Application of MC-MC Coupled Method in Neutron Shielding Analysis of Reactor Pit in Reactor Building

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Abstract. As one of the areas with highest radiation risk in nuclear power plant with neutron and gamma ray radiation emitted from the reactor core, the reactor pit in reactor building has great radiation impact on personnel radiation safety by neutron. In order to further improve the shielding design effort of reactor pit, the MC-MC coupled method is developed. By applying the MCNP code and SuperMC code, the variance of the effective dose rate and thermal neutron flux inside the reactor pit is lower than 5% by consuming 6.7 h. The result shows that the MC-MC coupled method can effectively solve neutron shielding problem in large scale or in complex space.

Keywords: Neutron design · Reactor pit · Monte carlo · MCNP · MC-MC coupled method

1 Introduction

The reactor pit in reactor building, as one of the area with highest radiation risk of neutron radiation in reactor building, is a neutron shielding design object with main focus [1, 2] Accurate neutron shielding analysis is carried out in order to minimize the neutron radiation risk and meet the requirements for personnel access to reactor pit. Hence, in order to analyze neutron shielding design problem of large-scale model accurately, a new analysis methodology that meets shielding design requirements should be established.

2 Development of Neutron Shielding Design for Reactor Pit

2.1 Methodology Development of Neutron Shielding Analysis

At present, two neutron shielding analysis method is widely used in the field of nuclear engineering: the deterministic method represented by the discrete ordinate method and the Monte Carlo (MC) non-deterministic method [3]. The discrete ordinate method is mostly used to deal with engineering problems that can be simplified to one-dimensional or two-dimensional problem, while Monte Carlo method is for stochastic simulation of neutron behavior by computer. Although Monte Carlo method can simulate complex

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geometry with high accuracy, slow convergence speed greatly influences its engineering application.

Due to the structural complexity and large-scale characteristics in nuclear power plant (NPP), neutron shielding problems with large and complex geometric is usually solved by coupled multi-dimensional calculation method [4–6]. Therefore, the MC-MC coupled method, which can decompose one model into continuous parts, is established to solve this type of problem. The MC-MC coupled method not only has high reliability on calculation results, but also has low accuracy loss during continuous calculation. Hence, MC-MC coupled method has been widely used to solve neutron shielding problem of large nuclear facilities due to the above advantages.

2.2 Development of Neutron Shielding in Reactor Pit

In order to shield the neutron generated by the reactor core, different NPPs adopt different structural design and shielding design to establish a sufficient shielding effect.

As shown in Fig. 1, the reactor pit and its passage in EPR NPP is built up by concrete wall with a watertight door at the egress of the reactor pit passage. Since the egress is on the second floor above the reactor pit passage, the access process of personnel includes stair climbing activity.

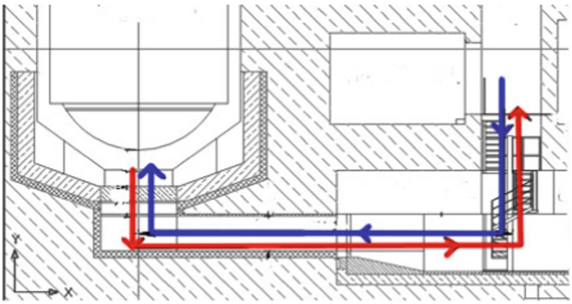

Fig. 1. Access Process of Reactor Pit in EPR NPP

Similar to EPR NPP, the access process in AP1000 NPP which is shown in Fig. 2 also includes stair climbing.

As for CPR NPP, an iron-made shielding door with 7 cm thick is installed at the egress of the reactor pit passage. Furthermore, unlike the reactor pit in other types of NPP, the egress and the passage of CPR is on the same floor (Fig. 3).

For the reactor pit in HPR1000 NPP, the design experience of EPR has been referred and a airtight door instead of watertight door is installed at the egress of the reactor pit passage (Fig. 4).

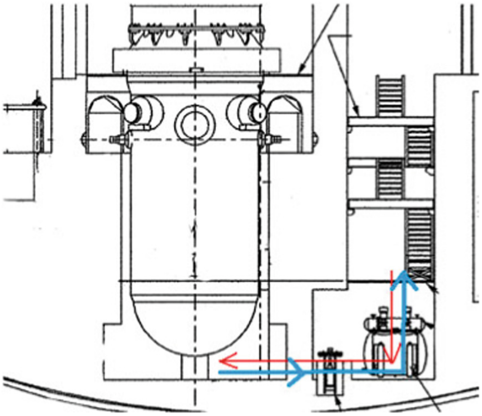

Fig. 2. Access Process of Reactor Pit in AP1000 NPP

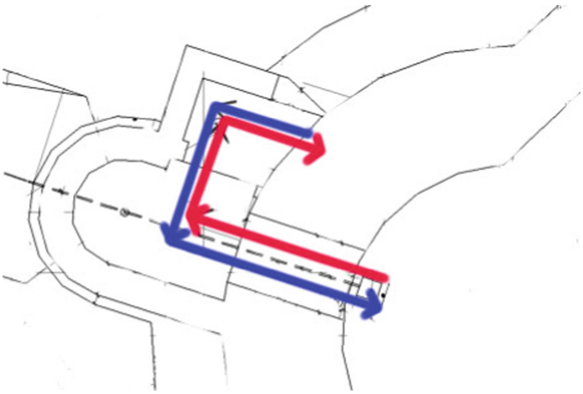

Fig. 3. Access Process of Reactor Pit in CPR NPP

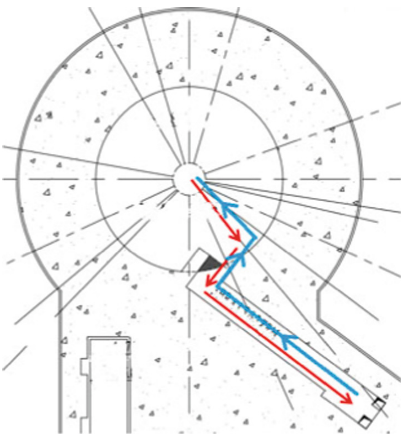

Fig. 4. Access Process of Reactor Pit in HPR1000 NPP

3 Establishment of MC-MC Coupled Method

3.1 Process of MC-MC Coupled Method

The current simulation program capable with MC-MC coupled method are MCNP code and SuperMC code developed based on Monte Carlo algorithm. These two programs can not only read and write standard surface source, but can also record accurately the particle trajectory information passing through the continuous surface as source distribution input for the continuous calculation. The MC-MC coupled method can greatly reduce calculation time and save computing hardware on the premise of ensuring result accuracy. Hence, these two programs are both suitable for solving large-scale neutron shielding problems such as reactor pit in NPP. The process of MC-MC coupled method is shown in Fig. 5.

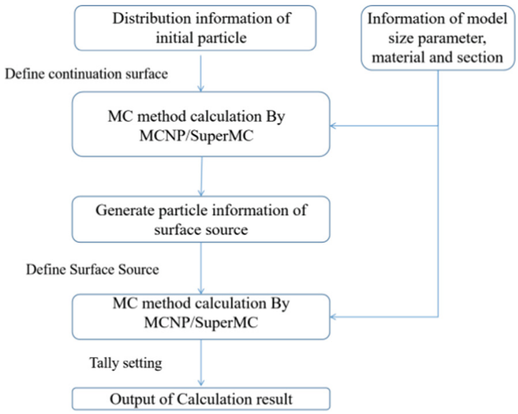

Fig. 5. The Process of MC-MC Coupled Method

3.2 Selection Principle of Continuation Surface

The reliability of the calculation result by MCNP or SuperMC code depends on its statistical error. In addition, as the statistical error is transitive in the process of MC-MC coupled method, the statistical error of the surface flux density should be controlled to ensure that the influence of statistical error transfer is acceptable on the reliability of the results when selecting the continuation surface.

The integral calculation is taken as an example to briefly describe the calculation process of the Monte Carlo sampling calculation method. The integral to be computed is treated as the mathematical expectation of a random variable *f(x)* obeying the distribution density function *p(x)*:

$$F = \int f(x)dx \tag{1}$$

With

$$\int p(x)dx = 1 \tag{2}$$

Through sampling experiments, sub-samples $x_1, x_2 \ldots x_n$ can be drawn from the distribution density function *p(x)*, and the arithmetic mean $\overline{f}$ of the corresponding random variables $f(x_i)$ is the Monte Carlo estimation of the integral value F of the values:

$$\overline{f} = \frac{1}{n}\sum\nolimits_{i=1}^{n} f(x_i) \tag{3}$$

The unbiased estimate of statistical error σ^2 is calculated by formula (4):

$$\sigma^2 = \frac{1}{n}\sum\nolimits_{i=1}^{n} \left[f(x_i) - \overline{f}\right]^2 \tag{4}$$

As statistical error is transitive in formula (3), the total value of the statistical error after n times of consecutive calculation is approximately as formula (5):

$$\sigma = \prod\nolimits_{i=1}^{n} (1 + \sigma_i) - 1 \approx \sigma_1 + \sigma_2 + \cdots + \sigma_n \tag{5}$$

Hence, the calculation results by MCNP code or SuperMC code is reliable only when the statistical error of calculation result is within 10%. According to the above analysis, for a two-step connection problem, the area with the statistical error of the surface flux density less than 5% should be selected as the spatial connection surface to ensure that the final result of the connection calculation is reliable.

4 Application of MC-MC Coupled Method for Reactor Pit Shielding Design

Take one Chinese NPP as an example. By applying MC-MC coupled method, the neutron shielding design problem of reactor pit can be evaluated accurately with less time consumed.

The general shielding design objectives for the NPP during normal operation is to ensure that the radiation exposure to personnel is ALARP and within the limits of radiation regulations. As the neutron radiation emitted from the reactor core is one of the predominant contributors to the reactor pit and its passage, specific shielding design objectives of reactor pit egress are as follows:

1. the area dose rate should not exceed 1 mSv/h;
2. the thermal neutron flux should not exceed 1.00E + 05 n/(cm^2.s).

4.1 Calculation Assumption and Modeling

As the reactor pit is complex and is in large scale, the MC-MC coupled method is suitable to solve the neutron shielding problem of this area. Based on conservative assumption, the neutron passing through the continuation surface are assumed to be all reflected into the reactor pit passage. By referring the actual structure of the reactor pit, the calculation model is established in Fig. 6 by SuperMC code.

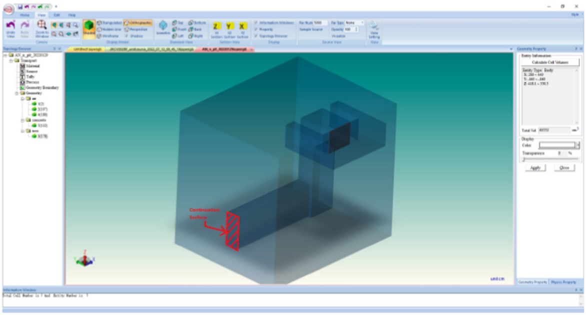

Fig. 6. Three-Dimensional View of Reactor Pit Passage Model

4.2 Calculation Process and Result Analysis

Based on the calculation assumption and modeling analysis in Sect. 4.1, the environmental effective dose rate and neutron flux are analyzed by MC-MC coupled method. The advantage of the methodology would be found through the calculation process.

A) Neutron Energy Distribution by First Continuation Calculation
As the continuation surface is selected in the red flame in Fig. 6, by applying MCNP code, the neutron energy distribution on continuation surface is calculated and is shown in Fig. 7.

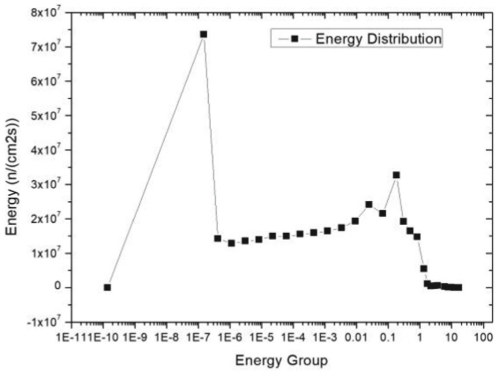

Fig. 7. Neutron Energy on Continuation Surface

B) Effective Dose Rate by Secondary Continuation Calculation
The environmental effective dose rate and neutron flux in reactor pit passage are analyzed by applying the neutron energy distribution on continuation surface in the secondary continuation calculation as calculation input. Neutron particle with a number of 1.00E + 06 is simulated in calculation with about 101 h consumed.

As shown in Fig. 8 and Fig. 9, the effective dose rate and neutron flux at the egress of reactor pit could not meet the radiation protection requirement if no shielding door is

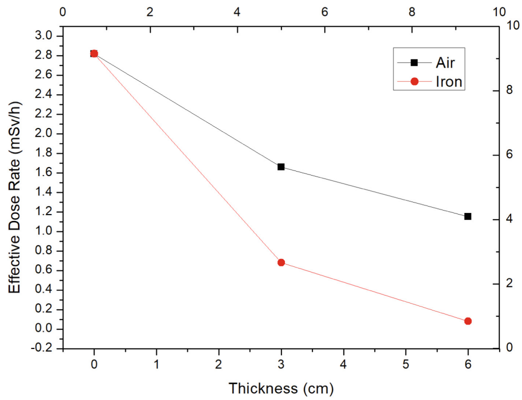

Fig. 8. Effective Dose Rate at the Egress of Reactor Pit

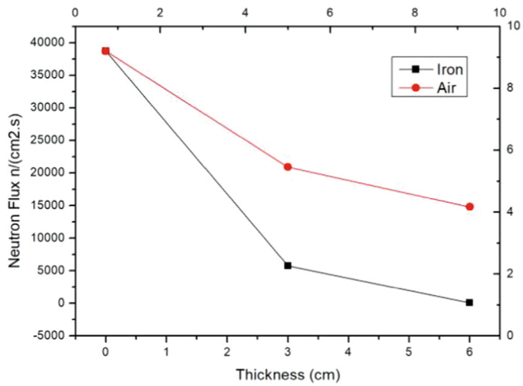

Fig. 9. Neutron Flux at the Egress of Reactor Pit

installed. An iron shielding door with thickness of 3 cm at the egress of reactor pit could ensure that the effective dose rate and neutron flux meets the design objective.

C) Three-Dimensional View of Environmental Effective Dose Rate and Neutron Flux

In order to evaluate the three-dimensional dose rate field and neutron flux field as well as for the process acceleration, SuperMC code is applied to optimize the calculation.

SuperMC code is a computer aided design (CAD)-based Monte Carlo program for integrated simulation of nuclear system developed by Chinese FDS team [7–9]. Compared with MCNP code, SuperMC code can generate global mesh weight window to speed up the three dimensional dose rate field calculation, which is practical to analyze the environmental effective dose rate variance tendency [10].

By using 64 CPU in parallel calculation, the global mesh weight window is generated with 1.0E + 13 neutron after consuming 2.8 h. The effective dose rate and neutron flux are then calculated with the help of generated global mesh weight window by consuming 6.8 h with 1.00E + 09 neutron. The three-dimensional view of neutron effective dose rate field and neutron flux field are shown in Fig. 10, Fig. 11 and Fig. 13.

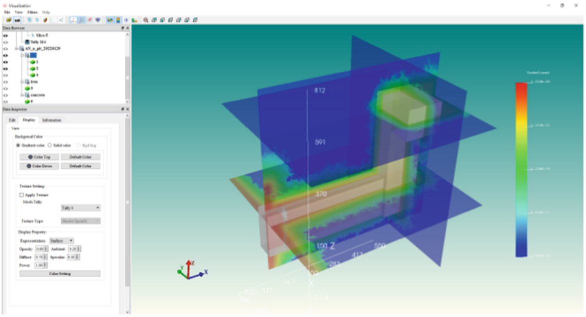

Fig. 10. Effective Dose Rate Field of Reactor Pit Passage Contributed by Neutron

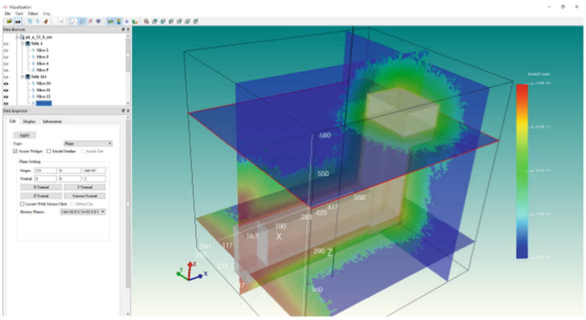

Fig. 11. Effective Dose Rate Field of Reactor Pit Passage Contributed by Secondary Photon

As shown in Fig. 12, the effective dose rate is calculated alongside the exit direction of reactor pit egress. As the origin point is set at the inner surface of shielding door, the neutron area dose rate decrease to 1 mSv/h at the reactor pit egress shielded by the iron in the airtight door.

Meanwhile, the three-dimensional view of neutron flux is shown in Fig. 13. Similar to the variance tendency of effective dose rate, the neutron flux decrease to 1.00E + 05 n/(cm^2.s) at the reactor pit egress which is presented in Fig. 14.

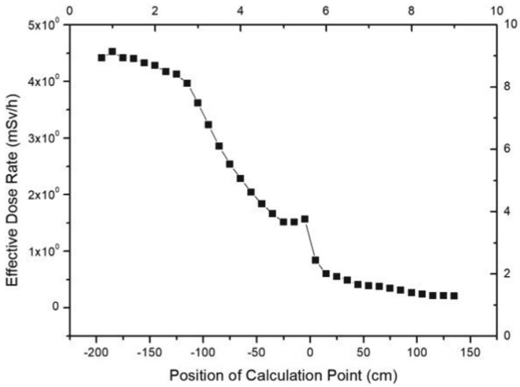

Fig. 12. Effective Dose Rate Variance Tendency of Reactor Pit Contributed by Neutron

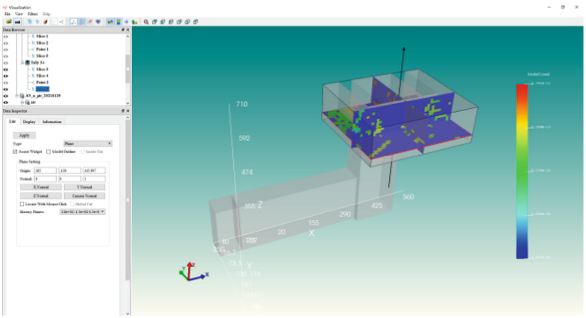

Fig. 13. Neutron Flux Field of Reactor Pit

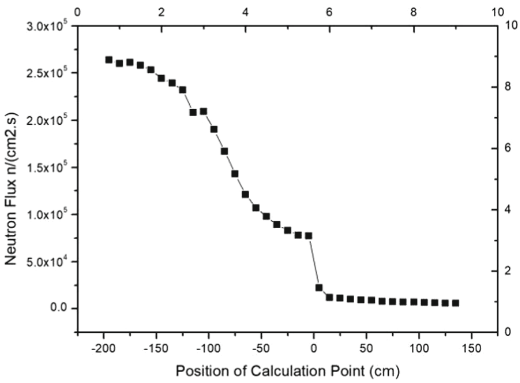

Fig. 14. Neutron Flux Variance Tendency of Reactor Pit

5 Conclusion

By applying the MC-MC coupled method, the neutron shielding problem in large scale can be effectively analyzed. In addition, the superiority of this methodology is proved in the application of reactor pit neutron shielding analysis. Furthermore, by applying SuperMC code, the calculation process of neutron shielding problem in large-scale model can be further optimized by ensuring the calculation accuracy with less time consumed.

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