , so that we can obtain the actual result by multiplying at each converged tally cycle:

$$\Psi_{real} = \frac{1}{1-ks}\Psi_n \tag{13}$$

3. Implementation

This section will introduce the implementation of neutron noise calculation flow within the NECP-MCX neutron transport code platform with the direct and K-S iteration method.

3.1 The NECP-MCX Code and its neutron noise calculation.

NECP-MCX is a Monte-Carlo-deterministic coupled particle transport simulation software developed by the NCEP team of Xi'an Jiaotong University with completely independent intellectual property rights.

The neutron noise calculation module was recently developed in an in-house developing version of the NECP-MCX code, including functions of generating noise source, direct and K-S iteration method, and further abilities like sampling based on source particles and different noise source type is being actively developed. Fig 1 demonstrates the Flow chart of the generation of noise source in NECP-MCX.

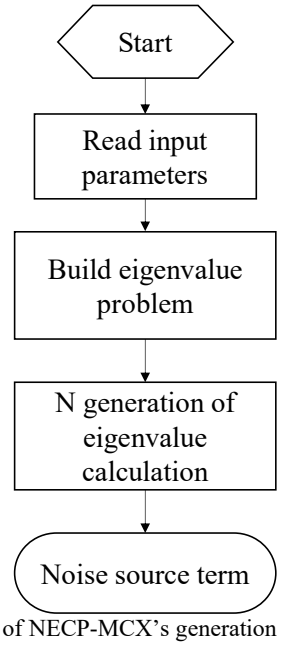

Fig. 1. Flowchart of NECP-MCX's generation of noise source

The direct method will use the traditional fixed-source calculation part, which will trace the history of the source particles one by one in different batches, with secondary particles created and banked during the calculation. Fig 2 shows the flow chart of fixed-source calculation.

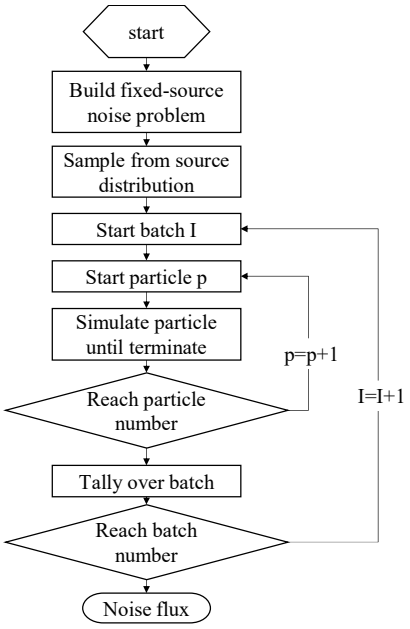

Fig. 2. Flowchart of NECP-MCX's direct method

The K-S iteration method has a similar scheme to traditional eigenvalue calculation but only needs to change the neutron banked in the fission bank. The flow chart of neutron noise calculation is shown in Fig 3.

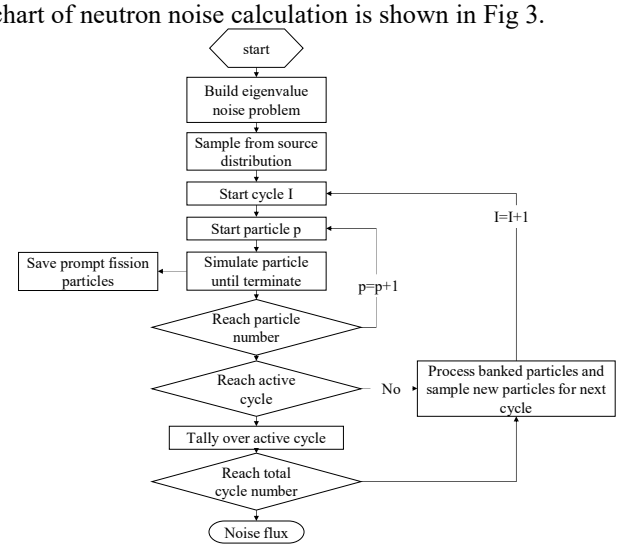

Fig. 3. Flowchart of NECP-MCX's K-S iteration method

4. Numerical results

The numerical results of the proposed K-S iteration method will be presented in this section. We introduce the simplified 2-D neutron noise benchmark problem, which has been used to test and compare different neutron noise solvers. Then the numerical results and a comparation between direct method and iteration method were given.

4.1 The Simplified 2-D Neutron Noise Benchmark

In order to compare different neutron noise calculation methods, a neutron noise benchmark is selected. This problem has been used to compare other neutron noise solvers developed under the structure of the CORTEX project [26]. The problem's configuration is a simplification of UOX fuel assembly for Pressurized Water Reactors, with 17×17 homogeneous square fuel assembly. The layout of this system is presented in Fig 4. The assembly pitch is 1.26 cm, the size of the fuel pin is 0.7314 cm×0.7314 cm, with 25 water holes of 1.26 cm×1.26 cm. Reflective boundary conditions are applied to the system. The cross-sections and the group constants are shown in table II.

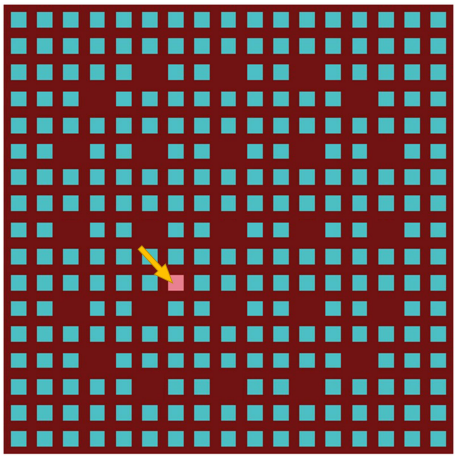

Fig. 4 Geometry of the Noise Problem

Table II: Parameters of the Noise Problem

Material	Moderator		Fissionable material	
	Fast group	Thermal group	Fast group	Thermal group
Total cross-section/cm-1	0.25411	1.2182	0.3779	0.55064
Scattering cross-section (matrix)/cm-1	0.22519143	0.028124	0.35128029	0.00086471
	0	1.188884	0	0.39276
Fission neutron production cross-section/cm-1	0	0	0.014940710628	0.2751820296
Fission cross-section/cm-1	0	0	0.0057671	0.10622
Fission neutron spectrum	-	-	1	0
Group constants	Fast neutron velocity/cm·s-1	Thermal neutron velocity/cm·s-1	Share of delayed neutron	Decay constant/·s-1
	1.82304E7	4.13067E5	0.00535	0.0851

The perturbation is defined as the percentage oscillation of the macroscopic cross-section of one of the fuel pins, which is pointed out with an arrow in Fig 4. The parameters of the perturbation are shown in table III.

Table III: Parameters of the perturbation

Total cross-section	0.041
Scattering cross-section	0.034
Fission cross-section	0.021

The frequency of the noise source is chosen as 1 Hz, and $\eta = 1$ is set by default.

4.2 Numerical results

We compared the results from both iteration method and direct method. Results from direct method is set as the reference. The calculation is performed on Beijing Super computer platform. The computational nodes are equipped with AMD EPYC 7452 32-Core Processors, 64 cores, 256GB, nodes are related with the ib net. All calculations using 4 nodes with 256 cores.

First, we simulate normal eigenvalue problem to get the noise source distribution and its strength. 2000 cycles are simulated and each cycle contain 100,000 particles with the first 30 cycle as non-active cycle. After then the calculated noise source is transformed into a complex-valued noise source by a python script and prepared to be used in the future.

With the procedures mentioned in section 2, the results from direct method are obtained as the reference value, and compared with the iterative one. Fig 5-8 shows the Amplitude and phase of neutron noise for fast and thermal group at the frequency of 1 Hz.

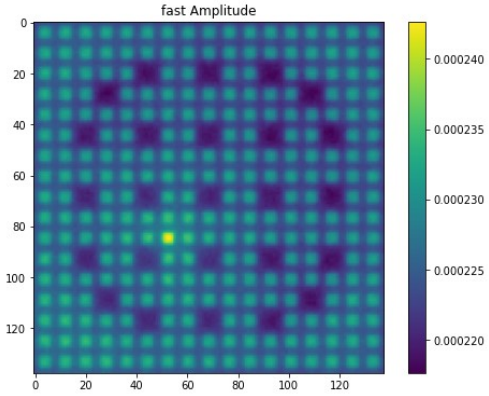

Fig. 5. Fast Amplitude calculated by Iterative method

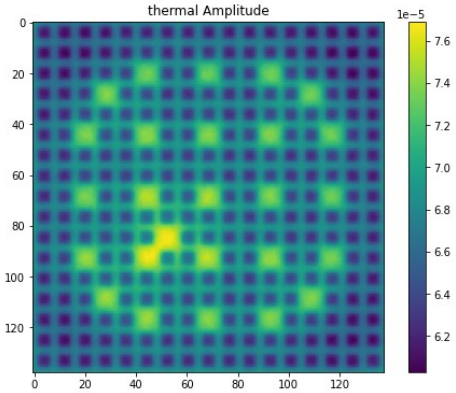

Fig. 6. Thermal Amplitude calculated by Iterative method

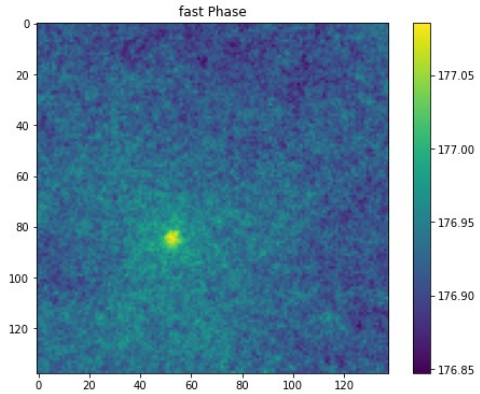

Fig. 7. Fast Phase calculated by Iterative method

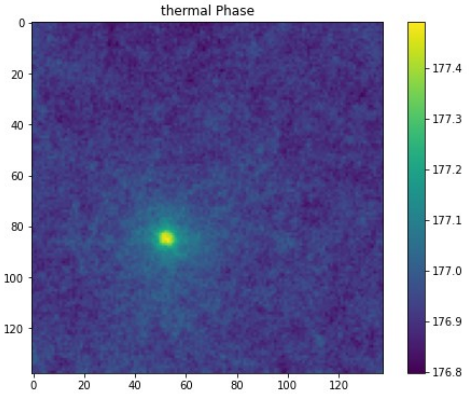

Fig. 8. Thermal Amplitude calculated by Iterative method

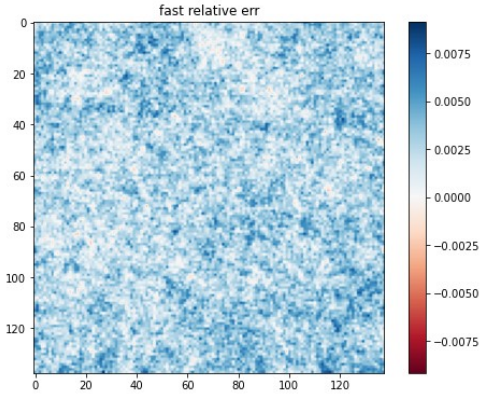

Fig. 9. Thermal Amplitude calculated by Iterative method

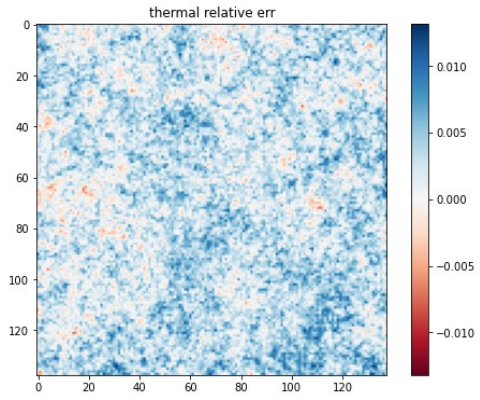

Fig. 10. Thermal Amplitude calculated by Iterative method

Fig 9 an Fig 10 shows the relative err of the direct Monte-Carlo method and iterative Monte-Carlo method. Results shows the new method has the ability to solve the neutron noise equation accurately.

5. Conclusions

In this work, we developed neutron noise calculation modules in the NECP-MCX code. A new Monte-Carlo-based iterative method to solve the neutron noise equations was proposed and compared with the direct method. The new method was tested with the simplified neutron noise benchmark, and the numerical results show that the new method could solve the neutron noise equation correctly.

Further works on the new method will be done in the future, especially its performance in the non-plateau region, and more focus on improving the convergence of the new algorithm.

Acknowledgements

This work is supported by the National Natural Science Foundation of China (NO. U2067209), the Innovative Scientific Program of CNNC and the Young Elite Scientists Sponsorship Program by CAST (NO. 2019QNRC001).

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