Analysis of PWR Primary Circuit Response Characteristics under SBLOCAs Considering Steam Bypass Discharge

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Abstract

Small-break superposed station blackout (SBO) accidents are the basic design accidents of nuclear power plants. Under the condition of a small break in the cold leg, SBO further increases the severity of the accident, and the steam bypass discharging system (GCT) in the second circuit can play an important role in guaranteeing core safety. To explore the influence of the GCT on the thermal-hydraulic characteristics of the primary circuit, RELAP5 software was used to establish a numerical model based on a typical pressurized water reactor (PWR) nuclear power plant. Five different small breaks in the cold-leg superposed SBO were selected, and the impact of the GCT operation on the transient response characteristics of the primary and secondary circuit systems was analyzed. The results show that the GCT plays an indispensable role in core heat removal during an accident; otherwise, core safety cannot be guaranteed. The GCT was used in conjunction with the primary safety injection system during the placement process. When the break diameter was greater than a certain critical value, the core cooling rate could not be guaranteed to be less than 100 K/h; however, the core remained in a safe state.

Full Text

Preamble

Response Characteristics of PWR Primary Circuit Under SBLOCAs Considering Steam Bypass Discharging

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Abstract

Small-break loss-of-coolant accidents (SBLOCAs) superimposed with station blackout (SBO) represent basic design-basis accidents for nuclear power plants. Under cold-leg small-break conditions, SBO further exacerbates accident severity, while the steam bypass discharge system (GCT) in the secondary circuit plays a crucial role in ensuring core safety. To investigate the influence of GCT on the thermal-hydraulic characteristics of the primary circuit, a numerical model based on a typical pressurized water reactor (PWR) was developed using RELAP5 software. Five different small-break sizes in the cold leg combined with SBO were selected, and the impact of GCT operation on the transient response characteristics of both primary and secondary circuit systems was analyzed. The results demonstrate that GCT plays an indispensable role in core heat removal during accidents; without it, core safety cannot be guaranteed. GCT operates in conjunction with the primary safety injection system during the mitigation process. When the break diameter exceeds a certain critical value, the core cooling rate cannot be maintained below 100 K/h; however, the core remains in a safe state.

Keywords: Steam bypass discharging; Pressurized water reactor; SBLOCA; Numerical simulation

Nomenclature

- **PWR:** Pressurized water reactor
- **SBO:** Station blackout
- **GCT:** Steam bypass discharge system
- **SG:** Steam generator
- **ASG:** Auxiliary feedwater system
- **HHSI:** High-head safety injection system
- **MHSI:** Middle-head safety injection system
- **LHSI:** Low-head safety injection system
- **RRA:** Residual heat removal system
- **SDCS:** Steam dump control system
- **FWCS:** Feedwater control system
- **ATWS:** Anticipated transient without scram
- **VDA:** Steam direct emission
- **SGTR:** Steam generator tube rupture
- **SBLOCA:** Small-break loss-of-coolant accident

1. Introduction

The GCT represents a critical component of nuclear power plant safety systems. When steam turbine load decreases sharply, reactor power cannot be reduced synchronously, resulting in a mismatch between instantaneous core power and turbine load. Under such conditions, the steam bypass system provides an “artificial” load for the reactor, maintains power balance between primary and secondary circuits, and ensures plant safety [1]. Extensive research has been conducted to understand the mechanisms of steam bypass discharge.

Jia [2, 3] performed comparative analyses between passive nuclear power plants and improved second-generation plants, demonstrating that control logic and objectives of passive plant steam discharge systems were enhanced. Duk et al. [4] simulated steam discharge valve capacity changes under power-uprated conditions, identifying optimal setpoints of 0.6 for the feedwater control system (FWCS) and 15.0 for the proportional band of the steam dump control system (SDCS). Lee [5] investigated steam emission control system parameter changes under load rejection with insufficient main feedwater, exploring optimal setting parameters. Wang et al. [6] developed a real-time simulation model of steam discharge control systems that showed good agreement with actual nuclear power plant operational data, establishing a solid foundation for subsequent parameter optimization and semi-physical simulation. Lu et al. [7, 8] simulated steam discharge in second-generation nuclear power plants during shutdown accidents, verifying the applicable scope and safety of steam discharge systems under various operating conditions. Relevant studies [9–11] have proposed reasonable response recommendations through simulation and analysis of steam discharge system control logic to address power reduction, reactor shutdown, and other accident conditions. Fan et al. [12, 13] analyzed transient response characteristics of full exhaust condensers in PWR nuclear power plant steam discharge systems, providing significant insights for steam turbine bypass discharge control system design. Zhao et al. [14] examined system response when GCT121VV failed to open normally under full-power turbine trip conditions, discussing generator unit control under such fault conditions.

Studies on AP1000 nuclear power plant GCT [15–17] demonstrate that the steam bypass system combined with reactor power control enables AP1000 designs to withstand load rejection or turbine power transitions without reactor trip, atmospheric discharge, or activation of pressurizer and steam generator (SG) safety valves. Tian [18] employed list analysis methods to describe steam dump valve site layout and the design, structure, and solenoid valve control of SDCS, enhancing personnel understanding of AP1000 steam dump valves. Dong [19] elaborated on steam emission system operation modes, control, and lockout functions, conducting accident analysis under anticipated transient without scram (ATWS) conditions to provide guidance for operation and maintenance. Zhou [20] controlled primary circuit average temperature and main steam header pressure, simulated steam discharge processes, and obtained improved solutions. Zhang [21] introduced an automatic control scheme for atmospheric discharge valves in which HPR1000 automatically triggers atmospheric discharge valves for regulation using safety injection signals, enabling rapid reactor coolant system cooldown and significantly reducing safety risks from human error. Sui et al. [22] used RELAP5 software to analyze HPR1000 primary circuit response characteristics under different steam direct emission (VDA) conditions after SBLOCA, demonstrating that VDA emission capacity is proportional to reactivity feedback and recommending VDA actuation before a critical point. Song et al. [23] studied steam generator tube rupture (SGTR) accident conditions in APR1400 nuclear power plants, revealing that atmospheric steam discharge systems can significantly slow core melting progression. Chen et al. [24] used RELAP5 to simulate SBLOCA in a million-kilowatt nuclear power plant, finding that GCT-a combined with pressurizer spray could effectively reduce primary circuit pressure, enable safety injection tank operation, replenish primary circuit coolant, and effectively control accidents. Cai et al. [25] studied AP1000 turbine bypass control functions, demonstrating that it could meet any unit transient operating conditions when used with reactor power control without generating emergency shutdown or safety valve actions. Liu et al. [26] used RELAP5 code to develop a control model and performed variable load dynamic simulations on SGs, showing that the control model responded quickly and exhibited good stability effects on coolant temperature and pressure.

In summary, numerical simulation studies of steam discharge characteristics under accident conditions have received significant attention and achieved notable results. However, parameter coupling characteristics between primary and secondary circuits during steam discharge under SBLOCA conditions require further analysis. This study establishes a complete nuclear power plant model using RELAP5 software to investigate primary circuit response characteristics—including pressure, flow rate, temperature, and other parameters—when operating the steam discharge system and primary circuit safety injection system under five different SBLOCA scenarios combined with whole-plant power failure.

2.1 RELAP5 Modeling

A typical three-loop PWR nuclear power plant system model was developed using the RELAP5 code. The model included a reactor pressure vessel and core, three coolant loops, pressurizer (232), break (502), residual heat removal system (RRA; 702 outlet; 721, 741 inlet), safety injection system (510–590), main steam and main feedwater system (280–486), auxiliary feedwater system (600–620), steam bypass system (295–495), steam generators, and other required components.

The RELAP5 code employs control volumes as computational units. As node count increases, calculation deviation gradually decreases until becoming negligible, while computational time increases [27, 28]. After comprehensive consideration, the optimal node configuration was determined as shown in Fig. 1 [Figure 1: see original paper], with the core key section divided into 20 nodes (active zones ranging from nodes 5 to 16).

Reactor coolant flows from the pressure vessel inlet (101) into the downcomer annulus (105) to the vessel bottom (120), then enters the reactor core through the core support plate (124) and lower grid plate (126), where heat is generated through nuclear fission of fuel elements. The coolant flows through the upper grid plate (138) and mixes with core side flow (135), guide tube side flow (110–

145), and core spoke side flow (101–140). The mixed coolant enters the three hot legs from the pressure vessel outlet (140). Using the first loop as an example, coolant flows from the pressure vessel outlet to the SG lower head (219), enters the U-tube heat transfer region (221) for heat exchange, then flows through the pump (225) from the lower head (223) back to the pressure vessel. Secondary main feedwater (282) mixes with fluid (272) and separates in the SG steam-water separator (371), enters the SG bottom through the U-shell side (270) to exchange heat with the primary circuit, then the generated steam enters the main steam line (286) through the steam-water separator and dryer (380) before returning to the main feedwater cycle after turbine power generation and subsequent processing.

2.2 Treatment of GCT Cooling Strategy

To ensure a safe core cooling rate during rapid cooldown, the average cooling rates at the reactor core inlet and outlet were set and maintained at 100 K/h following GCT actuation [29]. As shown in Fig. 2 [Figure 2: see original paper], three groups of parallel electric control valves (GCT1, GCT2, and GCT3) are installed in the GCT bypass discharge pipeline, with primary circuit steam pressure divided into three sections. Logic cards control these valves to operate within corresponding pressure intervals to achieve automatic adjustment: when one valve group opens, the other two groups close. This approach enables GCT to operate in conjunction with primary safety injection flow while meeting system requirements for stable core heat removal rates, as detailed in Table 1 .

2.3 Steady-State Verification

A steady-state simulation was first performed using 900 MWe nuclear power plant parameters as a benchmark. The results demonstrate matched heat transfer between primary and secondary circuits, with essentially identical parameters across the three loops. A comparison between numerical results and design values is presented in Table 2 . The basic parameters under steady-state conditions show good agreement with actual plant operating parameters, with relative deviations within reasonable ranges, confirming model reliability. Consequently, transient analysis under accident conditions can be performed.

3. Accident Sequence

The reference case was established as a typical small-break LOCA superimposed with station blackout (SBO), with the break located in the cold leg of the third loop (without pressurizer). Five small-break sizes were employed, with diameters of 10, 30, 50, 70, and 100 mm.

Assuming a small break occurs in the cold leg after 600 s of steady-state operation, primary pressure plummets, the control system intervenes, and an emergency shutdown is triggered. When pressurizer pressure falls below 13 MPa with superimposed SBO, the main pump stops operating and main feedwater supply ceases. Subsequently, the SG outlet steam line valve closes. Different break characteristics gradually manifest during the natural circulation stage. When pressurizer pressure drops below 11.7 MPa, the high-head safety injection system (HHSI) and GCT are actuated. With termination of natural circulation, the U-tubes on the SG secondary side become uncovered, SG heat removal becomes impaired, and core temperature increases unsafely. This represents the main phase of the accident. With supply from the auxiliary feedwater system (ASG), safety injection system, and RRA, the SG secondary side and core water levels are replenished, resulting in decreased cladding temperature. Finally, the reactor enters recirculation mode to achieve long-term core cooling. The transient simulation extends for 6500 s. Results under accident conditions demonstrate that transient processes are consistent with previous numerical studies by domestic and international experts [30–32]. The corresponding accident sequence is presented in Table 3 .

4. Results and Analysis

Fig. 3 [Figure 3: see original paper] illustrates transient pressures in primary and secondary circuits. A small break occurs in the coolant pipeline at 600 s, causing instantaneous primary circuit pressure drop, while secondary circuit response is slower. When pressurizer pressure reaches 13.0 MPa, emergency shutdown is triggered, and turbine stop valves and main feedwater valves close sequentially, leading to reactor coolant pump coastdown. As reactor coolant system pressure continues decreasing, GCT and primary safety injection systems are actuated, accelerating core cooling. When middle-head safety injection system (MHSI) operates, primary and secondary circuit pressures drop to lower levels and gradually decelerate. Following actuation of RRA and low-head safety injection system (LHSI), system pressure decreases further before entering the long-term cooling phase.

Under the 10 mm break accident condition, primary circuit pressure decreases slowly, requiring extended depressurization time before reactor shutdown initiation. As break size increases, system parameters change more rapidly, and each system responds earlier.

Figs. 4–5 present fluid temperatures at core inlet and outlet, SG secondary side fluid temperature, cladding temperature, and core minimum dimensionless liquid content, respectively. After reactor shutdown, the system primarily uses GCT and the break as heat sinks for heat removal. Larger breaks result in greater primary system heat loss through the break and faster core fluid temperature decrease. As injection systems are actuated, they further provide heat removal capacity for the core, causing significant core coolant temperature drops. When primary circuit pressure falls below 3.0 MPa and temperature below 180 °C, RRA is actuated. Coolant is extracted from the secondary circuit, cooled, and injected into the first and third loops. RRA actuation effects are significant, contributing to rapid primary circuit temperature reduction. LHSI follows closely, with RRA and LHSI providing stable heat sinks for the core.

Primary circuit temperature decreases rapidly and enters the unsaturated state after brief oscillations. Finally, the system enters long-term cooling stage. The cladding temperature curve in Fig. 4(d) [Figure 4: see original paper] shows that the core remained in a safe state throughout. As shown in Fig. 4(a), gas-liquid mixture persists at the core top throughout the process, resulting in essentially identical cladding temperature variations across different active zone positions.

During the accident progression, three break conditions (10, 30, and 50 mm) satisfy the requirement that core cooling rate does not exceed 100 K/h. However, under 70 and 100 mm break conditions, the large break size causes rapid reactor coolant system pressure drop. More heat is removed through the break, and the core cannot be cooled at a safe rate. Nevertheless, no abnormalities are observed in core cladding temperature.

Fig. 6 [Figure 6: see original paper] shows transient mass flow rates through the break, reactor core, and GCT. Main pump stoppage causes rapid core coolant flow reduction. After HHSI actuation, break flow decreases due to primary circuit pressure drop. Core coolant flow drops to low values and enters an oscillatory state. At this stage, the SG heat sink demand from the core increases. SG water absorbs heat, evaporates, and is rapidly discharged to atmosphere at high flow rate through GCT1. Primary pressure and temperature decrease rapidly, causing core coolant mass flow rate fluctuations. After MHSI actuation, primary circuit demand for secondary circuit heat sink weakens. GCT flow rate was reduced to satisfy stable core cooling rate requirements. When RRA and LHSI are actuated, loop oscillations occur due to flashing of newly injected coolant, causing GCT3 flow rate increase to meet long-term safety heat removal requirements.

Under 10 mm break conditions, primary circuit coolant loss is minimal. Due to pressurizer replenishment, coolant mass flow rate remains unchanged during the initial period, requiring extended pressure reduction to reach shutdown conditions.

Fig. 7 [Figure 7: see original paper] presents variations in core, pressurizer, and SG collapsed liquid levels. The pressurizer responds quickly to replenish water volume in the first loop after break occurrence, ensuring core submergence during the initial stage. After reactor shutdown, core collapsed level decreases gradually. When HHSI operates, slight oscillations occur in core collapsed liquid level. GCT opening forces SG working fluid evaporation, causing water level decrease and SG heat transfer efficiency deterioration. This accelerates core collapsed liquid level decrease rate. Subsequently, SG collapsed liquid level drops below 35%, triggering auxiliary feedwater (ASG) supply to refill the SG. As shown in Fig. 7(a), core collapsed level exhibits small fluctuations after auxiliary feedwater application. When MHSI operates, core coolant is replenished and core collapsed level decrease rate slows. Core collapsed level reaches a relative minimum by the time LHSI and RRA are actuated, followed by rapid core water level recovery to flooded state.

Under 10 mm break conditions, pressurizer liquid level decreases slowly to replenish coolant lost through the break, and the core remains submerged throughout the process. Under other break size conditions (30-100 mm), core collapsed liquid level decrease rate increases with break size, and minimum core collapsed liquid level values are lower. Larger break size accident conditions trigger injection and RRA systems earlier, resulting in earlier core water level recovery.

Fig. 8 [Figure 8: see original paper] shows maximum void fraction in the core active region. Compared with Fig. 7(a), void appears in the core active region as core collapsed liquid level gradually decreases. As shown in Figs. 3 and 4(b), coolant reaches near-saturated conditions after MHSI startup, and core collapsed liquid level decreases. Void fraction at the active zone top increases with break size, maintaining approximately 70% under 100 mm break conditions. Subsequently, void fraction increases slowly, reaching an instantaneous maximum of 85%. After RRA and LHSI actuation, the reactor enters long-term cooling, and void in the core active region gradually disappears.

For comparative analysis, corresponding conditions without GCT actuation were calculated. Fig. 9 [Figure 9: see original paper] shows primary and secondary circuit pressures, core coolant temperature, and core collapsed liquid level. After scram, the reactor system releases heat through the break and SG. Primary circuit temperature and pressure decrease normally, while secondary circuit temperature and pressure increase rapidly. Due to GCT absence in the secondary circuit, the secondary circuit cannot function as a heat sink when temperature and pressure increase to certain values. At this point, primary circuit heat cannot be released to the secondary circuit through the SG, and coolant loss through the break becomes the primary heat removal mechanism.

Under small-break conditions, the core loses minimal coolant through the break, and break heat removal capacity is limited; consequently, primary circuit temperature and pressure can no longer decrease after reaching certain levels, and MHSI and RRA cannot intervene because trigger conditions are not met. This results in inability to replenish coolant to the core for extended periods and failure to ensure core safety. Under larger break conditions, strong break heat removal capacity triggers middle-pressure injection system at later stages; primary circuit temperature and pressure continue decreasing but fail to reach low-pressure injection system and RRA actuation conditions. The core is not replenished with coolant, cannot enter long-term cooling phase, and cladding temperature continues increasing. Zirconium alloy reacts with water vapor when cladding temperature reaches 820 °C and melts at 1200 °C. Therefore, reactor safety requires maintaining fuel element temperature below 820 °C [33]. Based on top (minimum) dimensionless liquid content and envelope temperature of the core active zone in Figs. 9(e) and 9(f), the reactor core remains in an unsafe high-temperature, high-pressure state under 10 mm break accident conditions. Under 30-100 mm break conditions, the top of the core active zone becomes exposed to varying degrees, and cladding temperature exceeds fuel element safety temperature, potentially leading to core melt.

5. Conclusion

A PWR nuclear power plant numerical model was developed using RELAP5 code to analyze GCT effects on transient response during SBLOCA superimposed with SBO. The main conclusions are:

(1) When break diameter is less than 50 mm, GCT operates in conjunction with the primary safety injection system during the mitigation process, and core cooling can be maintained at a safe rate (100 K/h). When break diameter exceeds 50 mm, core cooling rate becomes excessively high; however, the core does not become substantially uncovered, and cladding temperature remains normal throughout the accident.

(2) Upon GCT actuation, it plays an important role in the initial stage with high GCT flow rate to provide sufficient heat removal capacity for the core. After MHSI actuation, demand for GCT weakens, and GCT mass flow rate decreases accordingly. It is recommended that GCT regulation capacity and response speed be improved to enable faster response during early accident stages.

(3) Without GCT operation, SG shell-side pressure exceeds limit requirements due to inability to export heat in a timely manner. Under large-break conditions, heat release through the break can reduce core temperature and pressure to varying degrees. However, due to inability of safety injection system and RRA to operate, fuel element overheating occurs. Therefore, GCT cooperation with safety systems is necessary for effective cooling.

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