*Transactions of the Korean Nuclear Society Spring Meeting*
*Jeju, Korea, May 22-23, 2025*

# Analysis of LOCA for a Conceptual Design of a Research Reactor with an Upward Flow in the Reactor Core

Suki Park [a*], Jong-Pil Park [a], Youn-Gyu Jung [a], Dongwook Jang [a], Donghyun Kim [a], Cheol Park [a]
[a]*Korea Atomic Energy Research Institute, 111, Daedeok-Daero 989 Beon-Gil, Yuseong-Gu, Daejeon, 34057, Korea*
[*]*Corresponding author: skpark@kaeri.re.kr*

***Keywords :** research reactor, loss of coolant accident, numerical simulation

## 1. Introduction

A reactor core flow is generally designed to be upward direction in high power and open pool type research reactors in order to meet the design requirement of net positive suction head for primary cooling pumps. In addition, the upward core flow has advantage for removing the residual heat in the core by natural convection in loss of flow events.

This paper deals with numerical analyses of loss of coolant accidents to set up the design bases of the primary cooling system and engineered safety features for a conceptual design of a high power and open pool type research reactor with an upward flow in the reactor core in normal power operation.

## 2. System Modeling and Results

### *2.1 RELAP5 Modeling*

The research reactor consists of the reactor pool, the primary cooling system (PCS), the secondary cooling system (SCS) and other fluid systems. The reactor structure is arranged at the pool bottom and its upper part is open to the pool water. The primary cooling system has two trains consisting of a primary cooling pump (PCP), a heat exchanger (Hx), flap valves (FVs), siphon break valves (SBVs), and check valves (CVs). Additionally, there is a decay tank at the upstream common pipe of the two trains. The primary coolant flows into the reactor plenum located under the reactor core. The fuel is cooled by an upward and forced flow in normal power operation.

Figure 1 is the RELAP5/MOD3.3 node diagram modeling the reactor pool, the reactor structure, the primary cooling system, and the secondary cooling system [1,2]. The secondary cooling system is simply modeled as time dependent volumes and junctions providing the cooling conditions. The reactor core is modeled as three parts of a hot channel (210), an average channel (220), and a gap (230) for cooling the neutron absorber rods and irradiation rigs. Figure 1 shows the elevation of major components from the reactor pool bottom.

The reactor has a siphon break pipe (SBP) and siphon break valves. The siphon break pipe is arranged between the reactor outlet pipe at the highest location and the reactor pool top. The siphon break valves are installed in the pipes connecting the reactor inlet pipe at the highest location to the reactor pool. They prevent a loss of pool water from siphoning in a break of primary cooling system pipes.

The flap valves installed in the pipe connected to the reactor plenum are opened as a loss of normal forced flow occurs. A natural circulation flow is established via the flap valves, reactor inlet pipes, reactor inlet plenum, reactor core, chimney, and reactor pool. The residual heat in the reactor core is removed to the reactor pool.

The check valves are designed to have a minimum flow area for a natural circulation flow through the primary cooling system and to prevent a reversal flow and maintain an upward flow in the reactor core even in case of a pipe break in the primary cooling system.

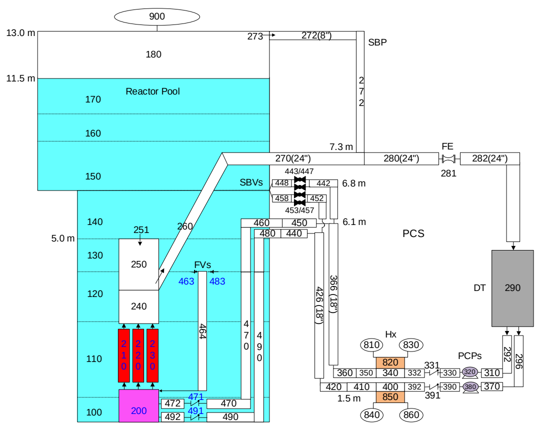

Fig. 1. RELAP5 Node Diagram

The major reactor parameters are summarized in Table 1. The thermal power is 20 MW. The fuel is $U_3Si_2$ and flat plate type.

Table I: Major Reactor Design Parameters

| Parameters | Design Values |
|---|---|
| Reactor power | 20 MW |
| Power peaking factor | 3.0 |
| Axial / Radial peaking factor | 1.25 / 2.4 |
| Fuel | Flat plate type fuel |
| Thermal design flow rate | 600 kg/s |
| Core inlet coolant temp. | 35 ℃ |
| Pool water level | 11.5 m |

### *2.2 LOCA with CVs at the Plenum Inlet Nozzle*

Investigated locations of pipe breaks are a downstream of the heat exchanger, an upstream of the pump, an upstream of the decay tank, and an upstream of the reactor structure in the pool. The most severe LOCA is the break of pipe downstream of the heat exchanger. Figure 2 and 3 show the maximum fuel temperature (MFT) and the minimum critical heat flux ratio (MCHFR). The MCHFR is calculated based on the CHF correlation developed in narrow rectangular channels covering upward and downward flows including a flow stagnation during a flow reversal [3].

The maximum break size is the same as the cross-sectional area of the pipe. The fuel is not damaged and the integrity remains. In this LOCA, the CV in the plenum nozzle prevents a reversal flow to the broken pipe, and the pump of intact train maintains an upward forced flow required for cooling the fuel right after the reactor trip. After the pump is off, the coast-down flow and natural circulation flow via the flap valve remove the core residual heat.

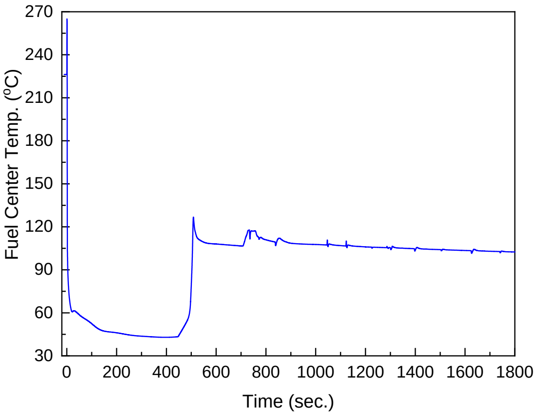


Fig. 2. MFT in LOCA with CVs at the Plenum Inlet

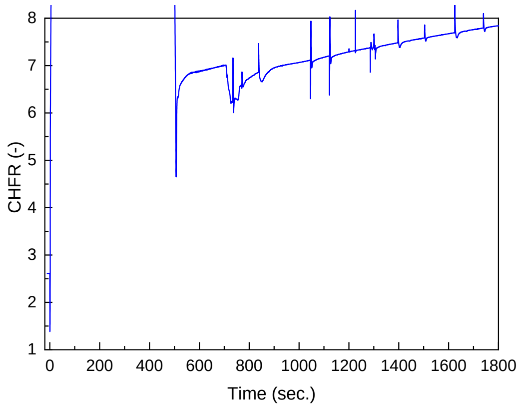


Fig. 3. MCHFR in LOCA with CVs at the Plenum Inlet

### *2.3 LOCA without CVs at the Plenum Inlet Nozzle*

The same break locations and sizes as those of the LOCAs with CV are investigated.

The most severe LOCA is the break of pipe downstream of the heat exchanger. The flow in the reactor core is abruptly reversed from upward direction to downward direction. The fuel temperature sharply increases and exceeds the melting temperature of Aluminum alloy. Figure 4 shows the maximum fuel temperature. The thermal property tables as a function of temperature are unreasonably extended to proceed with the calculation continually. However, the calculation is failed around 4.0 seconds after the pipe break. In order to ensure the fuel integrity in this study, the break size is less than around 20% of the cross-sectional area of the pipe.

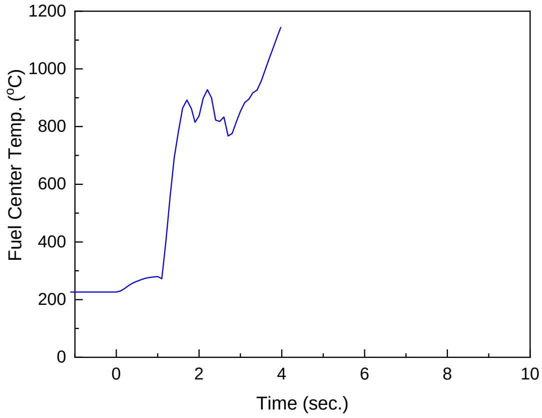


Fig. 4. MFT in LOCA without CVs at the Plenum Inlet

## 3. Conclusions

Analyses of loss of coolant accidents to set up the design bases of the primary cooling system and engineered safety features have been carried out by using RELAP5MOD3.3. The check valves at the plenum inlet nozzle ensure the fuel integrity even in a break of pipes upstream of the reactor structure in a high power and open pool type research reactor with an upward flow in the reactor core in normal power operation. However, the fuel integrity is not ensured without the check valves.

## ACKNOWELGEMENTS

This study was carried out with the financial support from the Ministry of Science and ICT of Korea (2710007312).

## REFERENCES

[1] RELAP5/MOD3.3, Code Manual Volume I: Code Structure, System Models, and Solution Methods, 2001.
[2] RELAP5/MOD3.3, Code Manual Volume II: User's Guideline and Input Requirements, 2001.
[3] M. Kaminaga, K. Yamamoto and Y. Sudo, Improvement of Critical Heat Flux Correlation for Research Reactors using Plate-Type Fuel, Journal of Nuclear Science and Technology, Vol. 35, No. 12, p. 943~591, 1998.