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Development of a Monte Carlo based PBR Fuel Management Code

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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 becomes a challenge because the movements of the fuel need to be accounted for in the analysis together with the change of the nuclide composition[1].

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[2].

As computer performance continues to improve, Monte Carlo (MC) based neutronic analysis has recently become commonly used in combination with some depletion codes. MC-based analysis provides more accurate neutronic analysis and is also flexible to model the complex geometry in PBR. For PBR depletion analysis, several MC-based depletion codes have been used and compared[3]. However, the MC-based depletion code comparison was limited to a single pebble fuel. A PBR depletion analysis code was also developed based on the MVP-BURN[4] code. Tran[5] performed PBR analysis of the Once-Through-Then-Out (OTTO) cycle using MVP-BURN based code. However, detailed validation of the depletion code and comparison with other depletion codes has not yet been performed or reported.

The development of a fuel management code should include the ability to perform neutron transport analysis to predict the neutron spectrum in the core, as well as the ability to perform depletion analysis to predict the nuclide change of the fuel material during the operation. In addition, for a PBR reactor we also need to accommodate the movement of the fuel in the core during the operation. The purpose of this study is to develop an MC-based depletion code for PBR fuel management analysis and to validate the code and compare it with other codes.

CODE DEVELOPMENT

PBR Fuel Management Concept

The MC-based code MVP-BURN was used in this development to perform the neutron transport and depletion calculation. The MVP-BURN code enables the burn up calculations by coupling the continuous-energy Monte Carlo code MVP[6] with the auxiliary code BURN which performs the depletion calculation. Statistical geometry model in MVP provide a simple and accurate model of double heterogeneity of PBR. Another module was developed so as to be able to simulate the fuel movement in the PBR core. Currently the code is limited to the OTTO cycle PBR. By this method, the code solved the standard depletion equation using the BURN code of MVP-BURN and then handled the movement of the core with the additional module. The developed code was named Monte Carlo Pebble Bed Reactor Analysis Code (MCPBR).

Computer Code Implementation

A flow chart of the computer code developed for the OTTO cycle fuel management analysis is shown in Fig. 1.

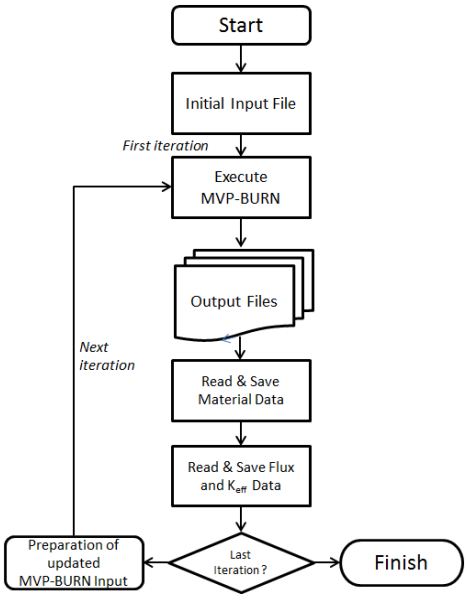

Fig. 1. Flow chart of MCPBR code.

CODE VALIDATION TEST

To validate the performance of the developed code, the results of equilibrium discharge burnup for a simplified PBR with the OTTO cycle from the MCPBR code were compared with the results of VSOP[7] and PEBBLE[8] code as reported by Gougar et.al[9]. The

simplified PBR only consisted of an active core and a 1 m reflector surrounding the active core as shown in Fig. 2. Top gas plenum (void), loading and discharge codes, the graphite inner reflector, and control elements were omitted for simplicity. The active core is 10 m high and 3 m in diameter. The core was divided into 5 axial flow channels of approximately equal cross-sectional flow area. Detailed conditions of the validation test are given in Gougar et.al[9].

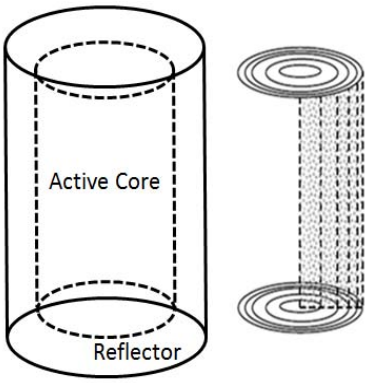

Fig. 2. Simplified PBR model. Active core and surrounding reflector (left), axial flow channel in active core (right).

RESULTS

Results for the equilibrium discharge nuclide of the simplified PBR are given in Table I. The results for MCPBR in Table I. include modeling with two-different axial mesh numbers, as well as different nuclide averaging methods for 10 axial mesh models. For the PEBBED results reported in Table I., the few group neutron fluxes were computed with the PEBBED diffusion solver. So in the above comparison, each of the three codes performed the transport calculation and burn up calculation using its own method.

In MCPBR, the single-step method includes only a single transport calculation for each axial region and the burn up calculation is performed using the neutron fluxes resulting from this transport calculation. In the double-step method, in each axial region, the transport calculation is initially executed then the burnup calculation is performed using the resulting neutron fluxes, the second transport calculation is executed again followed by another burn up calculation using its neutron flux output. The periods of each burn up in double method are half of the burn up period in single method. The residence time of the nuclide in the core is determined by the axial fuel velocity of the PBR. The double-step method shows improved accuracy compared to the single-step method using the same axial mesh, but if computation time is also considered; better accuracy can be achieved by doubling the axial mesh.

It was also investigated whether the linear-averaging method can improve the accuracy using the same axial mesh number. In the linear-averaging method the nuclide density used for the next fuel shuffling in each axial region is a linear average of the nuclide density resulting from burn up calculation in its own axial region and in the upper axial region. The computation time of the linear averaging method is the same as with the single-step method, this averaging method did not seem to improve on the results of the single-step method.

The results obtained using 20 axial meshes might represent the most accurate results from MCPBR. A higher axial mesh number give a better results but it will increase the computation time significantly. For 20 axial meshes, the equilibrium discharge nuclide density differences from those with VSOP were 3.7% for U-235, 0.9% for U-238, 15% for Pu-239, and 0.05% for PU-241. For strong absorbing fission product such as Xe-235 and Gd-155 the difference can be large.

TABLE I. Results of equilibrium discharge nuclide density

Nuclides	Initial Nuclide (Fresh Fuel)	Equilibrium Discharge Nuclide Density [atom per b.cm]					
		VSOP	PEBBED	MCPBR			
				10 Axial Mesh			20 Axial Mesh (Single-Step)
				Single-Step	Double-Step	Lin.Avg	
U-235	2.254E-03	4.170E-04	3.930E-04	3.633E-04	4.868E-04	6.750E-04	4.325E-04
U-238	2.094E-02	1.960E-02	1.960E-02	1.971E-02	1.981E-02	2.000E-02	1.978E-02
Pu-239	0	1.320E-04	1.330E-04	1.393E-04	1.610E-04	1.570E-04	1.524E-04
Pu-241	0	5.680E-05	5.700E-05	5.204E-05	6.042E-05	4.500E-05	5.677E-05
Xe-135	0	4.040E-11	2.630E-11	9.856E-11	4.125E-09	9.050E-10	8.499E-10
Gd-155	0	3.840E-07	4.840E-07	2.149E-07	5.728E-08	1.040E-07	1.298E-07

Comparison of the radial variation of the U-235 equilibrium discharge nuclide density between VSOP, PEBBED and MCPBR code can be seen in Table II. In general all codes give the same radial distribution. Table II. shows that difference between the results from MCPBR code compare to the results from VSOP and PEBBED are getting bigger near the reflector. The closer to the reflector the effect neutron transport method used in the code become more significant. MC based neutron transport, theoritically, should give a better results.

TABLE II. Radial variation of equilibrium discharge nuclide density of U-235

Code	Axial Channel				
	1	2	3	4	5
VSOP	3.909E-04	4.306E-04	4.627E-04	4.325E-04	3.418E-04
PEBBED	3.909E-04	5.326E-04	4.608E-04	4.306E-04	3.418E-04
MCPBR	3.949E-04	4.343E-04	4.758E-04	4.609E-04	3.717E-04

The differences in the results among VSOP, PEBBED, and MCPBR might be due to the method of the neutron transport calculation. The MC method applied in the MCPBR code should be more accurate in modeling the PBR because the physical features of PBR demand a more detailed transport method. The method used to model the core nuclide composition also might cause the differences. The MCPBR code models the nuclide composition exactly by accommodating the detailed geometry of the pebble ball and coated fuel particle layer, while the VSOP and PEBBED codes use a homogeneous nuclide composition.

CONCLUSION

A time-dependent fuel management analysis code for OTTO cycle PBR, called MCPBR, has already been developed. Code-to-code validation has already been performed by comparing the prediction of the equilibrium discharge nuclide density with that of VSOP and PEBBED codes. MCPBR applies the MC-based burnup calculation code MVP-BURN and an additional utility code to model the OTTO cycle movement of the PBR. Exact modeling of the material composition is applied. Those more accurate methods provide a better simulation of the PBR fuel management. A comparison of the equilibrium discharge nuclide density with the VSOP and PEBBED codes shows good agreement, in particular for the important heavy nuclides.

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