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The potential impact of Fully Ceramic Microencapsulated (FCM) fuel on thermal hydraulic performance of SMART reactor

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ABSTRACT

One of emerging advanced fuel materials for the next generations of the nuclear reactors is Fully Ceramic Microencapsulated (FCM) fuel. FCM fuel structure is comprised of TRISO particles dispersed randomly in a SiC matrix. In the present study, the thermal hydraulic performance of a SMART reactor is investigated and the results are compared with the case that conventional UO_2 fuel is loaded into the core. At the beginning of the cycle, the reactor is simulated in normal operational and transient conditions. MCNPX 2.6 stochastic code is utilised to calculate neutronic parameters. COMSOL and RELAP5 codes are used for thermal hydraulic analysis. The reactor dynamics is simulated based on six-group point kinetics and lumped parameters thermal hydraulic model. A ramp input reactivity is inserted into the core and the behaviour of the two fuel types are examined. As FCM fuel is heterogeneous structurally, detailed analysis is also performed. Homogenization of FCM fuel material might be a serious error source in related calculations. This is investigated numerically and the results are discussed in details.

1. Introduction

ORNL has suggested FCM fuel with enhanced accident tolerance features (Powers, 2013). FCM fuel is based on TRISO particles which has been operated in high temperature gas cooled reactors for decades. As shown in Fig. 1, TRISO particles consist of a spherical kernel bounded by four coating layers of PC, IPyC, SiC and OPyC (Dai et al., 2014). The TRISO structure acts as a multilayer barrier for fission product gases. In the FCM fuel concept, these TRISO particles are embedded in a SiC matrix in pellet form which is placed in a standard cladding material (e.g. Zircaloy, Steel, SiC). Therefore, fuel assemblies manufactured by FCM fuel rods might be utilised in the present and future generations of LWRs. FCM fuel retains fission products due to the presence of multiple barriers in its structure. The proliferation resistant feature is another advantage added by this type of fuel (Awan et al., 2018). The main advantage of FCM fuel is its high thermal conductivity. Integral configuration of FCM fuel along with improved safety features of SMRs may lead to enhanced reactor design which is inherently safe under severe accident circumstances (Terrani et al., 2012).

In the published literature, several studies are performed on FCM fuel temperature calculation using homogenization methods. Brown et al. (2013) has investigated different lumped models for FCM fuel. The homogenized thermal properties of the FCM fuel rod are calculated based on the material volume fractions (volumetric average model). As indicated by the author, this approach predicts acceptable fuel temperature only for slow transients (Brown et al., 2013). The fuel temperature is under estimated by the method since a high fraction of energy is deposited in the kernel of TRISO particles.

Another homogenization model for thermal analysis of FCM fuel, proposed by Lee and Cho (2015), is two-temperature homogenized thermal conductivity model. In this method, two regions are considered which are applied to heat conduction equations: one region represents TRISO kernels and the other region represents TRISO coating layers plus SiC matrix. Homogenized thermal conductivity is calculated for each region using particle transport Monte Carlo method employing HEATON program (Lee and Cho, 2015). The maximum fuel temperature obtained from this model is more accurate than that from volumetric average thermal conductivity model. However, a rough approximation of FCM fuel rod temperature distribution is estimated.

Thermal hydraulic codes commonly used for light water reactors are extensively validated. These best estimate codes include the US-NRC ones (RELAP5, TRAC, TRACE), the French code CATHARE-2 and the

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Nomenclature

t	time, the independent variable
$P(t)$	time dependent reactor power
P_o	nominal reactor power
$n_r(t)$	time dependent relative neutron density
$C_i(t)$	i_{th} group of delayed neutron precursors
i	delayed neutron group number ($i = 1, 2, \ldots, 6$)
j	fuel assembly number ($j = 1, 2, \ldots, 57$)
β_{eff}	effective delayed neutron fraction
$\beta_{i,\,eff}$	i_{th} group of effective delayed neutron fractions
λ_i	i_{th} group delayed neutron decay constant
k_{eff}	effective multiplication factor
S_f	deposition fraction of fission power in fuel
ℓ	prompt neutron lifetime
Λ	neutron mean generation time
q°	fuel rod heat generation rate
q''_H	conductive heat flux per unit area
$q^{o'''}$	volumetric heat generation rate of fuel rod
$T_f\,(R_f)$	outside fuel temperature
T_{fave0}	average fuel temperature at normal operational condition
T_{fave}	average fuel temperature
$T_f\,(r_f)$	fuel temperature at r_f
T_{fmax}	fuel centre temperature
T_{Clad}	average clad temperature
T_{Clad0}	average clad temperature at normal operational condition
T_{Cool}	average coolant temperature
T_{Cool0}	average coolant temperature at normal operational condition
T_{in}	core coolant inlet temperature
T_{out}	core coolant outlet temperature
A_g	heat transfer area of fuel gap
A_{Cool}	heat transfer area of coolant
ρ	reactivity
$\rho_{inserted}$	inserted reactivity
$\rho_{feedback}(T_{fave}, T_{Cool})$	total feedback reactivity which is a function of T_{fave} and T_{Cool}
$\alpha_f\,(T_{fave})$	fuel temperature feedback coefficient at T_{fave}
$\alpha_{Cool}\,(T_{Cool})$	coolant temperature feedback coefficient at T_{Cool}
h_{Cool}	convection coefficient of the coolant
h_g	heat conduction coefficient of the gap
k_f	fuel thermal conductivity
C_f	fuel material heat capacity
C_{Clad}	clad material heat capacity
C_{Cool}	coolant material heat capacity
m_f	fuel mass
M_f	total fuel mass
M_{Clad}	total mass of the clad
M_{Cool}	total mass of the coolant
M^o_{Cool}	coolant mass flow rate
V_f	fuel rod volume
R_f	fuel rod radius
L_f	active fuel length
r_f	the variable of fuel rod radius
DNBR	departure from nucleate boiling ratio
FARP	fuel assembly relative power $= \frac{\text{Power of a Fuel Assembly}}{\text{Average Power of Fuel Assmblies}}$
FCM	fully ceramic microencapsulated
IPyC	inner pyrolytic carbon
OPyC	outer pyrolytic carbon
ORNL	oak ridge national laboratory
PC	porous carbon
PPF	power peaking factor
SiC	silicon carbide
SMART	system-integrated modular advanced reactor
SMR	small modular reactor
TRISO	tristructural isotropic
UN	uranium nitride

German code ATHLET developed by GRS CEA-Grenoble, 2008). Structural differences of FCM and conventional UO_2 fuel rods are shown in Fig. 2. These codes can be adopted to predict fuel temperatures in FCM fuel rods based on effective thermophysical properties. However, using the effective thermophysical properties cannot return the details of the temperature distribution within an individual FCM fuel rod and its TRISO particles. It may cause a considerable deviation in the results as FCM fuel rods are heterogeneous structurally.

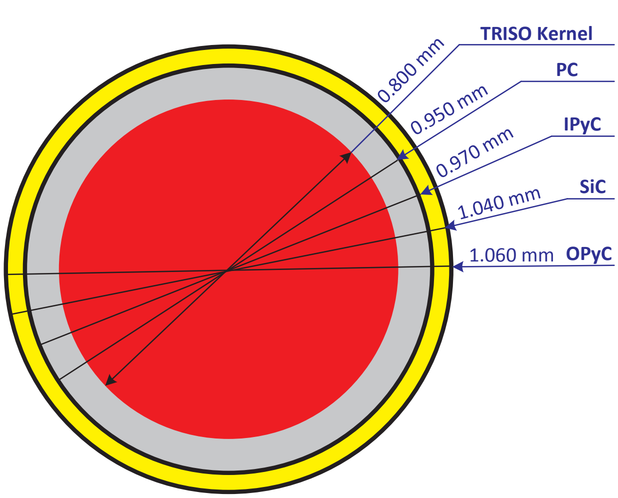

Fig. 1. A TRISO particle structure (Besmann et al., 2014).

In this work, at the beginning of the cycle, the whole reactor core is simulated using RELAP5 code (RELAP5/MOD3, 1998) with homogenized FCM fuel rods and detailed calculations are performed heterogeneously for hot section of the FCM hot fuel rod utilizing COMSOL code (Comsol, 2012). Reactor dynamics during the transient is modelled through a coupled neutronic and thermal hydraulic code developed for this purpose. The results of normal operation and ramp input reactivity accident for both cases are finally compared with each other.

2. TRISO particle and FCM fuel characteristics

TRISO particle failure depends on its kernel composition and layers thicknesses. Failed TRISO particles deteriorate the fission products retention capability of the FCM fuel rods (Zhou and Tang, 2011). Accordingly, Besmann et al. (2014) has studied thermomechanical aspects of TRISO particles with UN kernels for LWRs. It is stated that UN kernels with 0.80 mm in diameter can survive at 1600 °C with 178.22 GWd/THM burnup. As SiC matrix occupies a fraction of the core volume, higher enriched fissile material is required in FCM fuel design to have the same amount of fissile material mass in comparison with conventional UO_2 fuel. Since UN density is higher than UO_2, utilizing kernels containing UN material reduces the required fuel enrichment. Table 1 shows specifications of TRISO particles and FCM fuel rods simulated in the present study (also see Fig. 1). The data shown in this table is the material used in Step 1 of the procedure shown in Fig. 4.

Packing fraction is an important parameter that should be considered in FCM fuel rod design. The packing fraction is the volume ratio of TRISO particles to the FCM fuel rod. The undesirable particle contacts may lead to damages. In order to minimize TRISO particle