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Analysis on LOCA for CSR1000

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ABSTRACT

In CSR1000 (Chinese Supercritical Water Cooled Reactor 1000), based on the existing SCAC-CSR1000 code, by incorporating break flow model and passive safety system, CSR1000-DP02 code for LOCA analysis is developed. 50% hot leg break loss of coolant accident (LOCA) is analyzed. The results show that, when LOCA happens, the reactor scrams immediately. The main feed water mass flow, and coolant mass flow of the two fuel channels overall both increase in LOCA. In the flow of passive containment cooling system curve, there is a peak from 2 s to 4 s, and there is a critical flow phenomenon from 5.7 s to 12 s. When LOCA occurs, the cladding temperature gradually decreases, the maximum cladding temperature is below the safety criterion of 1260 °C.

1. Introduction

Supercritical water-cooled reactor (Cheng and Liu, 2008) (SCWR) is one of the latest nuclear energy system which has the highest development prospects, it has key merits of high thermal efficiency and simplified system thus having low cost for power generation. At present, SCWR core design and system research are the hot spots of domestic and international research. In 2009, Nuclear Power Institute of China proposed the conceptual design of China supercritical water reactor CSR1000 (Li et al., 2013). Validating its feasibility is necessary step to assess the safety of the new supercritical water reactor. Now, there are many safety analysis codes (Zhu et al., 2004) for the reactors, such as RELAP5, RETRAN, TRAC, CATHARE and ATHLET. After many years of development and improvement, the versions of these codes have been updated many times, and the models have gradually become mature. The codes are able to carry out transients analysis, including the transients due to reactivity introduction, transients due to loss of coolant accidents caused by reactor coolant pipes large break, etc. But for supercritical water cooled reactor, the heat transfer model of transcritical stage has not been able to be exactly described, and the models which reflex the safety features of the core under accident conditions are not accurate. Safety analysis for CSR1000 has not yet been performed in-depth thus it seems important to make CSR1000 as research object to work out a safety analysis code and analyze typical accidents. In order to analyze the safety features of CSR1000 under loss of coolant accident, safety analysis code is developed in a loss of coolant accident and abnormal transient response of CSR1000 under loss of coolant accident is determined. The calculating results have significance for improving performance of CSR1000.

2. The research object

2.1. The overall parameters

CSR1000 (Wu et al., 2014) is a thermal spectrum reactor, whose reactor core is cooled and moderated by the light water, and its thermal efficiency can reach up to 43.5%. To achieve this thermal efficiency, the coolant inlet temperature is 280 °C and the outlet temperature is 500 °C in the reactor core. Specific parameters are shown in Table 1.

In Table 1, a total of 177 fuel assemblies of CSR1000 core are divided into the first and second fuel channel assemblies, the first fuel channel assemblies number is 57 and the second fuel channel assemblies is 120. Number of fuel rods is 244 while number of water rods is 4 within a fuel assembly. It should be noted fuel assembly is designed with cross-shaped control rod and it is evenly distributed in the reactor pressure vessel.

2.2. Coolant fuel channel

Different from the conventional SCWR design, CSR1000 uses the flow scheme of flowing from inside to outside. Flow scheme of the coolant in CSR1000 is shown in Fig. 1.

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Table 1
CSR1000 reactor core parameters.

CORE	CSR1000
The inlet/outlet average temperature of coolant	280/500
The number of main coolant pipelines/main steam pipelines	2/2
Fuel rod diameter/spacing/cladding thickness (mm)	9.5/10.5/ 0.57
Water rod wall thickness (mm)	0.80
The number of fuel rods/water rods	224/4
The number of fuel assemblies/first fuel channel assemblies/ second fuel channel assemblies	177/57/120
Reactor core/main steam pressure (MPa)	25
Heat/electric power (MW)/thermal efficiency (%)	2300/1000/ 43.5
Reactor core coolant flow rate (kg/s)	1190

In Fig. 1, the coolant is divided into 4 portions (Liu et al., 2015) after main feedwater enters the core from the cold leg:(1) coolant of the first fuel channel is 35.9% of the total, from top to bottom; (2) moderator of the first fuel channel is 10.8% of the total, from top to bottom; (3) The water rod moderator of the second fuel channel is 30.0% of the total, from top to bottom; (4) cooling water along pressure vessel wall is 23.3% of the total, from top to bottom. All the coolant flows out of the core along the second fuel channel from bottom to top after mixed fully in mixing chamber of the reactor core's bottom.

2.3. Safety systems

Supercritical water cooled reactor (SCWR) puts high demands on the safety of the power plant. Considering technical characteristics and relevant safety requirements of SCWR, Nuclear Power Institute of China (Xiao et al., 2013) designed dedicated passive safety system. Safety system layout is shown in Fig. 2.

As can be seen from Fig. 2, safety system design of CSR1000 uses a combination of active and passive design. Active control systems include the following: automatic depressurization system (ADS), residual heat removal system (RNS) which is available after the accident. Passive control systems are as follows: high pressure makeup tank (RMT), passive residual heat removal system (ICS), passive containment cooling system (PCCS) and gravity driven cooling system of reactor core (GDCS).

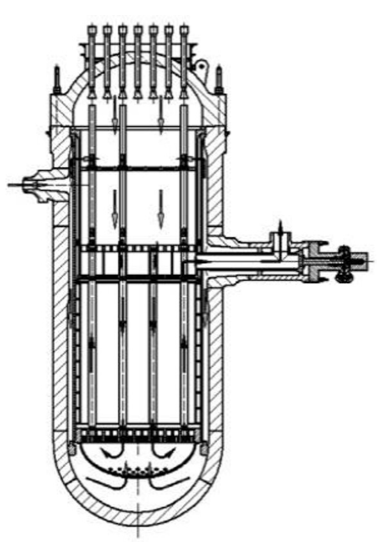

Fig. 1. The coolant flow scheme.

3. Computational model

3.1. Physical model

The six group point reactor equations of delayed neutron are used to calculate reactivity change over time in physical calculations, ignore the varying of reactivity with space. It is: the physical model using point reactor kinetics equation (Liu et al., 2015) as follows:

$$\frac{dN(t)}{dt} = \frac{(\rho(t) - \beta)}{\Lambda} N(t) + \sum_{i=1}^{6} \lambda_i C_i(t) + S \tag{1}$$

$$\frac{dC_i(t)}{dt} = -\lambda_i C_i(t) + \frac{\beta_i}{\Lambda} N(t) \quad (i = 1-6) \tag{2}$$

Among them, $N(t)$ is total fission power (neutrons/cm^3), β is the total effective delayed neutron fraction. Λ is the prompt neutron generation time (s). λ_i is delayed neutron decay constant of group i. β_i is delayed neutron effective fraction of group i. $C_i(t)$ is number of pioneer nucleus of group i (atomic/cm^3).

3.2. Conservation model

Ignoring the axial heat conduction in calculations, mass, energy and momentum conservation equations (Oka et al., 2010) are as follows:

$$\frac{\partial p(z,t)}{\partial t} + \frac{\partial G(z,t)}{\partial z} = 0 \tag{3}$$

$$\frac{\partial\{\rho(z,t)h(z,t)\}}{\partial t} + \frac{\partial G(z,t)h(z,t)}{\partial z} = \frac{1}{A_w}\{l_f Q''(z,t) - l_f Q''_w(z,t)\} \tag{4}$$

$$\frac{\partial\{\rho(z,t)h(z,t)\}}{\partial t} + \frac{\partial G(z,t)h(z,t)}{\partial z} = \frac{1}{A_w} l_w Q''_w(z,t) \tag{5}$$

$$\frac{\partial(\rho u)}{\partial t} + \frac{\partial(\rho u^2)}{\partial z} = \rho g \cos\theta + \frac{f}{2D_h}\rho u^2 \mp \frac{\partial P}{\partial z} \tag{6}$$

Among them, t is time(s). z is height of node, m. ρ is the liquid density (kg/m^3). G is mass flow rate (kg/s). u is the velocity (m/s). g is gravitational acceleration (m/s^2). f is frictional coefficient. D_h is equivalent diameter (m). P is pressure (MPa). h is enthalpy (J/kg). A_w is flow area, m^2. Eq. (4) is energy conservation equation of coolant, while Eq. (5) is energy conservation equation of moderator.

3.3. Critical flow calculation model

Critical flow (Zhang et al., 2015) can be divided into single phase critical flow and two phase critical flow.

3.3.1. Single phase critical flow model

In the stage of overheating and gaseous state supercritical phase, fluid physical properties tend to be ideal gas. The formula of critical pressure is:

$$p_c = p_0 \left(\frac{2}{1+\gamma}\right)^{\gamma/(\gamma-1)} \tag{8}$$

In the formula, γ is determined by the fluid physical parameters. p_0 is backpressure (MPa).

The break mass flow is given by:

$$W_i = \rho \sqrt{2c_p T_0 \left[\left(\frac{p_b}{p_0}\right)^{2/\gamma} - \left(\frac{p_b}{p_0}\right)^{(1+\gamma)/\gamma}\right]} \tag{9}$$

The critical mass flow is given by: