INTEGRAL EFFECT NON-LOCA TEST RESULTS FOR THE INTEGRAL TYPE REACTOR SMPART-P USING THE VISTA FACILITY

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ABSTRACT

The SMART-P a pilot plant of the integral type reactor SMART(System Integrated Modular Advanced Reactor) which has new innovative design features aimed at achieving a highly enhanced safety and improved economics. A test facility (VISTA) has been constructed to simulate the SMART-P which is a full height and 1/96 volume scaled test facility with respect to the SMART-P. The VISTA facility has been used to understand the thermal-hydraulic behavior including several operational transients and design basis accidents and finally it will contribute to verifying the system design of the SMART-P. During the past five years, several integral effect tests have been carried out and reported, including performance tests, MCP(Main Coolant Pump) transients, power transients and heatup or cooldown procedures. In the present study, the VISTA facility was subjected to the major safety related non-LOCA transient conditions in a primary and secondary system, including a power increase due to a CEDM (Control Element Drive Mechanism) withdrawal, a feedwater decrease and a steam flow increase in order to verify the safety analysis code for the SMART-P.

INTRODUCTION

The SMART is an advanced integral reactor with a power of 330MWt with 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 it eliminates an occurrence of a large break LOCA. A passive residual heat removal system (PRHRS) is also installed to prevent an 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 in it. The basic design of the SMART was completed in 2002 by KAERI and a prototypic SMART plant, the SMART-P with a rated power of 65MWt, will be constructed within six years in Korea. (Chang et al., 1999, 2002, Kim et al., 2003) The VISTA facility (Experimental Verification by Integral Simulation of Transients and Accidents) which was constructed to investigate the overall thermal-hydraulic behavior of the SMART-P has been widely used for a performance verification, supporting a design improvement, and a safety analysis. The reactor core is simulated by 36 electrical heaters with a capacity of 818.75kW which is a capacity of 120% of the scaled power. A schematic diagram of the VISTA facility is shown in Figure 1.

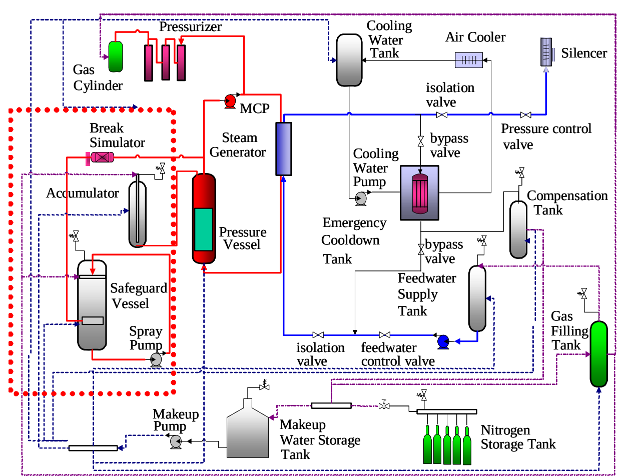

Figure 1. Schematic diagram of the VISTA

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 a pressurizer are connected by pipes for an easy installation of the instrumentation and a simple maintenance. The secondary system with 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 a safety injection will be installed after carrying out the performance tests such as the normal operation and operational transient tests as well as some of the design basis accidents. The PRHR system of the VISTA facility is composed of a single train for the cooling sub-system, which includes an emergency cooldown tank (ECT), a heat exchanger (HX), a compensating tank (CT), several valves and the related piping. It is connected to both the feedwater and steam lines of the secondary system to create a flow path with a natural circulation. Detailed information on the VISTA facility can be found elsewhere (Choi et al., 2003a, 2003b, Park et al., 2004a, 2004b).

During the past five years, several integral effect tests have been carried out and reported in literature, including normal performance tests, MCP transients, power transients, heatup or cooldown procedure, and PRHR system performance tests. (Choi et al., 2003c, 2004, 2005a, 2006a, Park et al., 2004c, 2004d). The normal performance tests are the individual performance tests of each component, such as the core heaters, a main coolant pump, a steam generator and the heat exchangers. The natural circulation performance was also investigated as one of the normal performance tests. The MCP transient tests imply a kind of anticipated operational transient (AOT) where the operation mode of the MCP changes. The power transient tests are mainly performed to help understand the thermal hydraulic performance of the in-vessel cold self-pressurizer. In the PRHR system performance tests, influences of the several design parameters, such as the opening time of the isolation and bypass valves, the initial water level and pressure in the compensation tank on the PRHR performance were investigated. Thermal hydraulic test programs with the VISTA facility are still underway to improve the thermal hydraulic performance and safety of the integral type reactor SMART-P and the main focus is now being put on safety related non-LOCA transient tests such as a feedwater increase/decrease, a loss of coolant flow, and control rod withdrawal accidents.

The most typical non-LOCA tests that need to be experimentally investigated include a loss of feedwater flow, a sudden increase in a steam flow, and a power increase due to a control rod withdrawal accident. Experimental results for a loss of feedwater flow and a power increase accident followed by an actuation of the PRHRS have already been carried out and reported in the literature (Choi et al. 2006b). In this paper, the other non-LOCA tests for a sudden steam flow and a power increase not followed by an actuation of the PRHRS are experimentally examined. The obtained data will be used to verify the performance and to assess the safety of the prototype reactor, SMART-P.

DESCRIPTION OF THE VISTA FACILITY

Fluid system

The fluid system of the VISTA facility consists of a primary system, a secondary system, a PRHR system and an auxiliary system. The primary system includes a reactor core, a main coolant pump (MCP), three pressurizers, and a helical-type steam generator. The primary components of the SMART-P are simplified to be a loop-type in the VISTA facility. The primary coolant circulates in the primary loop starting from the reactor core, through the MCP and the steam generator, to the reactor core. Three pressurizers, called an end cavity (EC), an intermediate cavity (IC), and an upper annular cavity (UAC), are simulated by three independent cylindrical vessels. Each vessel is connected by a separate pipe to simulate a surge flow. The volume of each cavity is scaled down at a ratio of 1/96 and the height and the elevation are preserved.

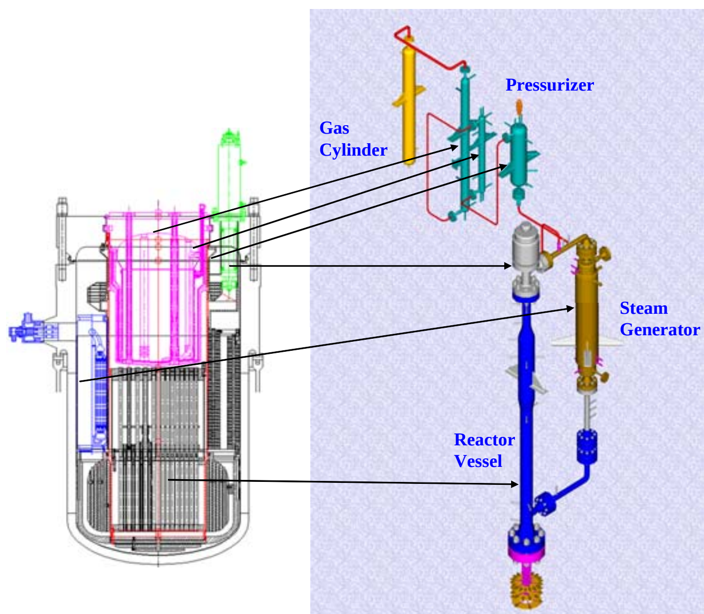

Figure 2. Arrangement of the major fluid system of the VISTA facility

The steam generator is scaled down to 1/8 with respect to one steam generator cassette of the SMART-P based on the same scaling 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 secondary system consists of the feedwater supply system and the steam discharge system. The arrangement of the major fluid system is shown in Figure 2 (Choi et al., 2003c, 2004, 2005a, 2006a, 2006b).

Instrument and control system

The VISTA facility has about 230 analogue input channels, including the pressure, differential pressure, water level, flow rate, electrical power, and temperature. Each instrument channel is carefully calibrated with a high precision calibrator. An automatic control system with a user friendly graphic interface has been installed in the VISTA facility to control the major thermal hydraulic parameters following an operator's instructions. In addition to this, automatic scenario control logics are implemented for each non-LOCA test. The sequence of events with respect to time such as the start of a transient, MCP trip, reactor core trip, and an opening of the valves of the PRHRS is controlled by the implemented sequence control logic.

During the non-LOCA transients in the SMART reactor, the core power depends on the moderator temperature coefficients, because the transients cause rapid variation in moderator temperature. Such power variation curve was achieved from a pre-calculation for the safety analysis code, TASS/SMR.(Hwang et al. 2005) Then, the power table with respect to time was implemented into the VISTA control system.

PRHR system

The VISTA facility is equipped with a passive residual heat removal system (PRHRS) in order to cooldown the whole plant system when needed. It is connected to both the feedwater and steam lines of the secondary system to provide a flow path for a natural circulation. It is designed to have the same pressure drop and heat transfer characteristics and it is arranged to have the same elevation and position as those of the reference system. Also the diameter, thickness, pitch, and orientation of the heat exchanger tubes of the VISTA facility are the same as those of the reference system. When an actuation signal is generated, the connecting valves to the PRHRS are opened coincidently with a closing of the isolation valves to the main feedwater and steam supply system in order to initiate the operation of the PRHRS (Park et al., 2004c, 2004d).

RESULTS AND DISCUSSION

In the present work, two accident scenarios are taken into account; a sudden increase in a steam flow and a power increase accident not followed by an actuation of the PRHRS when the reactor is tripped. The former case is one of the typical accidents in which the heat removal rate from the primary to the secondary system is increased and the primary system is cooled. The latter case is caused by a control rod withdrawal accident, in which the core power increases according to the negative moderator temperature coefficient. The initial and boundary conditions of both cases are conservatively determined in advance with a safety analysis code, TASS/SMR.

Table 1. Initial and boundary conditions for the test of a sudden increase in a steam flow

Parameters	Value	Remarks
Initial core power (%)	103.0	
Initial sec. flow rate (kg/s)	0.2575	103%
Initial core exit temp. (^{o}C)	315.5	
Initial primary coolant flow (%)	95	4.0 kg/s
Initial press. in the EC (MPa)	15.8	
Core power control	Yes	predefined table
MTC feedback effect	Yes	predefined table
Core decay power	ANS73x1.2	
High core power Rx trip setpoint (%)	122.2	
Closing time of the feedwater control valve (sec)	<1.0sec	
MCP operation after a reactor trip	coastdown	
Heat removal after a reactor trip		PRHRS works

Steam flow increase test

In a normal operating condition, the steam generated at the steam generator is introduced to the turbine system of the prototype plant at a designed steam pressure. In the VISTA facility, the turbine system is not simulated and the generated steam is dumped to the atmosphere for a simplicity. However, a steam pressure is preserved by controlling a steam pressure control valve which is installed downstream in the outlet pipeline of the steam generator. This transient starts with a sudden opening of the steam pressure control valve by an operator when the system reaches a steady state defined in Table 1.

Figure 3 shows the measured pressure trends at the exit of the secondary side of the steam generator and in the feedwater supply tank (FWST). The feedwater and steam flowrates are also plotted in the same figure. Before the transient starts, the steam pressure is maintained at about 3.7MPa, which is slightly higher than the designed pressure of 3.5MPa. As shown in this figure, the steam control valve is suddenly opened at about 346sec. It results in a sudden decrease in the steam pressure. The steam pressure temporally drops to about 2.5MPa. Decrease in the steam pressure brings about an increase in the pressure difference between the inlet and outlet of the steam generator. So, the feedwater flow rate suddenly increases and cools the steam generator temporally. An increase in the steam flow rate was not observed. It seems to be due to the sudden decrease in the steam pressure. In this transient, the core power is controlled by a predefined table with respect to the time and it is tripped when the core power reaches a high power trip setpoint. The observed reactor trip time was at 364sec. It took 18sec for the reactor to be tripped after the initiation of the accident. When the core is tripped, the PRHRS starts to work in order to remove the residual core power. The secondary pressure suddenly increases when the flow path is switched from the atmosphere to the emergency cooldown tank (ECT) in the PRHRS.

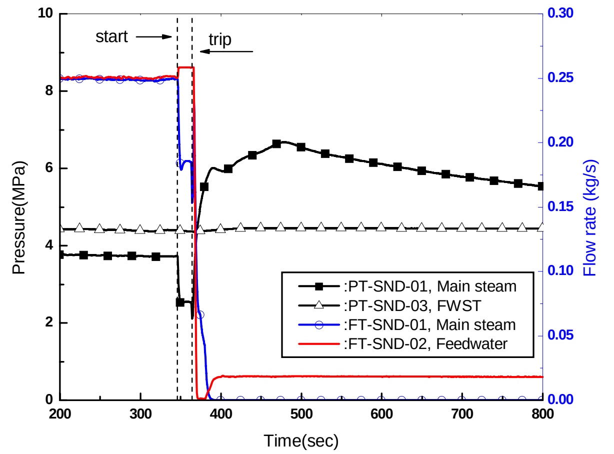

Figure 3. Secondary pressure and flow rate

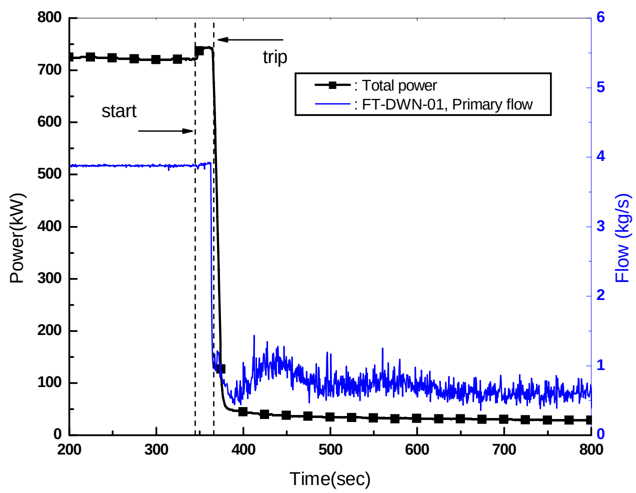

Figure 4. Core power and primary flow

Figure 4 shows the measure core power and the primary coolant flow rate. When the transient starts, the core power increases as shown in figure 4. When the core power control signal reaches its maximum value, the core power is programmed to linearly decrease to the ANS73 curve for 10 seconds and to follow a pre-defined decay power table, which was obtained by multiplying the ANS73 curve by 1.2 for conservative results. The transition to the conservative ANS73 decay curve is shown in Figure 4. After the reactor trip, the core decay power is removed by an actuation of the passive residual heat removal system (PRHRS).

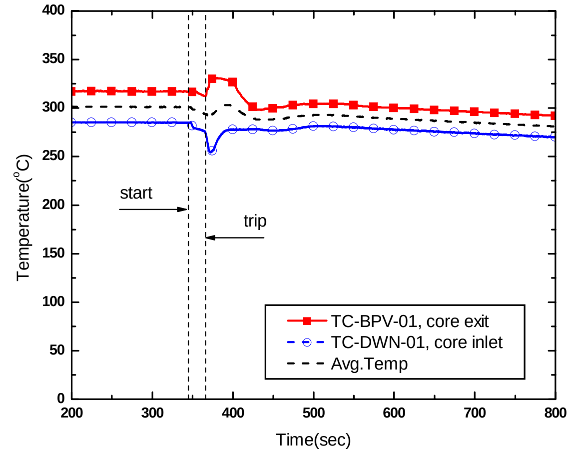

Figure 5. Core inlet and exit temperatures

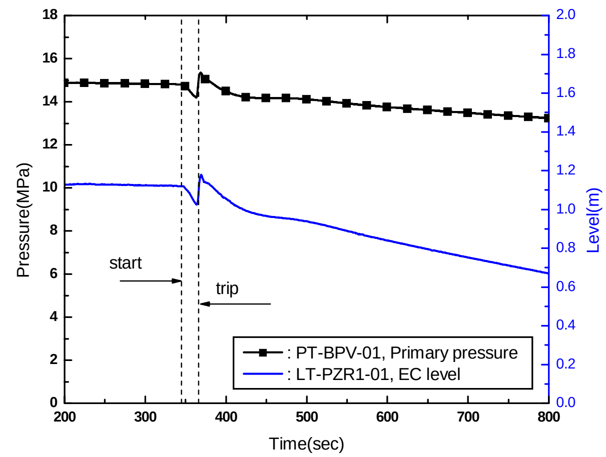

Figure 6. Primary pressure and the water level in pressurizer

Figure 4 also shows the primary coolant flow rate for the whole period of the present test. The primary coolant flow rate suddenly drops to a natural circulation level due to a simultaneous trip of the MCP with the reactor trip. The measured stable natural circulation flow rate is about 12.5% of the rated flow rate. It is noteworthy that the primary natural circulation flow rate shows a temporal increase at about 430sec and it decreases a little bit as shown in Figure 4. It can be explained by the temperature difference across the core as shown in figure 5. The major driving force of the natural circulation flow is a density difference across the core which is caused by a temperature difference. As shown in figure 5, the maximum temperature difference observed in the early period after the core trip causes an increase in the primary coolant flow rate. Before the core is tripped, the core inlet and exit temperatures drop considerably because the secondary heat removal capacity increases due to a sudden increase in the steam flow rate. When the core is tripped, the core inlet temperature decreases due to a reduction in the primary coolant flow rate, but the core exit temperature increases, resulting in a maximum temperature difference across the core.

The measured primary pressure is shown in Figure 6, along with the water level in the end cavity (pressurizer). When the transient starts, the primary pressure decreases due to an increased heat removal capacity in the secondary side. However, the primary pressure starts to increase again because the core power is increased according to the power table. The trend of the water level in the pressurizer agrees with that of the primary pressure.

Power increase test

A power increase accident resulting from a control rod withdrawal accident is experimentally simulated in the present study. Two kinds of power increase tests have been identified for the purpose of a safety analysis. Both cases have the same initial and boundary conditions, but utilize a different decay heat removal method. One is the case that the core decay heat is removed by an operation of the PRHRS after the core trips. The MCP is also tripped at the same time as the core trip. The primary coolant circulates in the primary loop by a natural circulation. A simulation of this case has already been carried out and reported in the literature (Choi et al., 2006b). The other is the case that the core decay heat is removed by an operation of the secondary system. The MCP is not tripped, but it is allowed to keep on working when the core is tripped. The PRHR system is not used, but the feedwater flow rate is controlled to decrease to 10% of the nominal value at 10%/sec when the core is tripped.

Table 2. Initial and boundary conditions in the power increase accident

Parameters	Value	Remarks
Initial core power (%)	97.0	
Initial sec. flow rate (kg/s)	0.2425	97%
Initial core exit temp. (^{o}C)	315.5	
Initial primary coolant flow (%)	95	4.0 kg/s
Initial press. in the EC (MPa)	13.5	
Core power control	No	
MTC feedback effect	considered	Pre-defined table
Core decay power	ANS73x1.2	
High pressure power trip setpoint (MPa)	16.44	
MCP operation after RX trip	Yes	
Heat removal after a reactor trip	10%/sec	2nd system works

The initial and boundary conditions of the accident are summarized in Table 2. The initial condition at which the accident happens is assumed to be a steady state condition of a power of 97%. The core power is expected to increase due to a moderator temperature coefficient (MTC) feedback effect during the accident. The power variation with respect to the time was obtained from a separate safety analysis code and it was implemented in the VISTA facility. As the core power increases, it reaches a high pressurizer trip setpoint, and then the reactor is tripped. When it is tripped, the core power is controlled to linearly decrease to the ANS73 decay power curve for 10 seconds and to follow a pre-defined decay power table, which was obtained by multiplying the ANS73 curve by 1.2 for conservative results.

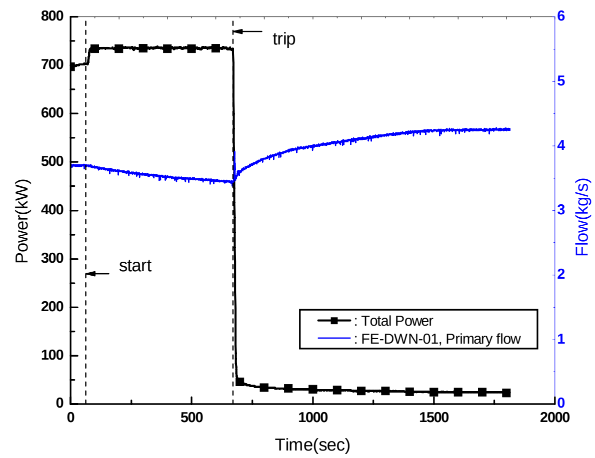

Figure 7. Core power and primary coolant flow rate

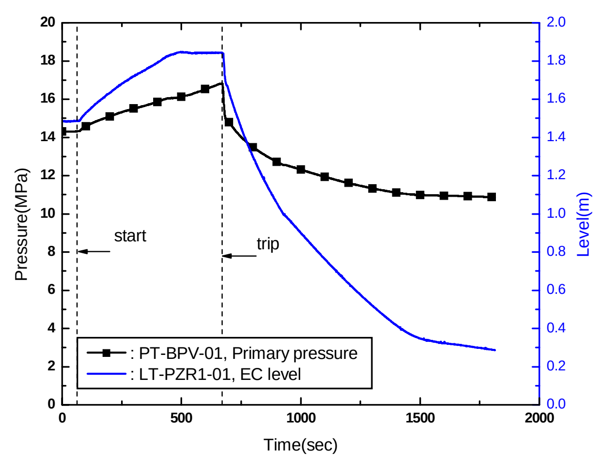

Figure 8. Primary pressure and water level in pressurizer

Figure 7 shows the measured core power and primary coolant flow rates in the same graph. The transient started at about 63sec by increasing the core power according to the programmed power table. The reactor was tripped at about 670sec when it reached a high pressure trip setpoint as shown in figure 8. When the transient starts, the core power reaches the maximum available power. The primary coolant flow continuously decreases. In this study, the primary coolant flow is a mass flow rate which is obtained from the measured volumetric flow rate multiplied by the measured density. Therefore, the continuous decrease in the primary coolant flow rate is due to a continuous decrease in the water density. When the core is tripped, the primary pressure and temperature start to decrease and the water density increases. So, an increase in the primary coolant flow rate is observed after the core trips.

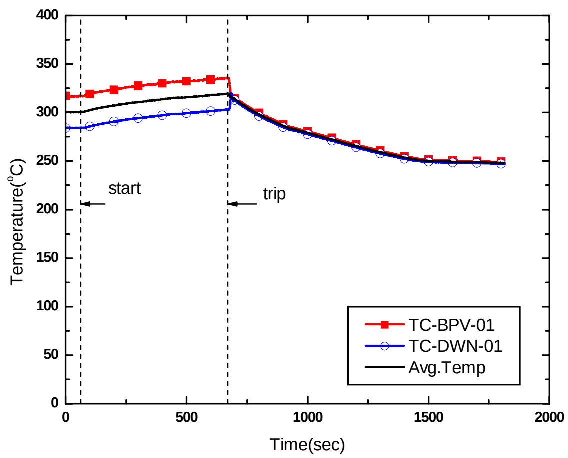

Figure 9. Core inlet and exit temperatures

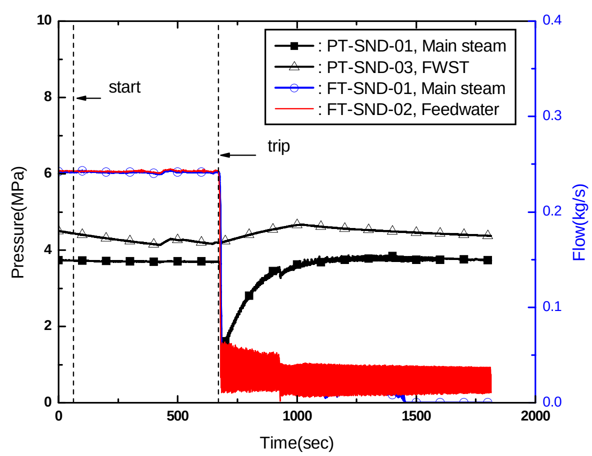

Figure 10. Secondary pressure and the secondary flow rate

Figure 8 shows the measured primary pressure, which continuously increases until it reaches a high pressurizer trip setpoint. The increase in the primary pressure is a result of an increase in the water level in the pressurizer by a volume expansion. A consistent increase in the water level in the pressurizer is shown in figure 8.

Figure 9 shows the measured core inlet and exit temperatures. The power growth results in an increase in the core inlet and exit temperatures. When the core is tripped, the core power suddenly drops and the MCP keeps on working. Therefore, the temperature difference across the core is minimized. Figure 10 shows the measured secondary pressure and flow rates. Before the trip, the secondary pressure maintains a designed pressure and the feedwater (or steam) flow maintains a constant value. When the core is tripped, the feedwater flow rate is controlled to decrease to 10% of the nominal value at 10%/sec. It is carried out by controlling the feedwater control valve installed at the inlet of the steam generator. During the transient, a severe oscillation in the feedwater flow rate is observed. It is due to simultaneous changes in the secondary pressure and feedwater flow rate. Decrease in the feedwater flow rate results in a decrease in the secondary pressure. In order to recover the secondary pressure to the designed value, the secondary pressure control valve starts to control the steam flow area automatically. As time goes by, the secondary pressure is totally recovered to the designed value as shown in figure 10. However, the feedwater flow rate experiences a severe oscillation.

CONCLUSION

The VISTA facility has been used to verify the performance and safety of the integral type reactor SMART-P. Several design basis accidents, such as an increase or decrease of the feedwater flow, a loss of coolant flow, a control rod withdrawal, and a limited case of a loss of coolant accident (LOCA) on the line to the gas cylinder are under investigation in order to understand the thermal-hydraulic responses and finally to verify the system design of the SMART-P.

In the present study, two accident scenarios, such as an increase in a steam flow, and a power increase caused by a control rod withdrawal accident, were investigated experimentally. The tests were successfully carried out at the pre-determined initial and boundary conditions. This data will be used to verify and validate the safety analysis code, TASS/SMR for SMART-P, which is under development.

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