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Experimental Study on Non-LOCA Transients of the SMART-P with the VISTA facility

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1. Introduction

The VISTA facility (Experimental Verification by Integral Simulation of Transients and Accidents) has been constructed to verify performance and safety of the integral type reactor SMART-P. The VISTA has the same height and 1/96 volume with respect to the SMART-P. The reactor core is simulated by 36 electrical heaters with a capacity of 818.75kW. Several design basis accidents, such as increase or decrease of feedwater flow, loss of coolant flow, control rod withdrawal, and a limited case of 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 this paper, simulation results on two safety related accidents, including loss of feedwater flow and power increase caused by a control rod withdrawal accident, are investigated.

2. Experiments

Two safety related accident scenarios are taken into account; loss of feedwater accident and power increase accident. The former case is one of the typical accident in which heat removal rate from the primary to the secondary system is reduced and so the primary system is heated. 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 the both cases are conservatively determined in advance with a safety analysis code.

3. Results and discussions

3-1. Loss of feedwater flow test

The initial and boundary conditions of the loss of feedwater accident are summarized in Table 1.

The feedwater flow rate abruptly drops to a zero level in a time less than 1.0sec. Loss of feedwater flow rate terminates the heat transfer from the primary to the secondary system in a steam generator and results in an increase in the core inlet temperature as shown in Figure 1(a). As a result, the core average temperature increases and it causes the primary pressure to increase. The measured primary pressure is shown in Figure 1(b), along with the water level in the end cavity (pressurizer). The abrupt increase in the primary pressure and the water level in the end cavity can be observed in Figure 1(b). In this accident simulation, the VISTA facility was programmed to be tripped when the primary pressure reaches 16.44MPa. The observed reactor trip time was at 131.5sec. It just took 6.5sec for the reactor to be tripped after the initiation of the accident. When the reactor is tripped, the core power is programmed to follow a pre-defined decay power table, which was obtained by multiplying 1.2 by the ANS73 curve for conservatism.

Table 1. Initial and boundary conditions in the loss of feedwater accident

Parameter	Value	Remarks
Initial core power(%)	103.0	
Initial secondary flow (kg/s)	0.2575	103%
Initial core exit temp. (^oC)	315.5	
Initial primary flow rate (kg)	4.0	95%
Initial press. in EC (MPa)	15.8	
Core power control	Not considered	Constant power
MTC feedback effect	Not considered	Constant power
Core decay power	ANS73 x 1.2	
High press. Rx. trip setpoint (MPa)	16.44	
Closing time of the feedwater control valve (sec)	<1.0sec	
MCP after Rx. trip	coastdown	
Heat removal after Rx. trip		PRHRS works

After a reactor trip, the core decay power is removed by actuation of the passive residual heat removal system (PRHRS). Figure 2(a) shows the primary coolant flow rate in the whole period of the present test. The primary coolant flow rate suddenly drops to a natural circulation level due to the simultaneous trip of the MCP with a reactor trip. The measured natural circulation flow rate is about 7.5% of the rated flow rate. The interesting founding is that the primary natural circulation flow rate jumps to higher level and the oscillation amplitude also increases about at 1636sec as shown in Figure 2(a). It is hard to find the exact cause of the transient of the primary natural circulation flow rate. However, during the transient, several tracing heaters were used in order to compensate for the heat loss to atmosphere and maintained at a constant power. Local voiding might occur somewhere in the corner region of the primary system due to high wall temperature. Therefore, the local voiding is considered to be the most possible cause of the transient of the primary natural circulation flow. This reasoning should be confirmed by comparing a counterpart test in which all the tracing heaters are turned off and the other thermal hydraulic conditions are the same. Even though the counterpart test has not been performed yet, in the similar pre-tests in which the tracing heaters were not used, it was observed that the primary natural circulation flow rate maintained a constant level during the entire transient. More detailed counterpart test will be carried out in the near future in order to confirm the inference.

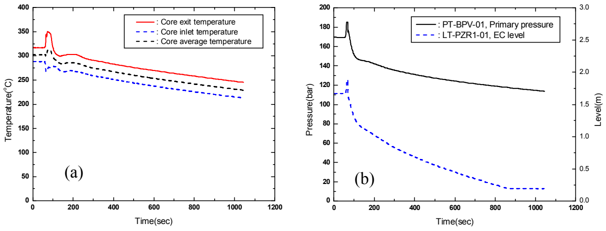

Figure 1(a). Core inlet and exit temperature 1(b). Primary pressure (H-FWDN-100-T)

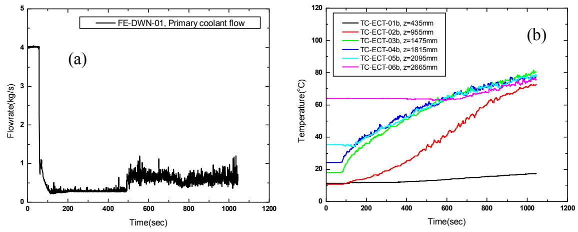

Figure 2(a). Primary coolant flow rate 2(b). Temperature in the pool of ECT 2 (H-FWDN-100-T)

3.2 Power increase test

A power increase accident resulting from a control rod withdrawal accident, is experimentally simulated in the present study. The initial and boundary conditions of the accident are summarized in Table 2.

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

Parameter	Value	Remarks
Initial core power(%)	97.0	
Initial secondary flow (kg/s)	0.2425	97%
Initial core exit temp. (^{o}C)	315.5	
Initial primary flow rate (kg)	4.0	95%
Initial press. in EC (MPa)	13.6	
Core power control	Not considered	
MTC feedback effect	considered	Predefined Table
Core decay power	ANS73 x 1.2	
High power trip setpoint (%)	122.2	Nominal 115
Closing time of the feedwater control valve (sec)	<1.0sec	
MCP after Rx. trip	coastdown	
Heat removal after Rx. trip		PRHRS works

The power growth results in an increase of the core exit temperature as shown in Figure 3(a). The increased core exit temperature makes the primary coolant volume expand so that the water level in the end cavity increases. It indicates that the primary pressure increases as shown in Figure 3(b). The observed pressure peak is about 17.5MPa, which is higher value than expected. It seems that the reactor reaches the high pressure trip setpoint earlier than the high power trip setpoint.

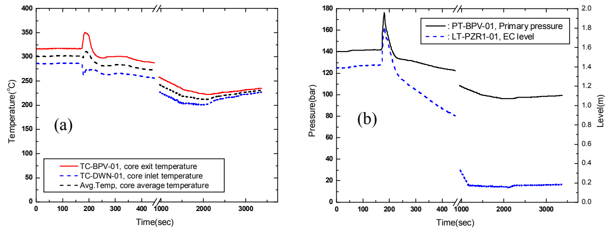

Figure 3(a). Feedwater flow rate 3(b). Core power (H-PowerUp-C1)

4. Conclusion

Two accident scenarios, such as loss of feedwater flow and power increase caused by a control rod withdrawal accident, are experimentally investigated. The tests are successfully carried out at the pre-determined initial and boundary conditions. These data will be used to verify and validate the safety analysis code for SMART-P, which is under development. The jump to high natural circulation level observed in the later period of the PRHR operation needs further study. Also, other safety related scenarios necessary for verification of the safety analysis code are scheduled to be carried out using the VISTA facility.

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