Study on reactor residual heat removal method based on steam controllable discharge in station blackout accident

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Abstract. The design of passive residual heat removal system (PRHRS) has the problems of uncontrollable coolant cooling rate and incomplete matching between the heat removal capacity and the reactor residual heat. To solve these problems, a residual heat removal method is proposed in this paper, which is controllable release of secondary steam into the environment at the early stage of reactor shutdown when the residual heat is large. The feasibility of this method is verified by the thermohydraulic analysis code. The results show that this method can discharge the residual heat in a period of time after the reactor shutdown, and the coolant system can establish a stable natural circulation. But when the steam generator is empty, the residual heat must be removed by other ways. After that, based on the critical flow theory of nozzle, a simplified method for calculating the steam discharge power and estimating the safety-related parameters by steam pressure is proposed, which can be a reference for operators to control the cooling rate, analyze the system status and handle the accident. The conclusions can also provide references for the study and improvement of residual heat removal method.

1. Introduction
If the residual heat cannot be effectively removed after the station blackout accident (SBO), it may cause coolant system overpressure and pipes damage. In serious cases, the fuel elements will overheat and burn down, and a large amount of hydrogen generated by metal-water reaction will explode, further resulting in excessive release of radioactive substances into the environment [1]. Therefore the residual heat must be removed by the passive residual heat removal system (PRHRS), which has high inherent safety [2]. Typically, the residual heat power is high within minutes after reactor shutdown, and decreases rapidly to 3% of the initial power after about 8 minutes, then declines slowly for a long time. In order to discharge the residual heat at the initial stage of shutdown and prevent the overpressure of the coolant system, the PRHRS is usually designed in accordance with the 3%FP (full power). As a result, the residual heat is mostly lower than this value when the PRHRS is running. Excessive design power imposes higher requirements on the size and structural strength of the system piping and equipment, and causes the cooling rate of the coolant to be too high [3]. Simulation results showed that the biggest coolant cooling rate can reach 200℃/h, far exceeding the limit. In that case, the thermal stress of pipe material will be large, which could affect the integrity of the primary loop pressure boundary and cause the coolant leaking.

The papers [4, 5] proposed an auxiliary residual heat removal way based on the steam generator auxiliary feedwater and steam release, which could effectively reduce the power design requirements of PRHRS, so that the cooling rate requirement was satisfied while residual heat was discharged...
effectively. However, this method requires sufficient water supply and emergency power, and their availability is difficult to guarantee in the SBO. The papers [3, 6] used surplus steam, generated by the residual heat, to drive steam-driven pump to provide feedwater for the steam generator (SG), so that SG would not be emptied while heat can be removed by the steam consumption. Although this method can provide feedwater driving force by itself, it needs to set additional steam-driven pump and condensing or processing equipment for its exhaust steam, thus the complexity of the system increases. Based on the design of AP1000 atmospheric relief valves, this paper proposes steam emergency discharge system, which can cool the reactor by controllably releasing the secondary steam into environment at the early stage of reactor shutdown. During the accident, since there is no water supply, the original water in SG is heated into steam by residual heat and discharged directly through the emergency discharge line, so as to remove the residual heat to final sink. In order to verify the residual heat discharge function of this system, the discharge process is numerically simulated by RELAP5 analysis code. Then a simplified method for controlling cooling rate and estimating safety-related parameters by steam pressure is proposed.

2. Design of steam emergency discharge system

A power-operated atmospheric relief valve, in conjunction with the startup feedwater system, can provide for controllable removal of residual heat when the condenser loses the function of heat sink and the turbine bypass system isn’t available. During reactor cooldown, the relief valve is automatically controlled by steam pressure, with remote adjustment of the pressure setpoint from the control room [7]. But its modulating steam relief function is implemented by nonsafety-related systems, of which steam pressure measurement, relief valve adjustment and remote control function may not be available after the SBO.

![Figure 1. Schematic diagram of steam emergency discharge system.](image1)

![Figure 2. Node diagram of steam emergency discharge system.](image2)

Considering the failure of power supply and display-control system, a steam emergency discharge system is established based on the simple manual operation, local measurement and display unit. As shown in Figure 1, the system is composed of steam generator, main steam line, emergency discharge steam line, manual control valve and local mechanical pressure gauges. Among them, the pressure gauge can be used for directly reading the steam pressure, to calculate the steam flow and discharge power, as the reference for controlling cooling rate. Because SG is the key component to connect the primary and secondary loops, which are strongly coupling in parameters, the safety related parameters...
can be calculated from the steam parameters which are relatively easy to measure when the measurement, display and control system are unavailable. Since the typical pressurized water reactors are similar in design, the reactor coolant system and PRHRS aren’t described. The detailed introduction can be found in the paper [8].

3. System function verification
Due to the low power and weak natural circulation capacity of primary system, there is strong coupling relationship among the parameters such as steam pressure, coolant temperature and flow rate. It is necessary to conduct numerical simulation and to discuss the residual heat removal capacity and the relationship between these parameters during the operation of the steam emergency discharge system.

3.1. Spatial discretization
Based on RELAP5 code, the thermohydraulic analysis model of reactor primary coolant system, steam generator and related steam line system is established, after a series of steady-state and typical accident condition verification as well as node sensitivity analysis. The node diagram of the emergency discharge system is shown in Figure 2, and the description of relevant nodes and component types is shown in Table 1. When the coolant system operates in the single-phase natural circulation, it has low driving force and low flow rate, and the flows in some SG U-tubes are prone to Ledinegg flow instability [9]. As soon as the gravity pressure head in the U-tube is lower than the sum of the resistance pressure drop and the pressure difference between the inlet and outlet, the reverse flow will occur. In order to accurately simulate this phenomenon, all of the U-tubes are divided into 6 groups according to tube length based on the lumped parameter method, and each group contains several tubes within a certain length range.

| Node number | Description                             | Component type |
|-------------|-----------------------------------------|----------------|
| 101, 120    | Inlet and outlet chamber of SG snglvol  |                |
| 103, 106,   |                                         |                |
| 109         | Primary side of SG U-tube pipe          |                |
| 112, 115, 118|                                         |                |
| 203         | Downward channel pipe                   |                |
| 205         | Secondary side of SG U-tube pipe        |                |
| 206         | Steam separator separatr                |                |
| 207         | Steam chamber snglvol                   |                |
| 209, 211    | Steam line pipe                         |                |
| 213         | Steam discharge line pipe               |                |
| 215         | Extern environment tmdpvol              |                |

3.2. Heat transfer formula of steam generator
Because of lack of feedwater due to power failure, the water inventory of SG will gradually decrease with the steam discharge. The secondary side of the U-tubes is gradually exposed from top to bottom, and its heat transfer mechanism will transform from the nucleate boiling heat transfer to the steam convective heat transfer.

3.2.1. Nucleate boiling heat transfer. In 1963, based on a large number of experimental data, J.C. Chen proposed an empirical formula for calculating the nucleate boiling heat transfer coefficient, which can be express as follow:

\[ h = h_{nc}F + h_{nc}S \]  \hspace{1cm} (1)

where \( F \) is the correction factor of two-phase Reynolds number; \( S \) is the inhibition factor of convection on boiling heat transfer; the first term on the right hand side of Equation (1) is the convective...
The second term is the boiling component. $h_{mic}$ can be calculated by Dittus-Boelter formula, and $F$ can be expressed as a function of Martinelli parameters. $h_{mic}$ is the pool nucleate boiling heat transfer coefficient, which is calculated by Forster-Zuber formula as follow:

$$ h_{mic} = 0.00122 \left[ k_f^{0.79} C_P f \rho_f^{0.45} \right] \frac{\Delta T_{sat}^{0.24} \Delta P_{sat}^{0.75}}{\sigma^{0.5} \mu_f^{0.29} h_{fg}^{0.24} \rho_g^{0.24}} $$  \hspace{1cm} (2)

where $\Delta T_{sat}$ and $\Delta P_{sat}$ are the wall superheat and corresponding differential pressure, respectively. The factor $S$ is calculated by two-phase Reynolds number [10].

3.2.2. Single-phase steam convective heat transfer. Like the liquid phase convection, the Dittus-Boelter formula is still used to calculate the steam convection heat transfer coefficient:

$$ h_g = 0.023 \frac{k_g}{\rho_g} Re_g^{0.8} Pr_g^{0.4} $$  \hspace{1cm} (3)

where the subscript $g$ represents the steam phase.

The heat transfer mode is determined according to void fraction. When the void fraction is less than 0.95, the heat transfer coefficient is calculated by Equation (1), and Equation (3) is used when the void fraction is greater than 0.99. The intermediate transition zone is calculated by interpolating the void fraction. Because the generation, separation and movement of a large number of bubbles have strong agitation on the adjacent fluid, which strengthens the heat transfer between the wall surface and the fluid, the boiling heat transfer coefficient is very large. When the surface is dried out, the heat transfer is seriously deteriorated and the steam convective heat transfer coefficient is small. Therefore, when the heat transfer mode changes to steam convection, the heat transfer coefficient will decrease, and when the void fraction is close to 1, the heat transfer coefficient will decrease rapidly.

According to Weisman’s experiment [11], the heat transfer coefficient between vertical rod bundles is proportional to the ratio of pipe spacing to diameter (P/D). For the external of vertical U-tubes rod bundle, the heat transfer coefficient is calculated by multiplying Equation (1) or (3) by the modified factor P/D.

3.3. Numerical simulation results and analysis

Before the accident, the reactor is operating at steady-state full power. When the blackout accident happens, the control rod, relying on its own gravity, falls rapidly to shutdown the reactor. In addition, the following assumptions are considered:

1. As all electric pumps and valves fail, coolant flow can only rely on natural circulation and steam generators lose feedwater;
2. The normal steam equipment is shut down, whose steam consumption is lost;
3. The operation of PRHRS and the heat dissipation of pipelines are not considered;
4. Considering the operator's reaction and operation delay, the control valve is manually opened within 60s after blackout accident occurrence.

Then numerical simulation of the steam discharge system is conducted by RELAP5 program. The response characteristics of the important parameters are shown in Figures 3-8. Among them, the power, pressure, mass flow, SG water inventory and steam pressure are referenced by their respective rated value, and the temperature is referenced by the rated temperature of the coolant system hot leg. During the accident, the coolant flow drops rapidly because the coolant pumps trip, and then it stabilizes due to the natural circulation between the steam generator and the reactor core. After opening the discharge valve, the heat removal mainly depends on the natural circulation of primary loop and steam discharge. With the continuous discharge of steam, the residual heat is taken out, and the coolant temperature and pressure gradually decrease. At the same time, the SG pressure and water inventory decrease as well. When all water in the SG secondary side is almost vaporized, the heat transfer capacity deteriorates seriously, which results in a significant decrease of the discharge power. Since the residual heat cannot be effectively discharged, the coolant temperature and pressure will rise,
and the natural circulation flow will also fluctuate and decline. After that, the reactor must be cooled by PRHRS or other means. Therefore, the steam emergency discharge system can remove the residual heat in a period of time to prevent the coolant system from overpressure discharge.

**Figure 3.** Steam discharger power and reactor residual power.

**Figure 4.** Primary system pressure.

It’s noted that coolant and SG saturation temperatures fluctuate intermittently in Figure 5. This is because the heating zone on SG secondary side is divided into four layers (as shown in Figure 2), in which water is emptied successively, and with heat transfer coefficient decreasing rapidly, the temperature difference between the two sides of U-tubes increases. However, the decreases of SG pressure and coolant temperature lead to the improvement of the heat transfer in SG secondary side and the corresponding decrease of the temperature difference. This process continues until the entire SG secondary side is completely emptied.

**Figure 5.** Coolant temperature and SG saturation temperature.

**Figure 6.** Natural circulation coolant flow.
4. Estimation of discharge power and coolant parameters

In order to meet the cooling rate requirements of the coolant, it is necessary to adjust the steam discharge power according to the reactor residual heat. However, as a result of SBO accident, the reactor display and control system may not be able to effectively measure, transmit and display the safety-related parameters remotely, and thus it is difficult to adjust the discharge power and deal with the accident. The method for determining the steam discharge power and primary system parameters from the steam pressure, directly gained from the local pressure gauges (as shown in Figure 1), is discussed below.

4.1. Discharge power

Treating the steam discharge valve as an ideal nozzle, we set the back pressure at the outlet of the nozzle as $p_b$, and the gas pressure before the nozzle as $p_0$. When $p_b$ is less than the critical pressure $p_{cr}$, corresponding to the nozzle throat, critical flow will occur and the throat velocity will reach the maximum. For an ideal gas, the critical pressure can be calculated as follows:

$$p_{cr} = p_0 \left(\frac{2}{\kappa+1}\right)^{\frac{\kappa}{\kappa-1}} \quad (4)$$

For dry saturated steam, if we take specific heat capacity ratio $\kappa$ as 1.135, then $p_{cr} = 0.577 p_0$. When $p_b$ is atmospheric pressure, as long as $p_0$ exceeds 0.175MPa, the throat flow can reach the critical condition. Because steam pressure is always greater than 0.175MPa in emergency discharge as well as external environment in atmospheric pressure, steam flow through the valve be calculated according to the critical flow. Then, the critical flow rate of the nozzle is

$$G = A \left(\frac{2\kappa}{\kappa+1} \frac{p_0}{v_0}\right)^{\frac{1}{2}} \quad (5)$$

According to the steam thermophysical parameters, the specific volume is approximately inversely proportional to the pressure within a certain range of steam pressure, then

$$G = Ap_0 \left(\frac{1}{k_1 \kappa + 1}\right)^{\frac{1}{2}} = kAp_0 \quad (6)$$

where $k_1$ is the proportionality coefficient. It should be pointed out that there are some differences between the steam and the ideal gas. However, Equation (6) can be used to calculate the critical steam flow rate in engineering. Here $\kappa$ doesn’t mean specific heat capacity ratio, but the coefficient $k$ can be derived from the experiment or the actual gas state equation. The power of discharge steam is:
Flow area $A$ can be calculated according to the valve opening, and $p_0$ is read from the local pressure gauge. Since the steam specific enthalpy $h_g$ is approximately constant, the power $P$ can be conveniently calculated from Equation (7). The coolant cooling rate can be calculated according to the energy balance between the heat removal power and the residual heat.

### 4.2. Primary system state parameters

As can be seen from Figure 5, due to the low coolant flow, the heat transfer power of the SG U-tubes is small, and the cold leg temperature of the primary system is close to the saturation temperature of SG secondary side, with the maximum deviation not exceeding 3%. Therefore, the cold leg temperature $T_c$ can be approximately obtained from the steam pressure.

Based on the experimental research and theoretical derivation, Wu et al. [12] proposed that for a given natural circulation loop, the relationship between the flow rate $G$ and power $P$ under the steady state conditions is:

$$ G = nP^m $$

where $n$ is the fitting coefficient, and $m$ is closely related to the flow state and resistance characteristics of the system. For the reactor primary system, the local resistance is usually much larger than the friction resistance, so $m$ can be 0.333 [12]. According to the energy conservation, the temperature difference between the hot and cold legs is

$$ \Delta T = T_h - T_c = \frac{P^{0.667}}{nc_p} $$

Where $c_p$ is specific heat at constant pressure. Therefore, according to Equations (8) and (9), the natural circulation flow and coolant temperature can be approximately calculated from the steam pressure and reactor residual heat.

### 5. Conclusions

In this paper, the steam emergency discharge system is designed, by which the reactor residual heat is removed, and its feasibility is demonstrated by RELAP5 program. The conclusions are drawn as follows:

1. The system can remove the residual heat within a period of time after the shutdown, and the primary system can establish a stable natural circulation. However, after emptying the steam generator, the residual heat must be removed by other ways.
2. The saturation temperature of the SG secondary side is approximately equal to the cold leg temperature. The natural circulation flow and the temperature of the coolant can be derived from the steam pressure and the reactor residual heat, which provides operational supports for the operators when safety-related parameters are not available.
3. The discharge valve flow can reach critical condition. Its flow rate, as well as discharge steam power, is proportional to the valve flow area and the upstream pressure, which is helpful for the operators to control the cooling rate.

In summary, if the steam emergency discharge system is taken as the controllable heat removal way prior to the PRHRS, the power requirements of the PRHRS can be reduced, and the problem that the heat removal capability and the reactor residual power is unmatched can be solved.

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