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Numerical approach to study the thermal-hydraulic characteristics of Reactor Vessel Cooling system in sodium-cooled fast reactors

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ABSTRACT

The main vessel plays an important role in containing the entire primary sodium for pool-type sodium-cooled fast reactors (SFRs). The Reactor Vessel Cooling System (RVCS) has great effect on cooling the main vessel. However, little attention has been given to the study on transient characteristics of RVCS in the previous SFR research. Thus, a home-made one-dimensional (1-D) code named Reactor Vessel Cooling system Analysis Code for Sodium-cooled fast reactor (VECAS) is proposed to evaluate the thermal-hydraulic characteristics for SFR. The detailed models of the developed VECAS are presented in this paper. Moreover, the developed models have been validated against an experimental study. Numerical data of the main vessel cooling circuit are compared with the measurements of the Demonstration Fast Breeder Reactor (DFBR). The simulation results are in good agreement with the experimental data. Furthermore, the validated VECAS is coupled with the Transient Thermal-Hydraulic Analysis Code for Sodium-cooled fast reactors (THACS). The transient characteristics of RVCS in China Experimental sodium-cooled Fast Reactor (CEFR) are simulated by the coupled code. Steady analysis shows that the main vessel is cooled effectively. The peak temperature appears at the top of the main vessel lower than the permissible upper temperature limit. During the transient analysis, VECAS has predicted a reverse flow in RVCS, which contributes to the core cooling. Furthermore, sensitivity analysis of the main parameter has also been performed. Therefore, it can be concluded that coupled VECAS has the ability to evaluate the thermal-hydraulic characteristics as well as the decay heat removal capacity of RVCS. The coupled code could provide references and technical supports for the design and optimization of the pool-type sodium-cooled fast reactor.

1. Introduction

CHINA is enhancing the development of generation IV advanced nuclear power systems (Zhang et al., 2018). As one of the 6 candidate Generation-IV reactors (Chen et al., 2018), Sodium-cooled Fast Reactor (SFR) is more effective in using uranium resources and transmuting long-life high level radioactive waste, which contributes to the sustainable development of nuclear fission energy (Ma et al., 2015). Thus, more and more importance is attached to SFRs by countries all over the word.

Fig. 1 presents the reactor structure of Indian Prototype Fast Breeder Reactor (PFBR), which is a typical pool-type sodium-cooled reactor (Velusamy et al., 2010). Fig. 2 shows another typical pool-type SFR China Experimental sodium-cooled Fast Reactor (CEFR). As shown in Figs. 1 and 2, the entire primary sodium is contained in the main vessel, which is a critical component for pool-type sodium-cooled reactors. The main vessel is very close to the hot sodium inevitably. The core outlet temperature of PFBR reaches up to 547 °C (Chetal et al., 2006). The sodium temperature in the hot pool is very high for the reason that the sodium heated in the core flows into the hot pool directly. The main vessel made of stainless steel will exceed its limiting temperature value easily. Therefore, it's of great necessity to cool the main vessel by cold liquid sodium.

The Reactor main Vessel Cooling System (RVCS) is designed to cool the main vessel and enhance its structural integrity. The schematic drawing of a typical RVCS is shown in Fig. 3. Generally, two concentric baffles are installed in the inner side of the main vessel, named inner thermal vessel and outer thermal baffle. What's more, inner vessel is

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Nomenclature

A	Area, m^2
De	Equivalent diameter (m)
f	Friction coefficient
H	Specific enthalpy ($J{\cdot}kg^{-1}$)
k	Thermal conductivity ($W{\cdot}m^{-1}{\cdot}{}^{o}C^{-1}$)
k_a	Coefficient of acceleration pressure drop
k_c	Coefficient of local resistance
L	Length (m)
l_i	Length of control volume i (m)
m_i	Mass of control volume i (kg)
P	Pressure (Pa)
q	Heat flux ($J{\cdot}s^{-1}$)
t	Time (s)
T	Temperature (K)
U	Wetting perimeter (m)
W	Mass flow rate, ($kg{\cdot}s^{-1}$)
z	Axial coordination (m)

Greek symbols

ρ	Density ($kg{\cdot}m^{-3}$)
ϕ	Heat flux ($J{\cdot}s^{-1}$)
Δp	Variation of fluid pressure (Pa)

subscripts

i	Control volume i
in	Inlet
N	Control volume number
out	Outlet

usually attached to the inner thermal baffle to create a gap filled with motionless sodium, which contributes to weakening the thermal stress of the baffles. The main vessel is surrounded by the guard vessel with insulting layer attached outside, and the gap between main vessel and guard vessel is filled with inert argon. The guard vessel provides a double guarantee to prevent the leakage of sodium. The Reactor Air Cooling System (RACS) in the pit of the reactor and the concrete in its outer side are also included in RVCS. The air flows upwards from the bottom of the reactor pit and removes the heat of the system. In the accident of black out, the natural convection of the air as well as the radiation between the concrete and the insulting layer also contribute to the decay heat removal.

As shown in Fig. 3, the main vessel and the thermal baffles have created two annuluses, i.e. outer annulus and inner annulus, to allow the cold sodium passing by. Under normal operating condition, cold sodium pumped from the cold pool flows upwards through the outer annulus and finally returns to the cold pool by flowing downwards in the inner annulus. The cold sodium in the annulus absorbs heat from the hot pool along its flow path. On one hand, the fluids take the heat into the cold pool, and on the other hand the main vessel is protected from being heated up by the hot pool. When accidents occur, the pump driving force disappears and the fluid in RVCS may flow reversely under the natural circulation condition. The reverse fluids would flow into the grid plate, one of whose outlets is connected to the inlet of RVCS originally under normal operating condition, and finally the reverse fluids would flow into the core under natural circulation.

Generally, the upward fluid in the outer annulus flows into the inner annulus in two ways as shown in Fig. 4. Many SFRs adopted the way of Path 1, i.e. climbing over the wall, including almost all the SFRs in Indian such as PHENIX (IAEA-TECDOC, 2013) and Demonstration Fast Breeder Reactor (DFBR) (Vivek et al., 2013). However, others such as MONJU (Ohira et al., 2013) and CEFR employed the way of the path 2, i.e. flowing through the hole. Both the two ways have their advantages and disadvantages. For path 1, when the flow is strong, the fluid in the upward flow climbs over the wall and then dumps into the inner annulus, acting as a waterfall. It's a very strong stress shock on the thermal baffle. While for the fluid flow in path 2, it can avoid the waterfall by flowing through the hole. However, the existence of the hole may affect the structural integrity of the material which will be a threaten to the safety, stability and ability to withstand stress of the thermal baffles.

Many countries have developed computer codes to calculate thermo-hydraulics for SFR, including the RUBIN and GRIF in Russia, the OASIS and TRIO-U in France (Tenchine et al., 2012), the SAS4A in America (Cahalan and Wei, 1990; Cahalan et al., 1994; Fanning, 2012) and so on. Generally, those codes aim at the whole system analysis, but little attention has been given to the study on transient characteristics of RVCS. In present work, the performed study on RVCS is mainly confined to the CFD simulation. Vivek v et al. proposed a CFD based approach to study the effect of ovality and the uniformity of sodium flow in the main vessel cooling system for pool-type sodium-cooled fast reactors (Vivek et al., 2013). Another CFD simulation proposed by T.C. Hung et al. also concluded that RACS is effective in removing decay heat after shut down (Hung et al., 2011). What's more, an experimental study on DFBR has been conducted and temperature distributions in the main vessel cooling system are carried out (Kamzaki et al., 1995).

In fact, it's difficult for CFD to simulate the whole system under a reasonable run-time (Feng et al., 2017). It's very hard to evaluate the impact that the RVCS plays on the transient characteristics of the whole system by CFD method. Thus, the one-dimensional approach is effective to simulate the thermal-hydraulic characteristics of RVCS as well as the thermal-hydraulic response to the transient process of the primary loop in SFRs.

In China, the research of the thermo-hydraulics for SFR is very few. Most studies are focus on the Pressurized Water Reactor (PWR). Feng T et al. proposed an innovative Direct Residual Heat Removal System (DRHRS) to study the capability of removing the residual heat of CPR1000 (Feng et al., 2016). Xi'an Jiaotong University has developed a program MIDAC to analyze the processes of in-vessel severe accident in CPR1000(Wang et al., 2014a,b). Cui S et al. also carried out some numerical research on the thermal-hydraulic characteristics of the Chinese Fusion Engineering Test Reactor (CFETR) (Cui et al., 2017). To support the development of sodium-cooled fast reactor in China, this paper has developed a home-made one-dimensional (1-D) code named Reactor Vessel Cooling system Analysis Code for Sodium-cooled fast reactor (VECAS) to evaluate the thermal-hydraulic characteristics for SFR. Not only can this code simulate the thermal-hydraulic characteristics of RVCS itself but also it can be coupled with a 1-D system analysis code THACS (Ma et al., 2015; Yue et al., 2015) with quick computing speed. This paper has performed the calculation of CEFR and sensitivity analysis of the main parameters in RVCS with the coupled code. This code has the ability of the parametric sensitivity analysis, optimal design of system, and evaluation of decay heat removal capacity.

2. Models and methods

The control volume of the main components in RVCS is shown in Fig. 5. The components in the model contain the fluid domain, the solid domain and the gas domain. The fluid domain consists of the upward fluid and the downward fluid. The main vessel, the thermal baffles in its inner side and the guard vessel, the insulating layer, the concrete in its outer side are included in the solid domain. The argon gap and the air cooling system in the pit of the reactor as the gas domain are also taken into consideration.

For the fluids in the annuluses, the convection heat transfer with the