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SPACE 3.0 Simulation of Natural Convection Experiment

Dongwook Jang[a*], Youn-Gyu Jung[a], Cheol Park[a]
[a]*Research reactor design division, Korea Atomic Energy Research Institute, Daejeon, Korea*
[*]*Corresponding author: dwjang@kaeri.re.kr*

1. Introduction

SPACE is a system code developed for the thermal hydraulic safety analysis of nuclear power plants and should be validated for applying to research reactor (RR) analysis. Among the various thermal-hydraulic phenomena for SPACE code validation, this study performed validation calculation for natural convection phenomena in RR operational condition. The experiment used in this work was loss of flow tests performed in IEA-R1 reactor. This experiment provided the data about temperature and mass flow rate of the cooling channel and fuel while the flow in the core changed from forced convection to natural convection.

The version of the code used for validation was SPACE 3.0 and the calculation using the RELAP5/MOD3.3 was also performed for comparative analysis.

2. Description of the IEA-R1 Loss of Flow Tests

2.1 IEA-R1 Reactor

The IEA-R1 is a pool-type research reactor with normal steady state power of 2 to 5 MW. It uses light water as coolant and moderator, graphite and beryllium as a reflector, and MTR (material test reactor) type fuels with low enrichment [1]. Table 1 shows the design parameters of the IEA-R1 research reactor.

Fig. 1 shows the schematic process diagram of the primary and secondary cooling systems. The primary cooling pump (P-101A or P-101B), the heat exchanger (HE-B), the secondary cooling pump (P1A or P1B), and the cooling tower (CTB) run for 5 MW operation. The other heat exchanger and cooling tower are operating only for low power operation less than 3 MW. The T1, T2, and T3 are the pool water temperature at the core inlet and the T4 is the coolant temperature at the reactor outlet. The FE is the flow element to measure the primary cooling flow.

Fig. 2 shows the reactor core and reactor pool. Major dimensions are described in the figure. The total volume of water in the pool is around 272 m^3 [2]. The primary coolant discharged to the pool bottom rises up to the fuel assembly top, flows down to the reactor core, passes the matrix plate, hopper, and coupling valve, and returns to the primary pump.

The coupling valve and major dimensions are shown in Fig. 3. By using the manually actuated pneumatic system, operators lift the header of the coupling valve to coupling the valve to the hopper before they turn on the primary cooling pump. After the primary cooling flow reaches the normal operating flow rate, operators turn off the pneumatic system. Then, the header of the coupling valve is kept coupled by the pressure difference between the inside of the coupling valve and the reactor pool. If the primary flow rate decreases below the set point, 90% of the normal flow rate, the reactor is shutdown and the header of the coupling valve falls by gravity force. Then, the residual heat is removed by natural circulation in the reactor pool.

Fig. 4 shows the cross section of two fuel assemblies. The thickness and width of the fuel plates are 1.52 mm and 67.1 mm, respectively. The gap thickness of the coolant channel between adjacent internal fuel plates is 2.89 mm. The gap thickness of the external channel is 4.52 mm.

Table 1 Design Parameter of the IEA-R1 Reactor

Reactor Parameters	Data
Steady state power level, MW	2 to 5
Fuel enrichment, %	< 19.75
No. of fuel assembly (FA)	24 = 20 (SFAs) + 4 (CFAs)
No. of fuel plates per FA	18 (SFAs), 12 (CFAs)
Thickness of fuel plates, mm	1.52
- Meat	0.76
- Cladding	0.38
Width of fuel plates, mm	67.1
Thickness of water channel, mm	2.89
Meat dimensions, mm	0.76 (T)x62.6 (W)x600 (L)
Total coolant flow, m^3/h)	772
- One FA	22.8
- Fuel assemblies	547.2
- FA Bypass	224.8
Maximum core inlet temp., °C	40
Temperature difference, °C	5.8
Pressure drop, kPa	7.84

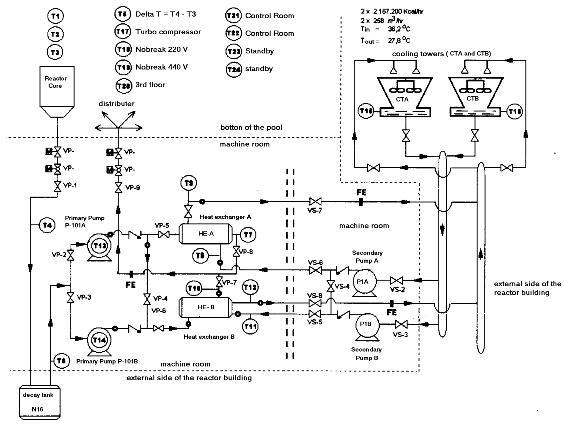

Fig. 1 Schematic Diagram of the Reactor Cooling System

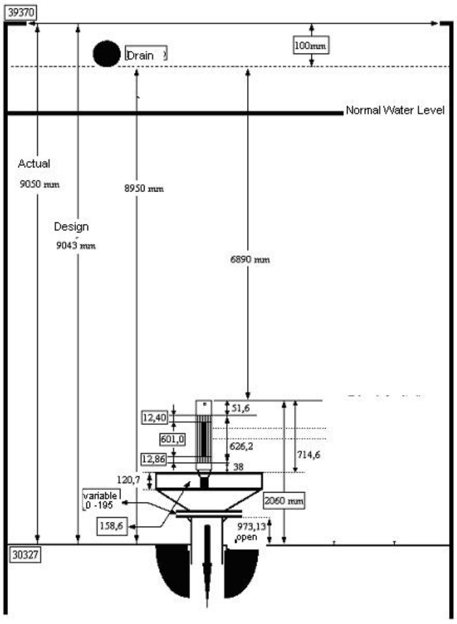

Fig. 2 Schematic Diagram of the Reactor Pool and Structure

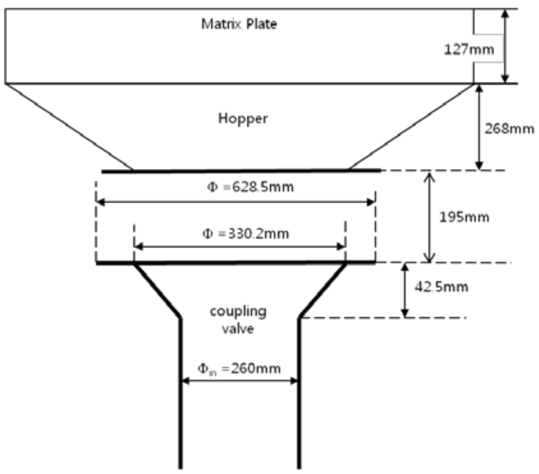

Fig. 3 Schematic Diagram of the Coupling Valve

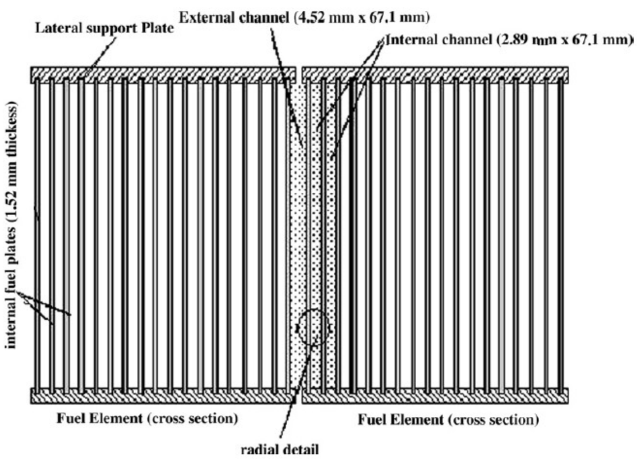

Fig. 4 Cross Section of Two Fuel Assemblies

2.2 Description of the Flow Test

For the loss of flow test, the coolant and cladding temperatures were measured at various locations of instrumented fuel assembly (IFA) as shown in Fig. 5. All the thermocouples are K type with a diameter of 0.5 mm. The error is less than 0.5℃ for temperatures lower than 50℃ and less than 0.8℃ for temperatures between 50℃ and 100℃.

The procedure of experiment was as follows:
(1) The reactor was operated at 3.5 MW and a coolant temperature of 32.7℃ at the pool inlet.
(2) The primary cooling pump was turned off at 7925 seconds.
(3) The reactor was turned off at 7928 seconds when the primary flow reached 90% of the normal flow rate.
(4) The coupling valve started to be decoupled at 46 seconds after the pump was turned off.

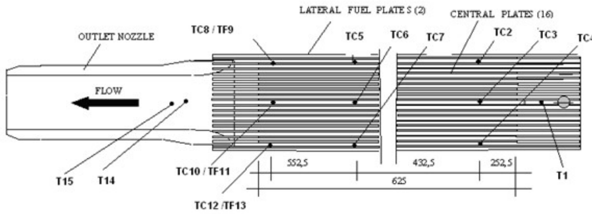

Fig. 5 Measurement Location of Temperatures

3. SPACE Modeling and Simulation Results

3.1 SPACE modeling

Fig. 6 shows the node diagram of the experiment for SPACE simulation. The reactor pool, core and coupling valve are modeled. The pumps, heat exchanger and other components in primary cooling system were not included in the modeling because the detail data of these components were not provided. The flow rate during normal and transient were modeled as TFBC of 910 and 920. The coupling valve was modeled as two valves (#245, #247) since the flow path through the two valves were possible when the coupling valve opened.

The core was modeled using the pipe component. The pipe component #211 to #213 were IFA. The pipe component # 221 and #231 represent non-instrumented fuel assemblies and fuel assembly bypass, respectively. The fuel plates were modeled using heat structure option with heat source. The axial power distributions of IFA were obtained from the experiment report [1].

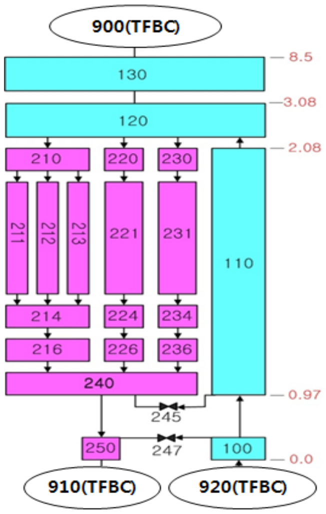

Fig. 6 SPACE Nodalization

3.2 Simulation Results

The flow rate at reactor core and pump are shown in Fig. 7. The reactor core flow rate is the same as the pump flow during the coupling valve closed. When the valve opens, the core flow rapidly decreases and reverses due to buoyancy force.

The coolant temperatures at the inlet and outlet of the IFA are shown in Fig. 8. In the Fig. 8, the first characters of the legends, S, R and E, indicate experimental data, RELAP and SPACE calculations, respectively. The calculation results are the same as the measured temperature until coupling valve starts to open because the thermal power of the IFA was estimated from the measured coolant temperature and flow rate. After the coupling valve opens, the calculated coolant temperature (ST14) at the IFA outlet increase and decrease much faster than the experimental data (ET14). The calculated peak temperature is higher than experimental one by 2 K. These tendency were more severe at the inlet of the IFA. The difference between the calculated peak temperature (ST1) and the measured one (ET1) is around 8 K. On the other hand, the difference between the SPACE and the RELAP is not significant.

Fig. 9 and Fig. 10 show the temperature at the coolant channel and cladding temperature. The behaviors of these two temperatures are also similar to those of the inlet and outlet temperature. All calculated temperatures increase and decrease more steeply than the experimental results and the calculated peak temperatures are higher than the experimental one during the flow reversal.

This behavior is directly related to the core flow variation after the coupling valve opens. The core flow rate decreases as the coupling valve opens. As the downward inertia flow decreases, the coolant temperatures at the flow channels increase. As the buoyancy force in the core overcomes the downward inertia force and flow resistance at the reactor core, grid plate, and hopper, the cold pool water flows into the reactor core via the coupling valve. Then, the cladding temperatures and the coolant temperatures at the channels including the IFA channels decrease. Therefore, the modeling of flow paths and loss coefficients is very important after the coupling valve opens. When the coupling valve opens, the flow behavior at the hopper and reactor pool around the coupling valve is supposed to be very complicated. It seems that the flow around the decoupled coupling valve and in the hopper shows three-dimensional characteristics. Since three-dimensional flow behavior cannot be modeled with one-dimensional code, there can be some discrepancies in predicting the core inlet flow.

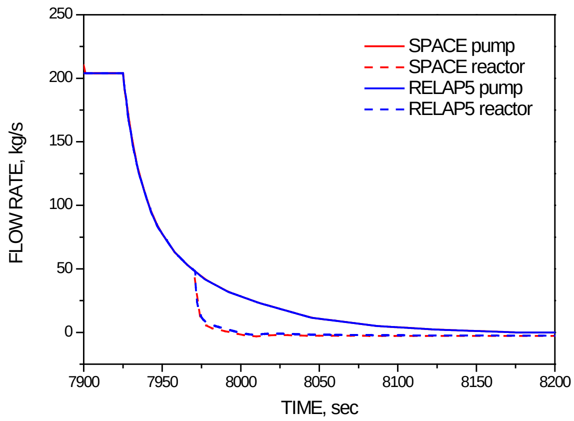

Fig. 7 Mass Flow Rate of Reactor and Pump

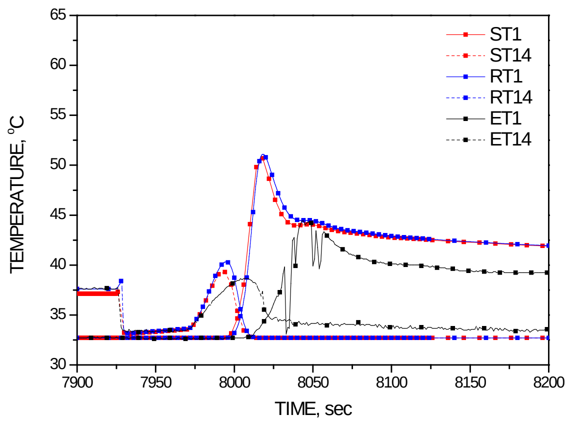

Fig. 8 Coolant Temperature at the Inlet and Outlet of IFA

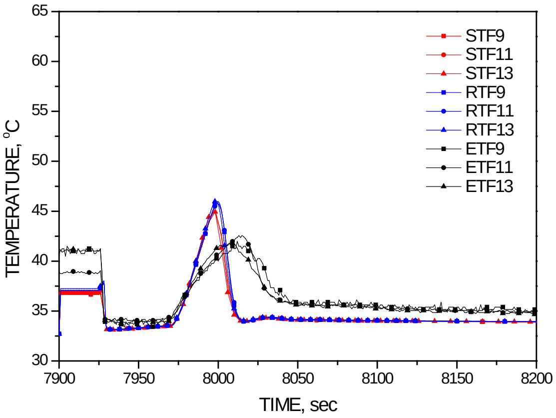

Fig. 9 Coolant Temperature in the Channel

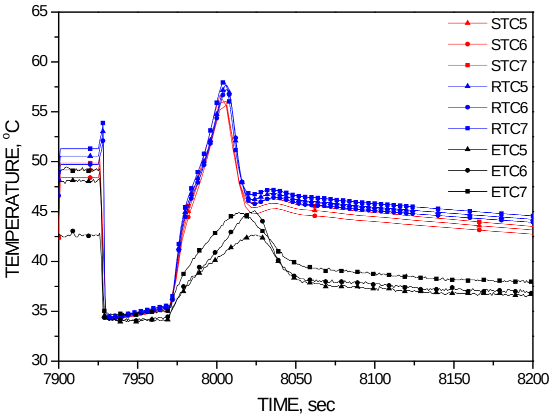

Fig. 10 Fuel Cladding Temperature

4. Conclusions

In order to assess the applicability of SPACE code to the safety analyses of research reactors, a loss of flow test conducted at the IEA-R1 reactor were simulated for the validation of natural convection phenomena in the flat plate fuel channel. It was concluded from the calculation results that:

a) SPACE 3.0 predict reasonably the overall trend of coolant and fuel cladding temperatures.
b) The calculated peak temperatures of the coolant and fuel cladding are higher than experimental results.
c) These discrepancies are supposed to be caused from the one-dimensional modeling of coupling valve and surrounding pool where the complex three-dimensional flow is expected.
d) The calculation results by SPACE 3.0 and RELAP5/MOD3.3 are almost identical.

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