The use of the neutronic calculation code CORNER for evaluating the protection of fast neutron reactor and CNFC equipment

M E Shekhanova
National Research Nuclear University MEPhI (31, Kashirskoe shosse, Moscow, 115409, Russia)
Nuclear Safety Institute of the Russia Academy of Sciences (52, B.Tulskaya st., Moscow, 115119, Russia)
E-mail: marie.altavista@gmail.com

Abstract. In this paper we propose a method of using neutronic calculation code CORNER to the analysis of experiments on the protection of fast neutron reactor and CNFC equipment. An example of Winfrith Graphite Benchmark experiment calculation using this approach is presented. This task can be considered as one step in the general theme of the safety analysis of FR with liquid metal coolant, their fuel cycles and related equipment. CORNER implement a solution of the kinetic equation with a source in the three-dimensional hexagonal geometry based on Sn-method. The purpose of this paper is a demonstration of the application of CORNER’s possibilities for the analysis of the actual reactor problems.

1. Approximations

Neutron transport equation in multi-group approximation in some spatial domain \( G \) for the energy group \( g, g = 1, 2, ..., N_g \), has the form:

\[
(\tilde{\Omega} \cdot \nabla \Psi^g (\vec{r}, \tilde{\Omega})) + \sum_{\phi} \Phi^g (\vec{r}, \tilde{\Omega}) = R^g_{\phi} (\vec{r}, \tilde{\Omega})
\]

(1)

where the right-hand-side of the equation looks like:

\[
R^g_{\phi} (\vec{r}, \tilde{\Omega}) = \sum_{h=1}^{N_g} \int_{4\pi} P^{h+g}_{\phi} (\vec{r}, \tilde{\Omega} : \tilde{\Omega}') \Psi^h (\vec{r}, \tilde{\Omega}') d\tilde{\Omega}' + \chi^g_{\phi} \sum_{h=1}^{N_g} \int_{4\pi} \Sigma^h_{\phi} (\vec{r}) \Phi^h (\vec{r}) + S^g (\vec{r})
\]

(2)

\[
\Phi^g (\vec{r}) = \frac{1}{4\pi} \int_{4\pi} \Psi^g (\vec{r}, \tilde{\Omega}) d\tilde{\Omega}
\]

(3)

At the boundary \( \Gamma \) of spatial domain \( G \) values of angular flow set to zero \( \Psi^g (\vec{r}, \tilde{\Omega}) \) for directions \( \tilde{\Omega} \) inside this area:

\[
\Psi^g (\vec{r}, \tilde{\Omega}) \big|_{\Gamma, \tilde{\Omega} \in \partial G} = 0
\]

(4)
The system of equations (1) - (4) describes a stationary distribution of neutrons taking into account the fission process as well as the specified internal sources [1].

Calculation task of $K_{\text{eff}}$ (homogeneous problem) is described by the system of equations (1)–(4) with zero boundary conditions and with zero internal sources $S^g(\vec{r})$ and with a multiplier $1/K_{\text{eff}}$ in front of the second term in the right-hand side of (2).

In solving these problems to calculate the collision integral in the right side of the transfer equation is necessary to set the angular quadrature (Sn-approximation method of discrete ordinates) and Pm-approximation of the scattering function, which corresponds to the solution of the transport equation in SnPm-approximation.

2. Description of the code

Computer code CORNER is based on SN discrete ordinates method and PM approximation of scattering cross section. It allows to solve two types of stationary problems of neutrons and gamma rays transport in the three-dimensional hexagonal (HEX-Z) geometry: the problem on $K_{\text{eff}}$ (homogeneous) and the problem with the source (heterogeneous). To approximate the spatial dependence implemented WDD (Weighted Diamond Difference) and nodal scheme were realized.

CORNER software tool is developed in Fortran and has a modular structure. The main modules are: module of preparation of neutron constants in ANISN format; geometric module; module of preparation of angular quadrature; module of input data containing the parameters of the approximation and the control parameters; module of neutronic calculation and the calculation results processing unit.

Energy dependence is represented by multi-group approximation. Discretization of angular variable is carried out by introducing the angular quadrature. Using of LQn and Pn-Tn quadrature kits also provides the opportunity to quadrature user sets. Use an iterative process solutions, including external iteration on the source of division and inner iterations from the scattering source. The output of the iterative process, both in terms of accuracy, and the number of iterations is provided.

3. Calculation of the test problem

An experimental graphite protection composition irradiated at the installation ASPIS, shown in Fig.1, is considered for test calculation [2].

The initial model is described using a hexagonal geometry. The size of "turn-key", equal to 5 cm, chosen from consideration of the conservation area of the cross-section of individual elements of the protective composition. As a result, cartogram in the plane contains 4624 cells. Nonuniform grid with 26 layers is constructed to analyze the high-altitude partitioning (along the Z axis). Example of partitioning is shown in Fig.2. Energy dependence is represented by 299-group approach, using group constants training program CONSYST with ABBN- RF library. Calculation of the code CORNER carried out in S4 approximation method of discrete ordinates, scattering anisotropy is taken into account in P1 approximation [3]. Estimated time is ~ 50 min. As a result of the calculation the rates of the reactions $\text{S}32(n, p)\text{P}32$, $\text{A}127(n, a)\text{Na}24$, $\text{In}115(n, n ')\text{In}115m$ and $\text{Rh}103(n, n ')\text{Rh}103m$ were obtained, studied and compared with experimental ones.

An example of the comparison of CORNER result with the results of other calculation codes ones is presented in Fig.3.
Figure 1. Scheme of protection composition in the installation ASPIS for the graphite experiment. (1-14 – numbers of zones).

Figure 2. Example of partitioning in the input file "geom".

Figure 3. Comparing of the calculated result with the results of other codes on example of rate of Rh$^{103}$(n, n') reaction.
4. Conclusion
A method of applying the neutronic calculation code CORNER for evaluating experiment with graphite protection was developed. Similar calculations for the protection of fast-neutron reactor and CNFC equipment are planned during further development of this work. For example, the proposed method makes it possible to evaluate the protection of facilities and equipment of process stages of closing the nuclear fuel cycle (processing module, fabrication module, module of radioactive waste management).

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