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A Coupled Calculation of System Thermal-hydraulics and Three-dimensional Reactor Kinetics for a 12-Finger CEA Drop Event

J.J. Jeong[a]*, S.W. Lee[a], B.D. Chung[a], J.Y. Cho[a], G.C. Lee[b] and S.G. Chi[b]
[a] Korea Atomic Energy Research Institute (KEARI), 1045 Daedeok-daero, Yuseong-gu, Daejeon, Korea
[b] Korea Power Engineering Co., 1045 Daedeok-daero, Yuseong-gu, Daejeon, 305-353, Korea
**Corresponding author: jjjeong@kaeri.re.kr*

1. Introduction

The coupled system thermal-hydraulics (T/H) and three-dimensional reactor kinetics code, MARS/MASTER [1], has been developed to attain more accurate predictions for system transients analyses that involve strong interactions between neutronic and T/H phenomena. The MARS/MASTER code has been successfully verified against various problems including the OECD/NEA main steam line break (MSLB) benchmark problem.

In this paper, the 12-finger control element assembly (CEA) drop event in an OPR1000 plant under full power operation was analyzed, where the 12-finger CEA that is nearest to the hot leg of Loop 2 is incidentally assumed to drop. This instantaneously results in an asymmetric radial power distribution, yielding asymmetric loop behavior and, in turn, leading to a reactor trip due to the cold leg temperature difference. This event clearly requires a coupled calculation of system T/H and three-dimensional reactor kinetics.

2. MARS/MASTER Code Validation Using the CEA Insertion Test Data

During the startup test of an OPR1000 plant, a CEA insertion test was carried out [2]. Until the beginning of the test, the core power had been kept constant at 50 % of the rated power and, then, CEA #25 (see Fig. 1) was manually inserted at a constant speed and it was fully inserted in 300 seconds. Snapshots were taken at the beginning and in 600 seconds to store the plant data.

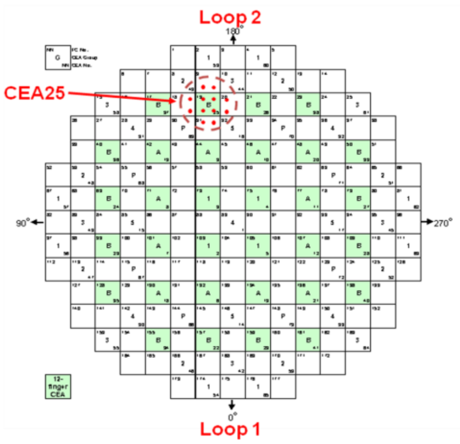

Fig. 1 The location of CEA #25 in the OPR1000 plant.

The CEA insertion test was simulated for the validation of the MARS/MASTER code. Figure 2 shows the MARS nodalization for the OPR1000 plant [3]. The MARS reactor vessel input model and the MASTER core input model are given in Fig. 3. Initial core nuclear data was used for the MASTER code. The insertion of CEA #25 is given as a linear function of time.

Table I shows the results of simulation. The core power and asymmetric T/H behaviors are predicted reasonably well, thus confirming the validity of the code.

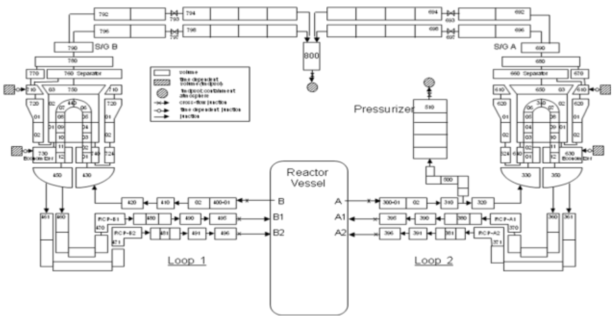

Fig. 2 The MARS nodalization for the OPR1000 plant.

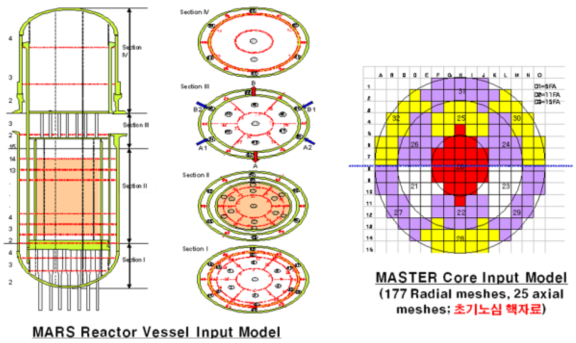

Fig. 3 The MARS reactor vessel input model and the MASTER core input model.

Table I. Results of the CEA insertion test simulation

Parameter	Unit	Test		Calculation		Comments
		t=0 s	t=600 s	t=0 s	t=600 s	
Calorimetric power*	%	49.9	33.5	50.0	32.1	
Neutron flux power	%	49.2	32.8	50.0	33.4	
Boron	ppm	855.0	862.5	855.0	855	Not controlled
Pressurizer pressure	MPa	15.30	15.60	15.30	14.6	Not controlled
Core mass flow	kg/s	16,696	16,719	15,160	15,245	
Hot leg 1 temp.	K	584.8	580.5	585.3	579.8	
Hot leg 2 temp.	K	584.5	576.9	585.3	576.4	
Cold leg 1 temp.	K	569.7	568.4	568.7	566.81	
Cold leg 2 temp.	K	569.7	568.4	568.7	566.84	
dT, loop 1	K	15.2	12.1	16.6	13.0	
dT, loop 2	K	14.8	8.4	16.6	9.6	
SG1 pressure	MPa	7.87	7.766	7.89	7.74	
SG2 pressure	MPa	7.88	7.773	7.89	7.73	
SG1 steam flow	kg/s	381.5	285.2	371.8	275.2	
SG2 steam flow	kg/s	373.0	183.1	371.8	201.5	

*Rated power: 2815 MW

3. The Coupled Calculation of the 12-Finger CEA Drop Event

Initially the plant was under normal operating condition at 100 % of the rated power. At t=0 s, CEA #25 was assumed to drop and, in 4.2 seconds, it was fully inserted. The protection and control system was not modeled in the calculation. However, the steam generator water level was controlled to maintain the wide range level. The governor valve was also regulated to match the core power. For the MASTER code, an equilibrium core data was used and xenon was not taken into account.

Figure 4 shows transient behavior of the core power, the heat removal from core, and the heat removal from the steam generators. Right after the CEA drop, the core power decreases to 82 % with the asymmetric radial power distribution as shown in Fig. 5. It is shown in Figs. 4 and 5 that both the intact and affected side powers decrease at the CEA drop. The heat removal reaches equilibrium sequentially. Figure 6 shows that the hot leg and cold leg temperature decrease until ~50 seconds. The intact cold leg temperatures increase again, but never exceeds the initial temperature. Figure 7 shows the cold leg temperature difference behaviors, where case 1 and 2 indicate the results of different governor valve and feedwater control.

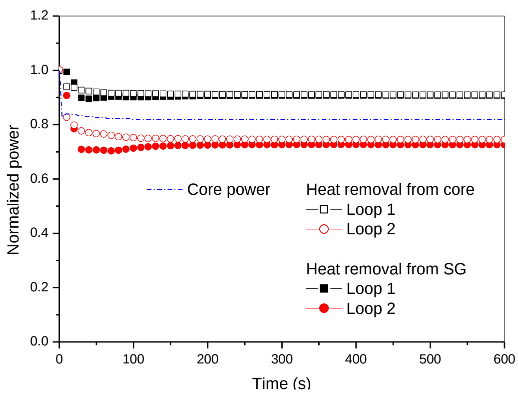

Fig. 4. Transient behavior of the core power, the heat removal from core, and the heat removal from the steam generators.

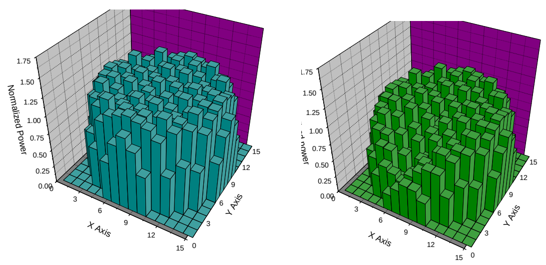

Fig. 5. The core power distributions at 0 s and 600 s.

4. Conclusions

The 12-finger CEA drop event in an OPR1000 plant under full power operation was analyzed using the MARS/MASTER code. The CEA drop instantaneously

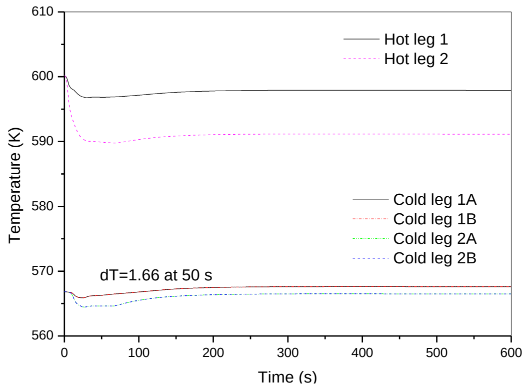

Fig. 6. The hot and cold leg temperature behaviors.

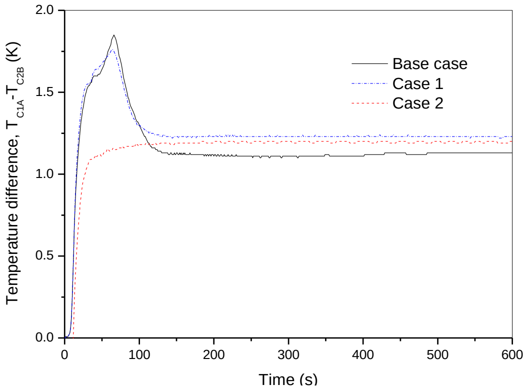

Fig. 7. The cold leg temperature difference behaviors

results in an asymmetric radial power distribution, yielding asymmetric loop behavior. The core powers at both the intact and affected sides decrease after the CEA drop. The hot leg and cold leg temperature decrease until ~50 seconds. The intact cold leg temperatures increase again, but never exceeds the initial temperature. Thus, the core seems safe from the departure from nucleate boiling. However, for further clarification, a detailed MARS/MASTER calculation using more realistic boundary conditions is necessary.

Acknowledgement

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References

[1] J.J. Jeong et al., "Simulation of A Main Steam Line Break Accident Using A Coupled 'System Thermal-Hydraulics, Three-dimensional Reactor Kinetics, and Hot Channel' Analysis Code, Annals of Nuclear Energy, Vol. 33, pp. 820-828 (2006).
[2] OPR1000 Startup Test Data.
[3] J.J. Jeong et al., KAERI-TR-1989/2001, KAERI.