Transactions of the Korean Nuclear Society Autumn Meeting
Gyeongju, Korea, October 29-30, 2015

A Preliminary Analysis of Reactor Performance Test (LOEP) for a Research Reactor

Hyeonil Kim [a*] and Su-Ki Park [a]

[a]*Research Reactor Core Design, Korea Atomic Energy Research Institute, 989-111 Daedeok-daero., Yuseong, Daejeon 305-353*
[*]*Corresponding author: hyeonilkim@kaeri.re.kr*

1. Introduction

Commissioning a research reactor is of great importance to verify and confirm the operational performance and the safety of the design of the research reactor. The final phase of commissioning is reactor performance test, which is to prove the integrated performance and safety of the research reactor at full power with fuel loaded such as neutron power calibration, Control Absorber Rod/Second Shutdown Rod drop time, I&C function test, Criticality, Rod worth, Core heat removal with natural mechanism, and so forth [1].

The last test will be safety-related one to assure the result of the safety analysis of the research reactor is marginal enough to be sure about the nuclear safety by showing the reactor satisfies the acceptance criteria of the safety functions (Fig. 1) such as for reactivity control, maintenance of auxiliaries, reactor pool water inventory control, core heat removal, and confinement isolation. After all, the fuel integrity will be ensured by verifying there is no meaningful change in the radiation levels.

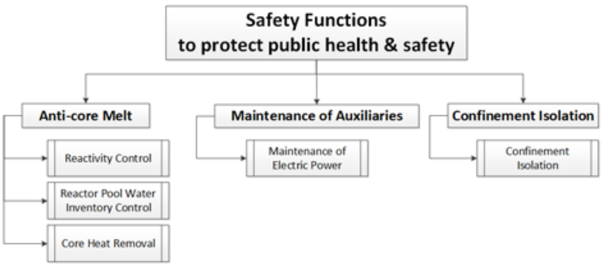

FIG. 1 Safety Functions Classification

To confirm the performance of safety equipment, loss of normal electric power (LOEP), possibly categorized as Anticipated Operational Occurrence (AOO), is selected as a key experiment to figure out how safe the research reactor is before turning over the research reactor to the owner.

This paper presents a preliminary analysis of the reactor performance test (LOEP) for a research reactor. The results showed how different the transient between conservative estimate and best estimate will look.

2. Method of Analysis

2.1 Modeling

A conventional open tank-in-pool research reactor is modelled by using the RELAP 5/Mod3.3 [2]. The model constitutes reactor structure assembly, the core, the primary cooling circuit, and so on.

For the analysis of LOEP, the calculation modeling is established only for the PCS. The SCS is only modeled as the boundary condition.

2.2 Loss of normal electric power

Loss of normal electric power can occur due to either electric load conditions such as overload in the system buses or natural & environmental conditions, such as flood, storms, earthquake, tsunami etc.. Terror of sabotage can also be the possibility.

For any reason, if a loss of normal electric power, also called as a loss of offsite power (LOOP), a possibly anticipated operational occurrence (AOO), occurs, incoming switchgear, intermediate switchgear, load center, and motor control center are tripped in a series. Therefore, the primary cooling system pumps, secondary cooling pumps and cooling tower blowers come to stop. And as soon as the electrical power to the reactor shutdown system is cut off, the reactor power decreases rapidly by the immediate insertion of control rods and second shutdown rods.

At the beginning, the reactor core is cooled by slowing down coolant through the PCS pipe by the inertial force of pump, flywheel and coolant itself. Due to the decay power after reactor trip, the flow at the core changes its direction from downward to upward as the natural convection is developed.

As the flow through the PCS decrease, the flap valves open, and pool water inflow to the pipe which connects to the core and a natural circulation through the flap valve is established using the pool as a huge heat sink. The siphon valves connected to the reactor outlet PCS pipe also open when the flow through the PCS decreases to a preset value.

2.2.1 Conservative estimation

Analysis method and major assumptions used in this analysis are as follows:

1) Reactor is tripped by the free drops of control rods due to de-energizing of the electromagnet

and the trip delay time is assumed to be about 0.1 seconds.

2) CARs are inserted with a condition that one CAR with the largest reactivity is extracted from the core.
3) Cooling water flow in the secondary side is assumed to be reduced to zero within 1 second following a loss of electric power.
4) Negative reactivity feedback effects by fuel and coolant temperature rises are not considered.
5) Flap valves are open when the pressure difference across the flap valves is smaller than 1.5kPa. In the simulation one of the two flap valves is assumed to open.

2.2.2 Best estimation

1) The trip delay time is not greater than 0.1 seconds.
2) All of SSR as well as all of CAR are inserted into the core due to LOEP.
3) The decay power is selected as a best estimate.
4) Negative reactivity feedback effects by fuel and coolant temperature rises are considered.
5) Flap valves works as designed.

3. Results

3.1 Conservative estimates

When a loss of normal electric power occurs, which is one of very common AOOs, the reactor is tripped by the free dropping of CARs even without considering the Reactor Protection System action.

In this case, core power (fig. 2) and PCS flow (fig. 3) decreases rapidly after the initiation of a loss of electrical power by insertion of control rods.

The flow through the flap valves are well established (fig. 4), where decay heat is removed by the natural circulation through the reactor pool for the long term cooling. The minimum critical heat flux ratio [3] in a hot channel is far from the design limits (fig. 5). The coolant temperatures at inlet/outlet of the core show the direction of flow path through the core changed from down (forced flow) to up (natural circulation).

Therefore, fuel cooling does not make any safety problem.

4.2 Best estimates

With reduced initial core power (fig. 2), increased PCS flow (fig. 3) at nominal condition and flow through both flap valves (fig. 4). The minimum critical heat flux ratio in a hot channel is far bigger than the result from the conservative safety analysis (fig. 5). The coolant temperatures at inlet/outlet of the core are lower than the results from the conservative analysis by about 3 degree.

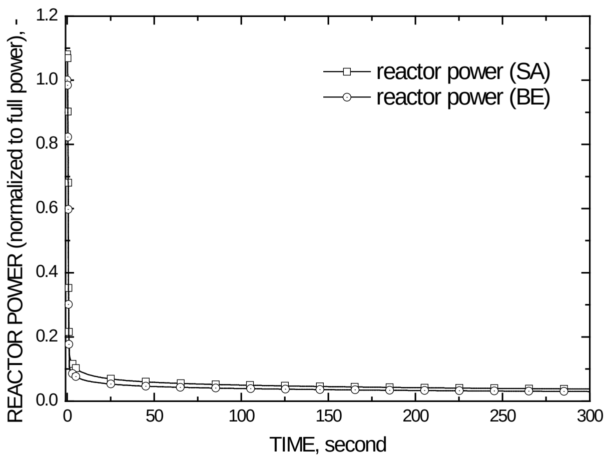

FIG. 2 Reactor power transient

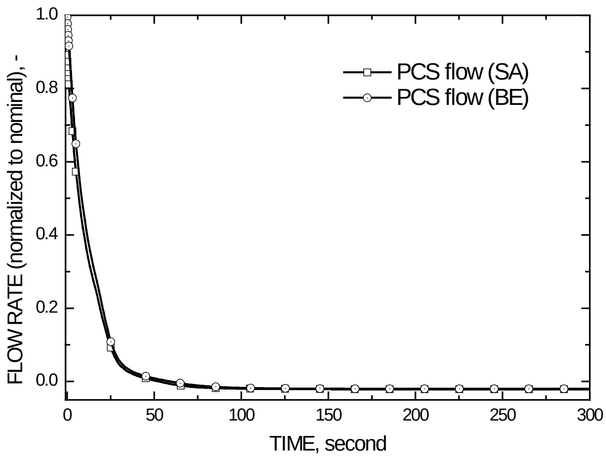

FIG. 3 PCS flow transient

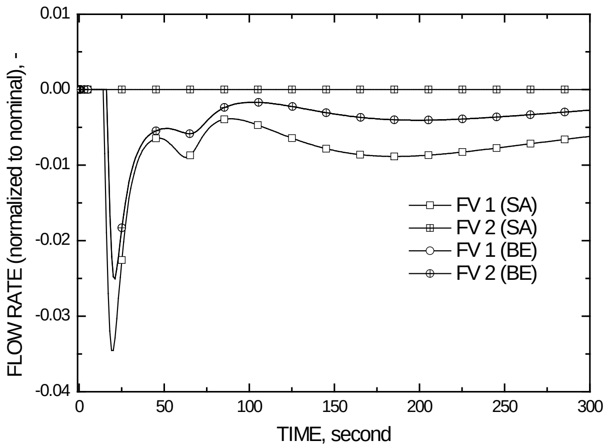

FIG. 4 Flow through valves

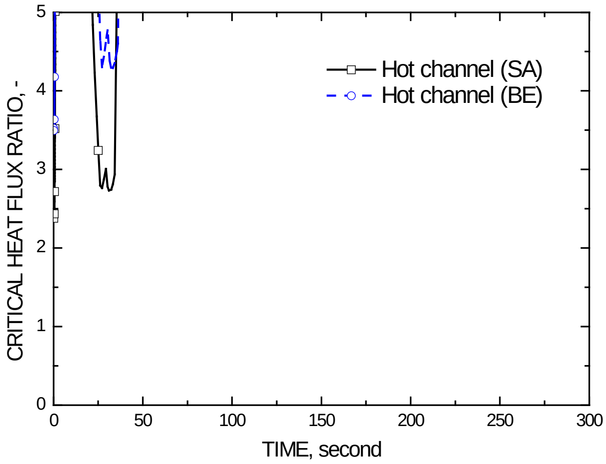

FIG. 5 CHFR transient

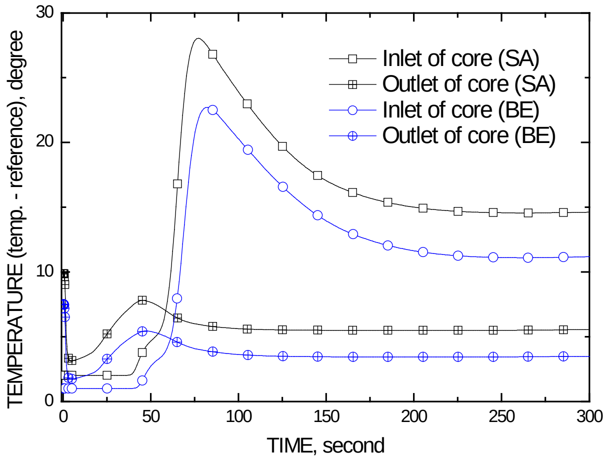

FIG. 6 Temperature transient

4. Summaries and Conclusions

Preliminary analyses have shown all probable thermal-hydraulic transient behavior of importance as to opening of flap valve, minimum critical heat flux ratio, the change of flow direction, and important values of thermal-hydraulic parameters.

A preliminary comparison to conservative estimation has shown that the nuclear reactor safety of the research reactor will be assured by verifying that the reactor power and the PCS flow rate are conservative.

REFERENCES

[1] Research Reactor (Design, Management and Utilization), KAERI, 2009.

[2] RELAP5-3D Code Manual Volume IV: Models and Correlations, INEEL-EXT-98-00834, Revision 1.1b, July 1999.s

[3] Kaminaga, M., Yamamoto, K., and Sudo, Y., Journal of Nuclear Science and Technology, 35 (12), pp. 943-951, 1998.