<Technical Report>

PARAMETRIC STUDIES ON THERMAL HYDRAULIC CHARACTERISTICS FOR TRANSIENT OPERATIONS OF AN INTEGRAL TYPE REACTOR

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Transient operations for an integral type reactor, SMART-P, have been experimentally investigated using a thermal-hydraulic integral test facility, VISTA (Experimental Verification by Integral Simulation of Transients and Accidents), in order to verify the system design and performance of the SMART-P, a pilot plant of SMART. The VISTA facility was subjected to various accident conditions such as feedwater increase and decrease, loss of coolant flow, and control rod withdrawal accidents in order to elucidate the thermal-hydraulic responses following such accidents and finally to verify the system design of the SMART-P. Full functional control logics have been implemented in the VISTA facility in order to control the required control action for an accident simulation. As one of the sensitivity tests to verify the PRHRS performance, the effects of the initial water level in the compensation tank are experimentally investigated. When the initial water level is 16%, the water is quickly drained and nitrogen gas is then introduced into the PRHR system, resulting in deterioration of the PRHRS performance. It is thus found that nitrogen ingression should be prevented to ensure stable PRHRS operation.

KEYWORDS : SMART-p, Integral Type Reactor, VISTA Facility, Safety-related Accidents, Transient Operations, PRHRS Performance

1. INTRODUCTION

The SMART reactor is an advanced integral type reactor with a power of 330MWt. It has several enhanced safety features, whose major RCS components, such as the main coolant pumps, helical-coiled tube bundle steam generators, and pressurizers, are contained in a reactor vessel. This integral design approach eliminates the large coolant loop piping, thus precluding occurrence of a large break LOCA. A passive residual heat removal system (PRHRS) is also installed to prevent over-heating and over-pressurization of the primary system during accidental conditions. The PRHRS removes the core decay heat through the steam generators by a natural circulation of the two-phase fluid. The basic design of the SMART was completed in 2002 by KAERI and a prototypic SMART plant, the SMART-P, having a rated power of 65MWt, will be constructed within six years in Korea. [1,2,3]

The VISTA facility (Experimental Verification by Integral Simulation of Transients and Accidents) has been constructed to investigate the overall thermal-hydraulic behavior of the SMART-P. It has the same height and 1/96 volume with respect to the SMART-P and is used to understand the thermal-hydraulic responses following transients and finally to verify the system design of the SMART-P. The reactor core is simulated by 36 electrical heaters with a capacity of 818.75kW. A schematic diagram of the VISTA facility is shown in Figure 1.

Unlike the integrated arrangements of the SMART-P, the primary components including a reactor vessel, a main coolant pump, a helical-coiled steam generator, and pressurizers are interconnected by pipes for easy installation of the instrumentation and simple maintenance. A secondary system having a single train is simply designed to remove the primary heat source. Besides these major systems, a make-up water system and a chilled water system are installed to control the feedwater supply and its temperature. Some of the safety-related systems to simulate a piping break and safety injection will be installed after carrying out performance tests such as normal operation and operational transients as well as some accidents. The PRHR system of the VISTA facility is composed of a single train of the cooling sub-system, which includes an emergency cooldown tank (ECT), a heat exchanger (HX), a compensating tank (CT), several valves, and related piping. The PRHR system is connected to both the feedwater and steam

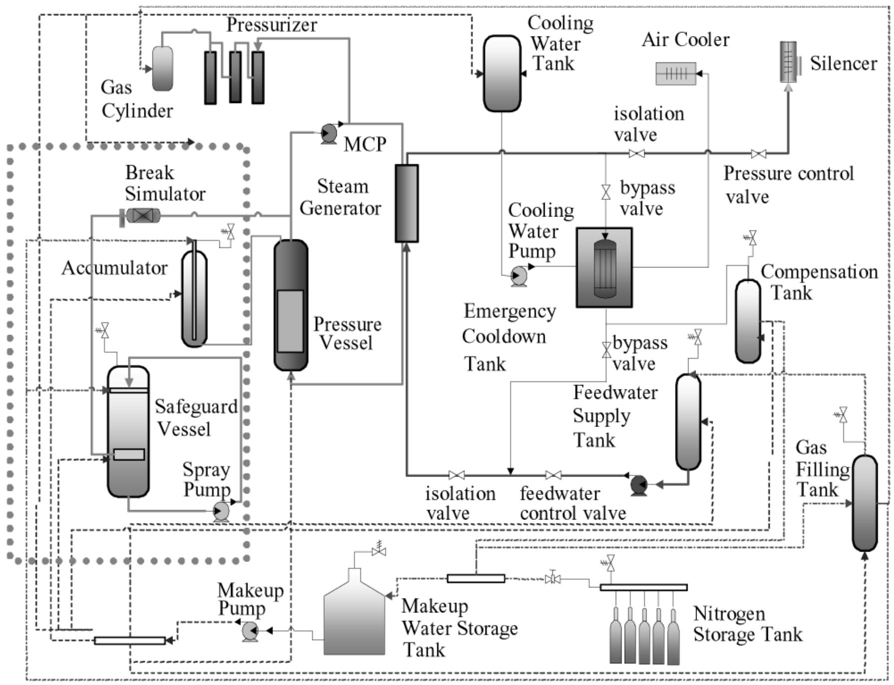

Fig. 1. Schematic Diagram of the VISTA Facility

lines of the secondary system to create a flow path with natural circulation.

An automatic control system has been installed at the VISTA facility to control the major thermal hydraulic parameters following the operator's instructions. In addition, several automatic control logics are implemented for startup, heatup, and transient operations. The same control logics as the prototype plant SMART-P are used in the VISTA facility for the core power control. The core power of the SMART-P is controlled by either a T-control or a T+N control method. The T-control method is a control method to adjust the core power according to the core exit coolant temperature and is designed to be used for high primary coolant conditions. On the other hand, the T+N control method is for low primary coolant conditions and it uses the core exit temperature as well as the core power itself as control inputs. Detailed information regarding the VISTA facility can be found elsewhere [4,5,6,7].

Thus far, several tests have been carried out and reported in the literature, including normal performance tests, transient response tests, and preliminary PRHR system performance tests. [8,9,10,11]. The normal performance tests imply individual performance tests of each component, such as the core heaters, the main coolant pump, the steam generator, and heat exchangers. The transient response tests include power step/ramp change operations and MCP speed change operations.

The main focus of this paper is safety related accident tests such as a feedwater increase/decrease, loss of coolant flow, and control rod withdrawal accidents. The main objectives of this study are to investigate the thermal hydraulic responses of the SMART-P during several accident situations and to verify the design and operation procedure of the SMART-P. Regarding the thermal-hydraulic behaviors of the PRHR system, limited research can be found in the literature. Chung et al. [12] investigated the thermal-hydraulic characteristics for the PRHRS in the SMART using the MARS code. They showed that the PRHRS fulfilled its functions in removing the heat transferred from the primary side in the steam generator when the heat exchanger is submerged in the emergency cooldown tank. In this paper experimental investigations of the heat transfer characteristics and the natural circulation performance of the PRHRS of the VISTA facility are reported in detail. In particular, the effects of the initial water level in the compensation tank on the heat transfer characteristics of the natural circulation are investigated.

2. DESCRIPTION OF THE VISTA FACILITY

2.1 Fluid Systems

The fluid system of the VISTA facility consists of the primary system, the secondary system, the PRHR system, and the auxiliary system. The primary system includes a reactor pressure vessel, a main coolant pump, three pressurizers, and a steam generator. The primary components of the SMART-P are simplified as a loop-type in the VISTA facility. The reactor core is simulated by three groups of electrical heaters. The main coolant pump supplies the primary system with forced convective flow to remove the heat generated from the core. One axial-type canned motor pump is used in the VISTA facility. The end cavity (EC), the intermediate cavity (IC), and the upper annular cavity (UAC) of the SMART-P are simulated by three independent cylindrical vessels. Each vessel is connected by a separate pipe to simulate a surge flow. The upper annular cavity is connected to the hot leg via a surge line. The volume of each cavity is scaled down at a ratio of 1/96 and the height and the elevation are preserved. The volumes of the connecting line between the cavities are designed to be larger than the scaled volume so as to avoid excessive friction. The steam generator is scaled down to 1/8 with respect to one steam generator cassette of the SMART-P based on the same scale law applied to the other components. The primary coolant enters the inlet of the SG and flows down through the shell side, forming a countercurrent flow with respect to the secondary feedwater flow inside the tube. The subcooled feedwater supplied from the feedwater supply tank (FWST) is heated in the helical tube by the primary coolant. When the feedwater flows into the tube it boils and finally exits in a superheated condition.

The secondary system consists of the feedwater supply system and the steam discharge system. Each system is simplified to have a single train. The feedwater supplies the demineralized water at a pressure of 4.55MPa and a temperature from 40°C to 70°C to the steam generator.

The steam discharge system is open to the atmosphere for dumping the steam generated in the steam generator. In the steam discharge system, there is a branch point at which the passive residual heat removal (PRHR) system starts. In the feed water supply system, there is also a branch point at which the PRHR connects. When the PRHRS starts to operate, the steam goes to the PRHR system, condenses in the heat exchanger tubes submerged in the emergency cooldown tank (ECT), and flows back to the feed water system in a liquid phase.

The PRHRS of the VISTA facility is composed of a train for the cooling subsystem, which includes an emergency cooldown tank (ECT), a heat exchanger (HX), a compensating tank (CT), several valves, and the related piping. The PRHRS of the VISTA facility should have the capability to simulate both passive and active cooling of the reference system. It is connected to both the feedwater and steam lines of the secondary system to provide a flow path for the natural circulation. Sufficient cooling water should be supplied to the ECT from the component cooling water system (CCWS) to remove the heat from the internal HX efficiently, and enough water and nitrogen gas should be supplied to the CT from the makeup water system (MWS) and the nitrogen supply system (NSS), respectively. The PRHR system is designed to have the same pressure drop and heat transfer characteristics and is arranged to have the same elevation and position as those of the referen-ce system. In addition, the diameter, thickness, pitch, and orientation of the heat exchanger tubes of the VISTA facility correspond with those of the reference system. Only one train of the cooling subsystem is installed in the VISTA facility. The PRHRS can be isolated from the secondary system using bypass valves in normal operating conditions. The location of the components, pressure drop characteristics, and the heat exchanging capabilities are properly calculated and reflected to have the same natural circulation capability as the reference system. During the PRHRS operation, the superheated steam generated from the steam generator secondary side is injected into and condensed in the PRHRS heat exchangers by natural circulation. The condensed water is drained through the PRHRS condensate line and returned to the feedwater line of the secondary system. The condensing heat is transferred to the emergency cooldown tank, the heat of which is removed by the component cooling water. The PRHRS heat exchanger is installed vertically in the emergency cooldown tank and is composed of 6 condensing tubes, as well as upper and lower headers. The steam from the steam line

Table 1. Test Matrix for Transient Operations

Test IDs	Description
H-FWUP-50-T	Feedwater increase (50% -> 100%)
H-FWDN-100-T	Feedwater decrease (100% -> 50%)
H-LOFA-100H-T H-LOFA-50H-T H-LOFA-50M-T	Loss of flow @ 100% with high MCP @ 50% with high MCP @ 50% with mid MCP
H-CRWD-100-T	Control rod withdrawal @100% power
PRHRS-P-R1a	Reference case (water level of 80%)
PRHRS-P-R1-C2	Initial water level in the compensation tank is 16% and the gas cylinder is isolated

of the secondary system is cooled and condensed in a heat exchanger submerged in the emergency cooldown tank, and the condensed water is re-circulated into the feedwater line of the secondary system and passes through the secondary side of the steam generator to cool the primary system. In order to simulate decay power in the core when a boiler is tripped, the electrical power is controlled in conformity with the ANS-73 decay curve.

2.2 Control Systems

The VISTA facility is designed to be operated by a combination of manual and automatic operation. Several initial operations such as inventory filling, startup, and cooldown are operated manually by an operator. Once the major thermal-hydraulic parameters reach a steady state condition, they are switched over to be controlled automatically by PID feedback control logics so as to maintain the achieved steady state condition. For automatic control, several PID control blocks are installed into a programmable logic controller (PLC) in the control system. Optimized PID values for each control component are determined by trial and error by considering the system's characteristics [4]. The controlled components include the electrical heating rod, the main coolant pump, the feedwater control valve, the steam pressure control valve, the FWST heater, and the makeup pump. During normal operation the electrical heater power is automatically controlled to give a constant core exit temperature, e.g., 300°C. A special combination of three heater groups is used to prevent the maximum heater surface temperature from exceeding the safety limit. During the PRHRS operation the predetermined power from the programmed ANS-73 decay curve is given to the core simulating heater.

3. RESULTS AND DISCUSSIONS

3.1 Test Matrix

Tests carried out in this study can be grouped into two categories, tests for safety related accident tests and for PRHRS performance tests, as summarized in Table 1. First, a feedwater increase event was investigated. Before the transient operation, the VISTA was set to operate at 50% power. The transient was initiated by increasing the secondary feedwater flow rate from 50% to 100% at 20%/sec. During the process, the core power was controlled by the T-control method. The second item, a feedwater decrease event, was also carried out after the VISTA reached a steady state condition at 100% power. Loss of coolant flow accidents (LOFAs) were also simulated in the present study. LOFAs can occur at several different initial conditions, depending on the core power and MCP operation modes. Hence, we determined three representative cases: a case at 100% core power with the MCP running at high speed, a case at 50% core power with the MCP at high speed, and a case at 50% core power with the MCP at medium speed. Finally, a control rod withdrawal accident was tested at 100% core power. As we are using electrical heaters to simulate the reactor core, neutronic effects such as the modeator temperature coefficients or Doppler reactivity are excluded in this study. The main focus of this work is on the thermal hydraulic aspects during the transient. The test was initiated by increasing the core power stepwise to the maximum value.

Table 1 also includes a sensitivity test item for verification of the PRHR system performance. Several test items for verification of the PRHRS performance have been identified by SMART designers; these include the effects of the opening delay time of the isolation valves, installation of a check valve downstream the emergency cooldown tank (ECT), initial water level of the compensation tank, initial nitrogen pressure of the compensation tank, and valve operation method downstream the compensation tank. Each test series is defined from A-series to E-series and conceptually described in Figure 2.

Among the identified test series, the effects of the initial water level in the compensation tank on the performance of the PRHRS are experimentally examined in the present study. The reference case indicates the design case, in which the water level in the compensation tank is 80%. Hence, in this case, the sensitivity test C2 was conducted with an initial water level of 16%.

3.2 Safety Related Accident Results

All the tests in the present study were initiated and controlled by automatic control logics developed by the authors. In particular, the safety related items require automatic reactor trip logics for initiating the PRHR system. Therefore, the corresponding trip logics and trip set points are programmed to control the transient progression.

3.2.1 Feedwater Increasing Accident

Figures 3 through 5 show typical results for the feedwater increase accident. As shown in Figure 3, the feedwater was rapidly increased from 50% to 100%. Due to the increased feedwater injection to the steam generator, the steam pressure sharply increases from the design value, 3.55MPa. Figure 4 shows the primary pressure and core power trend during this test. In this case, the reactor core is programmed to be tripped by either high reactor power (115%) or low pressurizer pressure (11.62MPa). It is found that the system reaches the high reactor power set point earlier than the low pressurizer pressure set point. The minimum pressurizer pressure is observed to be about 13.5MPa, which is still higher than the trip set point. It took around 60 seconds to reach the trip set point. Figure 5 shows the core temperature variation. It was observed that the core exit temperature suddenly drops slightly, but it recovers its previous value due to the increased power. After the reactor trip, the PRHR system is triggered to cool

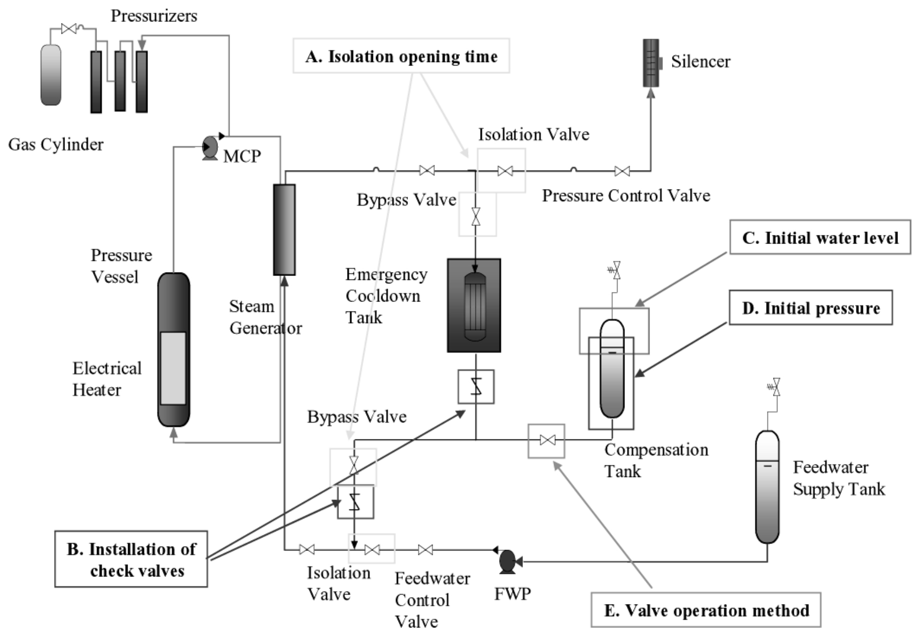

Fig. 2. Sensitivity Test Items for Performance Verification of the PRHRS

down the primary system. During the operation of the PRHRS, the primary pressure and temperature gradually decrease and the secondary pressure decreases as well.

3.2.2 Loss of Coolant Flow Accident

Figures 6 through 8 show the experimental results for a loss of coolant flow accident test, H-LOFA-100H-T. As an initial condition, the core power was set to 100% and the MCP was running in high speed mode. Figure 6 shows the primary coolant flow rate. The transition starts at about 111 seconds by switching off the MCP. Although the MCP is tripped, the primary coolant still circulates in the primary loop by natural circulation. The measured natural circulation flow rate is about 14% of the rated value. In this case, the reactor core was programmed to be tripped by a high pressurizer pressure (16.44MPa). Figure 7 shows the primary pressure and core power variation during the transition. The reactor core was tripped by the high pressurizer pressure trip set point 14 seconds after the transition. Figure 8 shows the core inlet and exit temperature. The core exit temperature initially shows a sharp peak due to a sudden loss of the coolant flow rate. The sharp peak directly results in a sharp volume expansion in the primary inventory and a sharp pressure spike, enough to trip the reactor core. After the trip, the PRHR system is initiated and continuously cools down the system.

Figures 9 and 10 show the experimental results for a loss of coolant flow accident test, H-LOFA-50H-T. In this case, the initial core power was set to 50% rather than 100%. Initially, the MCP is operated at the high speed mode. The reactor trip is programmed to be initiated by a high pressurizer pressure (16.44MPa). Sudden loss of the coolant flow can be observed in Figure 9. The primary coolant flow rapidly drops, as a result of switching off the MCP, but natural circulation flow can be observed even though the flow shows frequent oscillation. The variation of the core exit temperature during the transient is also plotted in Figure 9.

Figure 10 shows the variation of the primary pressure and the core power. Upon the transition, the core exit temperature rapidly increases due to a sudden reduction of the primary coolant flow. The increase of the core exit temperature leads to an increase in the primary inventory volume expansion, which is followed by an increase in the primary pressure. However, in this case, the maximum pressure does not reach the trip set point, 16.44MPa. On the other hand, the core power rapidly drops, because it is programmed to be controlled by a constant core exit temperature and the core exit temperature is well above the set point. After the primary pressure reaches the maximum value, it decreases below the nominal operating value and the core exit temperature also decreases to well below the set point. This causes the core power to increase so that the primary pressure and core exit temperature increase again until they surpass the set point. Therefore, oscillation features in the pressure and power are observed in Figure

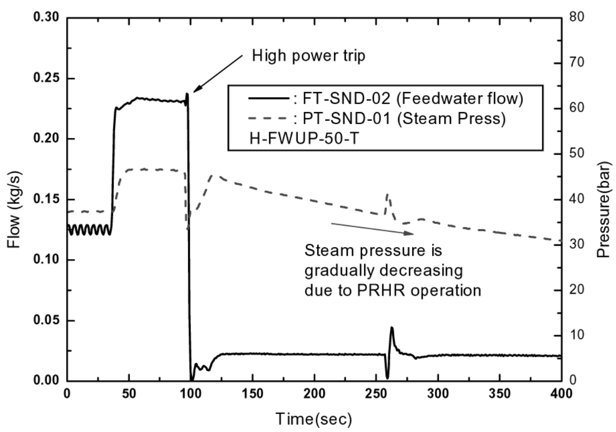

Fig. 3. Feedwater Flow and Steam Pressure (H-FWUP-50-T)

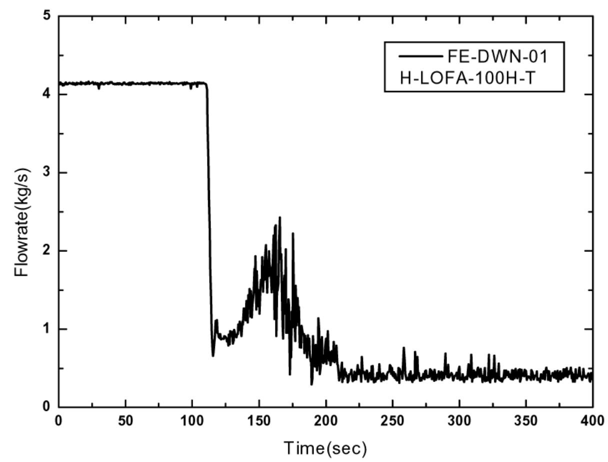

Fig. 6. Primary Coolant Flow (H-LOFA-100H-T)

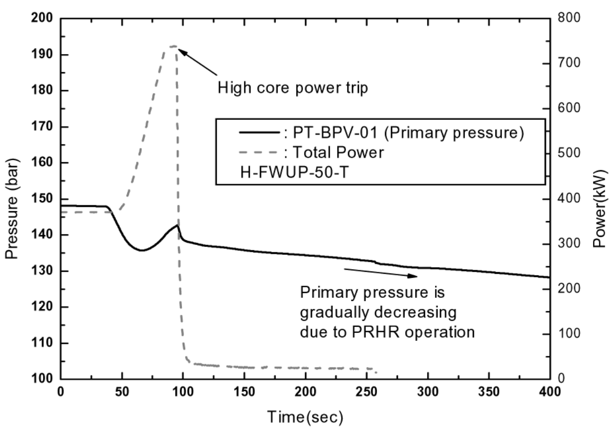

Fig. 4. Primary Pressure and Power (H-FWUP-50-T)

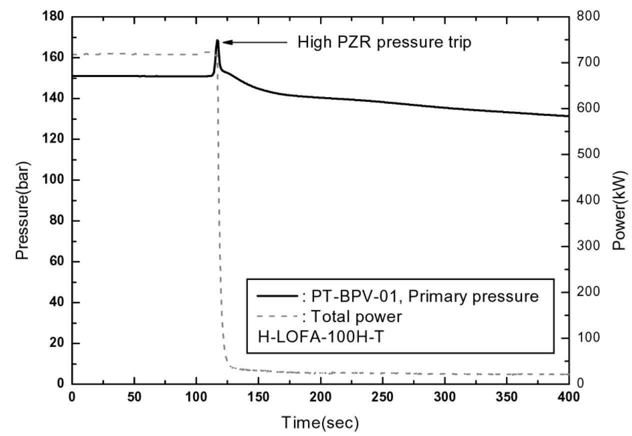

Fig. 7. Primary Pressure and Core Power (H-LOFA-100H-T)

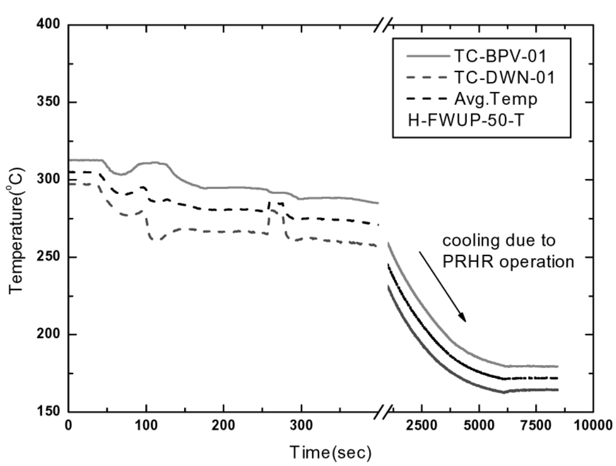

Fig. 5. Core Temperatures (H-FWUP-50-T)

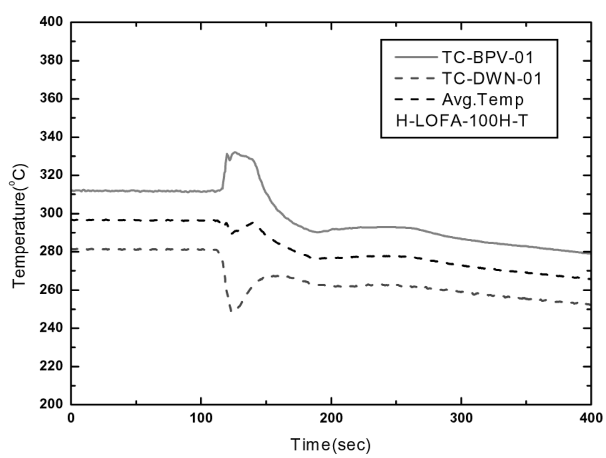

Fig. 8. Core Temperatures (H-LOFA-100H-T)

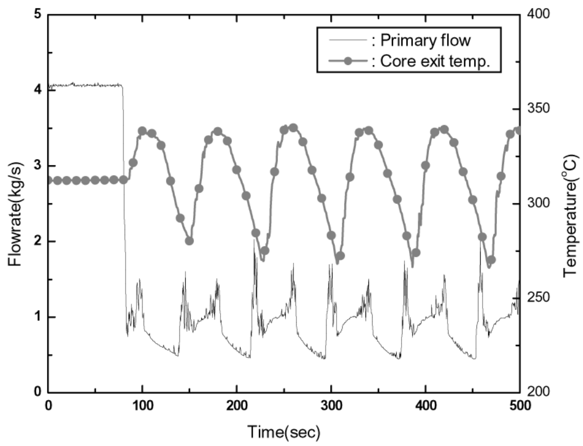

Fig. 9. Primary Coolant Flow and Core Exit Temperature (H-LOFA-50H-T)

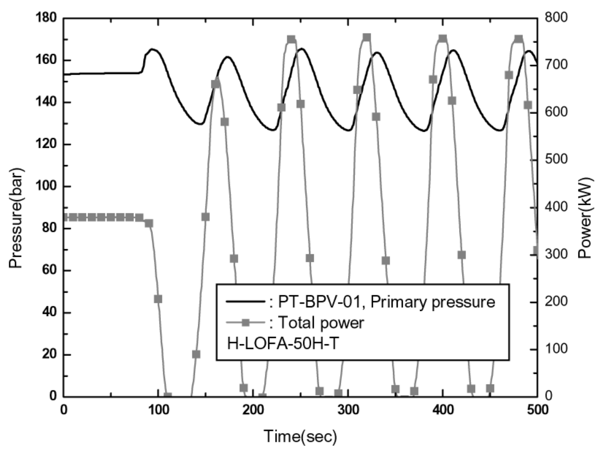

Fig. 10. Primary Pressure and Core Power (H-LOFA-50H-T)

10. The oscillation influences the coolant flow rate and results in the same oscillation in the coolant flow rate as seen in Figure 9.

3.2.3 Control Rod Withdrawal Accident

Figures 11 and 12 show the time variation of the major thermal hydraulic parameters for the control rod withdrawal accident test, H-CRWD-100-T. The initial core power is 100% and the MCP is operated in high speed mode. As noted in the previous section, electrical heaters are used to simulate the reactor core. Therefore, we cannot consider the neutronic effects such as the moderator temperature coefficients or Doppler reactivity in this study. In order to simulate the power excursion due to a control rod withdrawal accident, the core power is increased stepwise from 100% power to the maximum we can obtain in the VISTA facility. The reactor trip is programmed to be initiated by a high pressurizer pressure (16.44MPa). As seen in Figure 11, the primary pressure gradually increases when the core power is suddenly increased. In this case, it took 400 seconds for the primary pressure to reach the trip set point. During the pressure increase, the core temperature also gradually increases, as shown in Figure 12. After the trip, the PRHR system is initiated and continuously cools down the system.

3.3 PRHRS Performance Sensitivity Test Results

The effects of the initial water level in the compensation tank on the PRHRS performance are experimentally investigated. Table 2 summarizes the designed initial conditions for the PRHRS. Normally, the PRHR system is designed to operate at the conditions defined in Table 2. Two tests are carried out for initial water levels of 80% and 16% in order to examine the PRHRS performance.

Table 2. Designed Initial Conditions for the PRHRS

Parameter	Condition
Opening time of PRHRS Isolation or bypass valves (A-series)	Simultaneous opening and closing
Installation of the check valve downstream the ECT (B-series)	2 check valves are installed (Design fixed)
Initial water level of the compensation tank (C-series)	80%
Initial pressure of the compensation tank (D-series)	4.5MPa
Valve opening method downstream the compensation tank	Normally open
Type of the Emergency cooldown tank	Pool-type cooling
Initial filling in the PRHRS piping	Filled with water

Figures 13 through 18 show a comparison of the major thermal hydraulic parameters. “R1a” denotes the reference case defined in Table 2 and “R1-C2” indicates the case where the initial water level is reduced to 16% rather than 80% and the gas cylinder is isolated. Figures 13 and 14 show the pressures in the PRHRS and in the secondary system, respectively. Upon the transient, the pressure suddenly increases up to 6MPa and then decays. As the secondary pressure during the normal operation is 3.55MPa, the secondary pressure starts to increase from 3.55MPa and reaches a maximum peak value and then decays. There is no noticeable difference between the “R1a” and “R1-C2” cases.

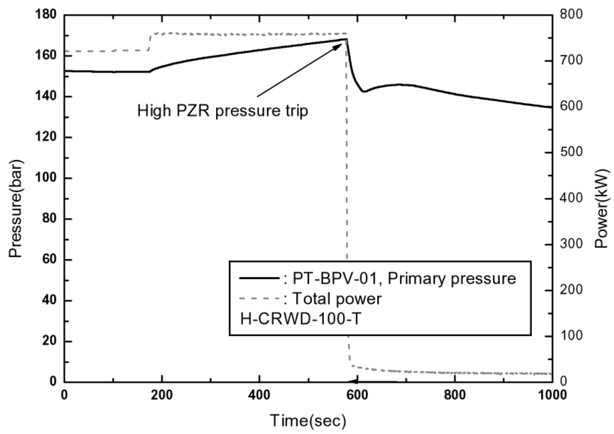

Fig. 11. Pressure and Core Power (H-CRWD-100-T)

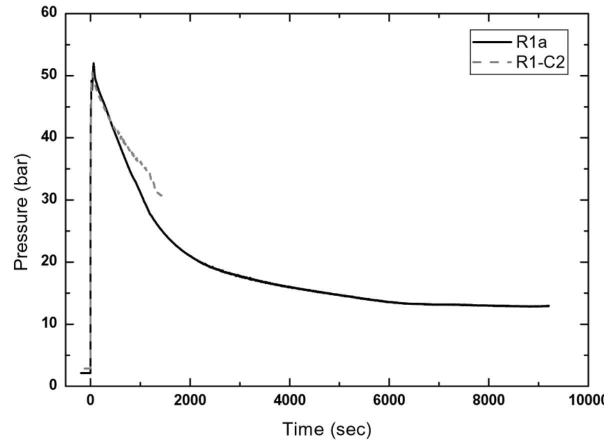

Fig. 13. Comparison of the PRHRS Pressure

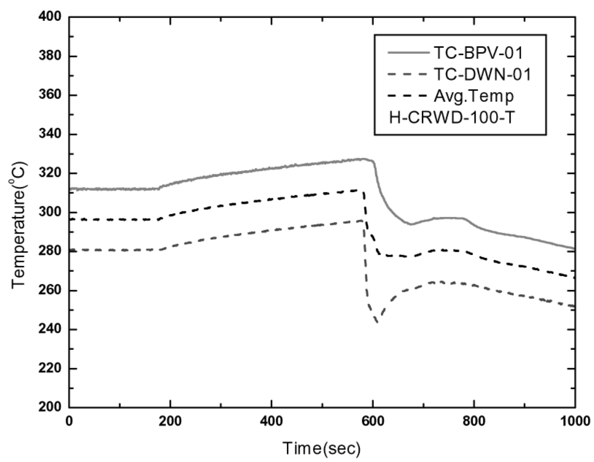

Fig. 12. Core Temperatures (H-CRWD-100-T)

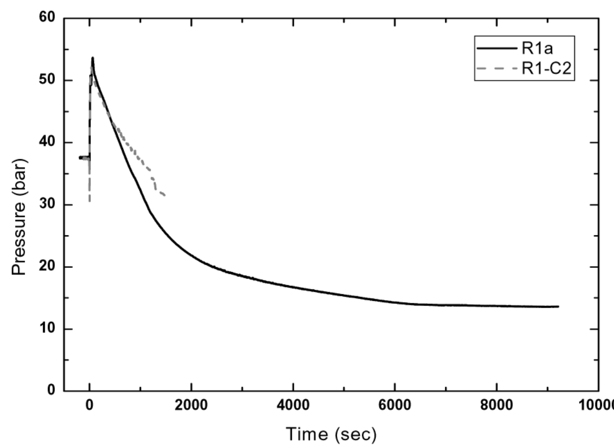

Fig. 14. Comparison of the Secondary Pressure

Figure 15 shows a comparison of the feedwater flow rate. Upon the transient, the feedwater supply is blocked by the isolation valve, and the bypass valve from the steam line to the PRHR system is opened. The “R1a” case shows a natural circulation flow rate of about 10% under the PRHRS operation. The “R1-C2” case also shows a similar natural circulation flow rate in the beginning of the transient. However, the natural circulation stops at about 1500 seconds. This time is consistent with the empting time of the compensation tank in Figure 16. As the initial water level is 16%, the water is quickly drained and finally the nitrogen gas in the compensation tank is introduced into the PRHR system. This results in a failure of the natural circulation flow due to the reduced heat transfer.

Figures 17 and 18 show the variations of the primary pressure and temperature, respectively. The “R1a” case shows gradual decreases in the pressure and temperature, implying that the PRHR system is efficiently removing the decay power in the primary system. The design requirement of the PRHR system is to reduce the primary temperature to below 200° C. As seen in Figure 17, it took 6000 seconds to meet the requirement.

The “R1-C2” case shows a faster decrease of the primary pressure than that the “R1a” case, as shown in Figure 17. This is attributed to the isolation of the gas cylinder from the pressurizer in the “R1-C2”case, which results in smaller gas volume in the pressurizer than in the “R1a” case. Also, the “R1-C2” case shows a minimum near 1000 seconds and the pressure and temperature start to increase again. This is due to the decay heat generated in the core, and also indicates that the PRHR system fails to work. The main reason for the failure of the PRHRS is the introduction of the nitrogen gas from the compensation

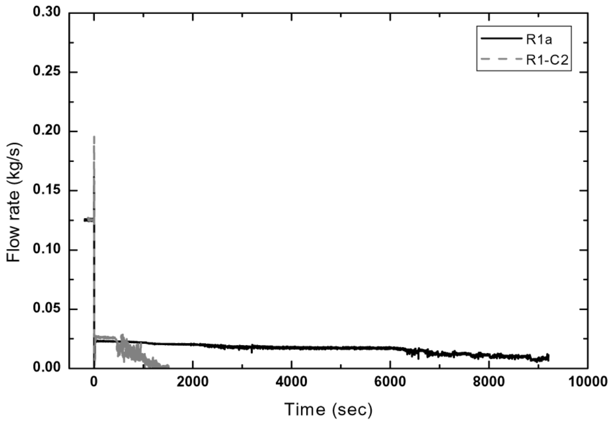

Fig. 15. Comparison of the Feedwater Flow

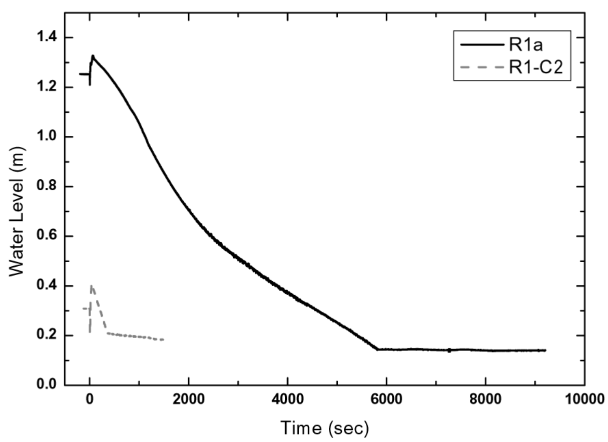

Fig. 16. Comparison of the Water Level in the Compensation Tank

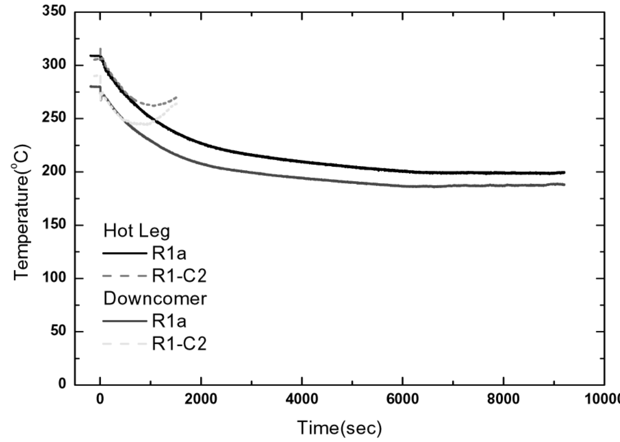

Fig. 18. Comparison of Primary Temperature

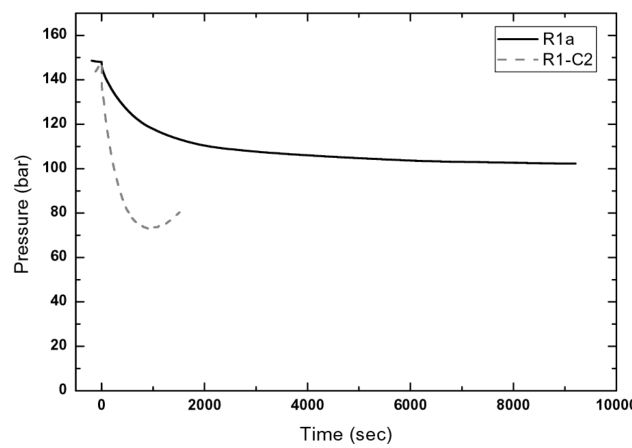

Fig. 17. Comparison of the Primary Pressure

tank. Therefore, it is found that the 16% initial water level in the compensation tank results in the deterioration of the PRHRS performance, and nitrogen ingression should be blocked in order to secure stable cooling performance.

4. CONCLUSIONS

Safety related accidents, including feedwater increase or decrease, loss of coolant flow, and control rod withdrawal accidents are experimentally investigated for design verification of the SMART-P using the integral test facility VISTA. The developed trip set points for the SMART-P are implemented into the VISTA facility. It is found that the system reaches the high reactor power set point earlier than the low pressurizer pressure set point in the case of a feedwater increase accident. Other accidents such as feedwater decrease, loss of coolant flow, and control rod withdrawal are tripped by the high pressurizer pressure setpoint.

The effects of the initial water level in the compensation tank on the PRHRS performance are also experimentally investigated. Two initial water levels of 16% and 80% are tested and compared from the viewpoint of PRHRS performance. It is found that, when the water level is 16%, the water in the compensation tank is quickly drained, and finally nitrogen gas is introduced into the PRHR system. The nitrogen gas ingression results in deterioration of the PRHRS performance and stops the natural circulation in the PRHR system. Therefore, the nitrogen ingression should be prevented in order to secure stable PRHRS operation.

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