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Neutronics analysis of a small PWR utilizing carbon coated fuel with diffusion theory code PRIDE

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ABSTRACT

Tri-structural Isotropic (TRISO) based fuel with SiC matrix has the potential to be used in Light Water Reactors (LWRs) for enhancing its safety features in extreme scenarios. In this paper, neutronic analysis of a conceptual compact sized PWR core utilizing TRISO fuel has been carried out using diffusion theory based code PRIDE which was previously performed with CITATION code (Hussain and Xinrong, 2010). WIMS-D/4 code has been used for generation of group constants. Results obtained from PRIDE and CITATION have been compared. Analysis reveals that results of both codes are in excellent agreement which also validates PRIDE code. Additionally, practically achievable packing fraction which is 44% has been used and operating life of the core has been compared with the operating life of core utilizing TRISO fuel without SiC matrix. The effect of increase in enrichment and size of fuel kernel on operating cycle of core has also been studied.

1. Background

Small and Medium Sized Reactors (SMRs) have gained high significance because of the compact size, reduced capital cost and increased safety features. These types of reactors are attractive and are potential source of energy for limited power production, desalination and heating. Carbon coated fuel is being considered in SMRs to increase safety barriers against fission product release in normal and abnormal scenarios. In recent past, different aspects of utilizing carbon coated fuel particles embedded in SiC matrix in LWR's have been studied. Detailed descriptions of this concept are presented, along with potential benefits, and issues with respect to their application in LWR environments, specifically from the point of view of materials, neutronics, operations, and economics (Terrani et al., 2012). A number of dedicated projects in the development of SMRs including "Design and Technology Development for LWRs with Coated Particle Based Fuel" are being encouraged and monitored by IAEA. IAEA also periodically produces reports and updates status of projects in the design and technology development for such reactors. The detailed research in SMR technology can be found in IAEA-TECDOC-1451 & 1652 (IAEA, 2005).

TRISO fuel particles consist of fuel kernel and successive layers of Pyro-carbon (PyC) and Silicon Carbide (SiC). First, a low-density PyC buffer coating is applied that provides void volume to accommodate fission gas, attenuates fission product recoils released from the fuel kernel and accommodates kernel swelling. This layer is surrounded by successive coatings consisting of an Inner PyC layer (IPyC), Silicon Carbide (SiC) layer and an Outer PyC layer (OPyC). The PyC coatings on either side of the SiC provide pre-stressing to assist in accommodating internal pressure. The SiC layer is the primary pressure vessel and is an effective barrier to fission product release. Generally, UO_2 with 8.0 to ~20% fuel enrichment is used as fuel kernel. Prior to use in power reactors, TRISO-coated fuel particles were extensively irradiated and tested in prototype and material test reactors. The AVR (Arbeitsgemeinschaft Versuchsreaktor) was a prototype HTGR with a pebble bed core having 46 MWt (15 MWe) power output with typical coolant inlet and outlet temperatures of 270 °C and 950 °C, respectively, at 1 MPa pressure. The AVR was operated between 1967 and 1988 in Germany (IAEA, 2010). The dimensions of different layers of pyrocarbon and SiC layers are given in Table 1.

In previous studies, neutronics and steady-state thermal hydraulic analysis of a conceptual compact sized PWR core has been performed (Hussain and Xinrong, 2010). In these studies, WIMS-D/4 and CITATION codes have been used to perform the neutronic's analysis. Reactivity control techniques of compact sized PWR have been suggested (Hussain and Xinrong, 2009) and reactivity

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Table 1
TRISO fuel particle composition and dimensions.

Material	Density (g/cc)	Outer diameter (mm)
UO_2 fuel Kernel	10.88	0.4
Porous carbon buffer layer	1.1	0.6
Inner PyC layer	1.9	0.7
SiC coating	3.2	0.77
Outer PyC layer	1.9	0.87

feedback coefficients have also been determined for conceptual compact sized PWR core (Hussain and Xinrong, 2011).

In continuation to the previous research work, in this paper, neutronics analysis of same core has been performed using PRIDE code instead of CITATION code. The results of PRIDE code are compared with well-proven diffusion theory code CITATION. Also, a design change, i.e. fuel rods consists of TRISO fuel particles embedded in dense Silicon Carbide (SiC) matrix, is suggested and subsequent burn up analyses have been performed.

2. Reactor design description

In this research paper, the criticality and burn up studies of a small PWR core utilizing TRISO fuel particles have been performed. The arrangement of fuel rods and fuel assemblies in the core is shown in Fig. 1. There are total of 2180 fuel and 45 control rods in the core with a reactor core diameter of 2.2 m. The height of the fuel rod is 1.5 m. Core is reflected by 8 cm thick beryllium layers from all the sides. This is a small reactor which has only 25 MWth.

Power output, the main parameters of design are given in Table 2

3. Materials and methods

3.1. Reactor simulation codes

In order to perform the criticality analysis of the core represented in Fig. 1, two codes have been used which are WIMS-D/4 and PRIDE. WIMS-D/4 is a neutron transport theory based code which generates the averaged homogenized group constants of a unit cell. These group constants are fed to the diffusion equation based code PRIDE to calculate the core effective neutron multiplication factor.

3.1.1. WIMS-D/4 code

WIMS stands for Winfirth Improved Multigroup Scheme. It is a general purpose transport theory code which performs the lattice calculations with certain approximations. WIMS has its own 69 energy group data library. WIMS first calculates the cell averaged

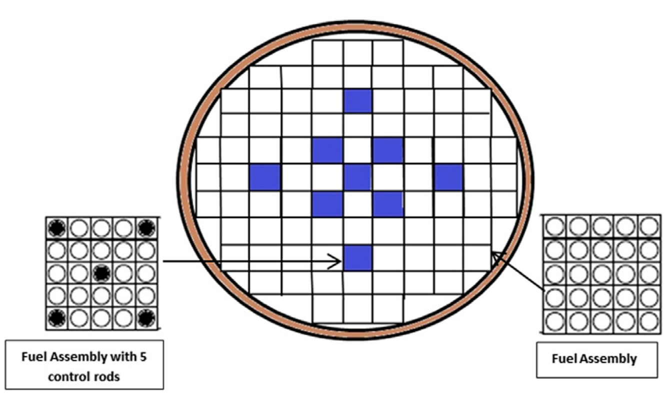

Fig. 1. Small PWR core utilizing TRISO fuel.

Table 2
Main design parameters.

Parameters	Dimension	Parameters	Dimension
Power output	25 MWth	No. of control rods	45
Core height	1.7 m	Control material	Hf
Core dia	2.2 m	Coolant, moderator	Water
Fuel rod height	1.5 m	Fuel type	TRISO
Fuel pitch	3 cm	Inventory of heavy metal	260 kg
Fuel rod dia	2 cm	Enrichment	9% by wt
Cladding thickness	1.5 mm	Fuel composition	UO_2
Number of assemblies	89	Reflector	Be
No of fuel rods/assembly	25	Reflector thickness	8 cm

homogenized group constants for all the energy groups and then these group constants are further utilized to generate the condensed group constants of the energy groups input to the code by the user. Important group constants which are used as input to the PRIDE code are diffusion coefficient (D), macroscopic absorption cross section ($\sum_a$), neutron production cross section ($\nu\sum_f$) and the scattering cross sections (Deen and Woodruff, 1995).

3.1.2. PRIDE code

PRIDE is the abbreviation for “Program for Reactor In-Core Analysis using Diffusion Equation”; it uses finite difference method to solve the diffusion equation. PRIDE is a user friendly code which provides formatting free input, easy input error detection system and flux and power density averaging etc. Accurate results can be obtained by using finer meshes of the core geometry, as it has no internal limit of meshes and regions. The common coordinate systems mostly used by researchers in reactor physics have been implemented up to three dimensions. These include one, two and three dimensional slab, the one, two and three dimensional cylinders and one dimensional sphere. Different boundary conditions including repeating boundary, inverted symmetry and 90° rotational symmetry conditions are provided. In addition to usual criticality calculation, a variety of problems including kinetic parameters calculation, buckling search for critical as well as for specified multiplication factor may be solved (Ahmed and Gull, 2012). In order to validate the results generated by PRIDE code, study has been performed on IAEA 3D benchmark problem and PRIDE has generated very accurate results (Ahmad and Sahibzada, 2012).

3.2. Methodology

In order to accomplish the task, work has been divided in two parts. In first part the criticality analysis i.e. effective neutron multiplication factor has been calculated of the core represented in Fig. 1. In the second part, burn up analysis has been performed to see the reactivity variation during the entire core cycle.

3.2.1. Criticality analysis

In order to perform the criticality analysis, cell averaged homogenized group constants are generated using different modeling options of WIMS-D/4 code. Whereas, the effective neutron multiplication factor has been calculated by PRIDE code using the group constants generated by WIMS-D/4 code. Group constants have been generated by using two different modeling options of WIMS-D/4 code, which are pin cell model and cluster model. Various libraries including ENDF-VI, ENDF-VII, IAEA, JEF-2.2, JEF-3.1, JENDL-3 and WIMS-86 have been used to generate the group constants. General methodology to perform the criticality analysis in the form of flow chart has been shown in Fig. 2.