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Partial flow blockage analysis of the hottest fuel assembly in SNCLFR-100 reactor core

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Abstract In this paper, we perform an unprotected partial flow blockage analysis of the hottest fuel assembly in the core of the SNCLFR-100 reactor, a 100 MW_{th} modular natural circulation lead-cooled fast reactor, developed by University of Science and Technology of China. The flow blockage shall cause a degradation of the heat transfer between the fuel assembly and the coolant potentially, which can eventually result in the clad fusion. An analysis of core blockage accidents in a single assembly is of great significance for LFR. Such scenarios are investigated by using the best estimation code RELAP5. Reactivity feedback and axial power profile are considered. The cross-sectional fraction of blockage, axial position of blockage, and blockage-developing time are discussed. The cladding material failure shall be the biggest challenge and shall be a considerable threat for integrity of the fuel assembly if the cross-sectional fraction of blockage is over 94%. The blockage-developing time only affects the accident progress. The consequence will be more serious if the axial position of a sudden blockage is closer to the core outlet. The method of analysis procedure can also be applied to analyze similar transient behaviors of other fuel-type reactors.

Keywords Transient analysis · Flow blockage · LFR · Natural circulation · RELAP5 code

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1 Introduction

The safety analyses of a lead-cooled fast reactor (LFR) usually entail simulations of design extension conditions (DECs), which are characterized by the failure of reactor scram, involving unprotected loss of flow (ULOF), unprotected loss of coolant transient (ULOC), unprotected transient over power (UTOP), and unprotected fuel assembly blockage accidents. Among these accidents, the flow blockage accident is the most dangerous for integrity of fuel assembly [1]. This will cause serious degradation of the heat transfer between the fuel assembly and coolant, eventually resulting in the clad fusion.

The flow blockage situation may be caused by swelling of the fuel or some materials falling into the reactor pool. For wire-spaced fuel bundles, the blockage generally occurs along the helicoid wire, because of the accumulation of debris from failed fuel pins or broken wires. For grid-spaced fuel bundles, the blockage growth is caused by particles which are spread around the sub-channel dimensions and collected at the spacer grid [2].

The flow blockage scenario has been studied extensively. Generally, system thermal–hydraulic codes like RELAP5 are commonly adopted, and the point kinetic model is utilized to observe the reactivity feedback effect [3, 4]. Lu et al. [3] studied the partial and total blockage of a channel in the IAEA 10 MW MTR pool-type research reactor core without scram by using RELAP5/MOD3.2 code. Khan et al. [5] compared flow blockage accident by using RELAP5 and NK/TH coupling codes; Adu et al. [6] developed blockage analysis in MNSR reactor and observed a safer steady again due to inflow of coolant from adjacent channels to the blocked channels. Reis et al. [7] developed similar analyses in a TRIGA reactor core. Ravi et al. used more complex models to investigate conjugate heat transfer of core damage propagation in fuel subassembly of a sodium-cooled fast reactor [8–10].

LFR is a promising candidate as the first kind of GEN-IV reactors to be industrialized, due to its outstanding technical, economic, and safety characteristics [11]. However, without an appropriate oxygen control system, blockages in LFRs take place more easily due to the oxide formation. Most of the known accidents occurred in the LFRs were caused by blockages.

Bandini et al. [12] performed RELAP5 simulations on different blockage accidents in ALFRED reactor. In the paper, foot blockage scenarios with blocked area fraction from 0.1 to 0.95 were simulated. Di Piazza and Console Camprini made a numerical analysis of flow blockage phenomena in ALFRED reactor fuel assembly by using computational fluid dynamics (CFD) code of CFX, and a first evolution of the neutronics feedback was given. The quantification was obtained by a coupling approach of the neutronics deterministic code ERANOS and CFX [13]. Similar CFD analysis was developed by Salama [14–16]. Also, an investigation in ADS MYRRHA was developed by Chen et al. [17] and a macroscopic differential model of the rod bundle flow was set up and implemented in SIMMER-III.

SNCLFR-100, a 100 MW_{th} small modular natural circulation LFR, was designed by USTC (University of Science and Technology of China). It can be served as an LFR research platform and a distributed power source. The initial steady-state and transient conditions, including UTOP and unprotected loss of heat sink transient (ULOHS), were analyzed RELAP5 code. The simulations of the two transient verified preliminarily the inherent safety characteristics of the LFR [18].

In this paper, flow blockage transients of the hottest fuel assembly are investigated based on SNCLFR-100 reactor core by using RELAP5 code. Reactivity feedback and axial power profile are considered. Key parameters affecting the blockage transients explored include the cross-sectional fraction, axial position, and developing time of the blockage.

2 Brief description of the reactor

The SNCLFR-100 is a typical pool-type fast reactor, with the thermal power of 100 MW_{th} and the refueling interval of 10 years without assembly reconfiguration. Some advanced design ideas, e.g., the integral arrangement and the modular design, do help simplify the system configuration, which improve the reactor safety performance and engineering feasibility. The overall structure design of SNCLFR-100 primary cooling system is depicted in Fig. 1.

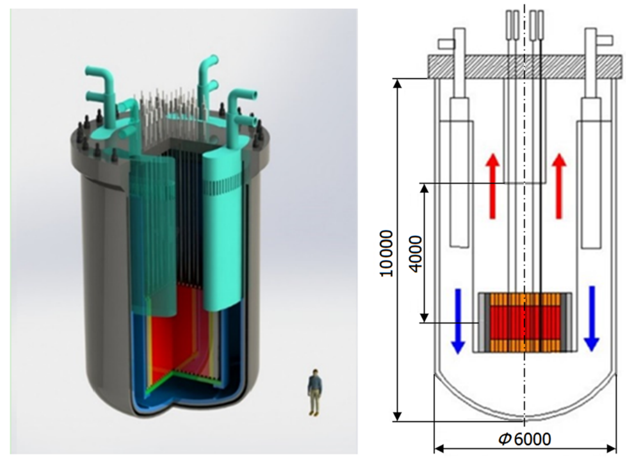

Fig. 1 (Color online) Schematic diagram of SNCLFR-100 primary cooling system

The primary system is cooled by a complete natural circulation of liquid metal lead. Figure 1 also shows flow path of the primary coolant. The lead, being heated in the core and getting light in density, flows upward, enters the upper plenum, and flows into four heat exchangers. The cooled lead flows downward into the lower plenum and returns to the core.

The reactor core consists of wrapped quadrilateral assemblies, containing 204 standard fuel assemblies, 36 control fuel assemblies, 48 reflector assemblies, and 84 shielding assemblies (Fig. 2a). A fuel bundle is fixed with six grid spacers (Fig. 2b). A fuel assembly (Fig. 2c) adopts 9 × 9 pins lattice. The control assembly (Fig. 2d) adopts 9 × 9 pin lattice, too, 72 pins plus the center beam box to hold a control rod [18]. The main design parameters are outlined in Table 1.

3 Calculation modeling and verification

3.1 RELAP5 model of SNCLFR-100

RELAP5 is a typical thermal–hydraulics system code in transient safety analysis. Figure 3a shows the RELAP5 nodalization for the primary system of SNCLFR-100. The major equipment and components considered include the core zone, the mixed up channel, the upper plenum, the heat exchangers, the annular down channel, the lower plenum, and the argon gas system. For simplification, the secondary side of the heat exchangers is modeled with the heat transfer pipe and proper inlet and outlet boundary conditions. The core active zone is lumped into four hydrodynamic channels, i.e., the average channel, the control rod channel (CR channel), the hot channel, and the bypass channel. Transverse interactions between the

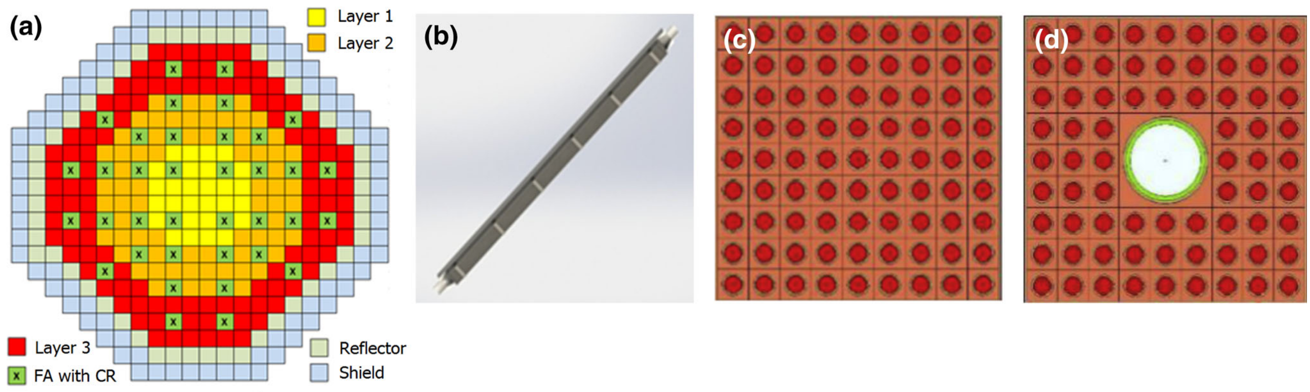

Fig. 2 (Color online) Scheme of SNCLFR-100 core. **a** Core layout, **b** 3D drawing of the assembly, **c** Fuel assembly, **d** Control rod assembly

Table 1 Main design parameters

Design parameters		*Fuel element dimensions*	
Thermal power (MW_{th})	100	Active zone height (mm)	1000
Refueling interval (*a*)	10	Equivalent core diameter (mm)	3460
Fuel assemblies	MOX fuel	Fuel pin diameter (mm)	9.8
Primary coolant	Lead	Cladding outer diameter (mm)	12.2
Secondary coolant	Water/steam	*Core kinetics*	
Steam generators	4 modules of straight shell-tube type	k_{eff} of BOL	0.99989
Inlet coolant temperature (°C)	400	Effective delayed neutron fraction (pcm)	368
Operating pressure	Barometric pressure	Prompt neutron generation time (s)	5.5206×10^{-7}
		Doppler feedback coefficient (pcm/°C)	− 0.19
		Axial expansion coefficient (pcm/°C)	− 0.17
		Coolant density feedback coefficient (pcm/°C)	− 0.42
		Coolant temperature feedback coefficient (pcm/°C)	− 0.49

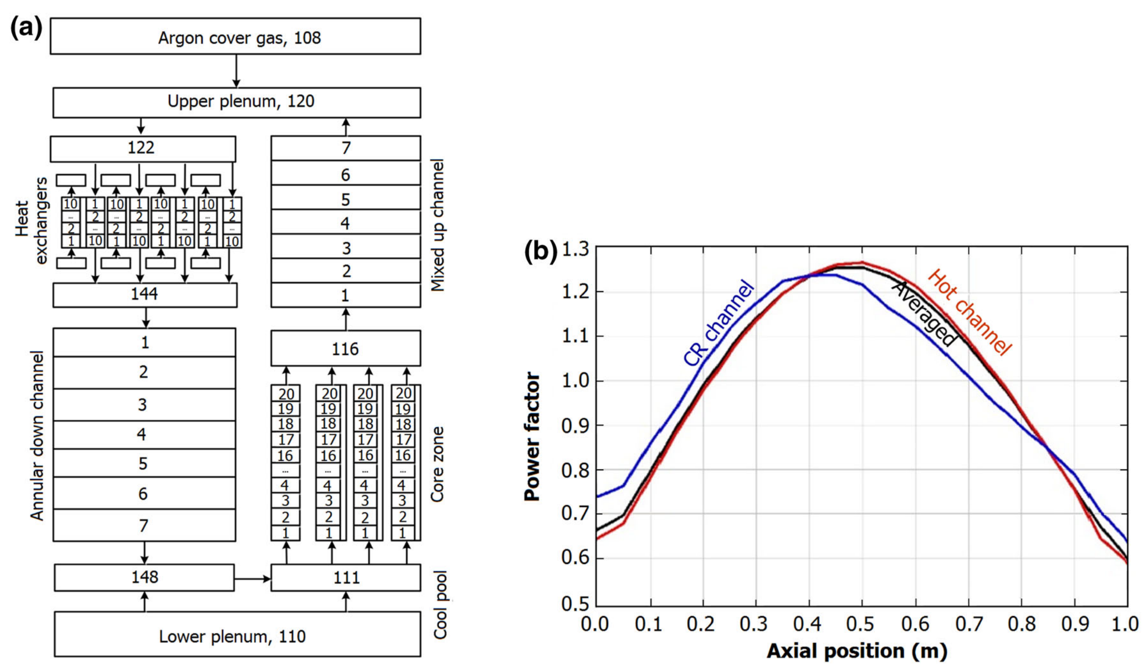

Fig. 3 (Color online) Model of SNCLFR-100 primary cooling system **a** Nodalization of RELAP5 and **b** Axial power density distribution

channels are not considered, because of closed fuel assembly. Each channel is divided into 20 volumes in axial direction. The axial power density distribution is shown in Fig. 3b. The hot channel is the hottest fuel assembly in the core with the radial power peaking factor of 1.40, which is chosen to carry out the flow blockage accidents.

3.2 Verification under steady-state condition

Steady-state calculation is performed to lay the groundwork for the following transient simulations of SNCLFR-100. The steady-state results under the full-power condition from RELAP5, and the design values [18], are listed in Table 2. The design values are calculated by LFR-SIN, a computational code based on single-channel model, developed by USTC for analyzing core thermal–hydraulic performance [19]. It can be seen that the maximum error between calculated and designed values is below 2%. It verifies that the steady-state results from RELAP5 are reliable, and the RELAP5 model for the primary system of SNCLFR-100 can be used for the transient analysis.

4 Transient analysis

The key parameters affecting the blockage transients include the blocked area fraction, the axial position of blockage, and the blockage time. To represent the flow blockage accident scenario, a RELAP5 motor valve component is set in the hot channel (Fig. 4). Blockages of different cross sections can be simulated by changing the valve stem position only. In a real situation, the flow blockage takes place gradually, which can be described well with a motor valve [20].

4.1 Transients with different cross-sectional fractions of the blockage

Consequences of the flow blockage accidents are analyzed with in Cases 0–11, defined as $\beta = 0.00$, 0.05, 0.10, 0.20, 0.30, 0.40, 0.50, 0.60, 0.70. 0.80, 0.90 and 0.95 ($\beta = A_{\text{blockage}}/A$, where A_{blockage} is the blockage cross section and A is cross section of the hot channel). The transient conditions are: (1) the reactor is in a nominal state before 50 s, (2) a sudden blockage occurs at the inlet of hot channel at 50 s; and the scram and safety systems are unavailable in the whole process.

Figure 5a shows total power level of the core changing with time at different β values. The impact of blockage accidents in the hot channel inlet on the total power was slight. At the beginning, the total power goes down immediately mainly due to the negative temperature coefficients of reactivity and then it increases to reach a stable value. As β increases, the power drop of the core becomes larger, with lower level of the final steady-state power and longer time to reach it. The point kinetic model was adopted in RELAP5; therefore, temperature changes in a single fuel assembly had limited influence on the total reactivity in the core, so the power drop was not so obvious.

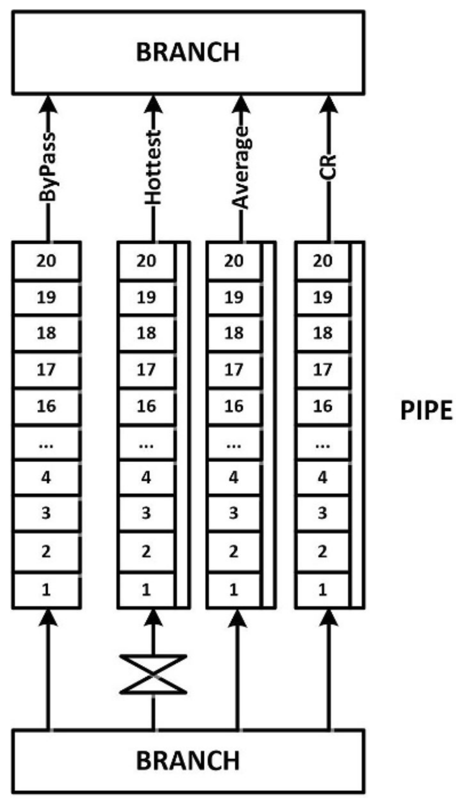

Fig. 4 Nodalization of the core zone for flow blockage

Table 2 Results of SNCLFR-100 in full-power steady-state condition

Parameters	Designed	Calculated	Error
Total power (MW_{th})	100	100	0
Mass flow rate in active zone (kg/s)	8528	8612	0.99%
Velocity of flow in active zone(m/s)	0.228	0.230	1.13%
Temperature at core inlet (°C)	400	400	0
Temperature at core outlet (°C)	480	480	0

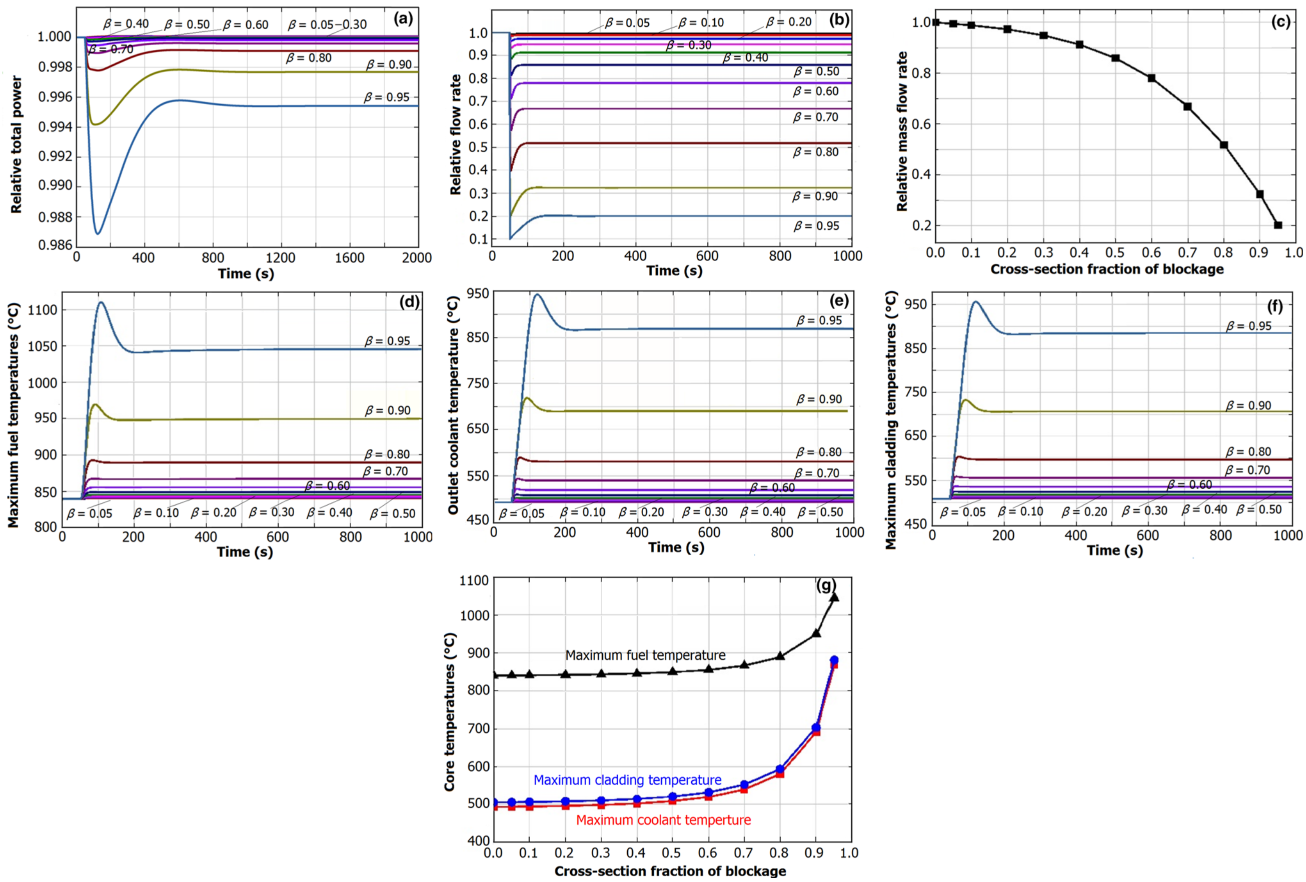

Fig. 5 (Color online) Simulation results of the thermal-hydraulic parameters in the hot channel at different fractions of flow area blockages

Table 3 Increase in outlet temperature in the hot channel

β	0.00	0.05	0.10	0.20	0.30	0.40	0.50	0.60	0.70	0.80	0.90	0.95
T_{in} (°C)	400	400	400	400	400	400	400	400	400	400	400	399
T_{out} (°C)	493	494	494	496	498	502	509	519	539	579	688	865
ΔT (°C)	93	94	94	96	98	102	109	119	139	179	288	466

In Fig. 5b, the flow rate trend in the hot channel changes with time in similar way as the total power. At the beginning, it decreases immediately and then increases to reached a new stable value, which is lower than the nominal value. When stable situation is achieved, the relative mass flow rate in the hot channel consistently decreased with increasing β (Fig. 5c). Compared to Case 0 (unblocked), the relative flow rate is 85.9, 51.8 and 20.0% at $\beta = 0.50$, 0.80 and 0.95, respectively.

In Fig. 5d, the maximum fuel temperature at different β values reaches a peak soon after the flow blockage accident, due to insufficient cooling capacity caused by the flow rate reduction. It recovers gradually to a new stable value. As β increases, the peak height, the steady-state value, and the time to reach it become greater. At $\beta = 0.95$, the maximum fuel temperature in the hot channel is 1110 °C at 106 s, below the safety limit of 2300 °C.

In Fig. 5e, the coolant outlet temperature rises to a peak, too, after the flow blockage accident, due to the flow rate decrease and the increase in fuel temperature. It recovers to reach a new stable value. The peak height, the steady-state value, and the time to reach it increase with β. At $\beta = 0.95$, the peak temperature is 942.9 °C at 120 s, below the boiling point of liquid lead.

The maximum cladding temperature (Fig. 5f) varies similarly. At $\beta = 0.95$, the peak temperature is 850 °C at 94 s, and 956.5 °C at 120 s, being less than the cladding melting temperature.

At new steady state, the core temperatures in the hot channel at different β values are shown in Fig. 5g, and the outlet temperature increases are given in Table 3. The core temperature increases are small at $\beta < 0.7$. Then, the temperature increase rate becomes fast. The maximum cladding temperature is out of the failure temperature at

Table 4 Results of blockage transients with blockages at different axial positions

Position	a	b	c	d	e	f	g
Power	0.99375	0.99375	0.99372	0.99368	0.99364	0.99361	0.99361
Flow rate	0.20225	0.20225	0.2014	0.2002	0.19904	0.19827	0.19827
T_{max}-fuel (°C)	1043.5	1043.5	1044.5	1046.1	1047.5	1048.6	1048.6
T_{max}-coolant (°C)	864.4	864.4	866.4	869.2	871.9	873.8	873.8
T_{max}-clad (°C)	880.8	880.8	882.7	885.6	888.3	890.1	890.1

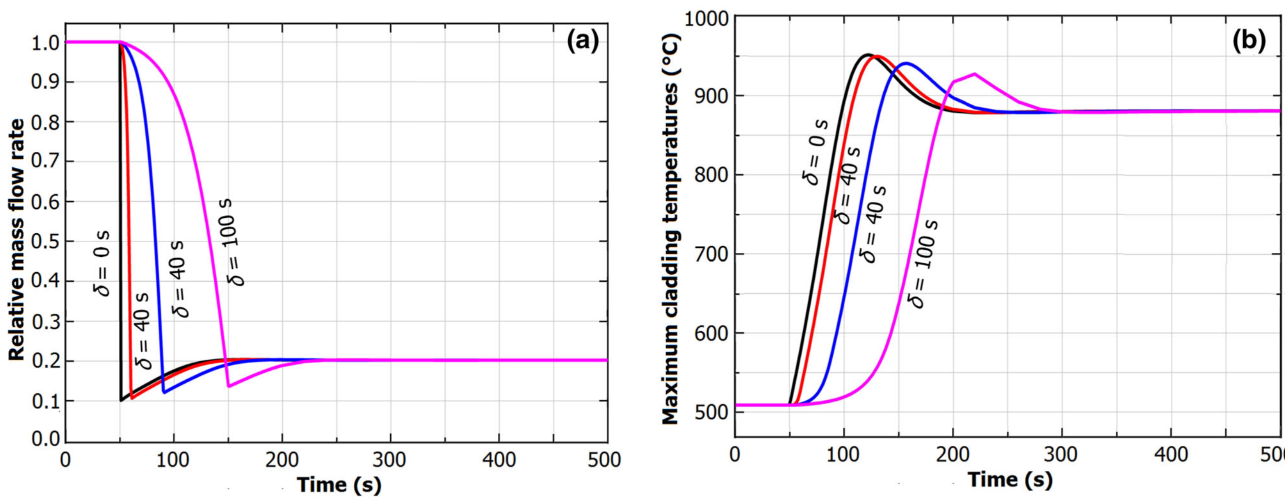

Fig. 6 (Color online) Relative mass flow rate (a) and maximum cladding temperature (b) at δ = 0–100 s in the hot channel, as a function of time

$\beta > 0.94$, because the loss of coolant flow is caused by obstruction of the hot channel, with a decreased Reynolds number. When the Reynolds number reduces to the upper critical value, the heat transfer became laminar convection from turbulent convection. Accordingly, the heat transfer between the fuel assembly and the coolant deteriorated sharply, hence the large increment in the maximum core temperatures with a large blocked area fraction.

4.2 Transients at different positions

Seven axial positions in the hot channel are chosen to analyze the flow blockage accidents: (a) core inlet, (b) active zone inlet, (c) quarter of the active zone from inlet, (d) middle of the active zone, (e) quarter of the active zone from outlet, (f) core outlet, and (g) active zone outlet. The transient conditions are as follows: (1) the reactor is in a nominal state before 50 s; (2) a sudden blockage at $\beta = 0.95$ occurs at Positions (a)–(g) at 50 s; and (3) the scram and safety systems are unavailable during the whole process.

Table 4 shows the results of blockage transients in the hot channel with blockages at different axial positions, after the primary cooling system recovers to stable status. When an axial position is located closer to the core outlet, the flow rate is smaller, the core temperature is higher, and the core power is lower. The calculated blockage transients between accidents at Positions (a) and (b), and Positions (f) and (g), are the same. This indicates that the consequence of blockage accident in a single fuel assembly becomes more serious as the position of blockage is closer to the core outlet. However, changing the blockage positions in inactive zone has little effect on the results.

4.3 Transients in different blockage-developing durations

This time effect is simulated with a motor vale. The blockage-developing durations are $\delta = 0$, 10, 40, and 100 s. The transient conditions are: (1) the reactor is in a nominal state before 50 s, (2) blockages of different durations, at $\beta = 0.95$, occur in the inlet of the hot channel at 50 s, and (3) the scram and safety systems are unavailable during the whole process.

Transient variations of the flow blockages taking place gradually in the different blockage-developing durations are shown in Fig. 6. As δ increases, the accident progress becomes slower, with increased minimum flow rate and decreased of peak value of the maximum cladding temperature. At all the δ values, the relative mass flow rate and the maximum cladding temperature recover to virtually the same stable value, i.e., the blockage-developing time affects the accident progress only. A shorter blockage-developing time leads to a more serious consequence. Thus, a sudden blockage is the most severe blockage accident in a single fuel assembly.

5 Conclusion

The blockage transient, where the flow is blocked in the hottest fuel assembly in SNCLFR-100, has been analyzed by using the system code RELAP5. The reactivity feedback and axial power profile have been taken into account in the current studies. Key parameters affecting the blockage transients, i.e., the cross-sectional fraction of blockage (β), the axial position of blockage, and the blockage-developing time (δ), are investigated. The main results can be summarized as follows:

For small blockages ($\beta < 0.5$), the flow rate decrease and the increase in core temperatures in the hottest fuel assembly are slight; for large blockages ($\beta > 0.7$, especially), the flow rate decrease and temperatures increase are much more significant.

At $\beta = 0.95$, the sudden blockage accident in a single fuel assembly is the most serious, and the closer is the blockage to the core outlet, the more severe consequence, but little position effect can be observed for blockages in inactive zone.

The blockage-developing time only affects the accident progress. A short δ in inlet of the hottest fuel assembly will cause serious consequence, but the δ has no effect on the asymptotic results of transients.

The biggest challenge is the cladding material failure. At $\beta > 0.94$, the flow blockage accident can affect integrity of the fuel assembly, because the clad temperature exceeds its failure limit.

The 3-D modeling shall include heat transfer between adjacent fuel assembly channels.

The analysis procedure in this study can be applied to other research reactors for studying similar transient behaviors.

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