Hindawi
Science and Technology of Nuclear Installations
Volume 2020, Article ID 4765046, 11 pages
https://doi.org/10.1155/2020/4765046

Research Article

Studies on Key Effect Factors of Natural Circulation Characteristics for Advanced PWR Reactor Cavity Flooding System

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Received 24 December 2019; Revised 26 March 2020; Accepted 29 June 2020; Published 1 September 2020

Academic Editor: Han Zhang

In order to enhance the ability of severe accident mitigation for Pressurised Water Reactor (PWR), different kinds of severe accident mitigation strategies have been proposed. In-Vessel Retention (IVR) is one of the important severe accident management means by External Reactor Vessel Cooling. Reactor cavity would be submerged to cool the molten corium when a severe accident happens. The success criterion of IVR strategy is that the heat flux which transfers from the corium pool must be lower than the local critical heat flux (CHF) of the reactor pressure vessel (RPV) outside wall and the residual thickness of the RPV wall can maintain the integrity. The residual thickness of RPV is determined by the heat flux transfer from the corium pool and the cooling capability of outer wall of the RPV. There are various factors which would influence the CHF and the cooling capability of outer wall of the RPV. In order to verify the optimized design which is beneficial to the heat transfer and the natural circulation outside the actual reactor vessel, a large-scale Reactor Vessel External Cooling Test (REVECT) facility has been built. A large number of sensitivity tests were carried out, to study how these sensitivity factors affect CHF value and natural circulation. Based on the test results, the structure of the test section flow channel has an obvious effect on the CHF distribution. The flow channel optimized can effectively enhance the CHF value, especially to enhance the CHF value near the "heat focus" region of the molten pool. The water level in the reactor pit has also a great impact on the natural circulation flow. Although natural circulation can be maintained with a low water level, it will lead to a decrease of the cooling capacity. Meanwhile, some noteworthy test phenomena have been found, which are also essential for the design of the reactor pit flooding system.

1. Introduction

The probability of severe accidents in nuclear power plants is much low, but once a severe accident occurs, it will cause the nuclear reactor core to melt. The core corium relocates to the lower head, forming a molten pool and transferring heat to the RPV wall. If the molten pool cannot be effectively cooled down, the pressure vessel may be melted through due to excessive thermal load, which greatly increases the possibility of radioactive material release to the environment. Then, how to maintain the integrity of RPV and limit the corium in the lower head become the focus of research on mitigation measures for severe accidents worldwide.

External Reactor Vessel Cooling (ERVC) to achieve In-Vessel Retention (IVR) is an important mitigation measure for severe accidents. Due to its advantage of low construction difficulty and good economy, IVR-ERVC is widely used in the 3rd generation million-kilowatt nuclear power plants. Chinese advanced pressurized water reactor mainly adopts this strategy to mitigate severe accidents.

In the design of IVR-ERVC, the reactor pit flooding system is one of the most important dedicated safety systems. In order to verify the effectiveness of IVR-ERVC, it is necessary to carry out related research to evaluate the factors that affect the cooling capability of the reactor pit flooding system. The factors include the design of RPV flow channel, the water level in the reactor pit, and so forth. In this paper,

the description of a new reactor pit flooding system and the REVECT (Reactor Vessel External Cooling Test) facility are introduced and the influence factors of cooling capability are studied. Based on the REVECT tests, a series of problems which have confused the designers in long-time are solved. The research results have been applied to the engineering design of reactor pit flooding system and the establishment of management guidelines for severe accidents in new nuclear power plants.

2. Research Status

After the occurrence of a postulated severe accident, corium will collapse into lower head of RPV. The disposal mode of corium becomes a common concern. Currently, there are mainly two treatment ways for corium cooling: (1) Ex-Vessel Retention (EVR): this method is mainly used in EPR. The specific method is setting a core catcher at the bottom of containment, which has functions of isolating molten debris and concrete as well as providing long-term cooling. EVR strategy is also adopted in Tianwan nuclear power plant (NPP) in China. (2) In-Vessel Retention (IVR): IVR strategy is implemented to remove decay heat from the RPV and to keep the corium in the RPV [1]. This strategy has also been used in VVER440, AP1000, HPR1000, and APR1400 [2].

In order to verify the effectiveness of the IVR strategy, some different test facilities which are used to simulate the flow and heat transfer characteristics have been built, such as ULPL test facility and HERMES-HALF test facility. The conclusions of the tests are as follows. (1) An aged copper surface exhibits a similar coolability performance as the external surface of the RPV wall, which is verified by ULPU and BETA-NC test [3–5]. (2) Optimized flow channel can enhance the CHF of the external surface of lower head. (3) The chemical properties of some coolants can affect the CHF. For instance, the coolant with boric acid can decrease the CHF to some extent.

In China, different types of CHF tests and related analyses have been performed in recent years. Based on a large-scale test facility, Yang [6] investigated the CHF characteristics of chemical solution boiling on a downward facing curved surface. It was found that the CHF of mixed solution of H_3BO_5 and Na_3PO_4 increases firstly and then decreases with the increase of Na_3PO_4 concentration. Through a two-dimensional full-scale facility FIRM, Wei [7] discovered that the aging effect could enhance the CHF of SA508 owing to the generation of Fe_3O_4 oxide. To study the CHF margin of External Reactor Vessel Cooling (ERVC) for Chinese AP1400, the FIRM subcooled flow boiling facility conducted by State Nuclear Power Technology Research & Development Center (SNPTRD) was built [8]. It was also shown that the concentration of Na_3PO_4 has an important effect on the CHF behavior. Based on the experimental data, a CHF correlation for IVR-ERVC was developed by Mei et al. [9] with considering the effects of surface orientation, thermal effusively and corrosion. Besides, Tan and Kuang [10] preliminarily determined nominal values of reactor vessel insulation design parameters according to ERVC related functional reliability criteria and related statistical analysis. Guo et al. [11] proposed a new method to study the transient feasibility of IVR-ERVC in which a theoretical CHF model was developed for the outer surface of the lower head. Jin et al. [12] performed the study on in-vessel and ex-vessel coupled analysis of IVR-ERVC phenomena for large-scale PWR by using MELCOR. Cheng et al. [13] used a CDF code Fluent coupled with a boiling model by UDF (User-Defined Function) to investigate the CHF of ERVC which was validated by experimental CHF values obtained by SNPTRD.

Even though a large number of tests and analyses have conducted, there are little test results about the effect of the key factors on the CHF distribution, such as the geometry of flow channel and the water level in the reactor cavity. The above key factors are also essential for the reactor flooding system design.

3. Design and Management of Reactor Flooding System for Advanced PWR

3.1. Reactor Pit Flooding System. The reactor pit flooding system is a dedicated severe accident mitigation system to achieve the IVR-ERVC. Generally, the reactor pit flooding system is composed of a reactor pit injection system and a natural circulation system. When a severe accident happens, cooling water can be injected into the reactor cavity to submerge the RPV via passive injection mode or active injection mode. The decay heat is finally removed through the natural circulation in the reactor cavity to maintain the integrity of RPV.

The passive reactor pit injection subsystem mainly consists of the reactor pit flooding tank located on a place higher than the main coolant pipes of Reactor Coolant System (RCP) in the containment, pipelines, and valves. During a severe accident after entering the severe accident management guideline (SAMG), the valves on the passive injection pipes are opened by the operator, and the water in the reactor pit flooding tank flows to the reactor pit by gravity with a large flow rate. When the water level in the pit flooding tank drops to a certain threshold, the injection is switched to small flow rate to compensate for the water loss due to evaporation, ensuring the reactor cavity to be kept flooded.

The active reactor pit injection is provided by active water injection pipelines which connect to the in-containment refuelling water storage tank (IRWST). When the reactor pit flooding tank is depleted, the cooling water is taken by pump from the IRWST and is injected into the reactor pit after cooled by Component Cooling Water System or Extra Cooling System. The schematic diagram of passive reactor pit injection subsystem and active reactor pit injection subsystem is illustrated in Figure 1.

Since the core outlet temperature reaches 650°C, which is a significant temperature alarm for the NPP to carry out the severe accident management guideline (SAMG), the operator opens the valves on the passive injection pipelines to submerge the reactor pit with a large flow rate. Then, the water injection flow is switched to small flow rate phase to make up the evaporated water. Before the IVR tank is empty, the operator starts up the pump to inject water from the in-containment refuelling water storage tank (IRWST).

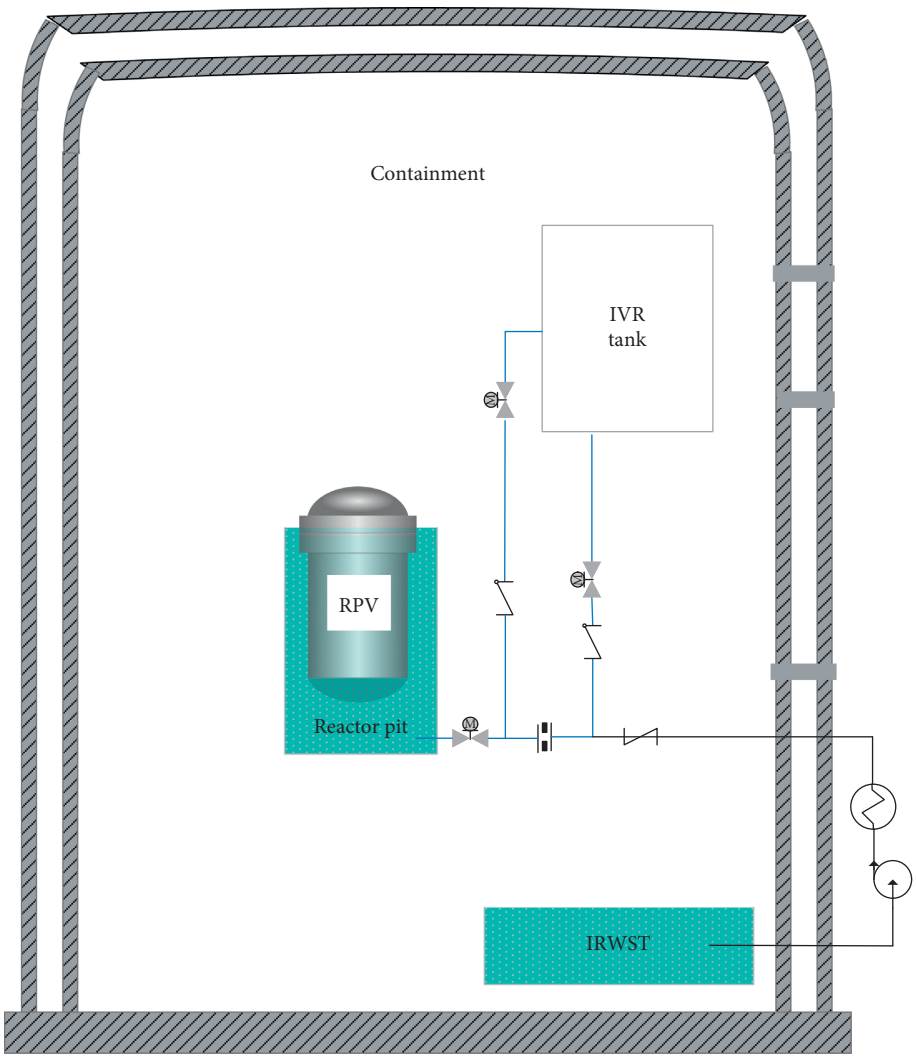

Figure 1: Design scheme of reactor pit flooding system.

3.2. Key Phenomena of Natural Circulation in the Reactor Cavity. The main phenomena and the cooling process related to the implementation of the reactor pit flooding system after severe accidents are as follows.

When the cooling water is injected into the reactor pit under severe accident conditions, the water inlets and steam outlets can be automatically opened. And the cooling water can enter the flow channel around the RPV to cool it directly.

During a severe accident, the molten corium will drop down and fall into the water pool in the lower plenum. The corium pool will heat up the internal wall of the lower plenum. And many bubbles on the external surface of RPV will produce under the heating of the lower head. Natural circulation will form due to the density difference in the circulation loop. The water-steam two-phase flow will flow upward to the steam vent ports. The water-steam separation will occur at the top of the cylinder space around the support ring. Finally, the water will flow back into the reactor pit by the recirculation pipes. The structural drawing of reactor pit is shown in Figure 2.

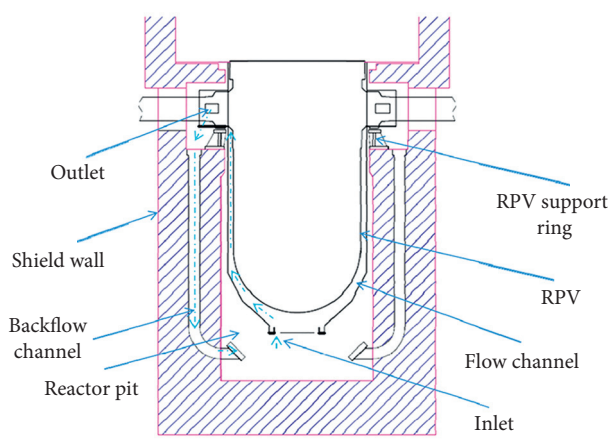

Figure 2: The natural circulation in the reactor pit.

Natural circulation forms in the flow channel after the reactor cavity being flooded. The specific cooling process after the severe accidents in the reactor pit is as follows.

(1) The cooling water is heated by RPV outside wall and many bubbles produce
(2) The water-steam flows upward along the gap between RPV outside wall and insulation and then vents out through the steam vent port of insulation
(3) The water flows back into the reactor cavity by backflow channel after steam-water separation

3.3. REVECT Test Facility

3.3.1. Presentation of IVR Test Facility. To study the influencing factors which determine the CHF distribution along the outer surface of the RPV and natural circulation flow, the REVECT (Reactor Vessel External Cooling Test) facility test facility for the advanced generation PWR has been built. The flowchart and the overview diagram of the REVECT test facility are shown in Figures 3 and 4. The test facility consists of electrical system, cooling system, power control system, primary loop and instrumentation and control (I&C) system, and so forth. The REVECT facility is a two-dimensional facility with a full-height circulation loop and a 1 : 1 radial scaled slice-type test section. The test section is the key component of the facility composed of a copper heating section and stainless steel flow channel, as the simulator of the RPV lower head and the flow path between the RPV and its insulation, respectively. Hundreds of cartridge heaters are inserted into the copper heating section to simulate the decay heat. The heating section is divided into more than 20 heating zones to realize independent heating power control of different zones. The stainless steel flow channel is bounded by several baffles which are used to simulate the insulation structure of the RPV lower head. The schematic diagram of the test section is shown in Figure 4. An integrated water tank is located under the test section in connection with the stainless steel flow channel, to simulate the reactor pit space under the RPV insulation. The upper water tank is located at the top of the facility, to simulate the annular region around the RPV support ring in the prototype. The main parameters of the REVECT facility are listed in Table 1.

The heating section is used to simulate the lower head of RPV after a large amount of molten corium relocated in it. The material of the heating block is copper. The width of heating block is constant in the test section. The slice-type heating section is divided into more than 20 independent power control zones to realize the simulation of the decay power distribution of the prototype along the lower head. Three baffles are installed in the flow channel to simulate the structure and the gap size of the RPV lower head insulation flow channel. In each heating zone, there are pairs of thermocouples installed in two concentric circles within the heating block to obtain the temperature data, so that the heat flux could be calculated according to the measured temperature under the Fourier heat conduction equation. The structure of the test section is shown in Figure 5. The main parameters of the test section are listed in Table 2.

3.3.2. Key Phenomena in the REVECT Facility Loop. The REVECT test facility can simulate the structure of circulation loops in the prototype. For example, the geometry of the test section is the same as that of the flow channel outside the RPV. It can simulate the flow and the heat transfer characteristic which is similar to the actual phenomena in the flow channel outside the lower head of the RPV. The height of the upward flow pipe is the same as that in the prototype and the resistance characteristic of the upward pipe and the steam exhaust ports are considered. The upward pipe and the steam exhaust ports can simulate the two-phase flow in the flow channel outside the RPV cylinder part and the steam vent ports. The integrated tank can simulate the cylinder space around the support ring of the RPV. The phenomenon of water-steam separation takes place in the integrated tank which also happens above the cylinder space around the support ring of the RPV. The resistance characteristic of the downward pipe is also modelled. It can simulate the recirculation pipes in the reactor pit shield wall.

3.4. Test Research on the Key Techniques for Reactor Pit Flooding System. In the design of the reactor pit flooding system for new type nuclear power plants, the research focuses on the natural circulation flow rate and CHF distribution. The reasons are the following.

(1) CHF distribution: the design of the flow channel has a great relation with the CHF value on the outer wall of the lower head of RPV. The specific influencing factors are as follows: the flow channel structure of RPV metal insulation, the area and the inlet position of RPV metal insulation, and the area of the outlet of RPV metal insulation for steam venting.
(2) Natural circulation: natural circulation flow rate is also closely related to the CHF value on the outer wall of the lower head. The larger the natural circulation flow rate is, the higher the value of CHF is. The factors which affect the natural circulation flow include the design of the backwater channel, water level, the makeup amount of water. The vibration frequency and amplitude of the two-phase flow which is important for the design of RPV metal insulation are also necessary to be considered.

3.4.1. Input Power. The power control strategy used in this experiment was proposed by Professor Theofanous et al. [1] at the University of California, USA. The similarity criterion that makes the flow of test section is similar to that of the prototype. The similarity criterion is successfully applied to IVR of AP600, AP1000, and APR1400. In the test, it mainly includes the following two aspects:

(1) The superficial vapor velocities of downstream position match up with the prototype for all $\theta > \theta_m$;
(2) The vapor flow rates build up gradually, so as to smoothly approach the value required at $\theta \leq \theta_m$, while allowing a "natural" development of boundary layer in all of the upstream region.

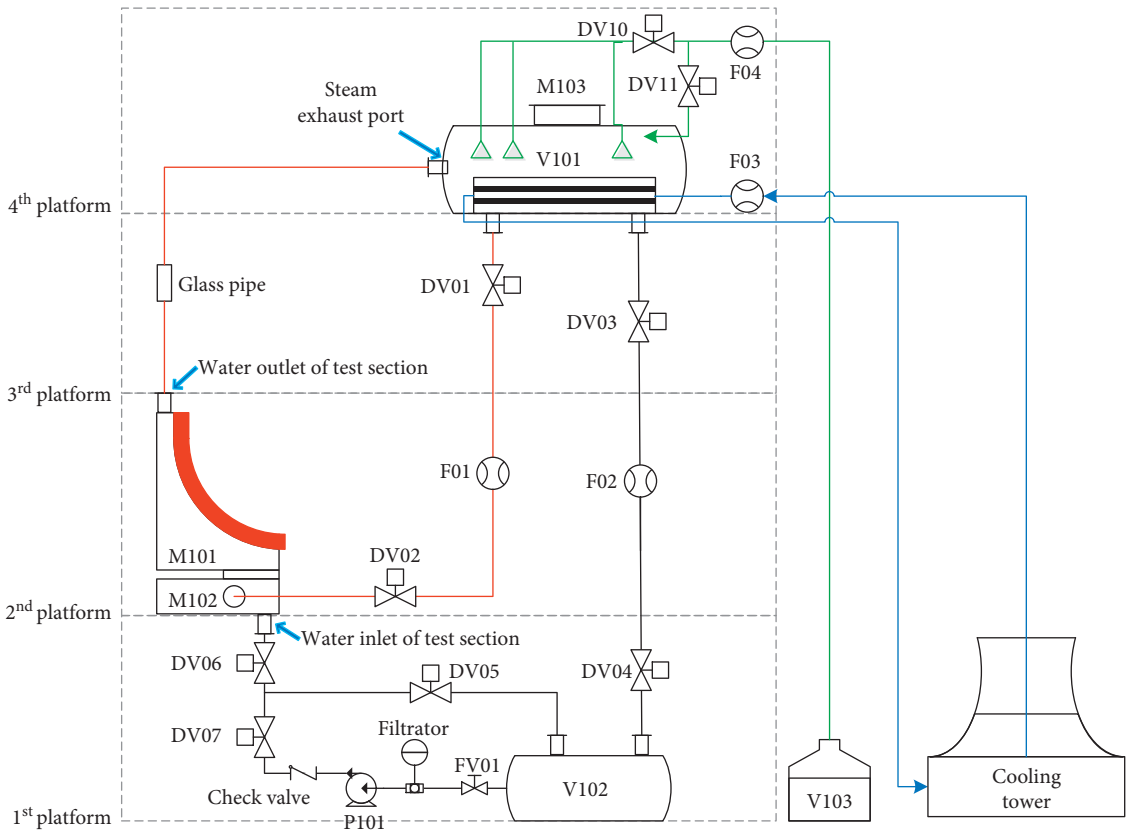

Figure 3: The flowchart of REVECT test facility.

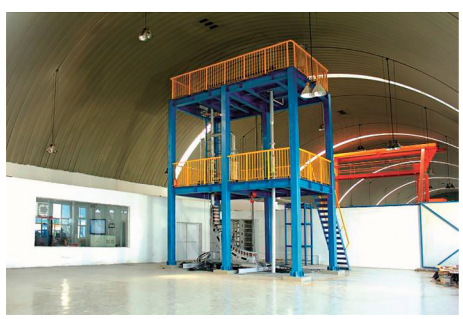

Figure 4: Overview of REVECT test facility.

The upstream θ power distribution calculation formula is

$$q_e(\theta) = q_p(\theta)\frac{\sin\theta}{\sin\theta_m}, \quad \text{for } \theta \le \theta_m. \tag{1}$$

For downstream locations of the test area, the power distribution calculation formula is the following:

$$\begin{aligned}\int_{\theta_m}^{\theta} q_e(\theta')\mathrm{d}\theta' &= J_p(\theta_m) + \frac{1}{\sin\theta}\int_{\theta_m}^{\theta} q_p(\theta')\sin(\theta')\mathrm{d}\theta' - J_e(\theta_m) \\ &= \overline{\overline{q_p}}(1-\cos\theta_m)\left(\frac{1}{\sin\theta} - \frac{1}{\sin\theta_m}\right) + \frac{1}{\sin\theta}\int_{\theta_m}^{\theta} q_p(\theta')\sin(\theta')\mathrm{d}\theta', \quad \text{for } \theta > \theta_m,\end{aligned} \tag{2}$$

Table 1: Key parameters of REVECT facility.

Parameters	Value/definition	Remarks
Test pressure	Atmosphere	Pressure in upper water tank
Water temperature at inlet of test section	~100°C	The saturation temperature corresponds to the pressure of the upper water tank
Target water level of the upper water tank	7300 mm	The elevation of the bottom of the heating section is selected as the 0 m level, the same as the prototype
Elevation of steam exhaust port	8000 mm	Relative to the bottom of the heating section
Circulation type	Natural circulation	The same as the prototype
Working fluid	Deionized water	
Steam/water outlet area of the upflow pipe	~0.01 m^2	About 1/100 of the prototype RPV insulation steam invent port area
Cooling water inlet area of the test section	~0.01 m^2	About 1/100 of the prototype RPV insulation cooling water inlet port area
Structure of baffles	3-part multilateral structure	With the same structure and "gap size" distribution as the prototype
Circulation resistance	Scaling as the prototype to ensure the consistence of the comprehensive resistance of facility and the prototype	Based on the geometry and resistance distribution analysis of the prototype flow path

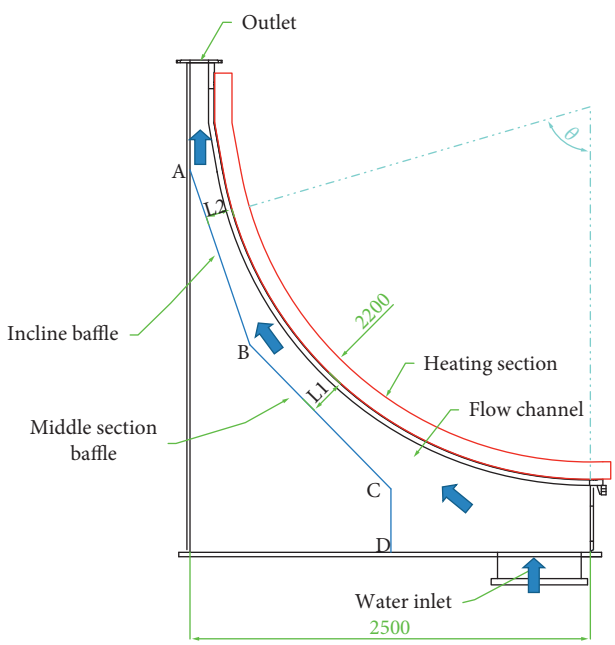

Figure 5: Schematic of the test section.

where $\overline{\overline{q_p}}$ is an average flux over $0 < \theta \leq \theta_m$ defined by

$$\overline{\overline{q_p}} = \frac{\sin\theta_m}{1-\cos\theta_m}\int_0^{\theta_m} q_p(\theta')\sin(\theta')\mathrm{d}\theta'. \quad (3)$$

The theoretical and actual input heat flux density distribution in the test heating zone 23 is shown in Figure 6.

4. Sensitivity Study on the Different Structure of the Flow Channel

4.1. Test Conditions. Nine test conditions (FC01-FC09) with different flow channel distances and different baffle structures were performed. The specific test conditions are shown in Table 3. The flow channel consists of three straight baffles and heating surfaces of the test section. The baffles mainly include a vertical section (CD), a middle section baffle (BC), and an inclined baffle (AB). The sensitivity tests were divided into three stages. In the first stage, the minimum distance from the middle section baffle (BC) to the heating surface is 250 mm, then, adjusting the incline baffle (AB) to different three positions. So three different flow channel structures are formed (FC01-FC03). Similarly, in the second and third stages, the minimum distances from the middle baffle to the heat section surface are set as 200 mm and 150 mm, respectively, then adjusting the incline baffle to different positions to perform the sensibility analysis.

4.2. Test Results of Sensitivity Study on the Different Flow Channels. Figure 7 shows the CHF distribution under three different test conditions (FC01-FC03).

The CHF value of FC01 changes in a "wave shape" with the increase of the azimuth angle, and the CHF value is relatively small. The CHF values of FC02 and FC03 show a "W-shape" change with the azimuth angle change from 35° to 85°.

Figure 8 gives the CHF test data in the second stage (FC04-FC06). The CHF values are similar to the test results of FC01-FC03. The CHF value of FC04-FC06 also changes in a "W-shape" from 35° to 90°.

Figure 9 shows the CHF test data in the third stage (FC07-FC09). The CHF values of these three tests (FC07-FC09) are higher than those in the former two stages, and the CHF values of most areas are greater than 1.4 MW/m^2. With the increase of the azimuth angle from 35° to 90°, the CHF value in the third stage increases firstly and then decreases until the position of 68° and then increases. The CHF values of the FC07 and FC08 change relatively smoothly, and the value is relatively large. The CHF values of most areas are larger than 1.5 MW/m^2. Besides, the baffle with a minimum distance of 100 mm (FC09) can increase

Table 2: Key parameters of REVECT test section.

Parameters	Value/ definition	Remarks
Circumferential scale of the test section to prototype	1 : 100	
Radial scale of the heating section to the prototype RPV	1 : 1	With the same outer diameter as the as the prototype RPV lower head
Height scale of the natural circulation loop to the prototype RPF system	1 : 1	
Number of independent heating zones	23	Cover the angular range of 0°–90°
Designed maximum surface heat flux	~2.5 MW/m^2	The actual heat flux is lower than 1.4 MW/m^2
Width of flow channel	150–250 mm	Effective width
Outer diameter of the heating section	4800 mm	The same as the prototype RPV lower head

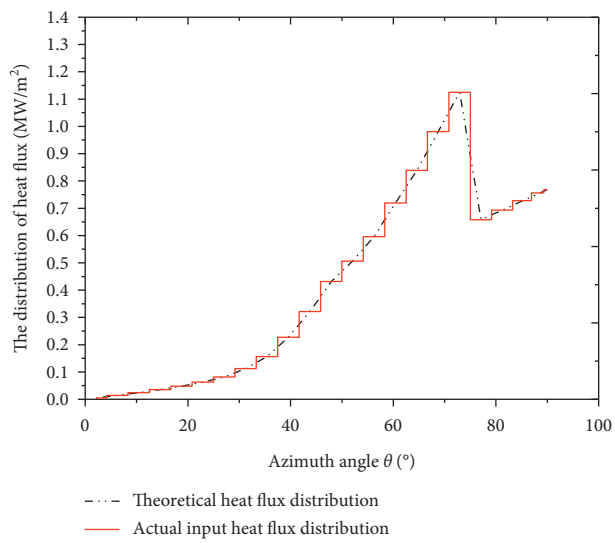

Figure 6: Theoretical and actual input heat flux distribution in zone 23.

Table 3: Test conditions for the flow channel optimization.

No.	Test types	Minimum distance of flow channel L1/L2	Temperature of the inlet	Pressure
1	Flow channel optimization test FC01	250/250 mm	~100°C	1 bar
2	Flow channel optimization test FC02	250/210 mm	~100°C	1 bar
3	Flow channel optimization test FC03	250/140 mm	~100°C	1 bar
4	Flow channel optimization test FC04	200/200 mm	~100°C	1 bar
5	Flow channel optimization test FC05	200/180 mm	~100°C	1 bar
6	Flow channel optimization test FC06	200/130 mm	~100°C	1 bar
7	Flow channel optimization test FC07	150/150 mm	~100°C	1 bar
8	Flow channel optimization test FC08	150/145 mm	~100°C	1 bar
9	Flow channel optimization test FC09	150/100 mm	~100°C	1 bar

the CHF value at the position between 52° and 64° and inhibit the CHF value in the upstream at the position between 68° and 72°.

The nine test results show that when the flow channel distance ranges from 250 mm to 200 mm and then further decreases to 150 mm, CHF value tends to increase; while the distance of flow channel continued to decrease to 100 mm (FC09), the CHF value begins to decrease at the upstream heating zone. It can be considered that the CHF value tends to increase with the space of flow channel decreases, but the value of CHF will be inhibited when the distance of flow channel decreases to a certain level.

4.3. Optimization of the Flow Channel Structure Outside the RPV. The above results show that different flow channel configurations lead to different CHF distributions. According to the test results, the optimal design of the structure for the

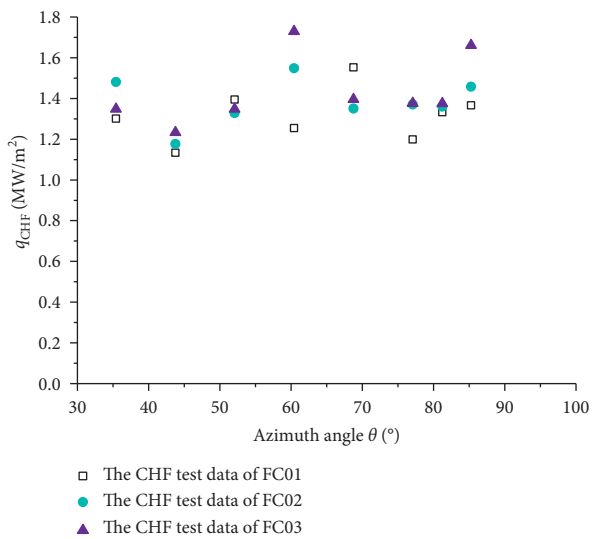

Figure 7: CHF distribution of the FC01-FC03 test conditions.

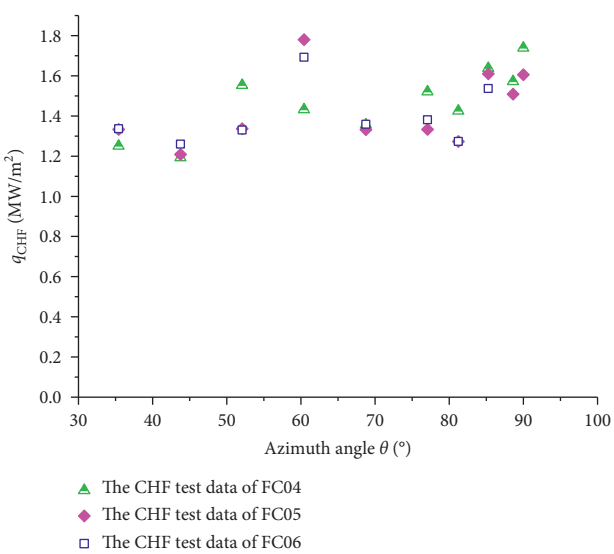

Figure 8: CHF distribution of the FC04-FC06 test conditions.

insulation is obtained and the verification test under the optimized structure is carried out. The test results show that even though the CHF date at the position of 52° is lower than that before being optimized, the CHF data with the optimized flow channel at higher angles (67°–90°) are obviously larger than the experimental data in the case that the flow channel is not optimized. The average increase amplitude is about 20%. The schematic of optimized flow channel is shown in Figure 10. That means that there is larger margin for the heat focus effect owing to the optimized flow channel. The comparison of CHF values between optimized and non-optimized flow channels (FC08) is shown in Figure 11.

The mechanism could be that when the distance changes from 250 mm to 140 mm, the velocity of the two-phase flow changes to higher, the film boiling is more unlikely to happen. While the distance further decreases to 100 mm, the

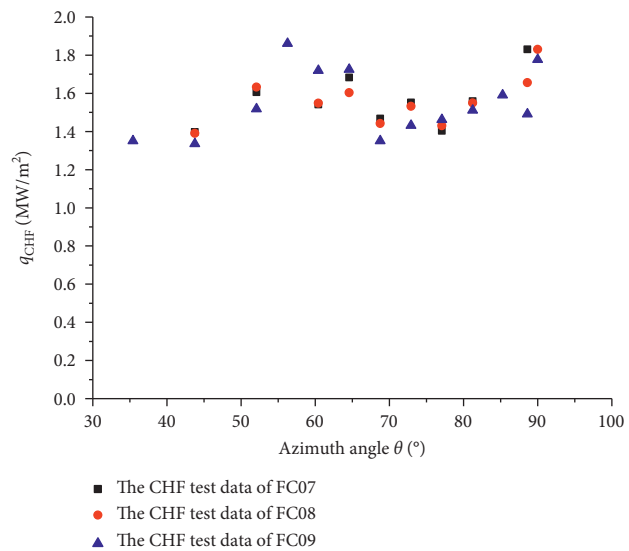

Figure 9: CHF distribution of the FC07-FC09 test conditions.

two-phase flow with many large bubbles is likely to block the flow channel, so the film boiling at the position of the minimum distance is more likely to happen.

5. Natural Circulation Flow and Evaporated Water

5.1. Test Condition. Tests of different water levels in the reactor cavity are also conducted to study the effect on the natural circulation characteristic. The details of test condition are shown in Table 4.

5.2. Test Results

5.2.1. The Flow Rate of the Nature Circulation and the Evaporation. The natural circulation flow exceeds 40 m^3/h. Due to 1 : 100 circumferential scale of the test section to prototype, the natural circulation flow rate is about 4000 m^3/h for the actual nuclear power plants. The test result shows that the natural circulation flow is very large once the steady natural circulation forms, so the backflow channel is essential. Besides, the test results also show that there is about 0.5 m^3 water evaporated per hour, so small flow rate (>50 m^3/h) is needed to inject into reactor cavity to make up the water and keep the level of water in the reactor cavity at the higher height.

5.2.2. Water Level Effect. According to the test results, the natural circulation can be maintained if the water level is between 8 m and 5 m, but the circulation flow rate is obviously decreased with lower water level. When the water level drops to 6.5 m, especially, the intermittent discharge occurs, causing the pressure in the flow channel to fluctuate correspondingly. The results are shown in Figures 12 and 13. Due to the phenomenon of intermittent discharge in the flow channel and the adverse cooling condition, the temperature on the outer surface of the lower head of RPV may rise at the same

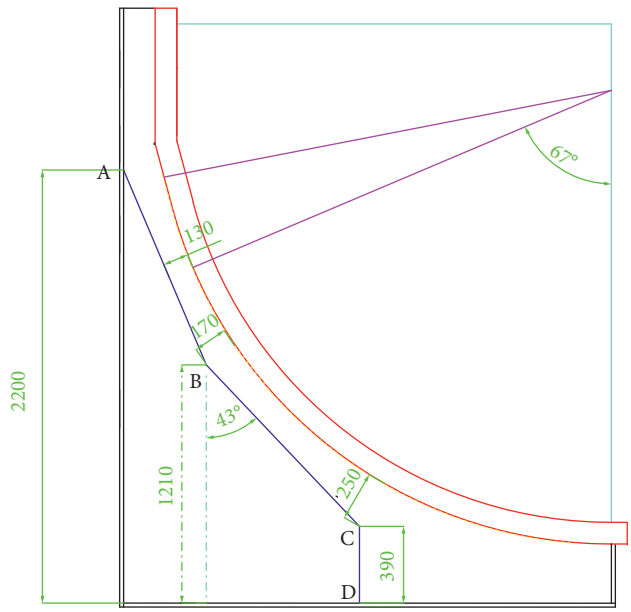

Figure 10: The schematic diagram with the optimized flow channel of the test section.

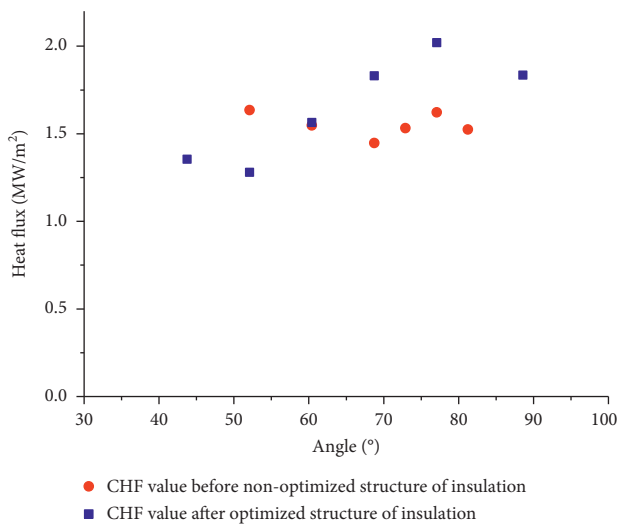

Figure 11: Comparison of CHF values between optimized and nonoptimized flow channels.

Table 4: Test conditions for the natural circulation characteristics.

No.	Test types	Water level	Temperature of the inlet	Pressure
1	Natural circulation characteristics tests NC01	8 m-5 m	~100°C	1 bar

time, which is not conducive to the cooling of the lower head. Therefore, in the design of reactor cavity flooding system, a clear request is made for the lowest water level of the reactor cavity after water injection. The water level of the reactor cavity must be kept as high as possible to ensure that the natural circulation can be maintained with larger flow rate and take away more decay heat from the corium pool.

5.2.3. Vibration Frequency and Amplitude. To obtain the variation of the vibration frequency of the two-phase flow in the flow channel with different water levels in the reactor cavity, the fluctuation pressure and frequency of the two-phase flow in the flow channel are considered.

When the liquid level decreases from 8 m to 7 m, the fluctuation pressure is within ±10 kPa. When the water level is

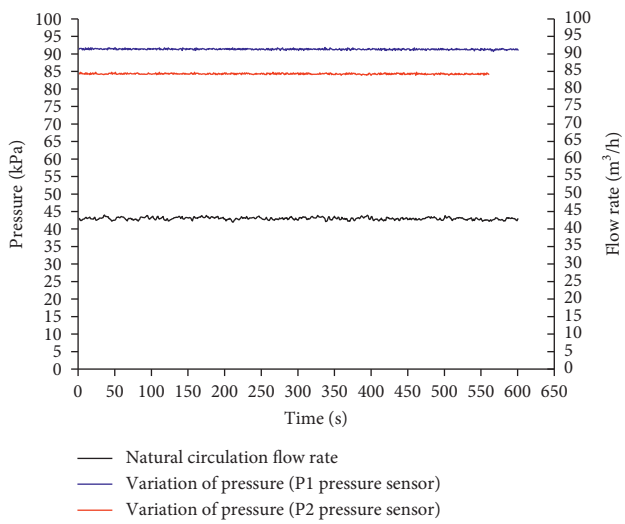

Figure 12: Fluctuation pressure is stable with water level of 8 m.

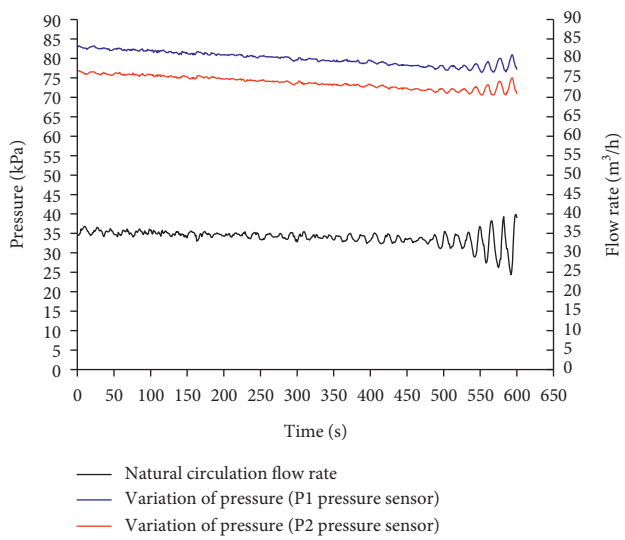

Figure 13: Fluctuation pressure is changing unstably with water level decreases.

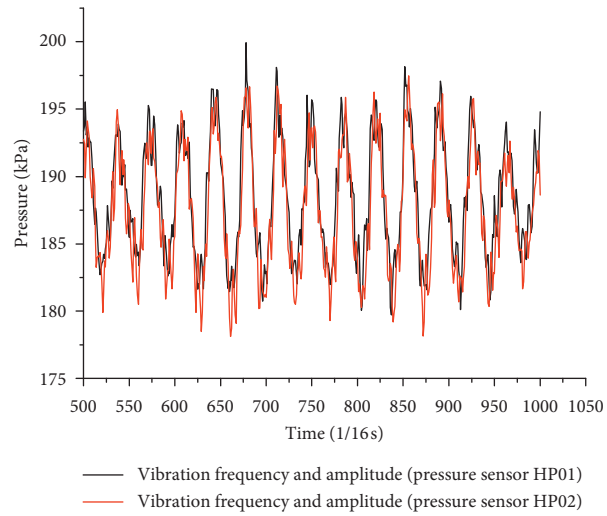

Figure 14: Vibration frequency and amplitude with water level of 8 m.

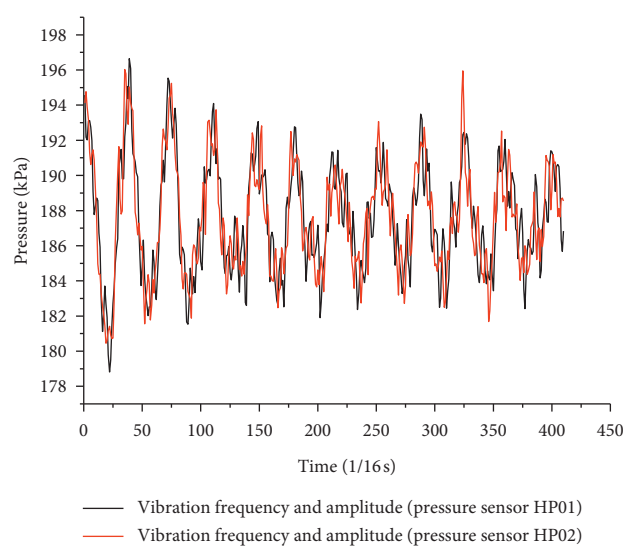

Figure 15: Change of vibration frequency and amplitude with water level decreases.

high, the pressure fluctuation is regular and the frequency is about 1.4 Hz. With the decrease of water level, the pressure fluctuation begins to appear irregular, but the fluctuation pressure is still in the range of ±10 kPa. So it can be found that the pressure fluctuation range shows a decreasing trend with the decrease of water level and the fluctuation frequency is about 1.1 Hz at a relatively regular fluctuation, as shown in Figures 14 and 15.

Based on test results, in order to prevent the occurrence of resonance phenomenon after reactor pit injection system is implemented, which may lead to the destruction of the insulation, the inherent frequency of the insulation is required to avoid 1.1 Hz–1.4 Hz and the antivibration pressure of insulation needs to be more than ±10 kPa at the same time.

6. Conclusions

Based on the above experimental results, the natural circulation characteristics of two-phase flow are investigated in detail when implementing the reactor pit flooding system after severe accidents and the following design requirements are proposed for the reactor pit flooding system for new type nuclear power plants. The specific requirements are as follows:

(1) The flow channel structure outside the RPV has a significant effect on the CHF value. The flow channel optimized can effectively enhance the CHF value, especially to enhance the CHF value near the "heat focus" region of the molten pool. Therefore, the

effectiveness of the IVR-ERVC strategy can also be further enhanced.

(2) Natural circulation flow can exceed 4000 m^3/h after reactor pit flooding system is implemented. Therefore, multiple backwater flow channels must be set to ensure that the water discharged from the outlet of insulation can flow back into the reactor pit again.

(3) The water level in the reactor pit has a great impact on the natural circulation flow. Although natural circulation can be maintained with low water levels, it will lead to a decrease in the cooling capacity. Therefore, the water level must be maintained as much as possible at higher position.

(4) The inherent frequency of RPV insulation is required to avoid 1.1 Hz–1.4 Hz and the antivibration pressure of insulation needs to be more than ±10 kPa at the same time.

Data Availability

The data used in the study have been included in the article.

Conflicts of Interest

The authors declare that there are no conflicts of interest regarding the publication of this article.

Acknowledgments

This work was supported by National Key R&D Program of China (Grant no. 2018YFB1900100).

References

[1] T. G. Theofanous, C. Liu, S. Addition, S. Angelini, O. Kymaelaeinen, and T. Salmassi, "In-vessel coolability and retention of a core melt," vol. 1, Office of Scientific and Technical Information, Oak Ridge, TN, USA, 1996.

[2] T. G. Theofanous, S. J. Oh, and J. H. Scobel, "In-vessel retention technology develoopment and use for advanced PWR designs in the USA and Korea," vol. 14, Office of Scientific and Technical Information, Oak Ridge, TN, USA, 2004.

[3] T. N. Dinh, J. P. Tu, and T. G. Theofanous, "Two phase natural circulation flow in AP-1000 in-vessel retention-related ULPU-V facility experiments," in *Proceedings of the ICAPP'04*, Pittsburge, PA, USA, June 2004.

[4] Y. H. Jeong, W. P. Baek, and S. H. Chang, "CHF experiments for IVR-RVC using 2-D slice test section," in *Proceedings of the Korean Nuclear Society Spring Meeting*, Kwangju, South Korea, May 2002.

[5] T. G. Theofanous, J. P. Tu, A. T. Dinh, and T. N. Dinh, "The boiling crisis phenomenon, part I: nucleation and nucleate boiling heat transfer," *Experimental Thermal and Fluid Science*, vol. 26, no. 6-7, pp. 775–792, 2002.

[6] S. Yang, "Experimental study on critical heat flux of chemical water boiling on a downward facing curved surface for IVR-ERVC strategy," *Nuclear Power Engineering*, vol. 6, pp. 23–27, 2016.

[7] W. Tao, "Experimental research on influence of real surface material and aging effect on ERVC-CHF of RPV," *Atomic Energy Science and Technology*, vol. 50, p. 1786, 2016.

[8] H. Chang, T. Hu, W. Lu, S. Yang, and X. Zhang, "Experimental study on CHF using a full scale 2-D curved test section with additives and SA508 heater for IVR-ERVC strategy," *Experimental Thermal & Fluid Science*, vol. 84, pp. 1–9, 2017.

[9] Y. Mei, Y. Shao, S. Gong, Y. Zhu, and H. Gu, "Effects of surface orientation and heater material on heat transfer coefficient and critical heat flux of nucleate boiling," *International Journal of Heat & Mass Transfer*, vol. 121, pp. 632–640, 2018.

[10] G. Tan and B. Kuang, "Geometric optimization and reliability assessment of reactor vessel insulation for IVR-ERVC based on passive system functional reliability concept," *Atomic Energy Science and Technology*, vol. 3, pp. 54–60, 2011.

[11] R. Guo, W. Xu, Z. Cao, X. Liu, and X. Cheng, "A new method to study the transient feasibility of IVR-ERVC strategy," *Progress in Nuclear Energy*, vol. 87, pp. 47–53, 2016.

[12] Y. Jin, W. Xu, and X. Liu, "In- and ex-vessel coupled analysis of IVR-ERVC phenomenon for large scale PWR," *Annals of Nuclear Energy*, vol. 8, pp. 322–337, 2015.

[13] X. Cheng, T. Hu, D. Chen, Y. Zhong, and H. Gao, "CFD simulation on critical heat flux of flow boiling in IVR-ERVC of a nuclear reactor," *Nuclear Engineering & Design*, vol. 304, pp. 70–79, 2016.