NUCL SCI TECH (2017) 28:10
DOI 10.1007/s41365-016-0154-y

Analysis of SBLOCA on CPR1000 with a passive system

Zi-Jiang Yang[1] · Jun-Li Gou[1] · Jian-Qiang Shan[1] · Pan Wu[1]

Received: 29 December 2015 / Revised: 1 March 2016 / Accepted: 3 March 2016 / Published online: 1 December 2016

Abstract Since the Fukushima accident in 2011, more and more attention has been paid to nuclear reactor safety. A number of evolutionary passive systems have been developed to enhance the inherent safety of reactors. This paper presents a passive safety system applied on CPR1000, which is a traditional generation II+ reactor. The passive components selected are as follows: (1) the reactor makeup tanks (RMTs); (2) the advanced accumulators (A-ACCs); (3) the passive emergency feedwater system (PEFS); (4) the passive depressurization system (PDS); (5) the in-containment refueling water storage tank (IRWST). The model of the coolant system and the passive systems was established by utilizing a system code (RELAP5/MOD3.3). The SBLOCA (small-break loss of coolant) was analyzed to test the passive safety systems. When the SBLOCA occurred, the RMTs were initiated. The water in the RMTs was then injected into the pressure vessel. The RMTs' low water level triggered the PDS, which depressurized the coolant system drastically. As the pressure of the coolant system decreased, the A-ACCs and the IRWST were put to work to prevent the uncovering of the core. The results show that, after the small-break loss-of-coolant accident, the passive systems can prevent uncovering of the core and guarantee the safety of the plant.

Keywords Passive safety systems · RELAP5/MOD3.3 · CPR1000 · SBLOCA

This work was supported by the National High-tech R&D Program of China (No. 2012AA050905).

✉ Jian-Qiang Shan
jqshan@mail.xjtu.edu.cn

Zi-Jiang Yang
yangzijiang@stu.xjtu.edu.cn

[1] School of Nuclear Science and Technology, Xi'an Jiaotong University, Xi'an 710049, China

1 Introduction

After the Fukushima accident, public safety requirements for nuclear power plants became much stricter, so that generation III nuclear power plants were constructed instead of prior generations. It is obvious that the safety systems of most existing reactors cannot satisfy the safety requirements for the generation III NPP [1]. Consideration has been given to adopting passive technologies to enhance reactors safety. According to the International Atomic Energy Agency, passive technologies are those that utilize natural forces, such as natural circulation and gravity. Furthermore, the passive systems should be composed totally of passive structures and components [2].

The advantages of passive safety systems are as follows. Firstly, when normally functioning, passive safety systems are not dependent on external power or pumps [3], which lowers the risk faced by the reactor during some accidents (such as SBO). Secondly, operator action is not needed for the successful performance of passive safety systems [4], which rules out human errors. Thirdly, when compared with "active" safety systems, passive safety systems are more economical because of their relatively concise layout.

Recently, passive safety systems have been widely applied in advanced reactor designs, such as the AP1000 in the USA, the Next Generation PWR in Japan, the WWER-1000 in Russia [5], the ESBWR in Europe, IRIS [6], and SMART [7], among others. Among these, the AP1000 reactor [8] adopts an entirely passive system, including an emergency core-cooling system, safety injection, etc. Its technology is relatively mature, and the AP1000 has already been constructed in China. The Next Generation PWR [9], which is proposed in Japan, is equipped with a passive safety system, one that acts as a backup to prevent damage when the standard safety system does not operate well. Its passive system consists of an automatic depressurization system, advanced accumulators [10], and a primary gravity injection system. The advanced accumulator method has also been applied in the APR1400, which contains multiple passive technologies, including safety injection systems, an IRWST, and a safety depressurization and vent system. This method has been adjusted specifically for the CPR1000 by Hu [11] in 2014. Unlike the AP1000, the APR1400 adopts a secondary passive heat-removal system, which is also used to improve the CPR1000 technology [12]. Some experimental research [13, 14] has already been conducted on this heat-removal system for the CPR1000. There is also other research being done on passive safety systems. Gavrilas [15] has presented some containment passive cooling design concepts. Chang [16] designed an integrated passive safety system to ensure ultimate safety. Achilli [17] designed two passive safety systems for light water reactors. A semipassive containment cooling system was designed by Byun [18] for a large-scale concrete containment. A passively cooled containment was designed by Gavrilas [19] for a high-rating PWR.

Although passive systems have become a common characteristic in designing advanced NPPs [20], they face the risk of functional failure as well [21]. It has been recognized that the combination of standard safety systems and passive safety systems will be the most optimized method of enhancing the safety of a reactor. This method can be applied in upgrades of existing mature reactor designs, which are mostly equipped with "active" safety systems.

This paper describes the application of a passive safety system, as a backup system for the original standard system, on the CPR1000 plant to enhance the plant's safety. As shown in Fig. 1, Chinese Pressurized Reactor 1000 (CPR1000) is a PWR with three loops that adopts proven technology. When the reactor is operated normally, the coolant is heated by the fission power and flows into the steam generator. After cooling by the steam generator, the coolant is pumped into the reactor vessel again.

According to the EPRI URD, the passive safety system should meet following requirements:

1. The safety system should have the ability to mitigate design-based accidents (DBAs).
2. The system should maintain safe shutdown conditions for at least 72 h without operator action or AC power in a DBA.
3. The annual core damage frequency should be less than or equal to 10^{-5} events per reactor year.
4. The safety system should have the ability to mitigate severe accidents.
5. The initiation of PDS valves should be avoided during non-LOCA accidents.

Utilizing the RELAP5/MOD3.3 code, an analysis of SBLOCA of this passive CPR1000 plant was carried out to illustrate whether the passive systems could mitigate the accident consequences. The SBLOCA accident was chosen for analysis because it triggered all components of the passive safety system.

2 Description of the proposed passive system

The components of the passive safety system are as follows: (1) the reactor makeup tanks (RMTs); (2) the advanced accumulators (A-ACCs); (3) the in-containment refueling water storage tank (IRWST); (4) the passive emergency feedwater system (PEFS), which are installed on the secondary side of the SGs; (5) the passive depressurization system (PDS).

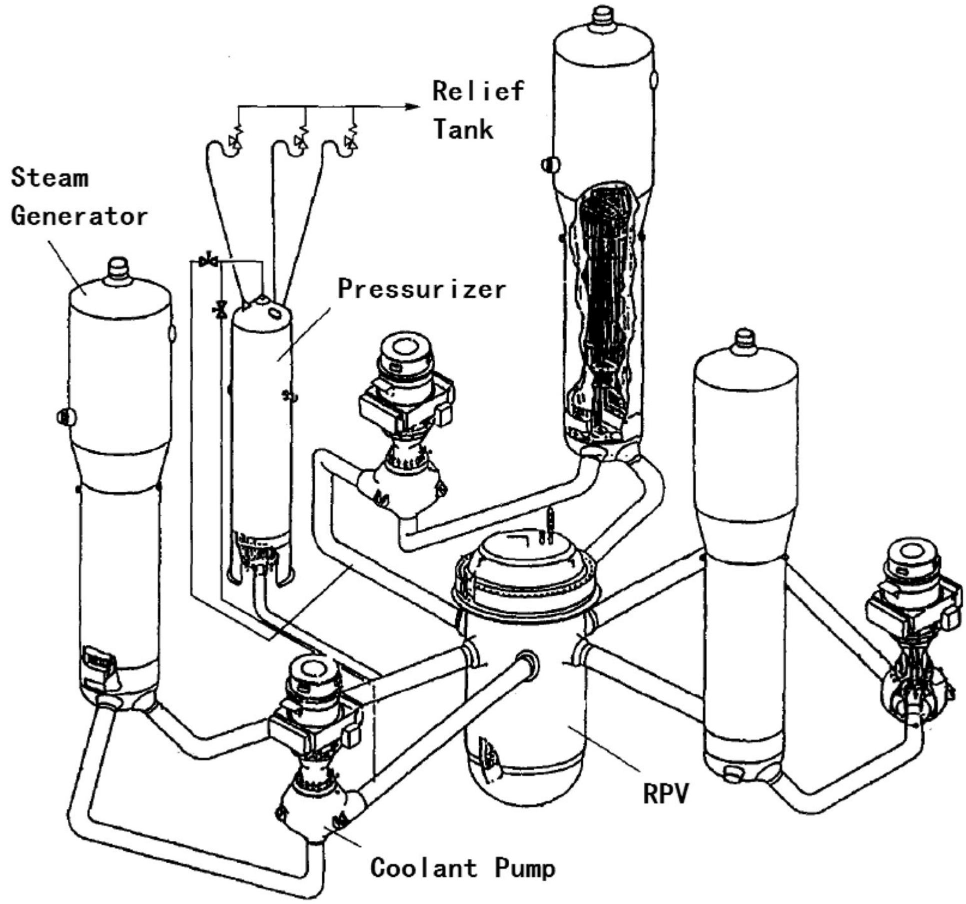
2.1 Reactor makeup tank (RMT)

Each loop contains a reactor makeup tank. The RMT (Fig. 2), filled with boron water, is located above the RCS loops. Its inlet pipe, which keeps the system pressure balance, leads to one of the cold legs. The outlet pipe of the RMT leads to the direct vessel injection line (DVI). Boron water will be injected into the RPV downcomer through the DVI, when the RMT is triggered. The temperature of water in the RMT is around 323 K. The RMT's height is 6 m, and its volume is 50 m^3.

2.2 Advanced accumulator (A-ACC)

Each loop contains an A-ACC. A flow rate control device is located in the lower portion of an advanced accumulator (Fig. 3). When the A-ACC is initiated, water flows directly through the outlet line to a DVI pipe. If the water level decreases below the standpipe, its flow channel is broken. Then the flow switches to the side line. A vortex is set up in this path, resulting in a reduced flow rate. The injection period is then prolonged by using this A-ACC technology. The A-ACCs will be put into operation when the system pressure drops below 4.5 MPa. The internal structure of the A-ACC is shown in Fig. 4. Its total volume is 90 m^3, its water volume is 72 m^3, and the standpipe is 1.4 m in height.

Fig. 1 Schematic of CPR1000 coolant system



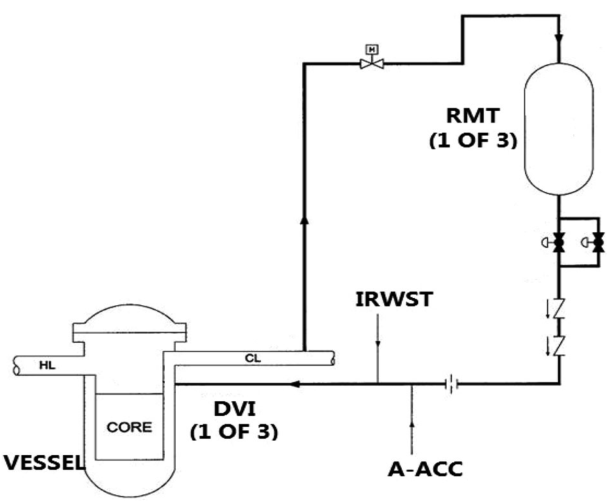


Fig. 2 Schematic of RMT

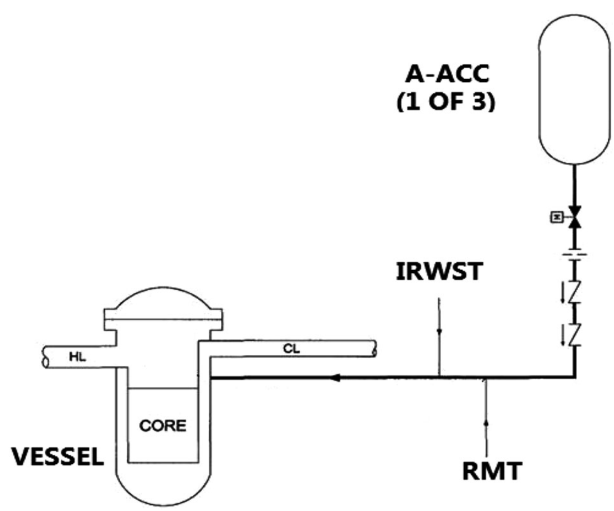


Fig. 3 Schematic of A-ACC

2.3 In-containment refueling water storage tank (IRWST)

The IRWST, as shown in Fig. 5, provides long-term injection water after the depressurization of the RCS. This large water tank is located above the vessel and is isolated from the coolant system by check valves. There are three pipes, connecting the bottom of IRWST to the DVI. The water level in the water tank is 8.5 m in height; the water volume is 2000 m^3.

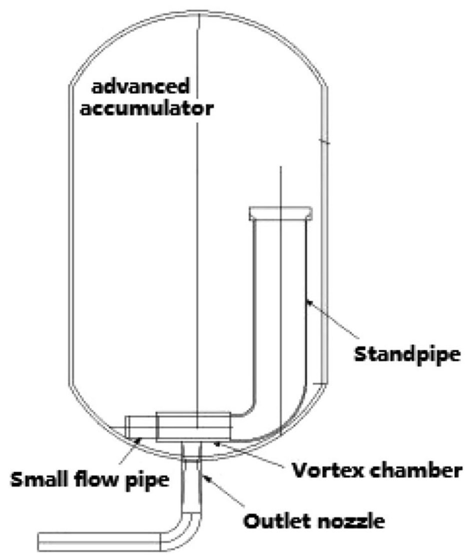


Fig. 4 Advanced accumulator

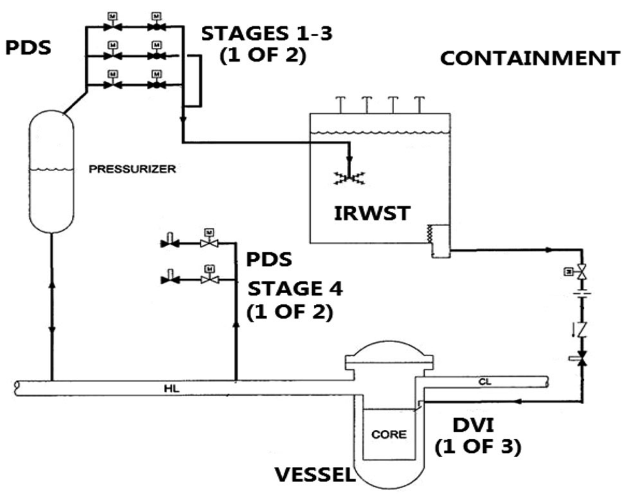


Fig. 5 Schematic of IRWST and PDS

2.4 Passive emergency feedwater system (PEFS)

There are three PEFSs connected to three SGs. The PEFS (Fig. 6) is placed on the secondary side of the SG. The PEFS is composed of heat exchangers, an outside water tank, and several pipes. When it is put into operation, the steam in the SG goes into the heat exchanger. The vapor is condensed in the heat exchanger and goes into the SG to cool down the primary system. There are two parallel isolation valves on each PEFS. The heat transfer area of each heat exchanger is 100 m^2. The water tank contains 2500 m^3 of water.

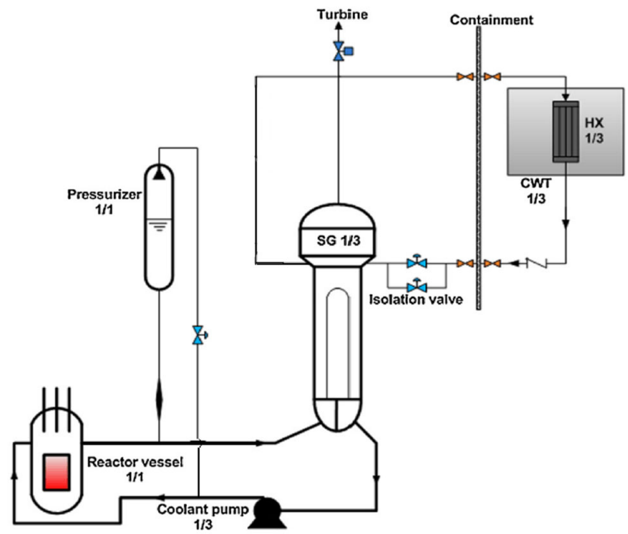


Fig. 6 Schematic of PEFS

2.5 Passive depressurization system (PDS)

The PDS system, as shown in Fig. 5, is composed of four-stage depressurization valves. Each stage is arranged into two identical flow paths [22]. Stages 1–3 of the PDS connect the top of the PRZ to the IRWST with a discharge pipe. Stage 4 (4A and 4B) of the PDS connects the RCS hot legs to the reactor containment. The valves of Stage 1 will open as the water level in the RMTs drops to a specific level (67.5%), and other valves of the PDS will open in sequence. During depressurization, the coolant will be firstly injected into the IRWST through the valves of Stages 1–3. As the valves of Stage 4 open, the coolant will mostly flow into the containment.

3 RELAP5 modeling

The best-estimate transient simulation code RELAP5/MOD3.3 is utilized to carry out the calculation presented here. Figure 7 shows the nodalization of the CPR1000 and the passive system. The CPR1000 has three loops, three pumps, and three SGs. There is a pressurizer on one of its hot legs. The components (RPV, core, PRZ, coolant pumps, SGs, RMTs, A-ACCs, IRWST, and PDS) are modeled specifically by the code. The core of the CPR1000 reactor is divided into three parts: an average channel, a hot channel, and a bypass channel. Other systems, such as the steam turbine and the main feedwater system, are modeled by time-dependent volumes and time-dependent junctions. The model for the PEFS is shown in Fig. 8. Utilizing the RELAP5 code, the HX, the water tank, and the pipes are simulated. The HX tubes are modeled by heat structures.

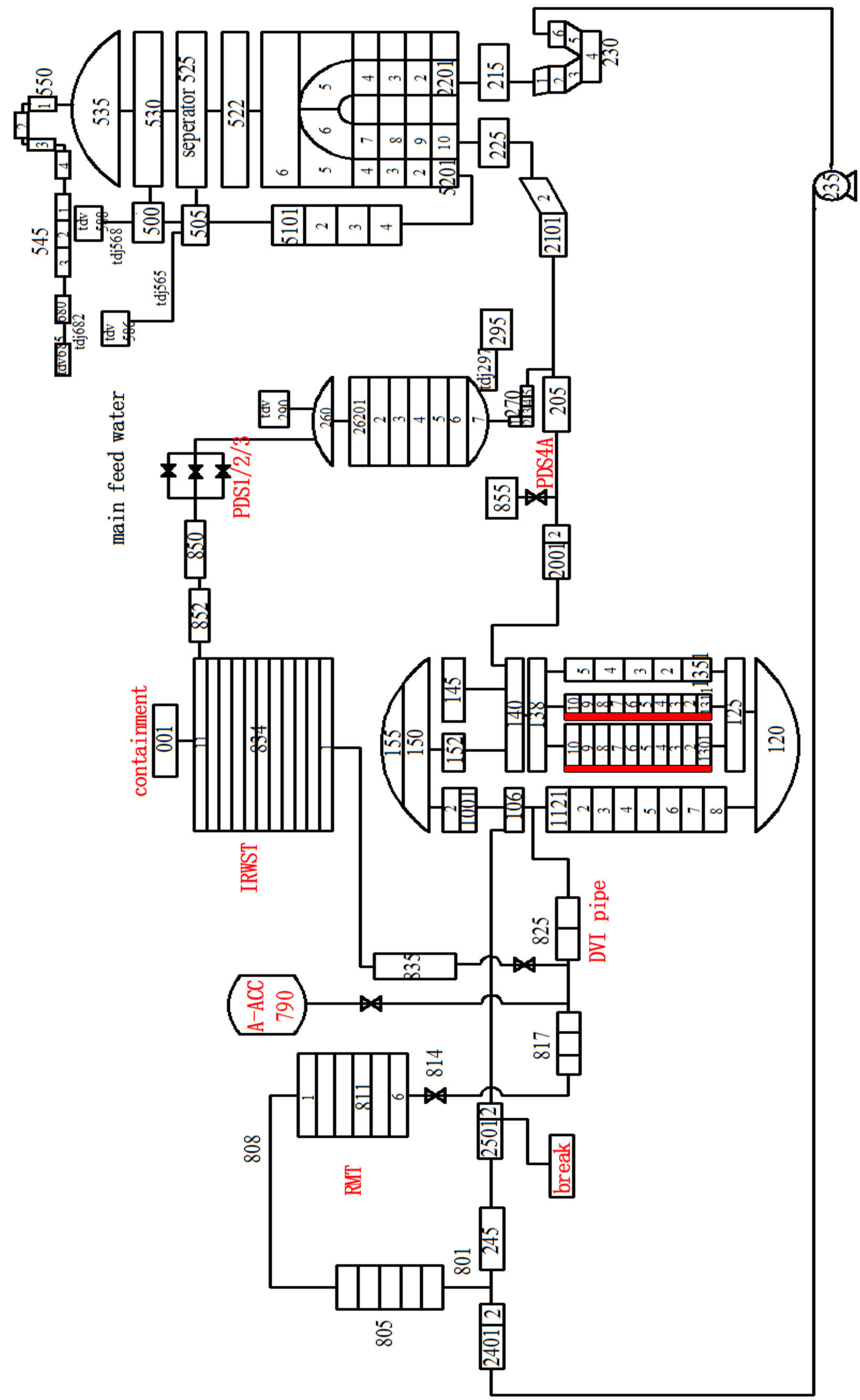


Fig. 7 RELAP5 nodalization of the passive system

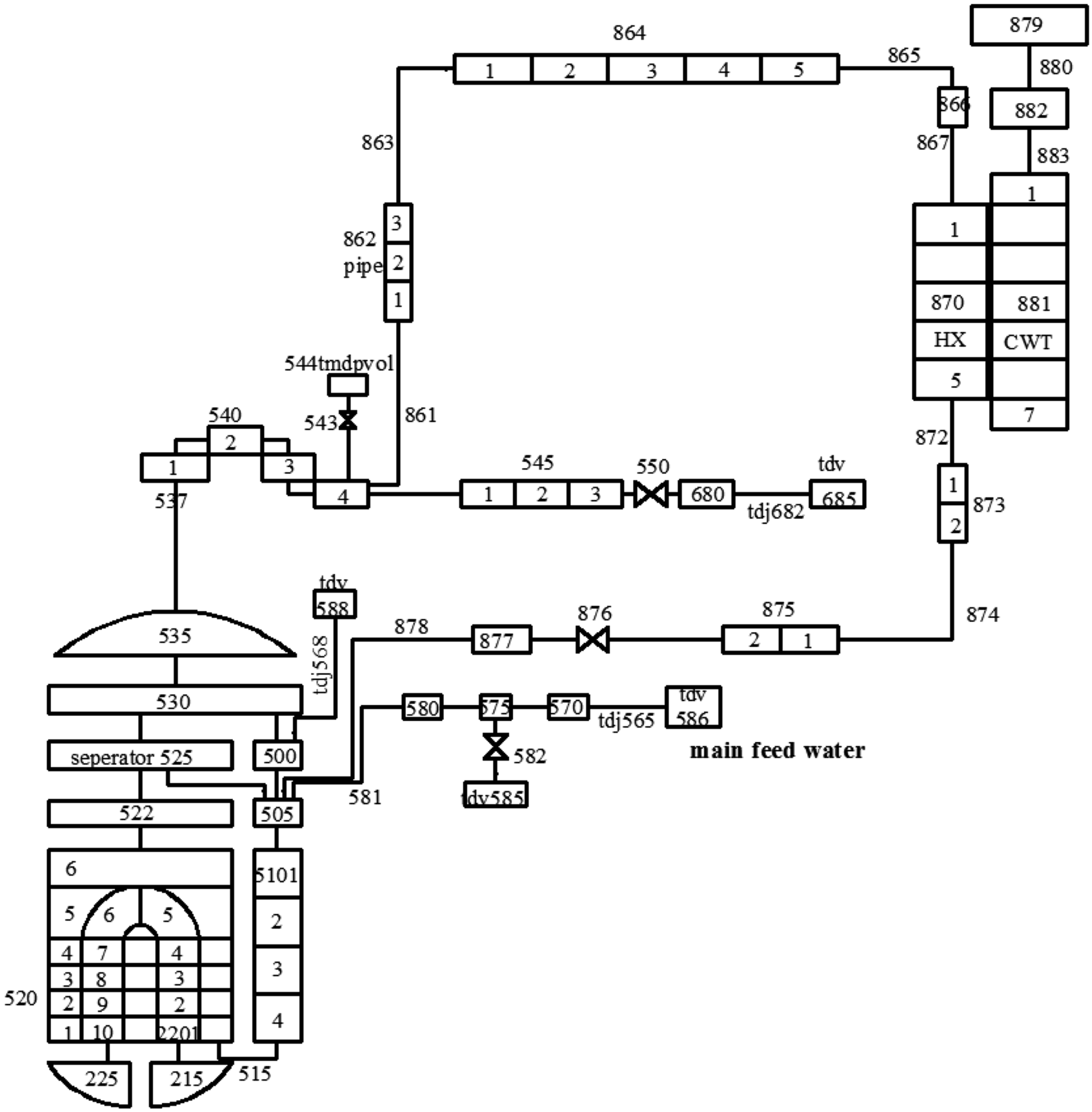


Fig. 8 RELAP5 nodalization of the PEFS

4 Results and discussion

4.1 Steady-state and actuation conditions

The initial-state conditions of the reactor are given in Table 1.

4.2 SBLOCA analysis

There are several assumptions in the analysis of the SBLOCA accident.

1. As the reactor shutdown signal was generated, the inlet valve of the steam turbine closed;

Table 1 Initial-state conditions

Parameters	Values
Reactor power (MWth)	2895.0
Coolant average temperature (°C)	310.0
Primary loop pressure (MPa)	15.5
Pump flow rate (kg/s)	4750.0
SG pressure (MPa)	6.56
Main feedwater temperature (°C)	226.0
Main feedwater flow rate (kg/s)	537.0

The actuation conditions used in the following accidents are given in Table 2

Table 2 Actuation conditions

Actions	Conditions
Reactor trip	1. Low system pressure (12.93 MPa)
Coolant pump trip	
Main feedwater pump trip	1. Loss of off-site power
Turbine trip	
RMT valves open	1. Low water level in the PRZ (27.7%)
RMT valves closed	1. High water level in the SGs (85% of narrow range)
A-ACC valves open	1. Low system pressure (4.5 MPa)
PEFS valves open	1. Low system pressure (11.6 MPa)
PDS Stage 1 valves open	Low water level in the RMTs (67.5%) (other valves will open in stage-number sequence)
IRWST valves open	1. PDS Stage 4 valves open

2. As the reactor shutdown signal was generated, the nuclear power plant lost its off-site power;
3. The break [23] ($d = 25$ mm) was on the cold loop;
4. One of the Stage 4 PDS valves failed to open after the accident.

The characteristics of the reactor are analyzed in Figs. 9, 10, 11, 12, 13, 14, 15, 16, 17, 18. The SBLOCA accident is assumed to occur at 0 s. The accident sequence of the SBLOCA is presented in Table 3.

After the SBLOCA accident occurred at 0 s, the coolant flowed out of the RCS, as shown in Fig. 9. The break flow rate changed with the system pressure. When the PDS valves were triggered, the flow rate decreased as the system pressure decreased. The loss of coolant decreased the PRZ water level and triggered RMTs (Fig. 10) at 180 s after the accident. As the RMTs initiated, a natural circulation was established. Subcooled water was injected into the RPV, and the hot water in the cold leg flowed into the RMT tanks. As the temperature in the RMTs increased, the

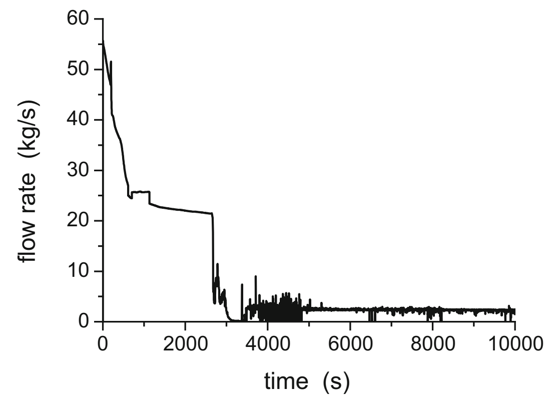


Fig. 9 Flow rate through the break

Table 3 SBLOCA accident sequence

Event	Time (s)
SBLOCA occurs	0.0
RMT triggered	180.0
Shutdown signal	186.0
Loss of off-site power	186.0
PEFS triggered	200.0
Reactor shutdown	200.4
PDS triggered	2650.0
A-ACC triggered	2750.0
IRWST triggered	3250.0

natural circulation flow rate decreased. After the system was depressurized by the PDS, the natural circulation was interrupted. The water in the RMTs was injected into the pressure vessel due to gravity. Because of the low system pressure, the hot water in the RMTs began to vaporize, and the RMT water tank dried up at about 3500 s after the accident.

Figure 11 shows the variation of system pressure. When the pressure in the PRZ decreased to 12.93 MPa, the shutdown signal was generated and the reactor lost its off-site power immediately. The main feedwater pump and the coolant pump were out of operation. The system pressure decreased drastically. As the system pressure decreased, the PEFSs were triggered at 200 s after the accident. The operation of PEFSs established a natural circulation in the primary loop (Fig. 12). Before the PDS triggered, the system pressure and coolant temperature decreased gently due to the operation of the PEFSs and RMTs. However, after the depressurization of the system, the system's water level decreased, resulting in the termination of the natural circulation. Then, the PEFSs could no longer remove the decay heat.

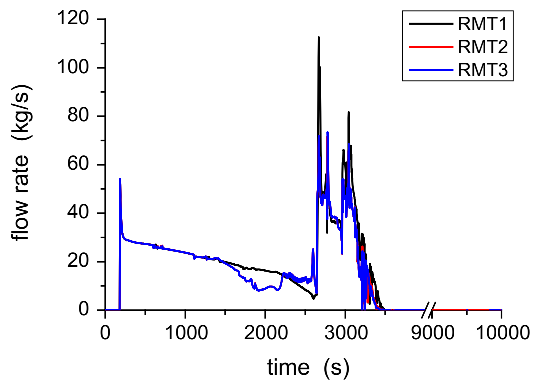


Fig. 10 Flow rate of RMTs

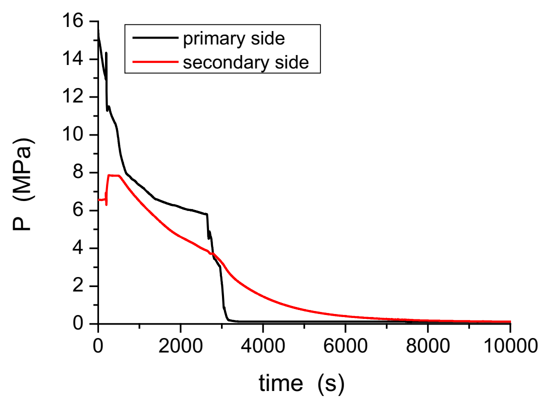


Fig. 11 System pressure

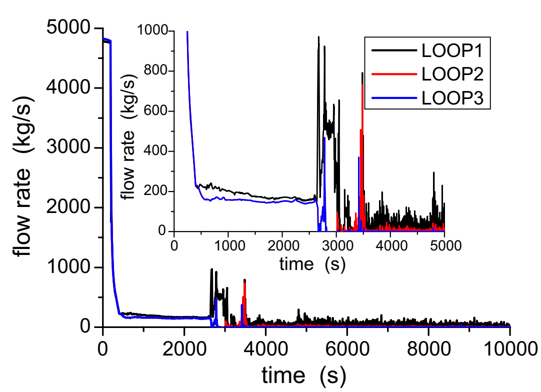


Fig. 12 Flow rate of the loops

At 2600 s after the accident, the low water level in the RMTs activated the PDS. The flow rate of each stage of the PDS is presented in Fig. 13. The Stage 1 valves open first, and the rest of the PDS valves open in sequence. According to the assumption, one valve of Stage 4A did not open after the accident. The system pressure decreased drastically because of the coolant loss. The initiation of the PDS also led to the fluctuation in heat transfer rate of the SGs and

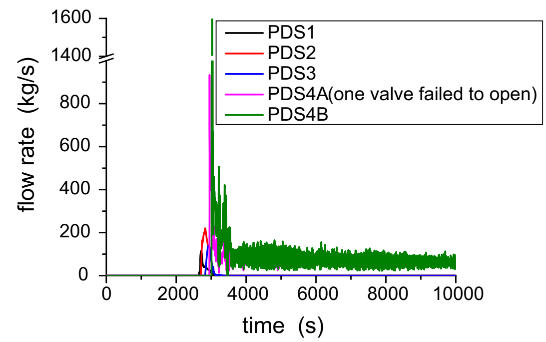


Fig. 13 PDS flow rate

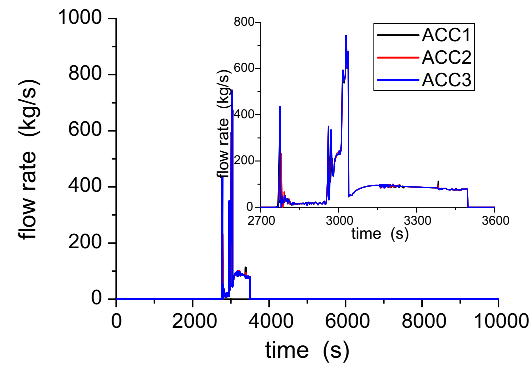


Fig. 14 A-ACC flow rate

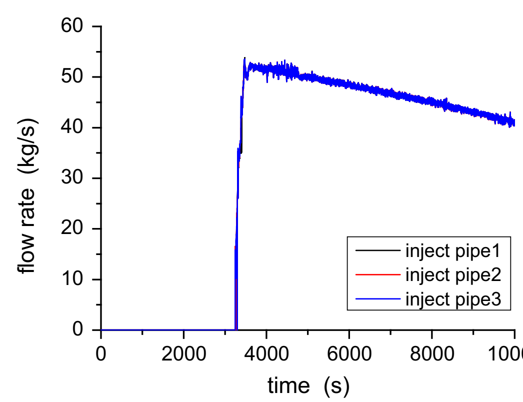


Fig. 15 IRWST flow rate

stopped the natural circulation in the loops. The first three stages of the PDS connected the PRZ and the IRWST, and the discharged coolant went into the IRWST and contributed to its water level. After the Stage 4A and 4B valves opened, the coolant flowed into the containment.

As the pressure decreased to 4.5 MPa, the A-ACC started to work. The flow rates of the A-ACCs are given in Fig. 14. Its large flow rate period lasted for about 300 s. When the water level in the A-ACCs decreased below the standpipe, the injection flow rate began to decrease. The

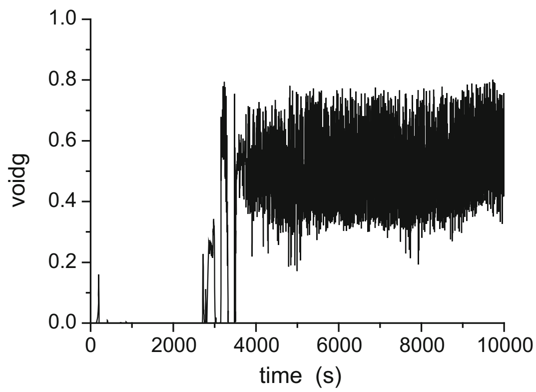


Fig. 16 Void fraction of the core outlet zone

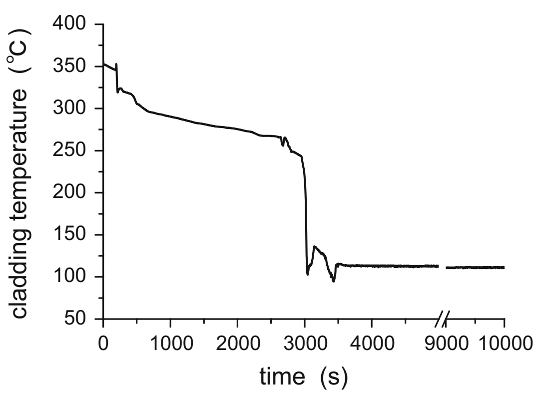


Fig. 17 Maximum cladding temperature

A-ACCs' small flow rate period lasted for about 500 s. The void fraction of the core outlet zone increased when the A-ACCs stopped injecting water into the system, which is illustrated in Fig. 16.

Triggered by the low system pressure, the IRWST started to inject water into the vessel at 3250 s after the accident. After 3500 s, only the IRWST provided subcooled water for the reactor core. Figure 15 shows the variation of the IRWST injection flow rate. As the water level in the IRWST decreased, the flow rate decreased. The system pressure, the coolant temperature, the flow rate of the PDS valves, and the flow rate of the break became stable within 5000 s after the accident. Figure 16 shows the variation of the void fraction of the core outlet zone. Figure 18 shows the variation of water level in the RPV. The water level decreased with the decrease in the system pressure. However, the operation of the A-ACCs greatly contributed to the water level. After the injection of the A-ACCs, the water level decreased and was then maintained by the injection of the IRWST. As presented in Figs. 16 and 18, the water level in the RPV was high enough to cover the core after the accident. The maximum

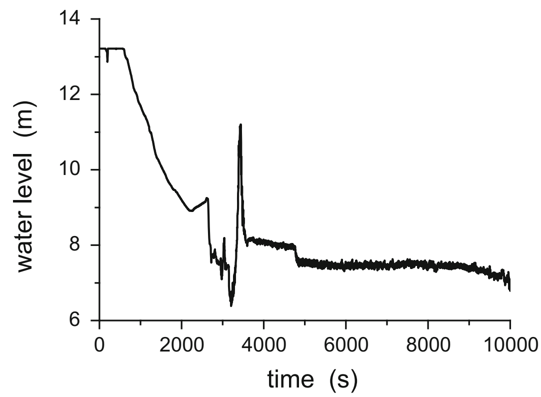


Fig. 18 Water level in the RPV

cladding temperature, which is given in Fig. 17, was controlled below the limit (1204.44 °C).

5 Conclusion

To enhance the CPR1000 NPP's safety, a passive safety system was added to the plant as the backup for the standard safety system. Utilizing the RELAP5/MOD3.3 code, the SBLOCA accident was studied. The following conclusions were reached.

After the SBLOCA accident, the components of the passive safety system were put into operation sequentially, which prevented the uncovering of the core and kept the cladding temperature below the limit. With the water injected by the IRWST, the reactor's thermal–hydraulic state became relatively stable within 5000 s after the accident. The analysis presented in this paper verified the reliability and effectiveness of the designed passive safety system after the SBLOCA accident.

The analysis of the SBLOCA described in this paper is just preliminary research. More accident analyses will be done to assess the passive safety system.

References

1. Z.J. Xiao, W.B. Zhou, H. Zheng et al., Experimental research progress on passive safety systems of Chinese advanced PWR. Nucl. Eng. Des. **225**, 305–313 (2003). doi:10.1016/S0029-5493(03)00178-X
2. P.E. Juhn, J. Kupitz, J. Cleveland et al., IAEA activities on passive safety systems and overview of international development. Nucl. Eng. Des. **201**, 41–59 (2000). doi:10.1016/S0029-5493(00)00260-0
3. A.K. Nayak, R.K. Sinha, Role of passive systems in advanced reactors. Prog. Nucl. Energy **49**, 486–498 (2007). doi:10.1016/j.pnucene.2007.07.007
4. A.K. Nayak, A. Chandrakar, G. Vinod, A review: passive system reliability analysis–accomplishments and unresolved issues.

Front. Energy Reasear **2**, 40 (2014). doi:10.3389/fenrg.2014.00040
5. B.T. Timofeev, G.P. Karzov, Assessment of the WWER-1000 reactor condition. Int. J. Pres Vessels Pip. **83**, 464–473 (2006). doi:10.1016/j.ijpvp.2005.11.008
6. M.D. Carelli, L.E. Conway, L. Oriani et al., The design and safety features of the IRIS reactor. Nucl. Eng. Des. **230**, 151–167 (2004). doi:10.1016/j.nucengdes.2003.11.022
7. K.H. Bae, H.C. Kim, M.H. Chang et al., Safety evaluation of the inherent and passive safety features of the smart design. Ann. Nucl. Energy **28**, 333–349 (2001). doi:10.1016/S0306-4549(00)00057-8
8. B. Sutharshan, M. Mutyala, R.P. Vijuk et al., The AP1000TM reactor: passive safety and modular design. Energy Procedia **7**, 293–302 (2011). doi:10.1016/j.egypro.2011.06.038
9. Y. Tujikura, T. Oshibe, K. Kijima et al., Development of passive safety systems for Next Generation PWR in Japan. Nucl. Eng. Des. **201**, 61–70 (2000). doi:10.1016/S0029-5493(00)00261-2
10. I.-C. Chu, C.-H. Song, B.H. Cho et al., Development of passive flow controlling safety injection tank for APR1400. Nucl. Eng. Des. **238**, 200–206 (2008). doi:10.1016/j.nucengdes.2007.07.002
11. H. Hu, J. Shan, J. Gou et al., Simulation of advanced accumulator and its application in CPR1000 LBLOCA analysis. Ann. Nucl. Energy **69**, 183–195 (2014). doi:10.1016/j.anucene.2014.01.037
12. Y. Zhang, S. Qiu, G. Su et al., Design and transient analyses of emergency passive residual heat removal system of CPR1000. Part I: air cooling condition. PROG NUCL. ENERG **53**, 471–479 (2011). doi:10.1016/j.pnucene.2011.03.001
13. J. Wu, Q. Bi, C. Zhou, Experimental study on circulation characteristics of secondary passive heat removal system for Chinese pressurized water reactor. Appl. Therm. Eng. **77**, 106–112 (2015). doi:10.1016/j.applthermaleng.2014.12.014
14. J. Wu, Q. Bi, C. Zhou, American Society of mechanical engineers: experimental investigations on temperature distribution and heat removal capability of residual heat exchanger, ASME 2013 Power Conference, 2013
15. M. Gavrilas, N.E. Todreas, M.J. Driscoll, Containment passive-cooling design concepts. Prog. Nucl. Energy **32**, 647–655 (1998). doi:10.1016/S0149-1970(97)00069-3
16. S.H. Chang, S.H. Kim, J.Y. Choi, Design of integrated passive safety system (IPSS) for ultimate passive safety of nuclear power plants. Nucl. Eng. Des. **260**, 104–120 (2013). doi:10.1016/j.nucengdes.2013.03.018
17. A. Achilli, G. Cattadori, R. Ferri et al., Two new passive safety systems for LWR applications. Nucl. Eng. Des. **200**, 383–396 (2000). doi:10.1016/S0029-5493(00)00256-9
18. C.S. Byun, D.W. Jerng, N.E. Todreas et al., Conceptual design and analysis of a semi-passive containment cooling system for a large concrete containment. Nucl. Eng. Des. **199**, 227–242 (2000). doi:10.1016/S0029-5493(00)00228-4
19. M. Gavrilas, N.E. Todreas, M.J. Driscoll, The design and evaluation of a passively cooled containment for a high-rating pressurized water reactor. Nucl. Eng. Des. **200**, 233–249 (2000). doi:10.1016/S0029-5493(99)00339-8
20. M. Hashim, H. Yoshikawa, T. Matsuoka et al., Quantitative dynamic reliability evaluation of AP1000 passive safety systems by using FMEA and GO-FLOW methodology. J. Nucl. Sci. Technol. **51**, 526–542 (2014). doi:10.1080/00223131.2014.881727
21. A.C.F. Guimarães, C.M.F. Lapa, F.F.L.S. Filho et al., Fuzzy uncertainty modeling applied to AP1000 nuclear power plant LOCA. Ann. Nucl. Energy **38**, 1775–1786 (2011). doi:10.1016/j.anucene.2011.02.005
22. M. Hashim, Y. Hidekazu, M. Takeshi et al., Application case study of AP1000 automatic depressurization system (ADS) for reliability evaluation by GO-FLOW methodology. Nucl. Eng. Des. **278**, 209–221 (2014). doi:10.1016/j.nucengdes.2014.06.040
23. China Nuclear Power Technology Research Institution, Final Safety Analysis Report (FSAR) of Ling'ao Nuclear Power Plant Phase 2161,CGNPC CHN, 2008