Thermal-Hydraulic Analysis for Supporting PSA of SBLOCA in APR+

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Abstract: The Advanced Power Reactor Plus (APR+), which is a Gen III+ reactor on the APR1400, is being developed in Korea. To enhance the safety of the APR+, a passive auxiliary feedwater system has been adopted for passive secondary cooling in the APR+. For estimating the safety of APR+ design, the probabilistic safety assessment is performed. This paper discusses the success criteria verified and decided for more realistic and accurate safety evaluation of APR+. The analysis is performed by the best estimate thermal-hydraulic code, RELAP5/MOD 3.3. The Sensitivity analysis was performed to decide the interaction of break sizes, number of high pressure safety injection pumps and the pilot operated safety and relief valves and the timing of operator's action for feed and bleed procedure. This study shows that the plant can be cool down without core damage as the one of the high pressure safety injection pump is available and the one of the pilot operated safety and relief valve is open at least in 80 minutes after the pilot operated safety and relief valve is first opened during small break loss of coolant accident without loss of offsite power. Also the most of the cases given scenarios except 1.97inch break analyzed to need to the additional action for preventing the core damage during small break loss of coolant accident with loss of offsite power. The analysis results can be used for contribute more realistic and accurate performance of the probabilistic safety assessment of APR+.
Keywords: PSA, Operator Action Time, feed and bleed, RELAP5/MOD 3.3, APR+

1. INTRODUCTION

The Advanced Power Reactor Plus (APR+), which is a GEN III+ reactor based on the proven APR1400, is being developed in Korea and standard design approval is in processing. APR+ is 2-loop PWR, which has one reactor vessel, two vertical U-tube steam generators, four vertical, shaft sealed reactor coolant pumps, one pressurizer, four pilot operated safety and relief valves (POSRV)[1].

The major design objectives are as follows;

- Net electrical power : 1,500MW
- Core damage frequency : 1.0E-6/RY
- Construction period (from first concrete to fuel loading) : 36 Months
- Economics : 10% lower than APR1400
- Safety systems : Hybrid safety systems (active + passive)

The large capacity of 1500MW has been adopted for enhancing the economics of APR+ and a passive auxiliary feedwater system (PAFS) has been adopted in the APR+ for enhancing the safety of APR+. The PAFS is the system which performs a function of cooling the primary side and removing the decay heat by introducing a natural driving force mechanism. It completely replaces the conventional active auxiliary feedwater system (AFWS). For estimating whether the design of APR+ satisfies the safety goal of APR+, the probabilistic safety assessment (PSA) is performed. For more realistic and accurate performance of APR+ PSA, this paper discusses the minimum success criteria verified and decided for feed and bleed (F&B) success during small break loss of coolant accident (SBLOCA).

2. Analysis Method

2.1. APR+ RELAP5 code model

For a realistic analysis, APR+ is modeled by using the best estimate thermal-hydraulic code, RELAP5/.MOD3.3. The RELAP5 code is a light water reactor transient analysis code developed for the U.S. Nuclear Regulatory Commission (NRC) which is the use of a two-fluid, nonequilibrium, nonhomogeneous, hydrodynamic model for transient simulation of the two-phase system behavior [2]. Fig. 1 shows the noding diagrams of the APR+ and the PAFS. The RELAP input model of APR+ describes the reactor coolant system with safety injection system and the important parts of the secondary system as main feed water system and PAFS to turbine control system. All essential control and protection systems are modeled for transient analysis. The model is developed in accordance with the design data and system configuration of the APR+ and the PAFS and the PAFS model is attached to the APR+ model. The PAFS consists of a passive condensation heat exchanger (PCHX), a passive condensation cooling water tank (PCCT), check valves, isolation valves powered by a battery (Class 1E DC), piping, instrumentation and control systems. The PCCT in the PAFS model is divided into six volumes to simulate natural convection in the PCCT. The heat exchanger is divided into 70 volumes in order to analyze in detail the condensation inside the heat exchanger tube[3]. It was assumed that primary heat load is transferred to secondary side. As Table 1, steady-state analysis is performed by using APR+ condition successfully.

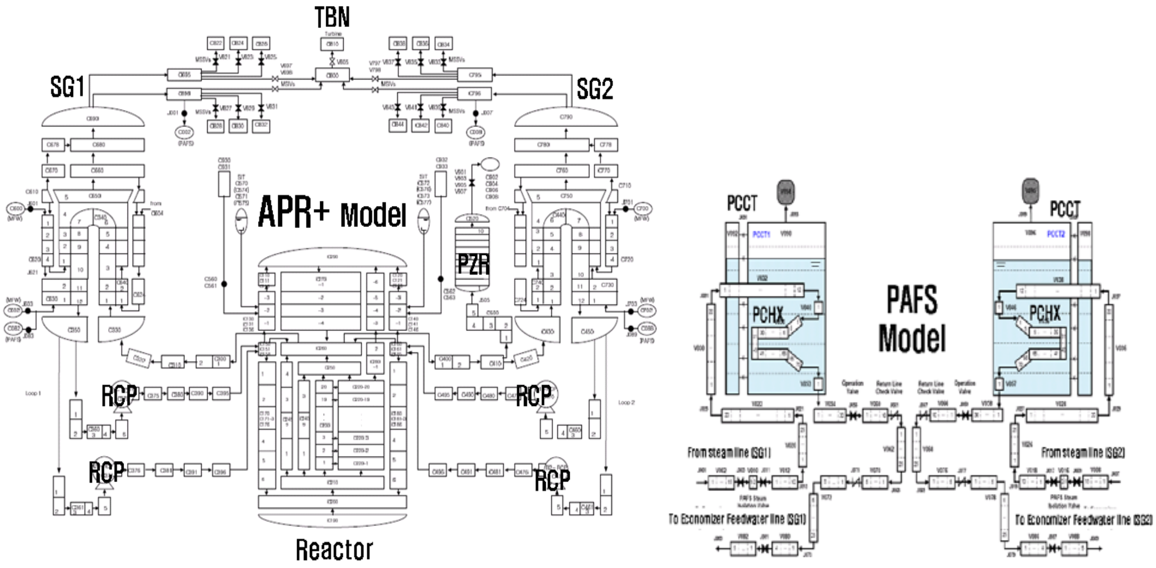

Fig. 1. Noding diagrams of APR+ and PAFS code model

Table 1: Calculation Results of Steady-state condition in APR+ code model

	Plant Parameter	**Design**	**RELAP5 calculation**
Reactor	Core Power (100%) [MWt]	4290.0	4290.0
Primary side	Cold Leg 1A Flowrate (minimum) [kg/s]	5247.8	5247.8
	Hot Leg Temperature [K]	597.05	596.6
	Cold Leg Temperature [K]	563.75	563
	Temperature Rise [K]	33.3	33.6
	PZR Level [%]	52.6	49.37
	PZR Pressure [psia]	2250	2250.09
Secondary Side	Steam Flowrate [kg/s]	1243.0	1237.0
	Steam Pressure [bar]	70.26	69.89
	SG Level [%] (NR)	-	44.10

2.2. Test cases and scenarios

The minimum required equipment for successful F&B managing during SBLOCA is decided from APR+ PSA [4].

- At least two out of four POSRV are opened within 40 minutes after secondary cooling failure for the RCS depressurization
- At least two out of four HPSI pumps has to start for F&B procedure

For accurate APR+ PSA evaluation, the sensitivity test was performed and the required equipment for F&B procedure was proposed [5].

- At least one out of four POSRV are opened after secondary cooling failure for the RCS depressurization
- At least one out of four HPSI pumps has to start for F&B procedure

It is necessary to verify whether the equipment conditions proposed is enough to manage the transient event without core damage and to decide the maximum available time to start the operator's action under the given accident condition [6][7].

For analysis, the transient scenarios and assumptions are as Table 2:

Table 2: Test Scenarios

	Case I	Case II
Break	0.38, 1.0, 1.5 and 1.97 inches of the cold leg break	0.38, 1.0, 1.5 and 1.97 inches of the cold leg break
System conditions	- 1 high pressure safety injection pump is available. - 4 safety injection tanks and 2 PAFS's are unavailable. - Without LOOP	- 1 high pressure safety injection pump is available. - 4 safety injection tanks and 2 PAFS's are unavailable. - With LOOP
Operator action	1 pilot operated safety release valve (POSRV) manually open in 60~90 minutes after POSRV's open first by operator.	1 pilot operated safety release valve (POSRV) manually open in 10~ 60 minutes after POSRV's open first by operator.

3. Result

3.1. Case I

The sensitivity analysis of Case I was performed to decide the interaction of break sizes and the timing of operator's action for feed and bleed procedure. Fig. 2 and 3 show the transient behavior of the PCT and RCS pressure according to the break size when 1 pilot operated safety release valve (POSRV) manually open in 60 minutes by an operator after POSRV's open first.

As the figures show, the limiting case is the break size of 0.38 inch. So the break size of 0.38inch in case I are analyzed for deciding the timing of operator's action. Following the reactor trip caused by pressurizer's low pressure signal, RCS pressure and temperature decrease. The RCS pressure decreases below the HPSI shut-off head, one HPSI is actuated as test scenarios. As considering the unavailability of all secondary systems, the RCS pressure and temperature start to increase. The pressurizer's pressure reaches at the setpoint pressure for opening of POSRVs. Since the RCS pressure is higher than the shut-off head of HPSI pumps during this transient phase, the primary coolant loss through the POSRVs can be not compensated by the HPSI inflow. To prevent the inadequate condition, the operator opens one POSRV shortly. For evaluating the maximum available time to start the operation under the given accident condition, the simulations by the changes of operation time is performed. As Fig. 4, the operator opens one POSRV at 60~90 minutes after first opening of POSRVs. As the RCS pressure decrease below the HPSI shut-off head before the operator's action, the cooling water by HPSI is injected to the RCS. But primary coolant loss from the break limited, the RCS pressure increases over the HPSI shut-off head. The POSRVs automatically open first and the operator manually opens one POSRV at 60~90 minutes after first opening of POSRVs. The core water levels are recovered by HPSI flow and the RCS pressure could be decreased without core damage by F&B procedures in 80minutes as Fig. 4.

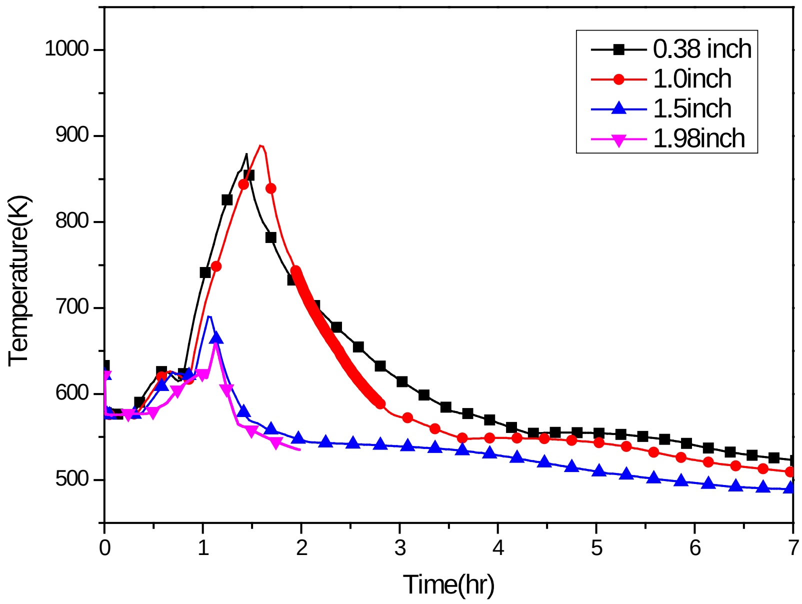

Fig. 2. PCT according to break size (Case I, Operator's action time : 60 min.)

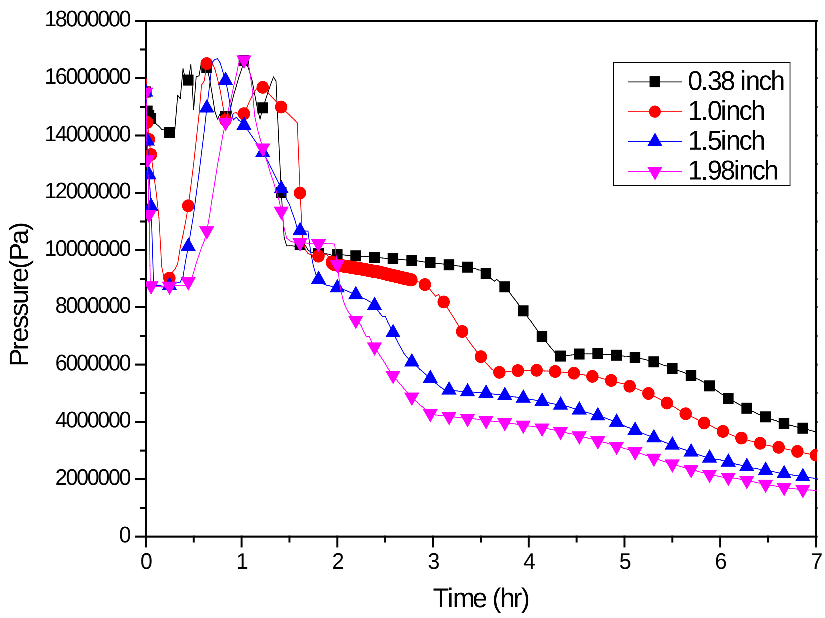

Fig. 3. RCS Pressure according to break size (Case I, Operator's action time : 60 min.)

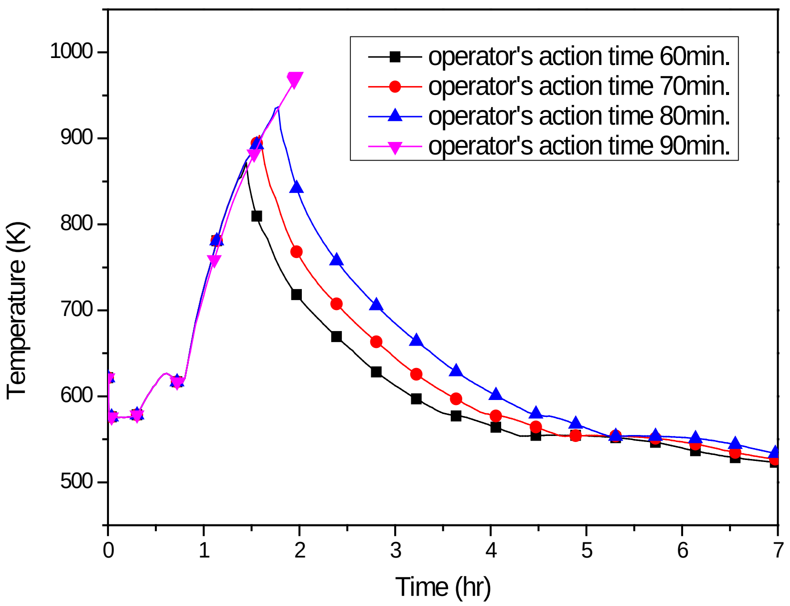

Fig. 4. PCT according to operator's action time (Case I, 0.38inch break)

3.2 Case II

The sensitivity analysis of case II was performed to decide the interaction of break sizes and the timing of operator's action for feed and bleed procedure.

Fig. 5 and 6 show the transient behavior of the RCS pressure and PCT according to the break size when 1 pilot operated safety release valve (POSRV) manually open by an operator after POSRV's open first.

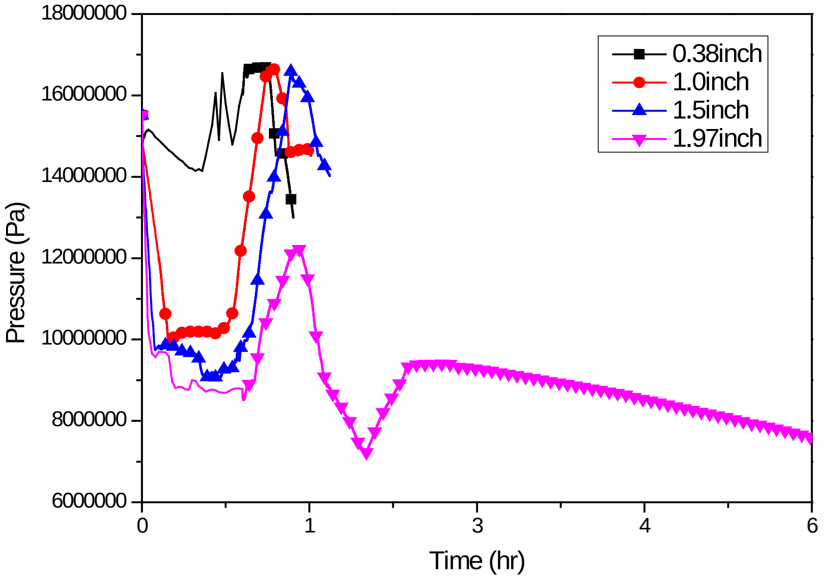

Fig. 5. RCS Pressure according to break size (Case II)

As the figures show, the core is uncovered before operator's action time by given scenarios in the most cases of small break size and the core temperature would increase to the peak temperature.

However the operator's action is not needed as the RCS could be fully cooled down by discharged flow from the break in 1.97 inch break case. In1.97 inch break case, the RCS pressure increase and HPSI pump stops temporarily by the difference the discharged flow from break and injected flow by one HPSI pump at the beginning of the event. But the RCS cool down enough by the discharged flow from break and the core is covered again.

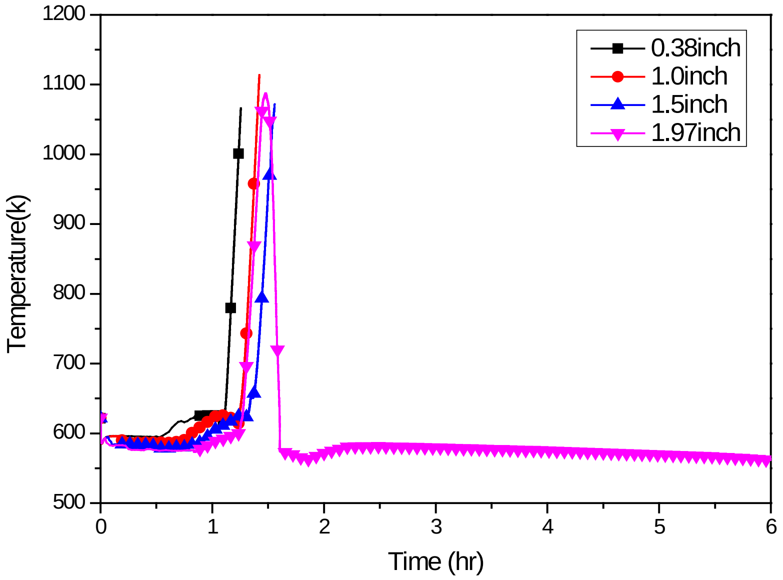

Fig. 6. PCT according to break size (Case II)

4. CONCLUSION

For verifying whether the equipment conditions proposed is enough to manage the transient event without core damage, the analysis is performed by RELAP5/MOD 3.3 code. In Case I, when the RCP pumps operate during SBLOCA event, the results in the smaller size break cases than 1.97inch show that the fast coolant circulation by RCP pump operation promotes the cooling rates of the RCS. There are a variety of the phenomena according to the break sizes. However, the operator's action time is allowed for 80 minutes in the limited case, 1.97inch, and the uncovered core could maintain the stable state for a long time, about a half or one hour by the positive effects given by the fast coolant circulation by RCP pump operation promotes the cooling rates of the RCS. In Case II, when the RCP pumps stop to operate by LOOP during SBLOCA event, the results in the smaller size break cases than 1.97inch show that the core is uncovered before operator's action time by given scenarios and the core temperature would increase to the peak temperature.
In simulations considering test scenarios, the minimum required equipment for successful F&B managing during SBLOCA without LOOP is verified that it is that F&B procedure could be performed successfully as if the operator would open at least one out of four POSRV within 80 minutes after secondary cooling failure for the RCS depressurization. But the most of the cases given scenarios except 1.97inch break during SBLCOA with LOOP analyzed to need to the additional action for preventing the core damage. These analysis results can be used for contribute more realistic and accurate performance of a APR+ PSA.

Acknowledgements

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