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Code development and safety analyses for Pb–Bi-cooled direct contact boiling water fast reactor (PBWFR)

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ABSTRACT

Pb–Bi-cooled direct contact boiling water fast reactor (PBWFR) can produce steam from the direct contact of feed-water and lead bismuth eutectic (LBE) in the chimney of 3 m height, which eliminates the bulky and flimsy steam generators. Moreover, as the coolant LBE is driven by the buoyancy of steam bubbles, the primary pump is not necessary in the reactor. The conceptual design makes the reactor simple, compact and economical. Owing to the large thermal expansion coefficient of LBE and good performance of steam lift pump, the reactor is expected to have good passive safety. A new computer code is developed to investigate the thermal–hydraulic behaviors and safety performance of PBWFR in the present work. Unprotected rod run-out transient over power (UTOP) and unprotected loss of flow (ULOF)/unprotected loss of heat sink (ULOHS) are simulated to test and verify its safety. The results show that PBWFR has very good inherent safety due to the satisfactory neutron and thermal–physical properties of LBE. Cladding materials turn to be the key factor to restrict its safety performance and UTOP is more dangerous for PBWFR. It's suggested that it should appropriately reduce the maximum value of the control rods to mitigate the consequence of UTOP due to good reactivity feedbacks in the core.

1. Introduction

Small reactors have attracted particular attentions because of their simple, compact, economical and reliable design features. Lead alloy cooled fast reactor (LFR) is more economical and safer than sodium cooled fast reactor (SFR) owing to the chemical inertness of lead alloy. It hardly reacts with air or water, so intermediate heat exchangers or intermediate circuits are not needed in LFR. As it has lots of merits, LFR has been selected to be as a candidate of the six types of advanced reactors in Gen-IV.

A more economical LFR called Pb–Bi/water reactor (PBWR) was proposed by Buongiorno (2001). Then Takahashi and et al. (2005a; Takahashi et al., 2008) redesigned the concept with a long-life core named Pb–Bi-cooled direct contact boiling water fast reactor (PBWFR) in 2003. The reactor could produce steam from direct contact of water and lead bismuth eutectic (LBE) in a 3 m-height chimney above the core, which could eliminate the bulky and flimsy steam generators (SGs). The coolant LBE is circulated by natural circulation of steam bubbles, which gets rid of the pumps in the primary system. Thus the accidents due to pump trip or SGs tube rupture can be avoided. The system pressure is set to be 7 MPa, the steam loops of which are almost the same as those in the conventional boiling water reactor (BWR). The structure of PBWFR is so simple that the economy is especially excellent.

The schematic diagram of PBWFR is shown in Fig. 1 (Takahashi et al., 2008) and the main parameters of the core are shown in Table 1. As shown in Fig. 1, the coolant LBE is heated in the core. Then it flows into the chimney and contacts directly with the cold water. In the chimney, the water is heated up to overheating by LBE. Meanwhile, LBE is cooled down. At the top of the chimney, LBE flows down through the down comer while the steam enters the separator and dryer. Owing to the two/three phases flow of LBE-water/steam, the density difference between the riser and down comer is large enough to establish a natural circulation.

As there are lots of advantages, the conceptual design of PBWFR is attractive and feasible. However, further studies should be worked out to evaluate its safety characteristics. Thus, a computer code for safety analyses of PBWFR is developed in the present work. And preliminary analyses of the transient accidents and safety features are carried out with the code.

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Nomenclature

A	Area of the flow channel, (m^2)
C_i	Decay neutron precursor concentration of group i
C_0	Distribution parameter
C_p	Specific heat capacity (J/(kg·K))
D_e	Equivalent diameter, (m)
d	Diameter, (m)
E_j	Decay heat fraction of group i
f	Friction coefficient
f_z	Axial power factor
f_R^N	Radial peaking factor
$f_{\Delta H}^E$	Engineering enthalpy rise factor
f_q^E	Engineering hot spot factor
Fr	Froude number
We	Weber number
g	Acceleration of gravity, (m/s^2)
h	Heat transfer coefficient, (W/m^2/K)
H_j	Heat of decay heat group j
hg	Gap heat transfer coefficient, (W/m^2/K)
H_w	Specific enthalpy of water/steam, (J/kg)
h_v	Volumetric heat transfer coefficient, (W/(m^3 K))
j	Superficial velocity, (m/s)
K	Form loss coefficient
K_a	Axial expansion coefficient, (1/K)
K_c	Coolant feedback coefficient, (1/K)
K_d	Doppler coefficient, (Tdk/dT)
K_r	Radial expansion coefficient, (1/K)
L	The length of a section, (m)
m	Number of computing sections
N	Fission power, (W)
n_d	Number of decay heat groups
Nu	Nusselt number
opt	Operation time, (s)
P	Total power (W) or pressure
P_{dec}	Total decay power, (W)
ΔP	Pressure drop, (Pa)
Pe	Peclet number
ΔP_{L0}	friction pressure drop, (Pa)
$\overline{Q}$	Average heat flux, (W/m^2/K)
Re	Reynolds number
Q_{ex}	Heat transfer rate between Pb–Bi and water/steam in chimney, (W/s)
t	Time, (s)
T	Temperature, (K)
V	Volume (m^3) or value opening
V_{ch}	Volume of chimney control volume, (m^3)
V_{gj}	Drift flux velocity, (m/s)
x	Ratio of diameter to pin pitch or steam quality in water
x_w	Water and steam quality in chimney
W	Mass flow rate, (Kg/s)
Δz	Length of the control volume, (m)

Greek symbols

α	Void of water/steam in whole fluid
β	Total delayed neutron fraction
β_i	Delayed neutron fraction of group i
Λ	Prompt neutron generation time, (s)
λ_i	Decay constant of precursor group i
λ_j^H	Decay constant of decay heat group i
$\overline{\rho}$	Average density in chimney, (kg/m^3)
ρ	Reactivity (\$) or density (kg/m^3)
ρ_{ex}	Reactivity due to the control rod, (\$)
φ_{L0}^2	Two phase friction multiplier

Subscripts

cs	Outer diameter of cladding
ci	Outer diameter of cladding
f	Friction
fs	saturated water
g	Gravitation
gs	Water and steam
l	Local
L	Laminar flow
$L0$	Full liquid phase
p	Pb–Bi
sat	Saturated
T	Turbulent flow
tr	Transition flow
TP	Two phase
w	Water and steam

Abbreviations

BWR	Boiling water reactor
LBE	Lead bismuth eutectic
LFR	Lead alloy cooled fast reactor
PBWFR	Pb–Bi-cooled direct contact boiling water fast reactor
SFR	Sodium cooled fast reactor
UTOP	Unprotected transient over power accident
ULOF	Unprotected loss of flow
ULOHS	Unprotected loss of heat sink
BWR	Boiling water reactor
LBE	Lead bismuth eutectic
LFR	Lead alloy cooled fast reactor
PBWFR	Pb–Bi-cooled direct contact boiling water fast reactor

2. Calculation models

The single channel model is chosen for the reactor core thermal–hydraulic calculation. All of the channels in the core are simulated as an average channel and a hot channel. According to the flow paths of coolant and water, the node diagram is constructed and shown in Fig. 2, which presents the flow direction of the coolant or water/steam in detail. The principal models of the system are established as shown below.

2.1. Core power model

As a system code, the point reactor kinetics model with six groups of delayed neutrons is adopted, with the reactivity feedbacks of the fuel temperature Doppler effect, coolant density changes, core radial expansion and core axial expansion respectively considered. The point reactor kinetics model shown follows is very simple for system analysis codes, thus it's widely used in the early computer codes.

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

$$\frac{\mathrm{d}C_i(t)}{\mathrm{d}t} = \frac{\beta_i}{\Lambda} N(t) - \lambda_i C_i(t) \quad i = 1, 2\cdots, 6 \tag{2}$$

The reactivity ρ of Eq. (1) contains two parts: the part due to the mechanical control rods, which can be simulated as a time function,