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Migration of Fission Products in a Fuel Element of a Prismatic High Temperature Gas-cooled Reactor

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1. Introduction

A prismatic high temperature gas-cooled reactor (HTGR) contains lots of prismatic graphite blocks which are equivalent to the fuel assemblies in a pressurized water reactor. There are many compacts and coolant holes in a block. Fission products are generated in the compacts of the blocks and then they migrate through the graphite and are released into a coolant. It is essential for the safe design and operation of a prismatic HTGR to quantitatively evaluate a fission product release into a coolant. This study set up a numerical method to estimate the migration of a fission product in a block of a prismatic HTGR.

2. Modeling for Fission Product Transport

A symmetrical element in a block of a prismatic HTGR looks like Fig. 1. It consists of a fuel compact, a gap between the compact and graphite, graphite, and a coolant hole. The fuel compact is a graphite cylinder containing many coated fuel particles. Fission products are generated in the coated fuel particles in a compact. They transport through the compact, the gap and the graphite, and finally they are released into the helium coolant. For a simple calculation, the symmetrical element is approximated by a one-dimensional slab as shown in Fig. 2 [1,2].

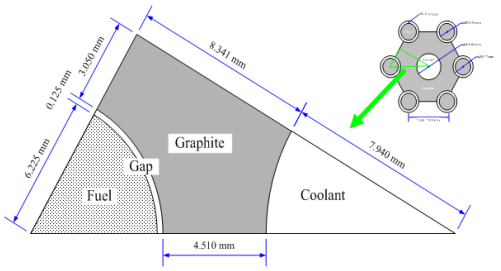

Fig. 1. A symmetrical element of a prismatic reactor core.

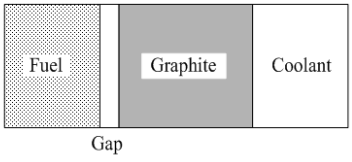

Fig. 2. One-dimensional slab model for a symmetrical element of a fuel block in a prismatic reactor core.

The fission product transport within the compact and the graphite can be described by the following Fickian diffusion equation [1,2].

$$\frac{\partial C(r,t)}{\partial t} = \dot{S}(r,t) - \lambda C(r,t) + \frac{1}{r^z}\frac{\partial}{\partial r}\left[r^z D(r,t)\frac{\partial C(r,t)}{\partial r}\right], \quad (1)$$

where C = concentration (μmol/cm^3), $\dot{S}$ = source term (μmol s^{-1} cm^{-3}), D = diffusion coefficient (cm^2/s) = D_0 exp(-Q/(RT)), D_0 = pre-exponent factor of a diffusion coefficient (cm^2/s), Q = activation energy (J/mol), R = gas constant (8.314 J mol^{-1} K^{-1}), T = temperature (K), λ = decay constant (s^{-1}), r = distance from compact center (cm), t = time (s), z = 1 in the fuel compact and 0 in the other region of Fig. 2, respectively. Source term is the volumetric generation rate of the fission products from the coated particles and heavy metal contamination.

The initial concentration is zero over the entire region. The current at the center of the compact is zero. It can be assumed that the so-called sorption isotherms are formed between the gap pressure and the concentrations at the compact surface and the inner surface of the graphite [1]. The fission products evaporate on the outer surface of the graphite. The concentration on the outer surface of the graphite is in equilibrium with the vapor pressure on the graphite side of the boundary layer which forms between the graphite surface and the coolant. The boundary conditions are given by the mass transfer through the layer and the Freundlich sorption isotherms between the concentrations at the graphite surface and the vapor pressures at the graphite side of the boundary layer [2].

A finite difference method was used to solve Eq. (1) and the related initial and boundary conditions. The fuel and graphite regions of the one-dimensional slab were divided into N_f and N_g intervals, respectively. It can be assumed that the diffusion coefficient and source term are only dependent on the time within each interval. Then the resulting system of equations becomes nonlinear because of the sorption isotherms in the gap and the boundary conditions at the outer surface of the graphite. An iterative solution scheme was applied to the system of equations.

3. Estimation of a Fission Product Release from Compact and Graphite

Migration of an isotope Cs^{137} for a fuel block was estimated. The irradiation time was assumed to be 850

days. The corresponding maximum fluence was assumed to be 3×10^{21} n/cm^2. The coolant temperature was 900 °C over the entire irradiation period in a normal operation case, and 1200 °C between 400 and 450 days and 900 °C at other times in an accident case. The accident time interval was equivalent to 1.4×10^{21} to 1.6×10^{21} n/cm^2 for the fluence interval. The defect fraction of the coated particles was assumed to be 1×10^{-4} over the entire time. The heavy metal contamination fractions were assumed to be zero. The diffusivity, sorption isotherm parameters, and mass transfer coefficient in helium for Cs^{137} were extracted from references [2,3,4]. The densities and sizes of the fuel materials for the test calculations are presented in Table I.

The data relating to a prismatic reactor for the test calculations is given by Table II. The dimensions of the symmetrical part of a fuel block are given in Fig. 1. Both the compact and graphite slab of the symmetrical element were divided into 21 finite difference intervals. The kernel, buffer, IPyC, SiC and OPyC of a coated particle were 10, 4, 4, 4, and 4 finite difference intervals, respectively. It took about 37 seconds to calculate the migration and release of Cs^{137} during the entire irradiation period, 850 days, with a dual core Pentium personal computer. Fig. 3 shows the concentration distributions of Cs^{137} according to the time and coolant temperature within a symmetrical element in a fuel block. The concentration increases with the time and the inventory decreases with the temperature. Fig. 4 shows the fractional releases of Cs^{137}. The release increases abruptly at an accident interval.

Table I: Thicknesses and densities of the fuel materials

Materials	Thickness (μm)	Density (g/cm^3)
Compact	12.45×10^{3} 1)	0.8 ~ 1.2
OPyC	40	1.90
SiC	35	3.20
IPyC	40	1.90
Buffer	95	1.02
Kernel	500 1)	10.83

1) diameter

Table II: An example of a prismatic reactor

Design parameters	Values
Active core height (m)	7.93
Effective inner diameter of active core (m)	2.96
Effective outer diameter of active core (m)	4.83
Fuel material	UO_2
Enrichment (%)	19.8
Thermal power (MW)	600
Power density (W/cm^3)	6.6
Inlet/outlet temperature (°C)	590/950
Coolant mass velocity (kg/s)	320
Primary coolant pressure (MPa)	7.0
Number of fuel columns	102
Number of fuel blocks per column	10
Number of coolant holes per block	108
Number of compacts per block	3126
Number of coated particles per compact	4830
Compact diameter (mm)	12.45
Compact length (mm)	49.3

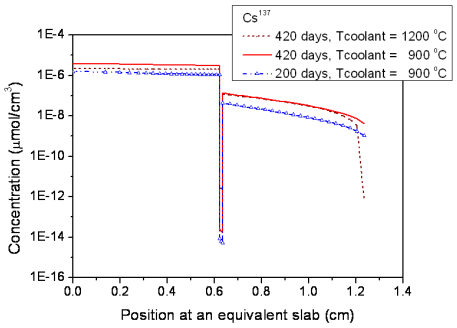

Fig. 3. Concentration distribution of Cs^{137} within a symmetrical element in a fuel block of a prismatic reactor.

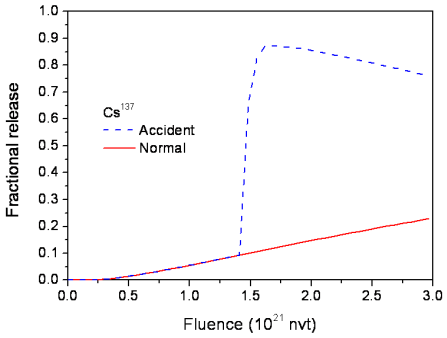

Fig. 4. Fractional release of Cs^{137} from a prismatic reactor core into a coolant.

4. Conclusion

A numerical method using a finite difference method was developed to estimate the transport of fission products through a block of a prismatic HTGR. The method described the effects of the time and temperature on the release of Cs^{137} correctly. It is necessary to verify and validate the present results with experimental data. The developed numerical method will be included in a fission product release analysis code which is being developed in KAERI.

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