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Comparison of transient analysis of LBE-cooled fast reactor and ADS under loss of heat sink accident

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ABSTRACT

A natural circulation LBE-cooled research reactor developed by Institute of Nuclear Energy Safety Technology (INEST) is proposed and designed to be 10 MW_{th}. The reactor has two operation modes, which are LBE-cooled critical fast reactor mode and LBE-cooled accelerator-driven sub-critical system (ADS) reactor mode. In this paper, protected loss of heat sink (PLOHS) and unprotected loss of heat sink (ULOHS) transient accidents were simulated for both the critical and ADS reactors above by NTC-2D code, aiming at the investigation of the safety advantages for the two reactor modes under this typical transient condition (loss of heat sink, LOHS). The results showed that during PLOHS for both the two modes, all the key parameters (core power, fuel, cladding and coolant temperatures in the hottest channel) decreased to very small values after the reactor scrammed, which meant the reactors under the two modes were both safe. For ULOHS, the fuel, cladding and coolant temperatures of the fast reactor increased smaller than those of the sub-critical one, which means the fast reactor had a better safety advantage under LOHS transient.

1. Introduction

The Lead-cooled Fast Reactor (LFR) is one of Generation IV Reactors, which can be used in electricity generation, fuel breeding and long-life nuclear waste transmutation (Tucek et al., 2006). LFR is an advanced reactor of inherent safety and lots of research work is in progress (Tucek et al., 2006; Moisseytsev and Sienicki, 2008; Sathiyasheela et al., 2011). The partitioning and transmutation technology was proposed in 1960s, which is used to deal with the long life high radioactive spent fuel. The accelerator driven sub-critical system (ADS) is a potential option for spent fuel transmutation, which is of inherent safety. In contrast with critical reactors, not only nuclear waste transmutation capacity of ADS is stronger, but the MA neutron economy is also better. The international design and study of ADS are in progress in several countries, such as MYRRHA, EFIT and XADS, see e.g. (Tichelen et al., 2002; Liu et al., 2010; Suzuki et al., 2005).

Based on the design experience of nuclear reactors (Wu et al., 1999, 2002, 2006, 2007a,b, 2009a, 2011; Qiu et al., 2000; Li et al., 2007), a natural circulation LBE-cooled reactor is proposed by Institute of Nuclear Energy Safety Technology (INEST) (Huang et al., 2007; Wu et al., 2009b, 2010, 2014) for pre-research of the research and development (R&D) work of LBE-cooled Fast and ADS reactors in China. The reactor has two operation modes, which are a LBE-cooled critical fast reactor mode and a LBE-cooled ADS reactor mode. The preliminary conceptual design of the reactor is already completed. The reactor is designed to be 10 MW_{th} power research reactor and the primary coolant of the reactor is liquid LBE.

In this paper, a safety analysis code NTC (Neutronics and Thermal-hydraulics Coupled Simulation Program) developed by INEST was employed to investigate the safety characteristics of the critical fast and ADS reactor modes during loss of heat sink (LOHS) transient, including protected and unprotected loss of heat sink accidents (PLOHS and ULOHS), which are the most typical transient accidents for the reactor safety analysis. And the simulation results of the two reactor modes under the same transients were compared, aiming at the safety characteristics investigations of fast and ADS reactors under LOHS transient.

2. Reactor concept

The LBE-cooled critical fast and ADS reactor modes mentioned above are designed to be pool type reactors. Almost all structure and thermal-hydraulics parameters of the two reactors are the same. The core thermal power is removed by 850 tons of liquid LBE of which the operation temperature is between 533 and

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663 K. The driven force of the primary coolant is chosen to be the natural circulation as the core power is 10 MW_{th}. The natural circulation is a kind of passive safety and leads to no loss of flow accident (LOFA). There are four primary heat exchangers in the main vessel. The secondary loop coolant is water and the final heat sink is air cooling. A Reactor Vessel Air Cooling System (RVACS) is used to remove the decay heat during accidents. The RVACS is designed to be composed of 40 U-tubes installed outside of the reactor main vessel to cool the reactor by thermal radiation. The heat removal by the RVACS is continued by means of natural circulation of air. The core assembly distributions of critical fast and ADS reactors are presented in Fig. 1. There are four functional regions of the reactor core from the center to the external part, which are steel rod region, active region, reflector, shielding for the critical fast reactor and target region, active region, reflector, shielding for the sub-critical one.

As the two reactors are designed to be research reactors, the steel rod region of the critical fast reactor is filled up with steel rods and the target region of the ADS is stuffed with a neutron source which plays as the spallation target in ADS. The fuel is UO_2 with 19.75% enrichment and the cladding material is 316Ti stainless steel. K_{eff} of the ADS reactor is designed to be 0.98. The spallation target of ADS reactor is chosen to be with window and cooled by the forced circulation, of which the coolant is also liquid LBE.

3. NTC code and calculation model

3.1. NTC code

There are many safety analysis codes for different kinds of reactors, most of which use the point kinetics and thermal-hydraulics coupled method, such as RELAP5, SIM-ADS, SAS4ADS and so on, see e.g. (Tucek et al., 2006). The superiority of the codes using point kinetics is the higher calculation rate, which is less calculation time than the physical time. SIMMER-III and -VI developed by KIT use a neutron transport equation and thermal-hydraulics coupled method, which is famous in liquid metal cooling reactor transient analysis, see e.g. (Tucek et al., 2006).

3.1.1. Outline of NTC-2D

NTC is also a neutronics and thermal-hydraulics coupled code and can be used in advanced reactor safety analysis (Bai, 2007; Ke, 2006; Gu et al., 2015a,b). And NTC-2D used in this paper is a two-dimensional version of NTC code. The code and function structure of NTC-2D is presented in Fig. 2. NTC-2D is a two-dimensional thermal-hydraulics code module based on fluid dynamics and structure heat transfer model coupled with neutronics code module based on a two-dimensional time-, space- and energy-dependent neutron kinetic model.

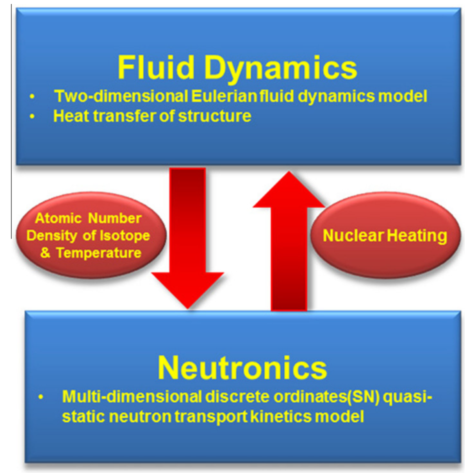


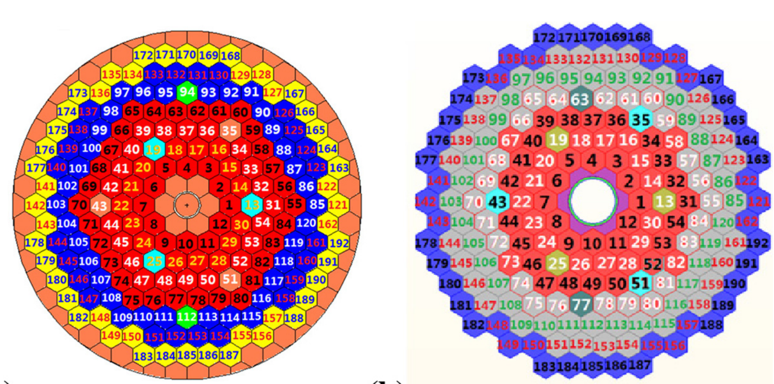
Fig. 2. Code and function structure of NTC-2D.

3.1.2. Neutronics model in NTC-2D

A time-, space- and energy-dependent neutron transport equation model as shown in Eq. (1) is adopted in the neutronics model of NTC-2D. In order to solve this equation, an improved quasi-static method is used. In this method, the space- and time-dependent neutron transport equation is factorized into two equations, which are shape function equations shown in (2), (3) and amplitude equations shown in (4), (5). Here the delayed neutron effect is considered. NTC-2D contains two cross section data bases, which are infinite dilute microscopic cross section and self-shielding factor.

$$\begin{aligned}\frac{1}{v}\frac{\partial\phi(\mathbf{r},E,\mathbf{\Omega},t)}{\partial t} &+ \mathbf{\Omega}\cdot\nabla\phi(\mathbf{r},E,\mathbf{\Omega},t) + \Sigma_t(\mathbf{r},E)\phi(\mathbf{r},E,\mathbf{\Omega},t)\\ &= \int_{4\pi} d\mathbf{\Omega}' \int_0^\infty \sum_s (E' \to E, \mathbf{\Omega}'\\ &\to \mathbf{\Omega})\phi(\mathbf{r},E',\mathbf{\Omega}',t)dE' + S(\mathbf{r},E,\mathbf{\Omega},t) \end{aligned} \tag{1}$$

$$\begin{aligned}\frac{1}{v}\frac{\partial\varphi(\mathbf{r},t)}{\partial t} &+ \frac{1}{vN}\frac{dN}{dt}\varphi(\mathbf{r},t) + M\varphi(\mathbf{r},t)\\ &= (1-\beta)\chi_p F\varphi(\mathbf{r},t) + \frac{1}{N}\chi_d\lambda C(\mathbf{r},t) + \frac{1}{N}S(\mathbf{r},t)\end{aligned} \tag{2}$$



(a) (b)

Fig. 1. Core assembly distributions: (a) fast reactor; (b) ADS reactor.