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Development of Monte Carlo-based pebble bed reactor fuel management code

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ABSTRACT

A fuel management code for pebble bed reactors (PBRs) based on the Monte Carlo method has been developed in this study. The code, named Monte Carlo burnup analysis code for PBR (MCPBR), enables a simulation of the Once-Through-Then-Out (OTTO) cycle of a PBR from the running-in phase to the equilibrium condition. In MCPBR, a burnup calculation based on a continuous-energy Monte Carlo code, MVP-BURN, is coupled with an additional utility code to be able to simulate the OTTO cycle of PBR. MCPBR has several advantages in modeling PBRs, namely its Monte Carlo neutron transport modeling, its capability of explicitly modeling the double heterogeneity of the PBR core, and its ability to model different axial fuel speeds in the PBR core. Analysis at the equilibrium condition of the simplified PBR was used as the validation test of MCPBR. The calculation results of the code were compared with the results of diffusion-based fuel management PBR codes, namely the VSOP and PEBBED codes. Using JENDL-4.0 nuclide library, MCPBR gave a 4.15% and 3.32% lower k_{eff} value compared to VSOP and PEBBED, respectively. While using JENDL-3.3, MCPBR gave a 2.22% and 3.11% higher k_{eff} value compared to VSOP and PEBBED, respectively. The ability of MCPBR to analyze neutron transport in the top void of the PBR core and its effects was also confirmed.

1. Introduction

A pebble bed reactor (PBR) with a moving fuel core provides flexibility in the fuel management of the core, including the advantage of online refueling. On the other hand, depletion analysis of this moving fuel core is especially challenging because the analysis must account for the movements of the fuel as well as the changes of nuclide composition.

Historically, the multi group neutron diffusion method has been used for the neutronic analysis of PBR fuel management codes. VSOP (Rutten et al., 2005) is a standard code used in PBR analysis which applies multi group neutron diffusion. The recently developed PEBBED (Terry et al., 2002) code for PBR analysis is also based on this diffusion approximation. Yet features of the PBR, such as the presence of the void at the top of the core and the heterogeneous nature of the core, demand the application of transport theory modeling (Tyobeka et al., 2007).

With recent advances in computer performance, Monte Carlo (MC)-based neutronic analysis has been coupled with depletion codes. MC-based analysis gives more accurate neutronic analysis, as well as the flexibility to model complex geometry in PBR. Several MCNP-based codes for PBR depletion analysis have been compared (Bomboni et al., 2010) although the report was limited to single HTR pebbles. A PBR depletion analysis code based on MVP-BURN (Okumura et al., 2006) code has also been developed. This depletion code was used to analyze OTTO-cycle PBR (Tran, 2012), but detailed validation of the depletion code and comparison to other depletion codes has not yet been performed.

The purpose of this study is to develop an MC-based depletion code for PBR fuel management analysis and to perform validation by performing comparison of the code with other codes. In this development, MVP-BURN code was used for neutronic analysis and depletion calculations. An additional utility code was developed to simulate the movement of the core.

As PBR fuel management, MCPBR can be used as a tool for design analysis of any sophisticated OTTO PBR design, fuel pebble recycling analysis, or other analysis. However, current study and discussion is focus on the development of the code. Details of the development of the code will be explained in Section 2, including the neutronic and depletion method used in MVP-BURN code. Calculation of discharge burnup at the equilibrium condition of OTTO cycle PBR will be used as a validation case. The results of our code will be compared with results from the VSOP and PEBBED code.

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Detailed validation conditions will be described in Section 3, including brief explanation of the method used in the VSOP and PEBBED code. Results of the validation case and its discussion will be given also in Section 3. Analysis of PBR core with top void using MCPBR and its results will be given in Section 4. The conclusion of this study will be given in Section 5.

2. Code development

2.1. PBR fuel management concept

Development of a fuel management code should include the ability to perform neutron transport analysis predicting the neutron spectrum in the core, also the ability to perform depletion analysis predicting the nuclide change of the fuel material during operation. In addition, for a PBR reactor we need also to accommodate the movement of the fuel in the core during operation.

The depletion equation in PBR after including the movement of the fuel as proposed by Massimo (Massimo, 1976) is as follows:

$$\frac{\partial N_k}{\partial t}+\frac{\partial N_k}{\partial z}\upsilon=\phi\sum_{i=1}^{m}N_i\sigma_{fi}y_{ik}+\phi\sum_{s=r}^{q}N_s\sigma_{as}\gamma_{sk}+\sum_{j=n}^{p}N_j\lambda_j\alpha_{jk}-\lambda_kN_k-\phi N_j\alpha jk \quad (1)$$

where

N_K = atomic concentration of isotope k; υ = axial ball velocity; ϕ = flux of the core region; σ_{fi} = fission cross section of isotope i; σ_{as} = absorption cross section of isotope i; λ_j = decay constant of isotope i; y_{ik} = yield of isotope k due to fission in isotope i; γ_{sk} = probability that neutron absorption in isotope s produces isotope k; α_{jk} = probability that decay of isotope j produce isotope k.

In the MCPBR, PBR fuel management is solved by simulating the physical process in a time-dependent manner. In this method, the core is analyzed from the initial core condition by performing neutron transport and depletion calculations for a specific period of time based on the axial speed of the fuel; then as fresh fuel is inserted in the top part of the core and the fuels in the bottom are discharged, a neutron transport and depletion calculation is again performed for the updated core condition. Eventually a steady state or equilibrium state of the core can be achieved. In this equilibrium state the neutron flux and nuclide fuel composition do not change. The advantage of using the time-dependent method, as in this study, is the ability to simulate the fuel management of a PBR for its whole lifetime, starting with the initial core, then the running-in phase, and finally the equilibrium condition.

Other methods directly calculate the equilibrium condition to avoid the long calculation times required to analyze the non-equilibrium phases. But these methods are unable to perform certain important fuel management analyses of PBR; for instance, the startup analysis of the core. This direct equilibrium method is applied in the PEBBED and PREC (Sekimoto et al., 1987) codes.

2.2. Computer code implementation

In this development, the Monte Carlo based code MVP-BURN was used to perform neutron transport and depletion calculation. The MVP-BURN code enables the burnup calculations by coupling a continuous-energy Monte Carlo code MVP (Nagaya et al., 2005) and an BURN code which performs the depletion calculation. A better simulation of the PBR core using MVP-BURN can be given by the statistical geometry method applied in the code which enables explicit simulation of the double heterogeneity of the PBR core. The additional auxiliary code was developed to enable simulation of the fuel movement in the PBR core. Currently, the code is limited to OTTO (Once-Through-Then-Out) cycle PBR. By this method, we are not solving the special depleted equation given in Eq. (1), but solving Eq. (2), the standard depletion equation, using MVP-BURN.

$$\frac{\partial N_i^z(t)}{\partial t}=\sum_{j\neq i}f_{j\to i}\lambda_jN_j^z+\sum_{k\neq i}Fact(t)\{g_{k\to i}C_k^z+\gamma_{k\to i}F_k^z+h_{k\to i}W_k^z\}N_j^z(t)-[\gamma_i+Fact(t)\{A_i^z+W_i^z(t)\}]N_i^z(t) \quad (2)$$

where,

i, j, k = Depleting nuclide number; N = atomic concentration of isotope k; Z = Burnup region number; λ, f = decay constant and branching ratio; g, γ, h = Yield fraction of each transmutation; F = Relative microscopic fission reaction rate calculated with MVP at time $t = t_n$; A = Relative microscopic absorption reaction rate calculated with MVP at time $t = t_n$; C = Relative microscopic capture reaction rate (=A−F) calculated with MVP at time $t = t_n$; W = Relative microscopic (n, $2n$) reaction rate calculated with MVP at time $t = t_n$; Fact(t) = Normalization factor to convert relative reaction rates to absolute ones.

MVP-BURN solves the above equation using Bateman's method with a modification for more accurate treatment of the cyclic chain caused by α-decay and so on. The reliability of the burnup calculation by MVP-BURN has been confirmed. The burnup performance of MVP-BURN code has been validated by applying the code to a high-conversion LWR lattice and a BWR lattice with burnable poison rods (Okumura et al., 2000). Several nuclide chains can be used in the MVP-BURN code, including the heavy nuclide chain which accommodates thorium.

In MCPBR, the active core is divided into some axial and radial region as shown in Fig. 1. Each region has the same neutron fluxes and nuclide compositions. For simplification, the bottom cone of the PBR core is not included in the analysis. This simplification is generally not a problem for fuel management analysis because the neutron fluxes at that region are very small. Based on the MVP-BURN code applied in MCPBR, statistical geometry model,

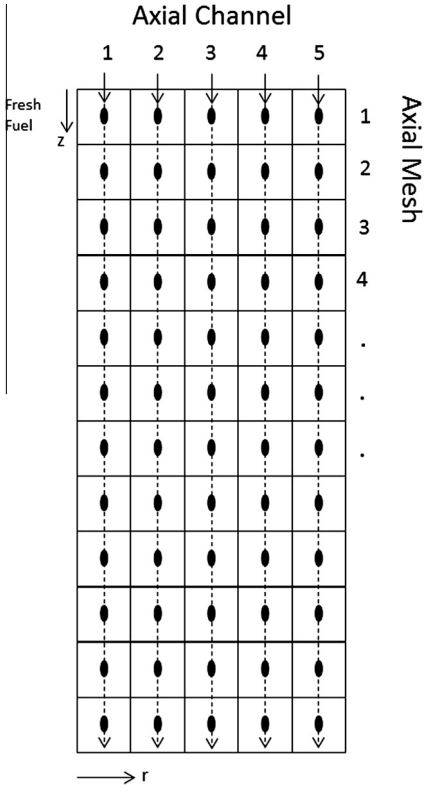

Fig. 1. Core model of the PBR in MCPBR.