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Analysis of core blockage scenarios during pump shutdown accidents for small size lead-cooled fast reactor using RELAP5-HD

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ABSTRACT

Two models of a designed small size lead-cooled fast reactor (LFR) were built by RELAP5-HD to simulate the behavior of the reactor system during pump shutdown accidents, along with different levels of core blockage. Two typical pump shutdown accidents of pump rotor seizure accident and loss-of-pump-power accident were selected to do the simulation and analysis of the behavior of the nuclear power reactor during pump shutdown accidents, when all the safety systems were assumed to be unavailable. A one dimensional vessel-one dimensional core model was built to simulate the long term cooling condition of the pump shutdown accidents under the reactor shutdown or operation conditions. The simulation results would help to analyze whether pump accidents could cause core damage. In order to show the heat removing capability of the coolant in the reactor system, a more detailed one dimensional vessel-three dimensional core model was built to simulate the long term cooling condition of the pump shutdown accidents along with different levels of core blockage accidents.

1. Introduction

Lead cooled fast reactor (LFR) was selected as one of the six Generation-IV reactor concept candidates which was expected to become the first to realize industrial demonstration of the Generation-IV nuclear power system (Y. Wu et al., 2014, 2015). With excellent neutron economy (Y. Wu et al., 1999, 2014), high transmutation efficiency, high power density and low pressure operation, LFR could be widely applied to nuclear fuel breeding (Y. Wu and FDS Team, 2011; Y. Wu, 2013), nuclear waste transmutation (Y. Wu et al., 1999, 2000, 2011), efficient power generation and other field (Q. Huang et al., 2004, 2011; J. Yu et al., 2004; Y. Wu et al., 2016a,b). Chinese Academy of Science (CAS) launched an engineering project to develop Accelerator Driven System (ADS) for nuclear waste transmutation in 2011. China Lead-based Reactor (CLEAR) was selected as the reference reactor in this ADS project. As a lead organization, Institute of Nuclear Energy Safety Technology (INEST) is in charge of the design and development of a series of lead cooled fast reactors.

One of the designed LFR is a 10MWth lead-bismuth cooled research reactor, with the primary coolant loop in a pool cycling by two mechanical pumps. The secondary loop, using pressurized water as coolant, contains of two loops with four straight flow heat exchangers connecting to the primary loop. The end heat sink is two air coolers. The coolant cycling schematic diagram shows in Fig. 1. Moreover, special configuration of the inlet holes on the fuel assembly spike which the coolant can flow through plays a great role in removing the whole core heat during the core blockage scenarios. The FA spike schematic diagram was listed in Fig. 2(G. Zhang et al. (2017). Meanwhile, the designed LFR has critical and sub-critical dual-mode operation capability for ADS transmutation system and fast reactor technology validations, respectively (G. Wu et al., 2015).

The blade of the main pump may fall off and reach the reactor core after the main pump shaft accident, which can cause the blockage accident. As a result, in this paper, two models were built by RELAP5-HD to simulate the behavior of the reactor system during pump shutdown accidents, along with different levels of core blockage (G. Zhang et al., 2015). Two typical pump shutdown (accidents of pump rotor seizure accident and loss-of-pump-power accident) were selected to do simulation and analysis of the behavior of the reactor system during pump shutdown accidents, when all the safety systems were assumed been unavailable. The results of the simulation would help to analyze whether the typical pump accidents could result in core damage. In order to show the heat removing capability of the coolant in the nuclear power reactor, a 1D vessel-3D core model was built to simulate the long term cooling condition of the pump shutdown accidents along with different hypothesized core blockage accidents.

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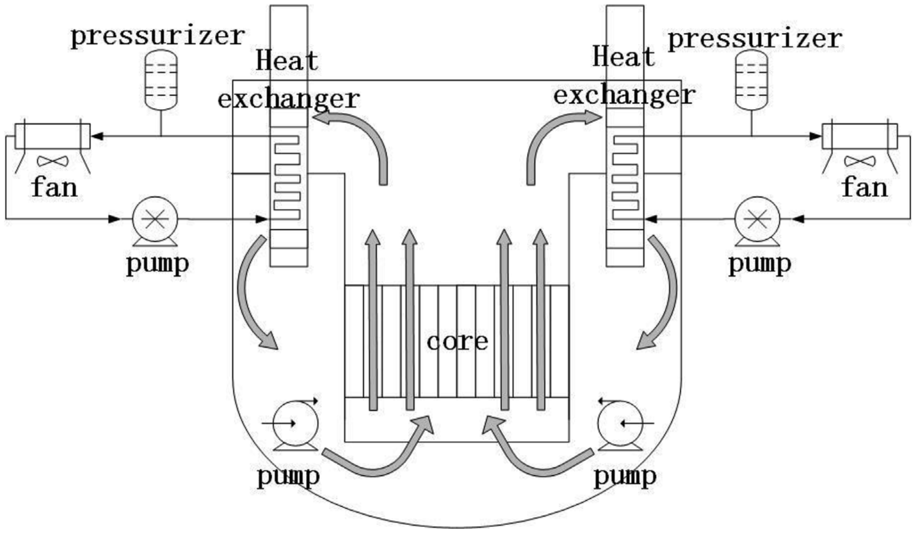

Fig. 1. Coolant cycling schematic diagram.

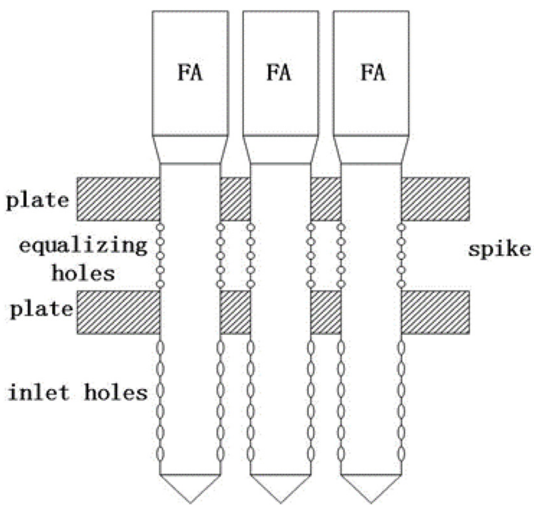

Fig. 2. FA spike schematic diagram.

2. Model description

2.1. Main parameters of the reactor

The main thermal design parameters and neutron parameters which would be used in the RELAP5-HD model are presented in Tables 1 and 2 (G. Zhang et al., 2017 and G. Wu et al., 2017).

Table 1
Lead cooled reactor thermal design parameters.

Items	value
Power (MW)	10
Fuel	UO2
Fuel assembly	86
Length of the FA/mm	1675
Length of the spike/mm	390
Inner diameter of the spike/mm	80
Inner diameter of the equalizing hole/mm	10
Primary coolant	Lead-bismuth
Core inlet/outlet temperature (°C)	300/385
Primary coolant mass flow (kg/s)	811.67
Secondary coolant	Pressurized liquid water
Secondary inlet/outlet temperature (°C)	215/230

Table 2
Lead cooled reactor neutron parameters.

Items	value
Effective delayed neutron fraction (pcm)	706
Doppler constant (pcm/K)	0.44
Coolant expansion coefficient (pcm/K)	0.53
Axial core expansion coefficient (pcm/K)	0.072
Radial core expansion coefficient (pcm/K)	0.788

2.2. RELAP5-HD model description

The RELAP5-HD thermal-hydraulic model solves three field equations for three primary dependent variables for single-phase flow. The primary dependent variables are pressure (P), specific internal energies (U), and velocities (v). For the one-dimensional equations, the independent variables are time (t) and distance (x). For the multi-dimensional equations, the independent variables are time (t) and distance (x, y, z for Cartesian; r, h, z for cylindrical.) The basic field equations for the RELAP5 model consist of continuity equations, momentum equations, and energy equations (Ransom, 1989).

Two models were built by RELAP5-HD to perform the simulation. The 1D vessel-1D core model was set up to perform the simulation of the pump shutdown accidents during the reactor shutdown or operation scenarios (N. Kolev et al., 2006). This model was selected because it can minimize the simulation time with the appropriate chosen of some flow paths and the nodalization.

Moreover, detail simulations were performed to analyze the thermal-hydraulic of the reactor system during the core blockage for different cases with the 1D vessel-3D core model. The simulation of the reactor core with three-dimensional modules could be conducted by the model, which allowed a more accurate description of the core blockage under different scenarios.

An overall structure of the full designed LFR system RELAP5-HD 1D vessel-1D core model nodalization adopted for the simulation is presented in Fig. 3.

The primary loop mainly contains the core (113, 123, 133), four heat exchangers (141, 142, 143, 144), two pumps (171,172) and other components. One-dimensional components were selected to perform the simulation of all the regions of the nuclear power reactor. Especially, the model of the reactor core was built by using three one-dimensional vertical pipe components where pipe 113 for the hot channel, pipe 123 for the average channels, and pipe 133 for the bypass channel containing the shielding layer and reflecting layer. Pipe 113

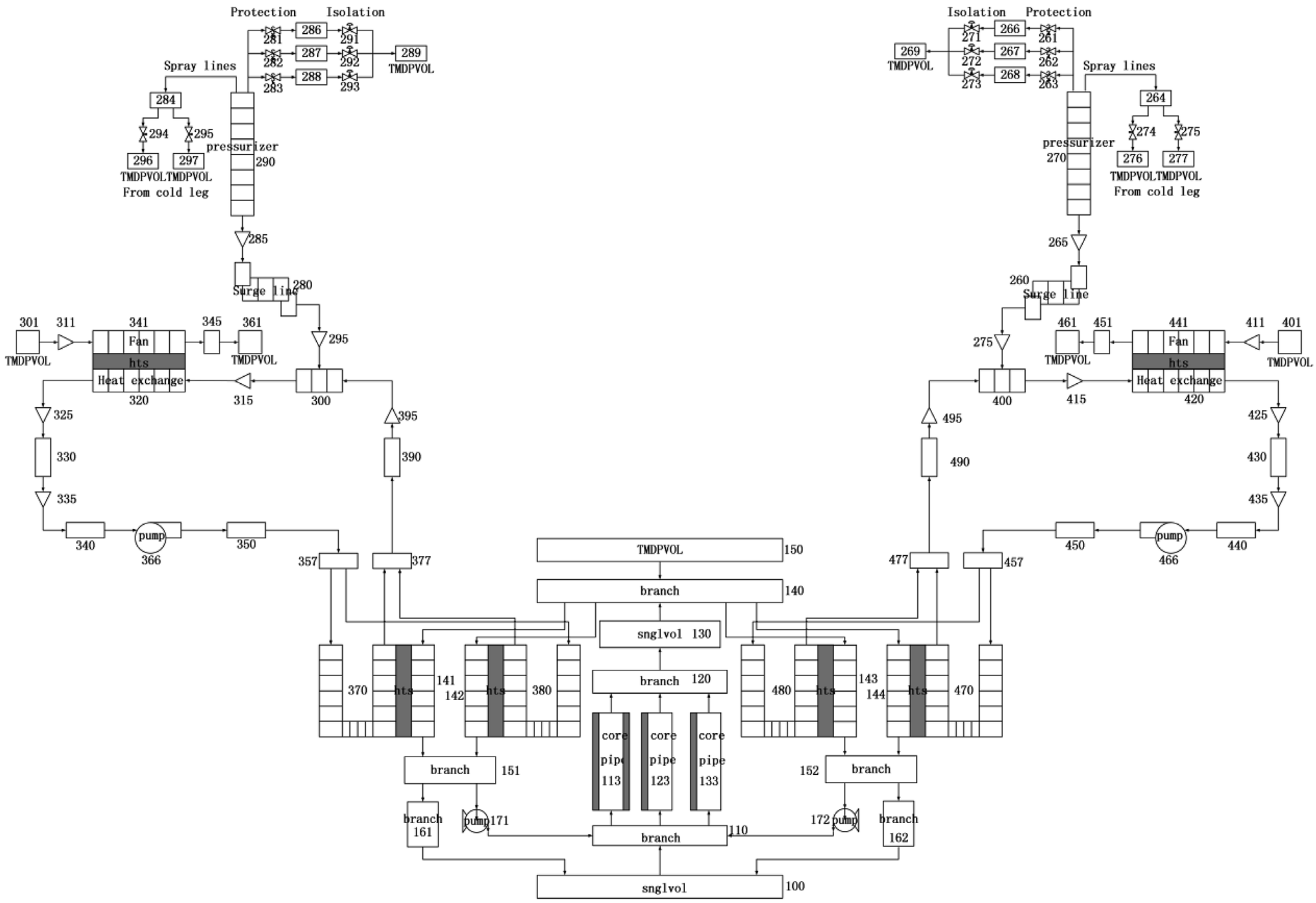

Fig. 3. 1D Nodalization of reactor system model.

which used to represent for the hot rod and other rods in the hot channel had been connected to two heat structures. One heat structure was connected to pipe 123 and another one was connected to pipe 133. The distribution of all reactor in the average channel had adopted proper radial peaking factors, so as well in the hot channel. The primary and secondary sides of the straight flow heat exchangers were simulated using pipe. Six axial nodes were used for the primary side of the heat exchangers (141,142,143,144) and sixteen axial nodes for the secondary side.

The secondary loop consists of two loops, mainly containing four heat exchangers (370, 380, 470, 480), two pumps (366, 466), two pressurizers (270, 290), two fans (320, 420) and other components.

The 1D vessel-3D core model was similar to the 1D vessel-1D core model except the part of the core. The nodalization of the 3D core model was in more detail description shown in Fig. 4 and Fig. 5 (A.Shkarupa et al., 2009). The core simulation was performed by using seven cylindrical three-dimensional modules (500, 501, 511, 521, 531, 541, 551). Therefore, 86 fuel channels, which were divided into 12 axial nodes respectively, could be represented by this nodalization for more accurate coolant flow. The power generated in each fuel channels was simulated by definition of ninety four heat structures in which a real power distribution had been imposed to represent the fuel elements.

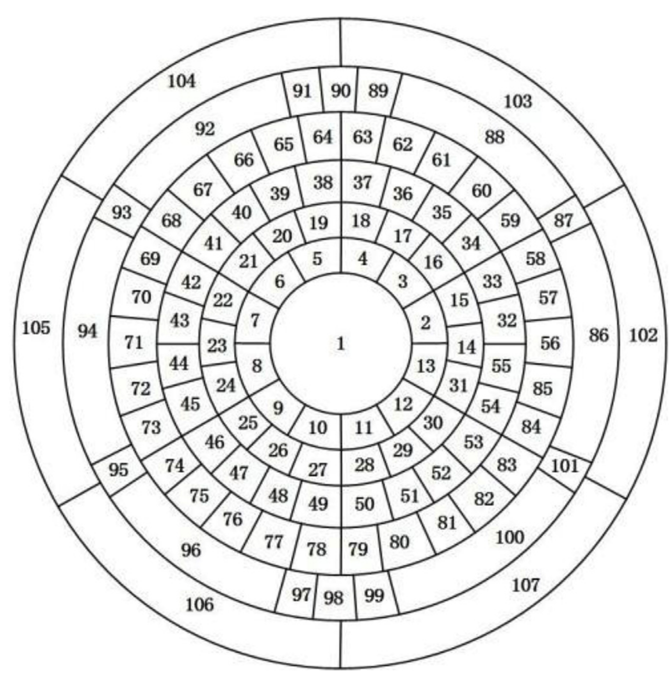

Fig. 4. Radial nodalization of 3D core model.

3. Results and discussion

3.1. The basic scenarios with 1D vessel-1D core model

As previously stated, the simulations of the basic scenarios were performed by the simpler 1D vessel-1D core model. Firstly, the reactor run in a steady-state at some point, and then the introduction of the pump rotor seizure accident or loss-of-pump-power accident would

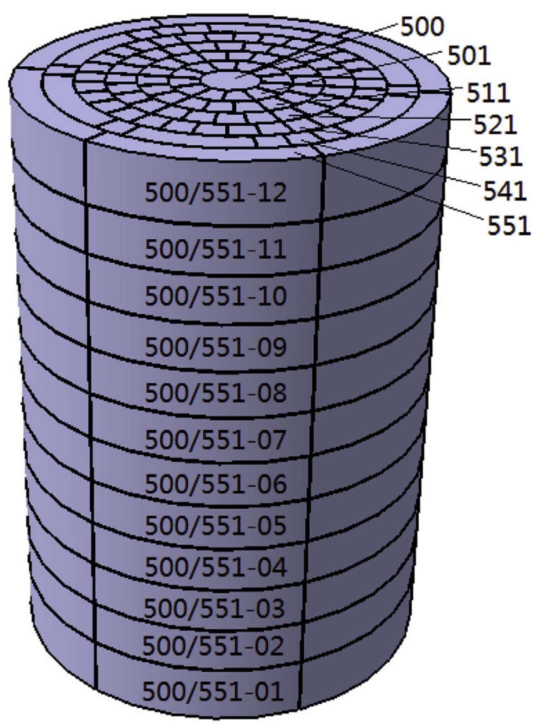

Fig. 5. Nodalization of 3D core model.

result in a new case which resumed from the antecedent step. . At the meantime, the reactor shutdown curve was brought in or not through the kinetic model of RELAP5-HD representing the reactor shutdown or operation. The simulation was extended for approximately 24 h after the accidents occurred. The parameters including reactor power, core inlet temperature, core outlet temperature, maximum peak cladding temperature and primary loop coolant flow rate were selected as results of the simulations, (Rodolfo Vaghetto and Yassin A.Hassan, 2006).

The maximum peak cladding temperature was an important thermal-hydraulic parameter that selected as deterministic factor to analyze whether a pump accident may lead to core damage. More detail parameters were chosen for plotting and analysis to make a better understanding of the appearance occurred after pump shutdown accidents. The results of the four cases were shown in Table 3, in which each case was identified as pass or fail depending on the maximum peak cladding temperature reached after pump shutdown accidents. The case which was defined as pass signified the maximum peak cladding temperature (PCT) would never exceed 800K (assumed as a peak cladding temperature limit value) after the induction of pump shutdown accidents.

The pump shutdown accidents, including the pump rotor seizure accident and the loss-of-pump-power accident, are the typical accidents in the running of the reactor. In the above cases, the coolant flow rate of the primary loop would decline rapidly which could result in no enough coolant flow for cooling to the top of the core. Then the maximum peak cladding temperature would continue increasing after accidents occurred. Thanks to the inherent designed characteristic of the LFR, the natural circulation was found to be established which could stop increasing the maximum peak cladding temperature by providing enough

Table 3
1D vessel-1D core model simulation results summary.

Accident/reactor shutdown or operation	Shutdown (PCT)	Operation (PCT)
pump rotor seizure accident	pass (714K)	pass (753K)
loss-of-pump-power accident	pass (721K)	pass (756K)

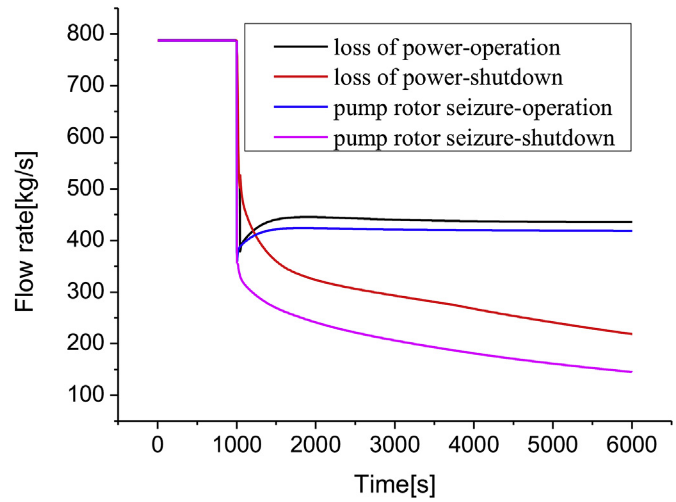

Fig. 6. Mass flow rate.

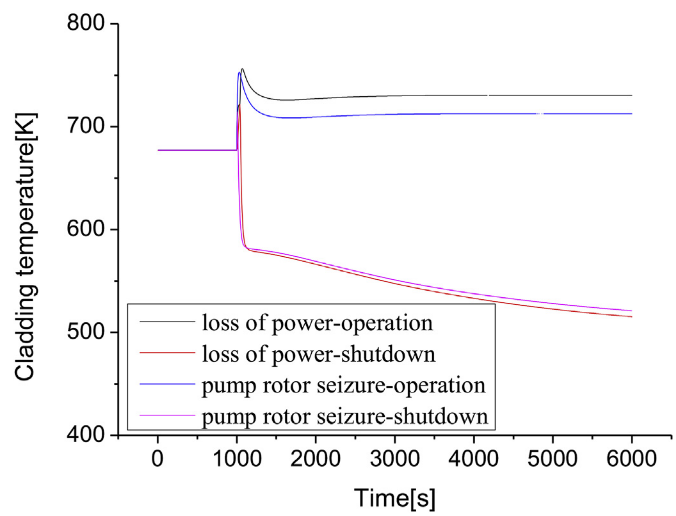

Fig. 7. Maximum peak cladding temperature.

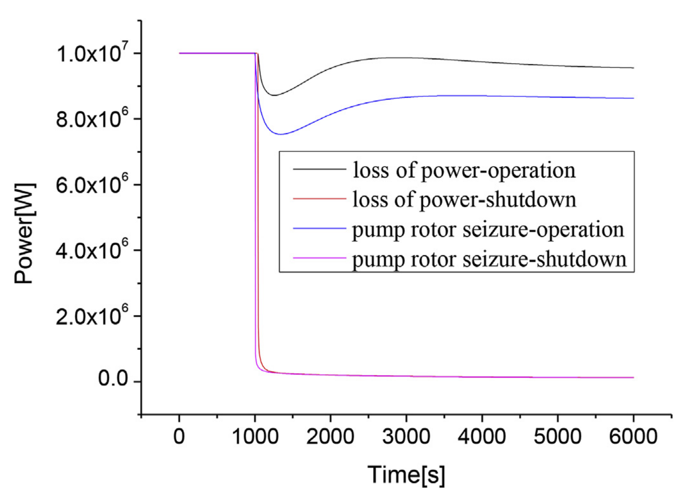

Fig. 8. Total power of the reactor.

coolant flow for the pump shutdown accidents during reactor operation and shutdown.

Figss. 6–9 showed the detail progress of the accident. In the reactor not shutdown case, when the pump shutdown accident occurred, the

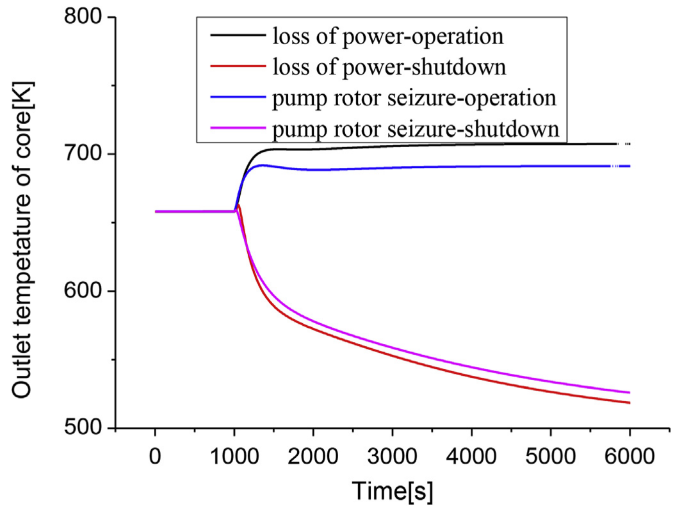

Fig. 9. Outlet temperature of the core.

flow of both loss-of-pump-power accident and pump rotor seizure accident had a rapid decrease, the maximum peaking cladding temperature suddenly rose. Due to the natural circulation established, the maximum peak cladding temperature gradually decreased to a steady state after the peak value, however, the maximum peak cladding temperature under 800 K during the whole progress. The total power of the reactor had a fluctuation because of the feedback of the coolant and fuel. Similarly, in the reactor shutdown case, though the natural circulation flow rate was low, the decay heat was removed effectively to guarantee that the maximum peak cladding temperature would not exceed the limit value.

3.2. Core blockage scenarios with 1D vessel-3D core model

Different core blockage accidents were performed with a more detailed core using the 1D vessel-3D core model. A simulation had been made with the 1D vessel-1D core model, and the results showed excellent safety characteristics of the reactor when the pump shutdown accident occurred. From the simulation results, we achieved that the pump rotor seizure accident may bring more serious consequence than the loss-of-pump-power accident. After, additional simulations with 1D vessel-3D core model were performed to study the core coolant ability under different levels of core blockage accidents during the pump rotor seizure accident time (Rodolfo Vaghetto and Yassin A.Hassan, 2006). The below cases were selected to analyze these accidents:

1. Full core Blockage + Bypass Blockage
2. Full core Blockage + Bypass Free
3. Full core Free + Bypass Free
4. Full core Free + Bypass Blockage
5. 1/2 Full core Free + Bypass Blockage
6. 1/3 Full core Free + Bypass Blockage
7. 1/6 Full core Free + Bypass Blockage

In order to study whether the decay heat could be removed effectively, case 2–7 only with bypass flow or limited core flow were selected to perform the simulations. Case 4–7 were selected to study the effect of different levels of core blockage when the pump shutdown accidents occurred. The scheme shown in Table 4 summarized the results for the seven cases simulated.

Case 1 presented that with the full core blockage and core bypass blockage, the maximum peak cladding temperature would continue increasing and exceed the limit value soon, which would finally result in core damage.

Case 2 presented that the core bypass flow provided enough coolant flow effectively to reach to the top of the core, which could decrease the maximum peak cladding temperature even with the full core blockage.

Case 3 presented no risk of the core damage, obviously.

Case 4–5 presented the decay heat was removed effectively by limited core flow to guarantee that the maximum peak cladding temperature would not exceed the limit value.

Case 6–7 presented that with more than 1/2 core blockage and core bypass blockage, the maximum peak cladding temperature would continue increasing and exceed the limit value soon which would finally result in core damage.

Analysis of the maximum peak cladding temperature of case 1–7 was plotted and represented in Fig. 10.

Table 4
1D vessel-3D core model simulation results summary.

Case	Description	PCT/K	Result
1	Full core Blockage + Bypass Blockage	–	Fail
2	Full core Blockage + Bypass Free	739	Pass
3	Full core Free + Bypass Free	714	Pass
4	Full core Free + Bypass Blockage	745	Pass
5	1/2Full core Free + Bypass Blockage	779	Pass
6	1/3Full core Free + Bypass Blockage	840	Fail
7	1/6Full core Free + Bypass Blockage	871	Fail

4. Conclusions

Two models were built by RELAP5-HD to simulate the behavior of the designed small size LFR under the pump shutdown accidents. In particular, the 1D vessel-1D core model was adopted to simulate the condition of the long term cooling under the pump rotor seizure accident and the loss-of-pump-power accident with the reactor whether shutdown or operation. The simulation results identified that during the two typical pump shutdown accidents, the peak cladding temperature always under 800K with the reactor shutdown or operation, showing an excellent inherent safety characteristics of the reactor. Different levels of core blockage scenarios were assumed for additional cases simulated with the 1D vessel-3D core model for the pump rotor seizure accident. The results showed that except for the full core blockage and bypass blockage case, the different levels of core blockage and bypass blockage scenarios were found efficient to remove the decay heat to guarantee that the maximum peak cladding temperature would not exceed the limit value.

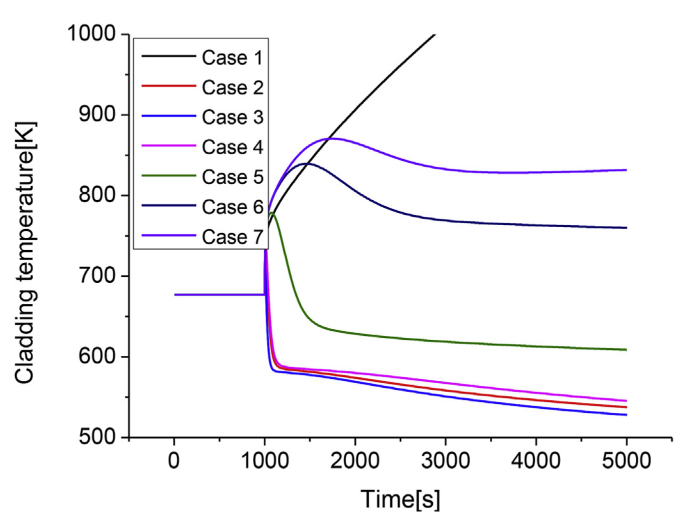

Fig. 10. 1D vessel-3D core model simulation results - peak cladding temperature.

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