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Neutronic-thermal hydraulic coupling analysis of the fuel channel of a new generation of the small modular pressurized water reactor including hexagonal and square fuel assemblies using MCNP and CFX

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ABSTRACT

The main purpose of this paper is neutronic-thermal hydraulics coupling analysis of the fuel channel of an advanced Small Modular Reactor (SMR), which is nominated as a near term option of the generation IV reactors. The CAREM25 is chosen as the reference SMR. By considering hexagonal and square lattices of the fuel assemblies in the SMR reactor core, neutronic-thermal hydraulic coupling of hot and average fuel channels were conducted via utilizing of the MCNPX and CFX codes. Central fuel assemblies are determined as the hottest fuel assembly in the proposed SMR cores with hexagonal and square fuel assemblies with power peaking factor of 1.778 and 1.674, respectively. The hottest fuel rod power peaking factor is calculated as 1.846 and 1.954 for hexagonal and square fuel channels, respectively. The calculated linear power densities of the hottest and the average fuel rod of hexagonal channels are $20.18\left(\frac{kW}{m}\right)and10.77\left(\frac{kW}{m}\right)$, respectively, as well as of square channels are$24.75\left(\frac{kW}{m}\right)$ and$10.36\left(\frac{kW}{m}\right)$, respectively. Results show that the maximum axial power along the fuel rods occurred below the mid-plane of the rod. The ratio of the hot to average rod axial power peaking factor as a safety parameter used to calculate the maximum heat flux in the hottest channel, is calculated close to 2 and 2.4 for the core with hexagonal and square fuel assemblies, respectively. The calculated power peaking factors and linear power densities of fuel rods are in suitable range and in good agreement with PWR safety issues.

The maximum temperature of fuel centerline in both types of the proposed SMRs is much lower than that of large PWRs (e.g., 2156 K for hot rod in VVER-1000). Between hexagonal and square fuel assemblies, the hexagonal FAs are preferred because of the two reasons; first, the axial fuel centerline temperature in the hottest hexagonal channel is lower than that of hottest square channel, so it is a good feature with respect of safety margins. Second, the outlet temperatures of the channels in the SMR core with hexagonal FAs are higher than the outlet temperature of the channels in the SMR core with square FAs. Thus the thermal efficiency of the reactor core with hexagonal FAs is more than that of the core with square FAs.

1. Introduction

The Small Modular Reactors (SMRs) have been under the remarkable attentions during recent years. A SMR, as defined by the International Atomic Energy Agency (IAEA), is a nuclear reactor that generate between 10 and 300 MWe of electricity (IAEA, 2014). The term "modular" is derived from the fact that small reactors can be manufactured in a factory completely and transferred to the site for installation. In specific, small modular pressurized water reactors, encapsulate all PWR primary coolant system components such as pressurizer, steam generators, pumps and control rod drive mechanism into a tall reactor pressure vessel (Ingersoll, 2009; NEA, 2011; Vujic et al., 2012; Yan et al., 2012). There are very wide

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Nomenclature

SMR	Small Modular Reactor
SG	Steam Generator
IAEA	International Atomic Energy Agency
RPV	Reactor Pressure Vessel
LWR	Light Water Reactor
BP	Burnable Poison
NEA	Nuclear Energy Agency
AE	Absorbing Element
VVER	Water-Water Energetic Reactor
PWR	Pressurized water Reactor
CFD	Computational Fluid Dynamics
BOC	Beginning Of Cycle
FA	Fuel Assembly
MDNBR	Minimum Departure from Nucleate Boiling Ratio
NPP	Nuclear Power Plant
HFP	Hot Full Power

varieties of SMR designs with distinct characteristics that are being developed. PWRs are the majority in SMR and the only type of LWRs, almost every country is pursuing this technology i.e. Argentina, Brazil, China, France, Republic of Korea, Russian Federation and United States of America. In order to ensure inherent and passive safety with compact reactor size we can find that most of LWRs designs are integrated pressurized reactors. (Alkan, 2013; Chang et al., 2000; Chung et al., 2003; Chung, 2008; Franceschini and Petrovic, 2008; Hibi et al., 2004; Hong and Song, 2013; Kim, 2011; Koroush and Kazimi, 2012; Mitenkov and Polunichev, 1997; Sahin et al., 2010; Sahin, 2009; Standring, 2009).

Although small modular light water reactors are currently receiving significant interest, there is not enough available operating experience especially for those new ones that include integrated phenomena and passive safety features. Therefore, it might be possible to initiate extensive investigations on these types of reactors for the purpose of improving the current performance level of these systems, significantly.

In order to study the behavior of a nuclear reactor core, several fields of physics need to be considered. The neutron flux distributions determine the amount of energy production by fission in the fuel. In a pressurized SMR the released energy is conducted through the solid fuel pins to the conjugate liquid water, and it is removed by the water flow, which is sustained via natural or forced circulation. The water is acting as coolant as well as moderator for the neutrons. In turn, the density of the water couples to the neutron distribution.

Besides, the fuel temperature, which depends on the power and the coolant conditions, gives another feedback to the macroscopic neutron cross-sections via the Doppler effect (Jareteg et al., 2016).

In order to consider the correct core reactivity feedback in the study of the nuclear reactor core, the coupled solution of the neutron transport equation, the equations of heat conduction in the fuel and the heat and momentum transport in the coolant are required. Usually, the codes for reactor analysis utilize one-dimensional hydraulic models for calculating the feedback, such as (Singh et al., 2013; Dürigen et al., 2013).

The neutronic - thermal-hydraulics coupling codes at various levels of spatial resolution is an effective method in reactor numerical analysis. The neutronic - thermal-hydraulics coupling algorithms have been widely used for many newly developed codes in recent years due to the great improvement of computer technology. In some simplified cases, an average fuel assembly neutronic model coupled with a sub channel thermal-hydraulics model can show almost the whole axial information of the reactor core (Shan et al., 2010; Yang et al., 2011; Sjenitzer et al., 2015; Wu and Kozlowski, 2015). While during more complex calculations like load following (Wang et al., 2013), as well as transient and accident safety analysis the whole information of the reactor core is required as completely as possible.

Although small modular light water reactors are currently receiving significant interest, there is not enough available operating experience especially for those new ones that include integrated phenomena and passive safety features. Therefore, it might be possible to initiate extensive investigations on these types of reactors for the purpose of improving the current performance level of these systems, significantly.

The main purpose of the present study is neutronic-thermal hydraulics coupling analysis of the fuel channel of an advanced SMR which is nominated as a near term option of the generation IV reactors. The MCNPX code is used for whole core neutronic simulation. Ansys CFX code is used for fully three-dimensional Computational fluid dynamics (CFD) and thermal hydraulics simulation of the fuel channel. By considering hexagonal and square lattices of the fuel assemblies in the SMR reactor core, neutronic-thermal hydraulic coupling of hot and average fuel channels were conducted via utilizing of the MCNPX and CFX codes.

The coupling study aims at the calculation of the temperature distribution of coolant and fuel along the hexagonal/square hot and average fuel channel, high qualified 3-D visualization of thermal neutron flux in the SMR reactor core including hexagonal and square fuel lattices, the fuel assemblies power peaking factors and axial power peaking factors of hexagonal/square hot and average fuel channels.

2. Material and methods

In order to carry out a detailed neutronic-thermal hydraulic investigation of an advanced pressurized SMR, firstly, the CAREM 25 is chosen as our reference case study. Secondly, the whole reactor core is simulated in details using MCNPX (Erfaninia et al., 2016). The MCNP code (Briesmeister, 2000) is the internationally recognized code for analyzing the transport of the reactor nuclear particles based on the Monte Carlo Simulations. It is severally used and benchmarked against experimental parameters of nuclear reactors (Mahler, 2009; Verboomen et al., 2006).

Using MCNPX, the whole core of advanced pressurized SMR including hexagonal-shaped and square-shaped fuel assembly is simulated, separately. The core with hexagonal and square fuel assemblies (FAs) is simulated so that the dimensions of the core as well as their FAs and the fuel rods are same as the reference core, just the enrichment of fuels and the number and the arrangement of fuel rods were adjusted to the certain values so that the simulated reactor is kept critical in the normal operating condition.

The Hot and the average fuel channels of the both type of simulated reactor core are determined via calculating the power peaking factors of the FAs and fuel rods by using tally (type F4) of MCNPX. Since the temperature and density of coolant considerably affect the neutron moderation and fission cross section of materials, the height of the core is divided into several equal-spaced zones and the axial fuel and coolant temperature changes from the core inlet to the core outlet are obtained by using Ansys CFX as a fully three-dimensional Computational fluid dynamics (CFD) code which is coupled with MCNPX.

Using MCNPX-CFX coupled study, this paper aims at calculating the temperature distribution of coolant and fuel along and across the hexagonal/square hot and average fuel channel, high qualified 3-D visualization of thermal neutron flux in the SMR reactor core

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