Simulation of the Safety Related Accidents of the SMART-P with the VISTA facility

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1. Introduction

Thermal hydraulic behavior for the safety related accidents of the SMART-P are experimentally simulated by using an integral effect test facility, VISTA, which is a full height and 1/96 volume scaled test facility based on the design features of the SMART-P.[1,2,3] The main focus is put on 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.

2. Control System

All the tests in the present study were initiated and controlled by automatic control logics developed by the authors, including a PID control and a sequence control logic. The developed control logics are installed into a programmable logic controller (PLC). 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. Results and Discussion

Table 1 shows a test matrix in the current study. First of all, a feedwater increase event (H-FWUP-50-T) was investigated. Before the transient operation, the VISTA was set to operate at a 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 (H-FWDN-100-T) was also carried out after the VISTA reached a steady state condition at a 100% power. The loss of coolant flow accidents (LOFA) were simulated at several different initial conditions, depending on the core power and MCP operation modes. We considered the representative three cases; a case at a 100% core power with the MCP running at a high speed, a case at a 50% core power with the MCP at a high speed, and a case at a 50% core power with the MCP at a middle speed. Finally, a control rod withdrawal accident (H-CRWD-100-T) was tested at a 100% core power. As we are using electrical heaters to simulate the reactor core, the neutronic effects such as the moderator temperature coefficients or Doppler reactivity are excluded in this study. The main focus is on the thermal hydraulic aspect during the transient.

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% -> 0%)
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

3.1 Feedwater increasing/decreasing accidents

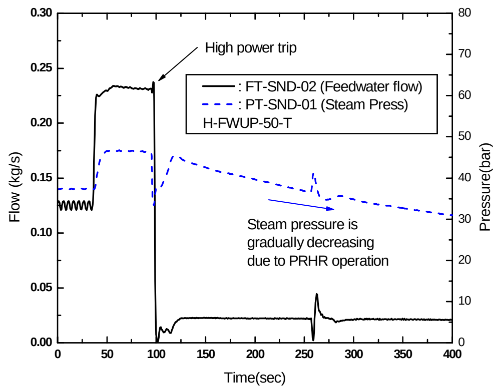

Figure 1 Feedwater flow and steam pressure (H-FWUP-50-T)

Figure 1 shows feedwater flow and the secondary pressure variation in the case of feedwater increasing accident. When the feedwater was increased from 50% to 100% the steam pressure sharply increases due to the increased feedwater injection to the steam generator. The primary pressure suddenly drops and the core power rapidly increases to compensate the reduction of the primary pressure. In this case, the reactor core is programmed to be tripped by either a high reactor power (115%) or a 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 core exit temperature suddenly drops a little, but it recovers its previous value due to the increased power. After the reactor trip, the PRHR system is triggered to cool down the primary system.

3.2 Loss of coolant flow accidents

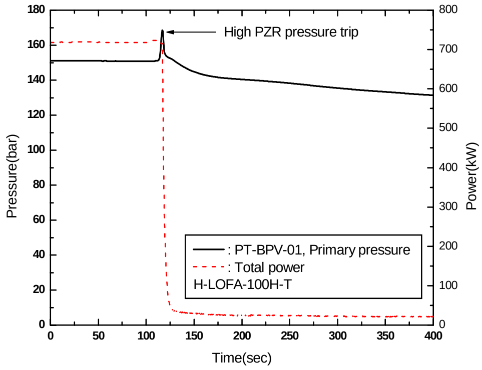

Figure 2 Primary pressure and core power (H-LOFA-100H-T)

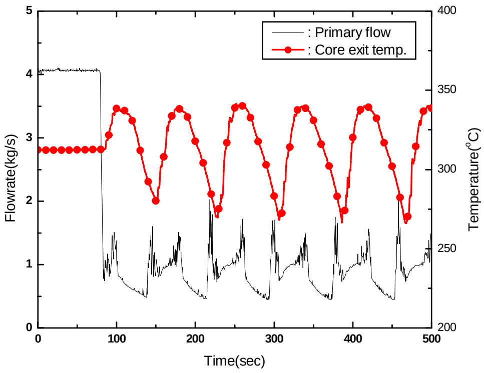

Figure 3 Primary coolant flow and core exit temperature (H-LOFA-50H-T)

Figure 2 shows a result when the initial core power was set to 100% and the MCP was running at a high speed mode. The transition starts at by switching off the MCP. Though the MCP is tripped, the primary coolant still circulates in the primary loop by a natural circulation. The measured natural circulation flow rate is about 14% of the rated value. The reactor core was programmed to be tripped by a high pressurizer pressure (16.44MPa). As seen in Figure 2, the reactor core was tripped by the high pressurizer pressure trip set point after 14sec from the transition.

Figure 3 shows a result when the initial core power was set to 50% rather than 100%. The primary coolant flow rapidly drops, due to switching off the MCP, but a natural circulation flow can be observed with an oscillation. 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. The primary pressure and the core power show similar oscillating behaviors.

3.3 Control rod withdrawal accidents

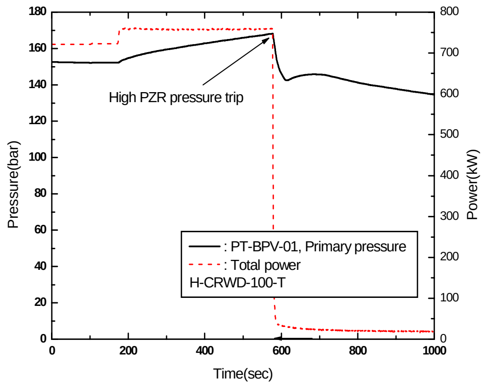

Figure 4 Pressure and core power (H-CRWD-100-T)

Figure 4 shows the time variation of the primary pressure and the core power for the control rod withdrawal accident test, H-CRWD-100-T. The initial core power is 100% and the MCP is operated at a high speed mode. In order to simulate the power excursion due to a control rod withdrawal accident, the core power is step-increased from a 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). The primary pressure gradually increases when the core power is suddenly increased. In this case, it took 400sec for the primary pressure to reach the trip set point. During the pressure increase, the core temperature is also gradually increasing. After the trip, the PRHR system is initiated and continuously cools down the system.

4. Conclusion

Safety related accidents, including a feedwater increase or decrease, loss of coolant flow, and control rod withdrawal accidents are experimentally investigated for a design verification of the SMART-P by using the integral test facility, VISTA. 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 the feedwater decrease, loss of coolant flow and the control rod withdrawal are tripped by the high pressurizer pressure setpoint.

REFERENCES

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[3] Choi, K. Y. et al., “Thermal-Hydraulic Characteristics during Transient Operation of the Integral Type Reactor,” *The 4th Japan-Korea Symposium on Nuclear Thermal Hydraulics and Safety (NTHAS-4)*, Sapporo, Japan, November 28-December 1 (2004).