Dynamic response characteristics of DHRS in small pressurized water reactor under SGTR accident

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Abstract

Small modular reactors (SMRs) are considered a promising direction for the future development of nuclear power generation due to their enhanced safety, lower initial costs, shorter construction periods, and flexible deployment options. Some typical reactors ensure robust performance during both normal operation and accident scenarios by employing passive safety mechanisms. To further understand the capabilities of these passive safety systems, based on the NuScale integral pressurized water reactor, a simulation model was developed using RELAP5 code to analyze the thermal-hydraulic response characteristics of the decay heat removal system (DHRS) during a steam generator tube rupture (SGTR) accident. The results show that the decay heat removal heat exchanger (DHRHX) on the faulted side has a limited contribution to the dissipation of the core decay heat during the accident process. In contrast, the DHRHX on the intact side can independently eliminate the core decay heat and effectively control the core inlet and outlet temperatures. When the leakage in the primary loop causes the water level to drop below the baffle of the pressurizer, significant flow oscillations occur, triggering fluctuations in other parameters. Under the conservative setting of decay power, the DHRS can still effectively prevent core melting.

Full Text

Dynamic Response Characteristics of DHRS in Small Pressurized Water Reactors Under SGTR Accidents

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Abstract: Small modular reactors (SMRs) represent a promising direction for future nuclear power development due to their enhanced safety, lower initial costs, shorter construction periods, and flexible deployment options. Some typical reactor designs ensure robust performance during both normal operation and accident scenarios through passive safety mechanisms. To further understand these passive safety system capabilities, a simulation model based on the NuScale integral pressurized water reactor was developed using RELAP5 code to analyze the thermal-hydraulic response characteristics of the decay heat removal system (DHRS) during a steam generator tube rupture (SGTR) accident. The results demonstrate that the decay heat removal heat exchanger (DHRHX) on the faulted side contributes minimally to dissipating core decay heat during the accident. In contrast, the DHRHX on the intact side can independently remove core decay heat and effectively control core inlet and outlet temperatures. When primary loop leakage causes the water level to drop below the pressurizer baffle, significant flow oscillations occur, triggering fluctuations in other parameters. Under conservative decay power settings, the DHRS can still effectively prevent core melting.

Keywords: SMR, SGTR, DHRS, Dynamic response

INTRODUCTION

Small modular reactors (SMRs) are recognized as a key future reactor type due to their high safety, low initial cost, short construction time, and modular features. Many countries are actively involved in their design and construction [1][3]. For instance, the ACP100 reactor, an innovative pressurized water reactor featuring passive safety systems and integrated design technology, is being developed by China National Nuclear Corporation. Relevant tests have been conducted on its passive safety systems and critical heat flux density of fuel assemblies [5][7]. The ACP100 is now under official construction and represents the world's first commercial small modular reactor. Dmitriev and Balunov conducted extensive research on KLT compact modular small reactors to verify the effectiveness of the emergency cooling system (ECS) design and the heat transfer capacity of two-phase natural circulation [8][9]. The Korea Atomic Energy Research Institute developed the TASS/SMR program for the 100 MW SMART integral reactor, employing a point-reactor model to calculate core thermal power and a one-dimensional drift-flux model to predict the primary system's thermal-hydraulic response during accidents [10][11]. The results demonstrate that SMART's design meets safety requirements. CAREM is an Argentine natural-circulation integral small reactor with 100 MW thermal power [12][13]. Marcel et al. studied the corresponding natural-circulation phenomena and stability, finding that existing thermal-hydraulic analysis methods for PWRs and BWRs are not applicable to CAREM [14].

Similarly, the NuScale reactor employs a fully integrated natural circulation modular design, eliminating the need for primary coolant pumps and reducing mechanical and electrically driven components, thereby enhancing design reliability [15]. This design has attracted widespread attention. Susyadi studied an alternative cooling solution during loss of coolant accidents (LOCA) using Relap5, proposing a two-step cooling strategy including rapid steam removal [16]. Fakhraei simulated the thermal-hydraulic behavior of the NuScale reactor during a station blackout (SBO) accident using RELAP5/SCDAP, finding that the NuScale reactor can effectively remove decay heat through passive safety systems during SBO [17]. However, flow oscillations in the primary loop during the later stages of the accident could potentially damage reactor components. Fakhraei also calculated critical heat fluxes (CHF) along the reactor core height based on the Groenveld look-up table and Bowring correlation during steady-state operation [18]. Zhexi performed a transient simulation of the NuScale PWR's turbine trip using ASTEC code, achieving well-matched results compared with reference data from RELAP5 [19]. Katarzyna simulated a severe accident in a NuScale SMR with RELAP/SCDAPSIM, finding that core damage begins in approximately 4.8 hours, emphasizing the importance of ECCS (Emergency Core Cooling System) [20].

Considering additional potential accidents, the performance of passive safety systems requires detailed investigation. This study simulates and evaluates the response characteristics of passive safety systems in NuScale reactors under SGTR accidents using the RELAP5 Mod 3.4 program.

VERIFICATION

Figure 1 [Figure 1: see original paper] shows a complete module of the NuScale small reactor, composed of a reactor core, pressurizer, and two steam generators, all housed within a pressure vessel enclosed by a containment vessel. Inside the pressure vessel, primary coolant flows through the steam generators, exchanges heat to become cold leg water, moves through the annular downcomer, enters the lower plenum, passes through the core to be heated, turns at the end of the riser section, and returns to the downcomer section containing the steam generators. The containment vessel and two decay heat removal heat exchangers outside it are both located in a large reactor water pool, which can remove residual heat through the decay heat removal system during accident conditions.

NUMERICAL MODELING AND STEADY-STATE

A numerical model based on the above module was built using RELAP5-SCDAP Mod3.4. Figure 2 [Figure 2: see original paper] presents the model node diagram. Key components are described as follows: On the primary side, component 100 represents the reactor lower plenum. The reactor core consists of average channels 121, hot channels 122, and bypass channels 123, with 115 set as the reflector channel. Each channel is modeled as a pipe divided into 20 nodes to ensure sufficient calculation accuracy. The coolant flows from the average channel, hot channel, bypass channel, and reflector channel to the risers 130 and 140, then to the pressurizer 210 and the downcomer 150, and finally returns to the lower plenum 100 via downcomer 160. The secondary loop consists of two circuits. Time control volumes 740 (742) and 741 (743) represent the feedwater tanks, 401 (403) and 402 (404) are the steam generator tube side, time control volume 800 (801) represents the turbine, 860 and 840 are the containment building, 824 (813) represents the residual heat removal heat exchanger, and 900 is the reactor pool. All control volumes marked in red have corresponding heat structures. Sensitivity analyses have been conducted for the mentioned node numbers to ensure calculation results are not affected by node numbers.

Based on the safety analysis report and relevant literature [22][30], key parameters involved in the modeling are summarized in Table 1 . As shown in Table 2 , calculation results under steady-state operating conditions are highly consistent with design values, with small errors. Therefore, the transient process is conducted based on these results.

III. RESULTS AND ANALYSIS

When SGTR occurs, primary coolant flows from the steam generator shell through the break into the tubes, then via the main steam line to the turbine, releasing radioactivity externally until the affected steam generator is isolated by the main steam isolation valve. In this study, the transient sequence of a SGTR accident in the NuScale reactor was simulated. The rupture was assumed to occur at the bottom of the steam generator tube, with a break area equal to 55% of the cross-sectional area of a single tube. Only DHRS was activated under the accident scenario. The accident sequence and simulated response times are summarized below.

Table 3 Accident Sequence

Event	Designed Value	Simulated Value
Maximum primary circuit pressure	-	-
Low pressurizer water level signal	1246 s	1069 s
Pressurizer heater isolation	1248 s	1071 s
Reactor trip	1249 s	1072 s
Low-low pressurizer water level signal	1282 s	1112 s
Secondary loop isolation signal	1283 s	1113 s
Main steam valve isolation	1285 s	1114 s
Main turbine isolation	1285 s	1114 s
Secondary side high-pressure signal	1293 s	1153 s
Decay heat removal system activation	1295 s	1154 s

The safety measures in this accident sequence are triggered by three signals: low pressurizer water level, low-low pressurizer water level, and secondary-side high pressure. Based on thermal phenomena during accident progression and triggering signals, the accident can be divided into five main phases: 1) Pseudo-transient phase of accident initiation; 2) Rapid cooling and depressurization phase after low pressurizer water level and reactor trip; 3) Secondary loop isolation phase after low-low pressurizer water level signal triggers; 4) Transient process and stable natural circulation phase after high pressure of the secondary loop and activation of the decay heat removal system; 5) Unstable flow phase.

Figures 3 to 10 illustrate the dynamic response of key parameters during the pseudo-transient phase following the heat transfer tube rupture. Before 6000 seconds, the system operates in a full-power steady-state condition, after which SGTR occurs. The pseudo-transient phase corresponds to the period from 6000 to 7069 seconds. Figure 3 [Figure 3: see original paper] shows the transient mass flow rate at the break. Initially, no safety systems are active. The break area is much smaller than the reactor core's downcomer. A critical flow forms near the leak, maintaining a stable leakage rate to the secondary side, which is also corroborated by the linear decline of the pressurizer water level shown in Figure 4 [Figure 4: see original paper].

Since the secondary loop is not isolated, the main steam line remains connected to the turbine, and the feedwater tank continuously supplies working fluid. The static head loss of the primary coolant during critical flow formation is converted into dynamic head, preventing significant pressure changes in the secondary loop. Meanwhile, the pressure in the primary loop does not exhibit a notable decrease due to the full-power operation of the pressurizer heaters. Although a slight pressure drop occurs at the moment of rupture, it is quickly offset by steam generated by the pressurizer heaters, leading to minor pressure recovery. The pressure trends in both loops are shown in Figure 5 [Figure 5: see original paper].

Figure 6 [Figure 6: see original paper] presents the changes in temperature at the core inlet and outlet. Due to the rupture, primary coolant directly contacts feedwater in the secondary loop, causing a decrease in core inlet and outlet temperatures. Therefore, based solely on the temperature decrease at the core inlet and outlet, one might conclude that the mass flow rate in the primary loop would decrease. However, as shown in Figure 7 [Figure 7: see original paper], it exhibits a slight increase. This indicates that the critical flow at the rupture location partially offsets the reduction in the density difference driving force. Figure 8 [Figure 8: see original paper] shows the water levels in the secondary side of two steam generators. During this phase, the secondary-side water levels experience only a slight increase. The decrease in temperature at the core inlet and outlet reduces heat absorption in the secondary loop, elevating the water level. Additionally, due to leakage of primary coolant into the tubes of one steam generator, the inlet water level of the heat transfer tubes on that side shows a slight increase and maintains a stable temperature under the influence of steady feedwater supply. The mass flow rate also increases to 19.5 kg/s, as shown in Figures 9 and 10. This phase lasts for approximately 1069 seconds, during which the majority of radioactive materials are released into the turbine.

Upon receiving the low pressurizer water level signal, the reactor enters the rapid cooling and depressurization phase following the reactor trip. The pressurizer heaters are also deactivated, causing the pressurizer water level to drop rapidly. As shown in Figure 4, there is a significant increase in the rate of decline of the pressurizer water level at 7076 seconds, indicating that the accident progression will soon enter the low-low pressurizer water level phase. The primary loop pressure decreases rapidly, while the secondary loop pressure remains relatively unchanged during this phase, as it is still connected to the external environment, as illustrated in Figure 11 [Figure 11: see original paper]. The leakage flow rate at the rupture site also decreases with the reduction of the primary loop pressure, stabilizing around 2 kg/s, as shown in Figure 3.

Figure 12 [Figure 12: see original paper] presents the temperature changes at the core inlet and outlet during this phase. Following the reactor trip, power rapidly drops to 11.8 MW, causing the temperature at the core inlet and outlet to decrease sharply, with the temperature difference narrowing significantly. The outlet temperature drops to only 525 K. Figure 13 [Figure 13: see original paper] shows the transient changes of the mass flow rate in the primary loop, which initially overshoots, dropping to 50 kg/s, and then stabilizes around 100 kg/s.

During the secondary loop isolation phase triggered by the low-low pressurizer water level signal, the isolation sequence involves closure of the main steam isolation valve and the turbine isolation valve. The feedwater isolation valve is designed to respond simultaneously with the main steam isolation valve, meaning that when the main steam valve is isolated, the feedwater isolation valve also closes. After a 1-second delay following the low-low pressurizer water level signal, the secondary loop isolation signal is triggered, and after another 1-second delay, the feedwater isolation valve, main steam isolation valve, and turbine isolation actions are initiated.

Once the secondary loop is isolated, decay heat from the core cannot be effectively removed. The primary loop continues to depressurize due to the reactor trip, while the baseline pressure of the secondary loop is significantly lower than that of the primary loop. Under these conditions, the pressure in the secondary side of the steam generator, now acting as a closed system, begins to rise. This pressure trend is illustrated in Figure 14 [Figure 14: see original paper]. With decay heat unable to be dissipated and secondary loop flow halted, the temperature at the core inlet and outlet exhibits fluctuations, as shown in Figure 15 [Figure 15: see original paper]. This phase lasts for approximately 41 seconds.

When the secondary loop pressure reaches 5.51 MPa, the high-pressure signal of the secondary loop is triggered, entering the transient phase of the residual heat removal system (DHRS) being put into operation and the stable natural circulation stage, continuing up to 12500 seconds. Figures 16 to 21 illustrate the variation patterns of key parameters during this phase. Firstly, Figure

16 [Figure 16: see original paper] shows the water level changes in the steam generators. After DHRS activation, water stored in the DHRS flows into the steam generators, causing water levels to rise. On the intact side, the water level reaches only 5 meters, while the water level on the rupture side rises to 6.87 meters, indicating that the steam generator is already in a full-water state. This significant difference occurs because primary coolant continues to leak into the secondary loop.

Correspondingly, the water level in the DHRHX also shows a significant difference. There is almost no liquid coverage on the intact side, while the heat exchanger on the rupture side, after initially supplying water to the steam generator by gravity due to the height difference, quickly returns to a full-water state due to leaking coolant, as shown in Figure 17 [Figure 17: see original paper]. There is no condensation transfer occurring in the faulted DHRHX; heat is removed solely through conduction. On the intact side, the heat exchanger removes heat by condensing saturated steam, driven by the significant density difference between low-density steam in the exchanger and high-density water in the steam generator. Meanwhile, on the rupture side, the driving force in the upper space is largely hindered by accumulated water in the heat exchanger, preventing low-density steam from condensing and effectively driving natural circulation. This results in the phenomenon shown in Figure 18 [Figure 18: see original paper], where the heat exchanger on the intact side exhibits a noticeable flow rate of approximately 2 kg/s, which gradually decays, while the flow rate on the rupture side decreases more rapidly, approaching zero around 7600 seconds. These results are consistent with findings in the NuScale Final Safety Analysis Report [31].

Figure 19 [Figure 19: see original paper] shows the temperature corresponding to the elevation of the DHRHX in the reactor pool. After absorbing heat carried away by the two DHRHXs, the temperature near the heat exchangers in the reactor pool rises to a maximum of 370 K. Figures 20 and 21 provide the transient temperature at the inlet and outlet for the faulted and intact DHRHX, respectively. It can be seen that the outlet temperature trends on the rupture side closely match temperature changes at the lower part of the reactor pool. Since the variation pattern of inlet and outlet temperatures on the faulted DHRS side is difficult to observe within a relatively short period, extended time results are presented in Figure 20(b), and the temperature in the reactor pool during the corresponding period is also presented in Figure 19(b). Due to the poor thermal conductivity of water, temperature changes on the steam generator side have less significant impact on the heat exchanger outlet compared to the influence of the reactor pool. Similarly, the inlet temperature on the rupture side aligns with temperature changes at the upper part of the reactor pool but exhibits fluctuations. This is likely due to heat exchange between saturated steam (nearly 500 K) and liquid at the heat exchanger inlet, causing temperature oscillations. On the intact side, the temperature difference between inlet and outlet remains around 60 K, with both temperatures decreasing synchronously.

During this phase, DHRS is activated, primarily removing decay heat through natural circulation in the residual heat removal heat exchanger of the intact loop. Within the system, the density difference between the reactor core and steam generator space drives natural circulation. Meanwhile, on the intact side of the secondary loop, the density difference between DHRS and the steam generator also establishes a natural circulation loop. These natural circulation loops interact and constrain each other, collectively facilitating removal of decay heat.

After 12,500 seconds, DHRS continues conducting normal natural circulation to carry decay residual heat. However, various parameters undergo drastic and fluctuating changes. Therefore, corresponding changes during this stage are separately analyzed in depth. Figures 22 to 25 illustrate key parameters during this phase. It can be observed that core inlet and outlet temperatures, primary loop flow rate, inlet and outlet temperatures of the residual heat removal heat exchanger (DHRHX), and mass flow rate of the DHRHX all exhibit oscillations after 12,500 seconds. It should be noted that during SGTR accident, the water level in the primary loop continuously decreases. This is due to both the presence of leakage and phase change caused by decreasing pressure and temperature, where saturated water turns into saturated steam, further lowering the water level. This inevitably leads to liquid separation between the riser and downcomer of the steam generator. As a result, natural circulation of liquid within the reactor is nearly halted, with only steam continuing to flow through the upper sections, as illustrated in Figure 26 [Figure 26: see original paper]. This results in oscillatory changes in parameters across various regions.

Here, it is further verified that the minimum critical heat flux density ratio at the core is 1.66. This confirms that the passive residual heat removal system can still effectively prevent boiling criticality in the core region. The flow oscillations in the core area will not lead to core meltdown.

IV. CONCLUSION

A detailed analysis of the thermal-hydraulic response of the NuScale small modular reactor (SMR) during a steam generator tube rupture (SGTR) accident was conducted using Relap5 code. The conclusions are as follows:

1. The faulted DHRHX has a limited contribution to dissipating core decay heat during the accident process. The DHRHX on the intact side can independently eliminate core decay heat and effectively control core inlet and outlet temperatures.

2. When leakage in the primary circuit causes the water storage volume to drop below the pressurizer baffle, significant flow oscillation occurs due to disconnection of the liquid-phase coolant in the upper space, thereby triggering fluctuations in remaining reactor parameters.

3. Under conservative decay power settings, DHRS can still effectively pre-

vent core melting, but long-term flow oscillation may have adverse effects on components inside the core.

In this study, we mainly analyzed the tube rupture accident at the SG bottom of the NuScale small modular reactor. For future work, further research will be performed to investigate the sensitivity of the rupture, including accident progression under different rupture heights, different rupture areas, and different numbers of ruptures, to reveal system response characteristics under various conditions and comprehensively verify the effectiveness of the decay heat removal system.

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Note: Figure translations are in progress. See original paper for figures.