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MCNP BASED CALCULATION OF REACTIVITY LOSS IN CIRCULATING FUEL REACTORS

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ABSTRACT

The molten salt reactor (MSR) concept seems to be one of the most promising systems for the realization of transmutation. In circulating fuel reactors the reactivity decreases due to the motion of the fuel and the moving of the delayed neutron precursor nuclei. A method was developed and implemented in MCNP4C that can make the code capable of performing criticality calculations on fluid fuel systems with cylindrical geometry. The method is based on the determination of the shift of the precursors from the velocity of the fuel and the sampled lifetime of the nuclide. There is an option to account for the stirring of the fuel outside the core. Calculations were performed for a hypothetical homogeneous cylindrical reactor. The change in reactivity was investigated for cases of different velocities and recirculation times. In order to investigate the effect of the stirring of precursors in the primary loop calculations were performed with the stirring option on and off.

Key Words: molten salt reactor, fluid fuel, delayed neutron, criticality calculation, reactivity loss

1. INTRODUCTION

The molten salt reactor (MSR) concept seems to be one of the most promising systems for the realization of transmutation and one of the proposed Generation IV reactor types [1]. In the molten salt reactors (or subcritical systems) the fuel circulate dissolved in some molten salt. The main advantage of this reactor type is the possibility for the continuous feed and on-line reprocessing of the fuel.

Due to the motion of the fuel, in a fluid fuel reactor the spatial distribution of the delayed neutrons differs from that of the prompt neutrons as the precursor nuclei change their position between fission and neutron emission and they may decay even outside the core. Therefore, the reactivity and effective delayed neutron fraction of these reactors decrease. The extent of the reactivity loss is primarilly influenced by the velocity field of the fuel and the time of fuel recirculation. Other effects, like the different “stirring” equipments, such as pumps, which cause

that the radial distribution of the delayed neutron precursor nuclei returning to the reactor become uniform, can be important as well.

Since all of the presently operating power reactors utilize solid fuel, little attention was paid to the calculation of the reactivity loss caused by the circulation of the fuel. In the 1960s measurements were performed on the Molten-Salt Reactor Experiment (MSRE) at the Oak Ridge National Laboratory and a theory was developed to determine the above mentioned reactivity loss [2]. When the interest toward such type of reactors started to grow a few years ago, research in this field gained a great impetus again. Efforts have been made to determine the reactivity loss by deterministic methods, like the point kinetics theory and the one dimensional diffusion model presented in [3-5]. Though the deterministic theories are capable of describing the phenomena in full detail, they usually contain simplifications.

The Monte Carlo method based calculational approach can be very fruitful due to its unlimited capabilities of modelling 3D geometries and simulate nuclear processes in full detail. Obviously, this is applicable only for steady-state situations. Since Monte Carlo codes are widely used in neutronics calculations nowadays, it is also very important to develop a method which makes these codes suitable for performing criticality calculations on circulating fuel systems without any approximation. This paper present such a method and its implementation to the MCNP4C[6] code. Furthermore, results obtained for an hypothetical reactor are analyzed.

2. THE METHOD

The method developed is based on the determination of the shift of the precursors from the velocity of the fuel and the sampled lifetime of the nuclide. The delayed neutron is born in this new position.

The above method is realized in MCNP4C. Though this is a highly flexible code and can handle the delayed neutrons in all detail, it cannot account for the motion of delayed neutron emitters. Therefore, modifications were introduced in to the code. With the aid of the modified version the multiplication factor can be determined at different fuel velocities and different circulation times, moreover there is an option to take into account the stirring of the fuel outside the core.

Criticality calculations in MCNP are carried out by source iteration. The predetermined number of neutrons are started in each cycle of the iteration. Fission is treated as absorption but the energy of produced fission neutrons are sampled. Data of these neutrons are stored and serve as source neutron data in the next cycle of the iteration. A few of the first cycles are used to settle the fission source. After that, the data collected during the remaining cycles are used to produce the multiplication eigenvalue and fluxes.

Two subroutines of the MCNP source code were modified: COLIDK (which generates the fission neutrons for the next cycle of the simulation) and SOURCK (which starts the required number of fission neutrons for the next cycle).

When MCNP creates a new fission neutron during the simulation process (in subroutine COLIDK) it decides on whether it is delayed or not. Then it samples an energy for the neutron and if the neutron is delayed, a lifetime for the precursor is sampled. The place of birth of the neutron is defined by the place of the fission event. This is where intervention is made to the usual simulation flow of MCNP and the own developed subroutine CRFUEL is called in order to determine the new position of the precursor at the time of the neutron emission.

The algorithm of the subroutine CRFUEL is described in the following. First, the time which is necessary for the precursor nucleus to exit the reactor is calculated. If this time is larger than the lifetime of the nucleus, the starting position of the newly born neutron is shifted with the distance it travels during this period. If this time is less than the lifetime, it means that the nucleus exits to the primary loop. In this case the time spent in the core is subtracted from the lifetime and it is checked whether the remaining lifetime is shorter than the time T_e to be spent in the loop. If so, i.e. the neutron is produced outside the core, the neutron is killed (weight is set to zero). If, however, it is larger than time T_e, the precursor can reach the core. In this case, provided that the user had turned on this option, the radial position of the nucleus is sampled again from uniform distribution in a plain to simulate the stirring in the primary circuit outside the reactor. Then there is a return to the beginning of the algorithm. The procedure is repeated until decision is made whether the neutron is killed or not. The procedure described above actually follows the whole life of the precursor nucleus. After subroutine CRFUEL modifies the starting position of the delayed neutron, MCNP continues on with the normal procedure.

In the present modifications the following assumptions were made:

- The velocity of the fuel is parallel to the z axis.
- The velocity (v) is constant in the core.
- The fuel exiting through the upper boundary plain of the system, which is perpendicular to the z axis, enters again, after traveling time T_e, through the lower boundary plain, which is also perpendicular to the z axis.
- The shape of the active zone is a cylinder (which constitutes a single cell in the MCNP geometry).

Obviously it is also assumed that the system is in a steady-state condition.

3. CALCULATIONS

Calculations were performed for an hypothetical homogeneous cylindrical reactor, in which 100% enriched $^{235}UF_3$ is dissolved in 32 mol% BeF_2 + 68 mol% ^{7}LiF molten salt with a concentration of 4.8 mol%. The diameter and height of the reactor are 80 cm and 100 cm, respectively.

The change in reactivity was investigated for cases of different velocities and recirculation times. The velocity was in the interval between 0 and 100 cm/s, while the recirculation time was changed between 2 s and infinity. For the case of T_e = 5 s the reactivity loss was determined with and without stirring.

In order to theoretically estimate what results can be expected, a simple formula was derived in a one dimensional model. During the operation of the reactor, a precursor nucleus born at height z of the core moves parallel to the axis of the core. The probability that it exits from the core is:

$$P(z) = e^{(z-H)\frac{\lambda}{v}} \tag{1}$$

where λ is the the decay constant of the precursor, v is the fuel velocity in the core, and H is the height of the core. Assuming that the spacial distribution of the precursors is uniform along the z axis (i.e. there is no change in the flux) we can easily obtain the following formula by integrating (1) over the entire core:

$$\Delta\rho = 1 - \frac{1}{\beta}\sum_{i=1}^{6}\beta_i\left[1 - \frac{v}{\lambda_i H}\left(1 - e^{-\lambda_i \frac{H}{v}}\right)\right], \tag{2}$$

where $\Delta\rho$ is the reactivity loss in \$, β is the delayed neutron fraction, β_i are the delayed neutron fractions, and λ_i are the decay constants of the individual delayed neutron groups.

4. RESULTS

The delayed neutron fraction was determined as the difference between the multiplication factors calculated with and without the assumption of delayed neutrons. In the case of the above described hypothetical reactor it was found to be 666 pcm.

First the results obtained from equation (2) was compared with the MCNP calculations. Since recirculation was not assumed in the theory, a very long, practically infinite recirculation time was chosen for the MC simulation. The delayed neutron data obtained from the ENDF/B-VI data library used for the calculations can be found in Table I. As it can be seen in Fig. 1, the trends of the results are similar but the calculated reactivity loss is higher, especially at higher velocities. This fact agrees to the expectations as the simple theory considers only the neutron loss due to ex-core emissions assuming a flat axial flux distribution while the reduction in the delayed neutron worth due to the change of the position is neglected. However, the results show that the latter effect is not negligible.

Table I. Delayed neutron parameters

Group	λ_i (1/s)	$T_{1/2}$ (s)	β_i/β
1	0.0133	51.98	3.5%
2	0.0327	21.17	18.1%
3	0.121	5.74	17.3%
4	0.303	2.29	38.7%
5	0.849	0.82	15.9%
6	2.85	0.24	6.6%

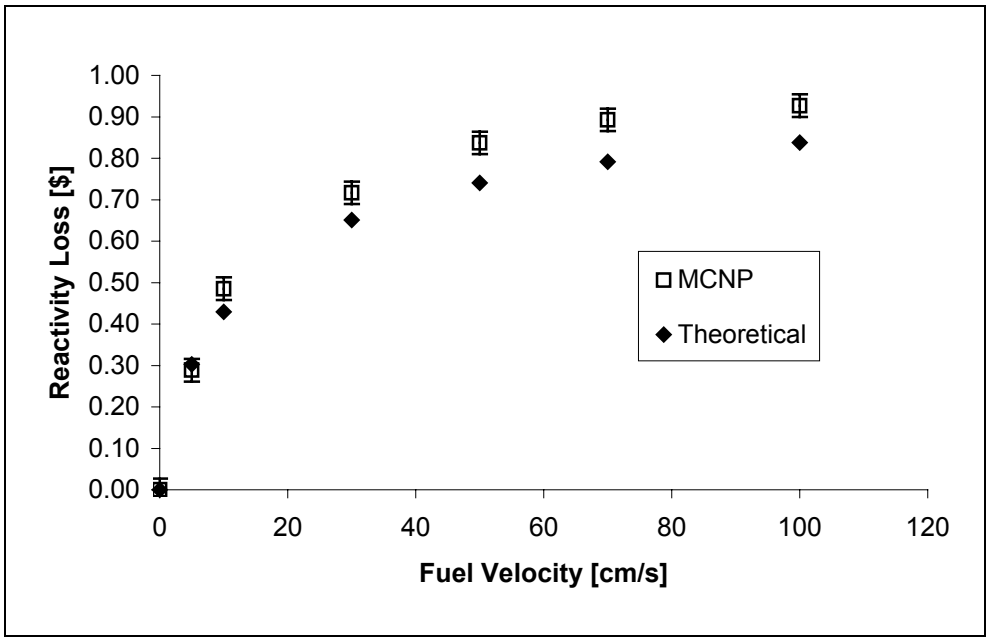

Figure 1. Calculated reactivity loss with recirculation time T_c = 100000 s

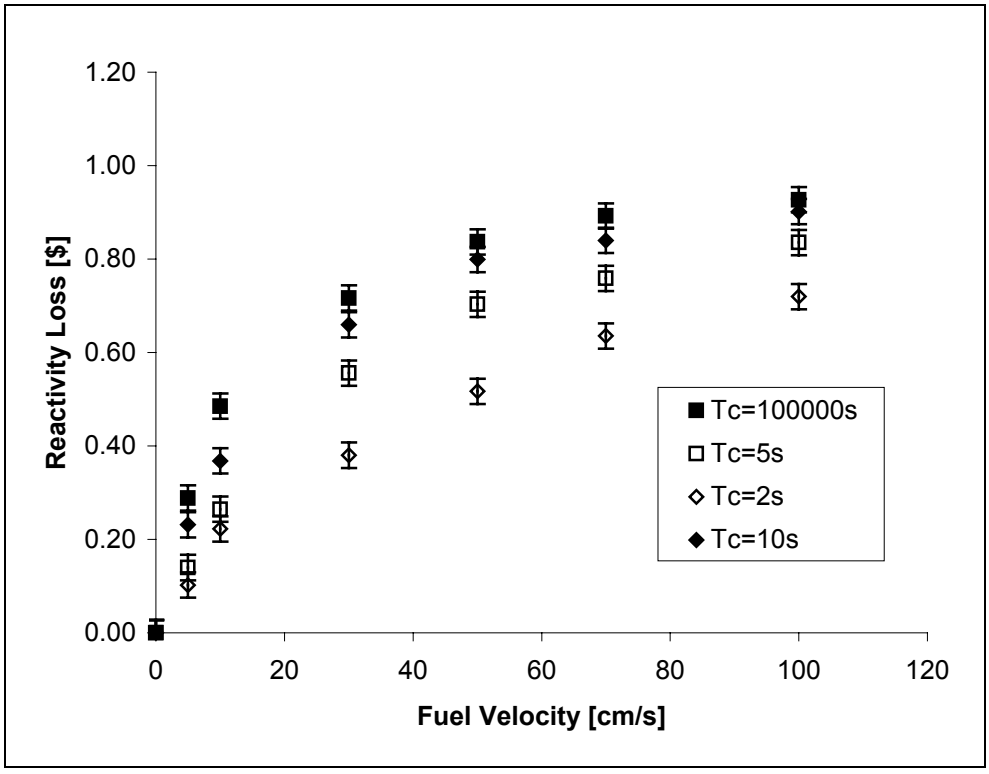

Figure 2. Reactivity loss as a function of fuel velocity and recirculation time

Next the effect of the recirculation time was investigated. From Fig. 2. one can conclude that if the recirculation time is 10 s or longer in the case of a reactor with this dimensions, only very few precursors return to the core and the reactivity loss is almost the same as without recirculation. Below 10 s the recirculation time can basically influence the effective delayed neutron fraction.

In order to investigate the effect of the stirring of precursors in the primary loop, calculations for recirculation time T_c=5 s were performed with the stirring option on and off. The curves obtained can be seen in Fig. 3. A further reactivity reduction can be clearly observed due to the stirring but the change is so low (about 5 ¢) that neglecting this effect seems an acceptable approximation.

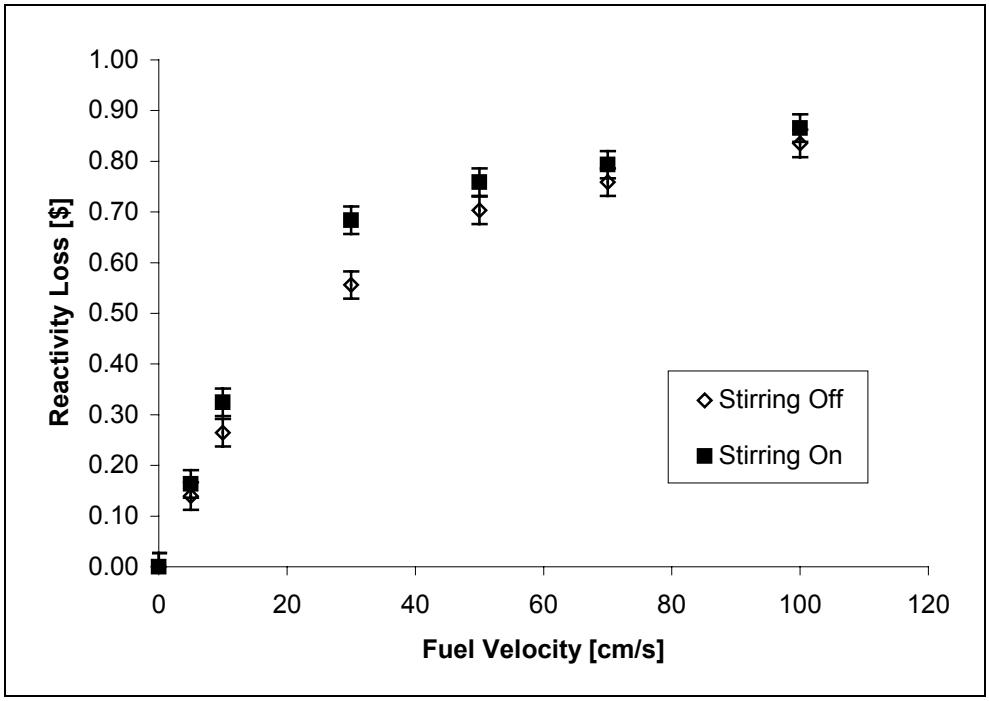

Figure 3. Reactivity loss with and without stirring the fuel at T_c=5 s

5. CONCLUSION

A method was developed and implemented in MCNP4C that can make MC codes capable to perform criticality calculations on fluid fuel systems with cylindrical geometry. Reactivity loss can be calculated as a function of the fuel velocity and the recirculation time. There is an option to take into account the stirring of the fuel outside the core.

Results obtained for an hypothetical molten salt reactor were analyzed. The comparison with theoretical results shows that the reduction in the delayed neutron worth due to the change of the position is significant. However, the additional reactivity loss due to the stirring of the fuel seems negligible.

Further modifications are planned to extend the code to be capable of performing calculations on more complicated geometries, e.g. lattices. The direct calculationof the effective delayed neutron fraction by starting only delayed neutrons from a fission source settled during a normal criticality calculation can make it possible to determine the reactivity loss in fluid fuel systems even more precisely, however this method requires basic modifications to the conventional MCNP criticality program flow.
In order to verify the program, calculations will be performed for the MSRE, in which case direct comparison with measurement results will be possible. Due to the inhomogenous core of this reactor the method requires further development.

6. REFERENCES

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