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Development of a transient thermal-hydraulic code for analysis of China Demonstration Fast Reactor

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ABSTRACT

The transient thermal-hydraulic code THACOS is under development for analysis of China Demonstration Fast Reactor. Applying modular technology, the code contains the core module, the pump module, the sodium pool module and the heat exchanger module and each module could operate separately. It can provide one-dimensional thermal-hydraulic simulation for the primary sodium coolant loop. The point reactor kinetics equations with six-group delayed neutrons have been applied to calculate the core power considering reactivity feedbacks caused by the Doppler effect, coolant density, axial expansion of fuel rods and radial expansion of the core. Multiple-channel model is applied to depict the core. Compressible homogenous flow model is used for the two-phase flow of sodium. The calculated results show that sodium boiling will occur quickly under the ULOF accident without any shutdown rods insertion. While, with the insertion of three hydraulically suspended shutdown rods, the core could be shut down safely and boiling will not occur in a short period of time. Obviously, the passive hydraulically suspended shutdown rods could keep the core safe under ULOF accidents.

1. Introduction

China Demonstration Fast Reactor (CDFR), as shown in Fig. 1, is a pool type sodium cooled fast reactor, which is under design by China Institute of Atomic Energy (CIAE). The thermal power is 2100 MW and the electrical power is 870 MW. CDFR contains three loops: the primary sodium loop, the intermediate sodium loop and the steam/water loop. The primary coolant system is submerged in a large sodium vessel which is divided into two large pools (hot and cold pools) by the thermal baffle, which increases the thermal inertia of the system greatly. The primary coolant, driven by the three main pumps, flows from the cold pool into the inlet plenum, then flows into the core to cool the core and finally flows into the hot pool. The intermediate heat exchanger (IHX) is a shell-and-tube heat exchanger, where the primary loop sodium flows through the shell side and the secondary sodium flows through the tube side. The primary loop sodium flow through IHX is driven by the difference of liquid level between the hot pool and the cold pool. CDFR, with a MOX fueled core, is designed in such a way that it has intrinsic negative reactivity feedbacks under unprotected accident conditions. A passive shutdown system with three hydraulically suspended shutdown rods has been designed to make sure that the core could be shut down safely under ULOF conditions (Alexandrov et al., 1995).

Computer codes developed for PWRs are not suitable for analyzing sodium cooled fast reactors (SFRs). In order to evaluate the dynamic behavior of the SFRs, it is necessary to develop specialized transient thermal hydraulic analysis codes for SFRs. Many transient thermal-hydraulic analysis codes have been developed in different countries for their fast reactor systems. In America, several specialized codes have been developed for different plants, such as IANUS for FFTF, DEMO for CRBRP (Albright and Bari, 1978) and NATDEMO for EBR-II (Mohr and Feldman, 1981). Also, some universal codes for SFRs have been developed. Brookhaven National Laboratory (BNL) developed NALAP, on the basis of RELAP3 code, by replacing the water property module with that of sodium (Martin et al., 1975), and also developed a general code SSC-L for loop type SFRs (Agrawal, 1978). For the safety analysis of severe accidents, Argonne National Laboratory developed the SAS4A code, which applied detailed steady-state and transient thermal-hydraulic, neutronic, and mechanical models to simulate the behavior of the reactor core and its coolant, fuel elements, and structural components under accident conditions (Cahalan and Wei, 1990). In Russia, Institute of Physics and Power Engineering (IPPE) developed GRIF for the simulation of SFRs dynamics, which could also be used for severe accidents conditions. Durham developed MELANI for modeling the Prototype Fast Reactor, in the UK (Durham, 1976). Recently, based on SSC-L, Korea Atomic Energy Research Institution (KAERI) developed the SSC-K code for analysis of the Korea Advanced Liquid Metal Reactor (KALIMER), which is a pool-type fast reactor (Chang et al., 2002). India Gandhi Center

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Nomenclature

A	area
c_i	neutron precursor concentration of group i
c_p	specific heat
D_e	hydraulic diameter
d	diameter
f	friction factor
g	acceleration of gravity
h	specific enthalpy or convective heat transfer coefficient
L_i	length of control volume i
Nu	Nusselt number
n	kinetic power
Pe	Peclet number
p	pressure
Q	heat flow or volumetric flow
q	heat flux
Re	Reynolds number
r	radius
T	temperature or torque
U	wetting perimeter
t	time
W	mass flow rate
x_e	equilibrium vapor quality
z	Cartesian coordinate along the height

Greek symbols

β	delayed neutron fraction
ρ	reactivity or density
Λ	neutron generation time
λ_i	decay constant of precursor group i
ω	rotation speed
ε	wall roughness
Φ_{f0}	two-phase frictional multiplier

Subscripts

c	clad
f	fuel or fluid
i	node number
IHX	intermediate heat exchanger
s	solid
W	wall

for Atomic Research (IGCAR) developed a computer code DYNAM for analyzing the dynamic behavior of FBTR (Vaidyanathan et al., 2010). Japan Atomic Energy Agency (JAEA) developed SIMMER-III and SIMMER-IV for the three-dimensional neutronics–thermohydraulics simulation of core disruptive accidents in SFRs (Yamano et al., 2009).

As previously mentioned, there have been many codes for simulating the dynamic behavior of SFRs, however, these codes couldn't be obtained publicly. The THACOS code is being developed to analyze the dynamic behavior of CDFR under accident conditions. As part of the work, models for the primary loop have been developed and applied in the present code successfully. The models and the calculated results will be presented in this paper in detail.

2. General description and models of THACOS

THACOS is a transient thermal-hydraulic analysis code, used to simulate the dynamic behavior of pool type sodium cooled fast reactors under steady-state and off-normal and accident conditions. It simulates the primary loop as shown in Fig. 2, containing the core, hot pool, three IHXs, cold pool and main pumps. Modular modeling technique has been applied to develop the code, which is convenient to expand, upgrade and transplant as shown in Fig. 3.

2.1. Core model

The core model is based on the assumption that the behavior of the whole core could be represented by parallel channels, where a channel includes of a single fuel pin and its associated coolant to represent one kind of subassemblies with similar characteristics.

2.1.1. Power generation

The heat generation in a nuclear reactor consists of two parts: fission and decay powers. The point reactor kinetics equations with six-group delayed neutrons have been applied to calculate the core fission power as follows:

$$\frac{dn}{dt} = \frac{\rho - \sum_{i=1}^{6}\beta_i}{\Lambda} n + \sum_{i=1}^{6}\lambda_i c_i \tag{1}$$

$$\frac{dc_i}{dt} = \frac{\beta_i}{\Lambda} n - \lambda_i c_i, \quad i = 1, \ldots, 6 \tag{2}$$

The total reactivity ρ includes the external input reactivity and the intrinsic reactivity caused by the Doppler effect, the change of coolant density, the axial expansion of fuel rods (Guppy, 1983) and the radial expansion of the core. The reactivity caused by the axial expansion of control rods is not considered, because it is small compared with the above. Detailed reactivity feedback models have been applied in this code, and the reactivity coefficients for each control volume are specified by users. Moreover, for the decay heat power, empirical correlation is used (Way and Wigner, 1948). The point kinetic parameters used in this code are summarized in Table 1. It needs to point out that the total sodium void reactivity of the CDFR core is −41.4 pcm, while that of most SFRs is positive, which is a very unique advantage of CDFR for enhancing the intrinsic safety.

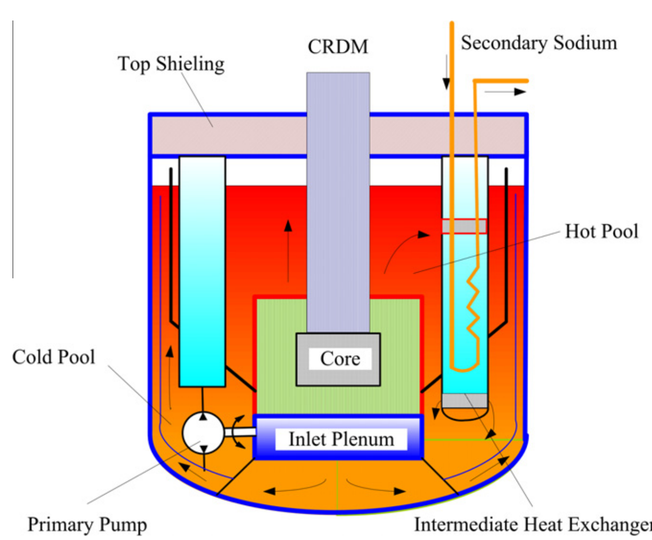

Fig. 1. Diagram of China demonstration fast reactor.