

Review of Neutron Flux Evaluation Program Based on Monte Carlo Simulation

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1. Introduction

When decommissioning a nuclear power plant, a radionuclide inventory evaluation is essential for estimating the amount of radioactive waste generated in the future. A significant portion of this radioactive waste is generated from contaminated sites and radionuclide-decommissioned structures. For this type of radioactive waste, a neutron activation radionuclide inventory evaluation can be used to estimate the final radioactive waste volume.

To estimate the amount of radioactive waste generated, neutron flux assessment of the target structures must first be performed. Based on this neutron flux value, a radionuclide inventory is calculated. Flux assessments typically utilize programs based on the Monte Carlo methodology. Therefore, In this study we compare and analyze the latest neutron flux evaluation programs based on Monte Carlo simulation.

2. Methodology of Neutron Flux Assessment

Neutron flux assessment programs should calculate the neutron flux for various areas and components of the structure. And the program should conduct a 3d-modeling of the source and the surrounding structures. The neutron transport equation is given by equations (1) and (2).

$$\frac{\delta N}{\delta t} + v\Omega \nabla N + \Sigma v N = \iint \Sigma' + f v N' dE' d\Omega' + Q \quad (1)$$

$$\frac{1}{v} \frac{\delta N}{\delta t} + \Omega \nabla \Phi + \Sigma \Phi = \iint \Sigma' + f \Phi dE' d\Omega' + Q \quad (2)$$

The neutron transport equation can be derived from the neutron balance within the packet. Considering ① the number of neutrons remaining in the packet, ② the number of neutrons entering the packet as a result of collisions, and ③ the number of neutrons coming into the packet from external sources, the results are derived as shown in equation (1).

vN can be expressed as an angular frequency Φ , as shown in equation (2). It can be broadly cassified into two methods: ① probabilistic (Monte Carlo) method

and ② deterministic method. In general, the probabilistic assessment method is mainly used.

3. Neutron Flux Evaluation Program Based on Monte Carlo Simulation

Monte Carlo simulation is a method of finding a solution by repeatedly generating random numbers based on a stochastic model. The general Monte Carlo simulation procedure is as shown in Figure 1.

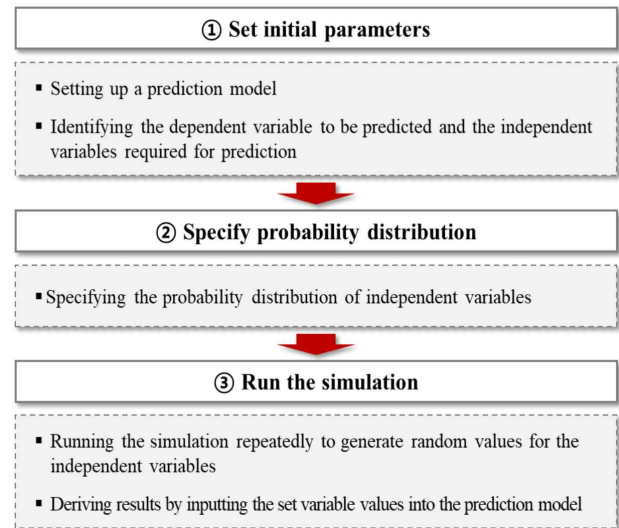


Figure 1. Process of Monte Carlo simulation

3.1 MCNP series

The MCNP (Monte Carlo N-particle) code is the most commonly used Monte Carlo simulation code. The MCNP modeling example and flow diagram of MCNP radiation transport are as shown in Figure 2 and Figure 3.

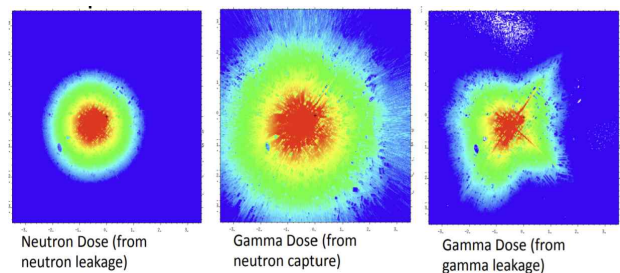


Figure 2. MCNP modeling example

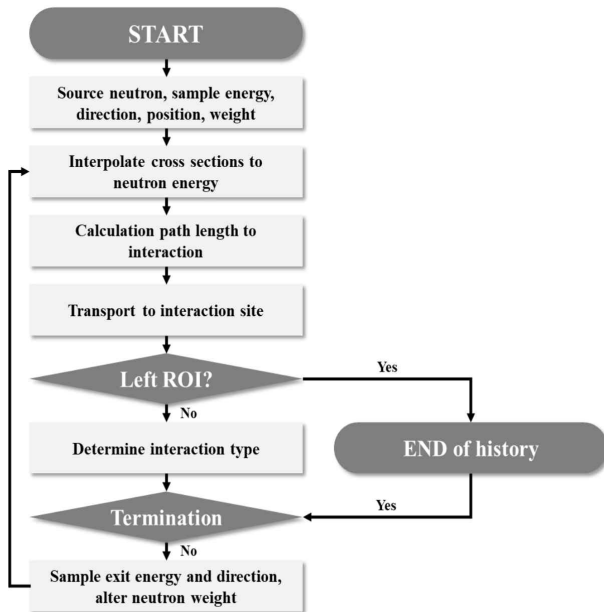


Figure 3. Flow diagram of MCNP radiation transport

3.1.1 MCNP(version 5)[1]

The MCNP(version 5) is a general code for calculating radiation transport for neutrons, photons, and electrons. It can calculate eigenvalues for critical systems. It also provides specific reaction cross sections for neutrons (e.g., ENDF/B) to calculate all reactions.

3.1.2 MCNPX(MCNP eXtended)[2]

The MCNPX (MCNP eXtended) is an extended version of the original MCNP, expanding tally card calculation options such as particle count and continuous energy expansion. It was developed to utilize the LA150 library developed at Los Alamos National Laboratory (LANL). Furthermore, it incorporates a Depletion/Burn module, enabling radioactivity analysis using the burn card option.

3.1.3 MCNP6[3]

The MCNP6 is a unified code that integrates the MCNP5 and MCNPX codes. It also incorporates variance reduction techniques and a state-of-the-art radionuclide library. Furthermore, the k_{eff} eigenvalue calculation function for nuclear fission is provided as a standard feature.

3.2 MCBEND[4]

The MCBEND is a program capable of representing the subtle energy of nuclear data and incorporates a flexible geometric modeling package. It can utilize various nuclear data libraries, including the following:

- ① United Kingdom Nuclear Data Library (UKNDL)
- ② Joint Evaluation Fission and Fusion Library (JEFF)
- ③ International ENDF format reaction cross-section library (ENDF/B)
- ④ Japan Neutron Data Library (JENDL)

3.3 TRIPOLI-4[5]

The TRIPOLI-4 is a program primarily used in radiation transport analysis in France. Its input data is linked to MCNP, allowing MCNP format data to be edited in TRIPOLI-4. It can utilize various nuclear data libraries, including the following:

- ① Joint Evaluation Fission and Fusion Library (JEFF)
- ② International ENDF format reaction cross-section library (ENDF/B)
- ③ Japan Neutron Data Library (JENDL)
- ④ IAEA/NDS Reaction Cross-section Library (FENDL)

4. Conclusion

In this study, we reviewed the ① methodology of neutron flux evaluation and ② neutron flux evaluation program based on monte carlo simulation. As a result, the MCNP computer code was confirmed to be the most widely used, as it can adequately model various structures within nuclear facilities and minimize statistical uncertainty by applying appropriate decomposition methods. Unlike other codes, It can model a variety of curved surfaces and support detailed material card definitions, including multicomponent mixtures, anisotropic materials, and temperature-dependent data. It also facilitates analysis of radiation transport and energy distribution at specific points within complex structures. Furthermore, the MCNP computer code is highly compatible with codes used for assessing activation radionuclide inventory, making it an optimal choice. The results of this study can be useful as basic data for evaluating the activation radionuclide inventory of major structures during future decommissioning of domestic nuclear power plants.

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