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Reactor core simulation during a SLB accident by ASTRA and CUPID coupling

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1. Introduction

Multi-dimensional physics code system is required to analyze the realistic asymmetric core power behavior caused by the design basis accidents such as a steam line break (SLB) accident and a control element assembly ejection accident. In this perspective, thermal hydraulic code CUPID (Component Unstructured Program for Interfacial Dynamics) and three-dimensional neutron kinetics code ASTRA (Advanced Static and Transient Reactor Analyzer) was coupled.

CUPID was developed by Korea Atomic Energy Research Institute to analyze two phase flow behavior in nuclear power plant components such as reactor vessel, steam generator and containment etc. The CUPID code adopts a two-fluid, three-field model for two-phase flows, and the governing equations are solved over unstructured grids with a semi-implicit two-step method [1, 2]. ASTRA was developed by KEPCO Nuclear Fuel as a nuclear design code for commercial reactor core. ASTRA employs semi-analytic nodal method for the accurate and efficient analysis of two group or multi-group diffusion problems [3].

In this paper, the coupling scheme of ASTRA and CUPID is introduced and the simulation results of core power behavior using the coupled code are described.

2. Numerical Methodology

2.1 Coupling Scheme

ASTRA code was coupled with CUPID through the dynamic link library (DLL) method. CUPID gives core thermal-hydraulic condition to ASTRA and ASTRA returns core power to CUPID. The core power calculated by ASTRA is based on the core thermal-hydraulic condition given by CUPID. The parameters to be transferred between CUPID and ASTRA are shown in Fig. 1. CUPID transfers reactivity feedback parameters such as moderator temperature, moderator density and fuel temperature to ASTRA. ASTRA gives three-dimensional core power to CUPID.

ASTRA simulates core power with 1/4 fuel assembly scale and CUPID employs a fuel assembly scale model. The radial node mapping for the coupled code is shown in Fig. 2. The reactor core model with 26 nodes in axial direction is adopted in both codes.

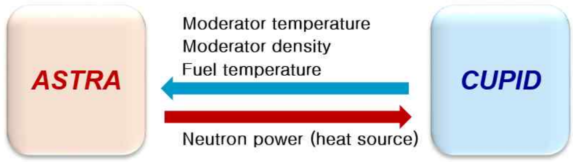
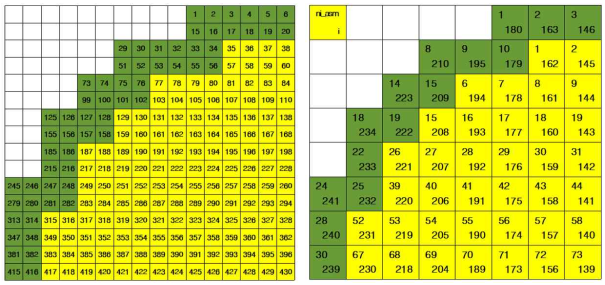


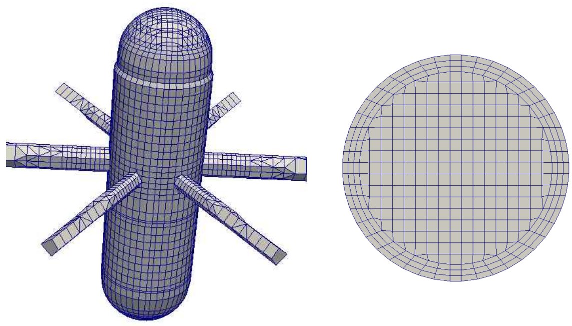
Fig. 1. Coupling scheme between ASTRA and CUPID



(a) ASTRA (b) CUPID

Fig. 2. Radial node mapping between ASTRA and CUPID

2.2 Reactor modeling



(a) Side view (b) Contour view

Fig. 3. Reactor core modeling (A fuel assembly scale for OPR1000)

The porous media approach is adopted because of the complexity of fluid and structure region of reactor core [4]. The geometry and mesh for a fuel assembly scale of OPR1000 are shown in Fig. 3 with a total of 21,618 nodes.

3. SLB accident simulation

The preliminary analysis of a SLB accident for OPR1000 was performed using the coupled code. According to a SLB accident scenario, a steam line break occurs, the other steam lines are isolated.

Therefore, excessive stream releases through the break area. The asymmetric heat removal between steam generators results in asymmetric thermal-hydraulic condition in the reactor coolant system and power redistribution in the core. If moderator continues to cool down after reactor trip, core may have a chance to reach re-criticality due to positive reactivity addition.

3.1 Assumptions

Main assumptions considered in the preliminary analysis are described in Table 1 and the CEA configuration is shown in Fig. 4. It is assumed that a SLB (right side loop) occurs during full power operation (HFP) and the single scram rod with the highest reactivity worth, R41, is not inserted into the reactor core despite of reactor trip signal. Core kinetics parameters are adjusted to maximize the positive reactivity insertion caused by moderator cool-down. The scram rod worth is one of the main parameter to determine reaching re-criticality after reactor trip. In Case 2, scram rod worth is adjusted to an extremely small value to artificially reach re-criticality. The cold-leg thermal hydraulic condition which was calculated by the system performance code is set as the boundary condition of CUPID.

Table 1. Main assumptions

	Case 1	Case 2
Initial core power, MWt	2,871.3	
Moderator temp. coefficient	Most negative	
Doppler coefficient	Most negative	
Core burnup	End of cycle	
Axial shape index	+0.3 (bottom skewed)	
CEA worth on trip, %Δρ	9.4	5.0

					o	R1	o	R1	o					
			o	o	R2	o	R3	o	R2	o	o			
		R3	o	B	o	B	o	B	o	B	o	R3		
	o	o	R4	o	P	o	R5	o	P	o	R41	o	o	
	o	B	o	A	o	A	o	A	o	A	o	B	o	
o	R2	o	P	o	o	o	o	o	o	o	P	o	R2	o
R1	o	B	o	A	o	R12	o	R12	o	A	o	B	o	R1
o	R3	o	R5	o	o	o	R4	o	o	o	R5	o	R3	o
R1	o	B	o	A	o	R12	o	R12	o	A	o	B1	o	R1
o	R2	o	P	o	o	o	o	o	o	o	P	o	R2	o
	o	B	o	A	o	A	o	A	o	A	o	B2	o	
	o	o	R4	o	P	o	R5	o	P	o	R4	o	o	
		R3	o	B	o	B	o	B	o	B	o	R3		
			o	o	R2	o	R3	o	R2	o	o			
					o	R1	o	R1	o					

R41 **Stuck CEA**

Fig. 4. CEA configuration and a stuck CEA position

3.2 Results

After 30 seconds of steady state calculation, steam line break is simulated. Core power increases because of the positive reactivity insertion due to moderator cooldown caused by the excessive steam released through break area. As a result, core power reaches reactor trip setpoint (103.5 % of nominal power) and scram rods drop into the reactor core.

As shown in Fig. 5, after reactor trip, Case 2 reaches re-criticality, however, Case 1 does not reach re-criticality. It can be explained through the different assumption of scram rod worth. Contour map of core power and moderator temperature of Case 2 are presented in Figs. 6 and 7. The asymmetric core power behavior and stuck rod effect with time are clearly shown in these figures.

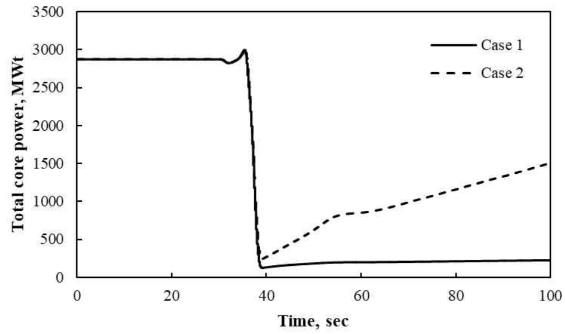


Fig.5. Core power vs. Time

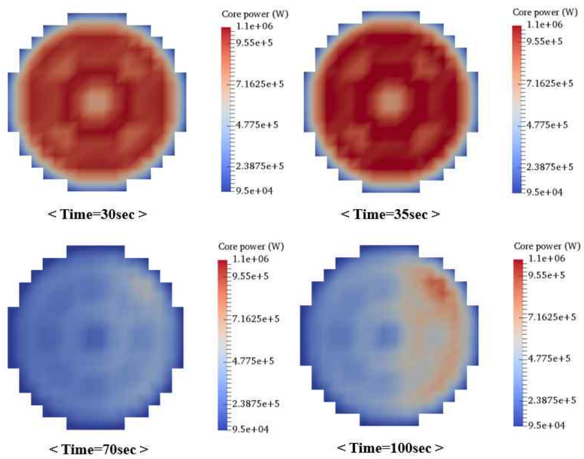


Fig.6. Contour map of core power (Case 2)

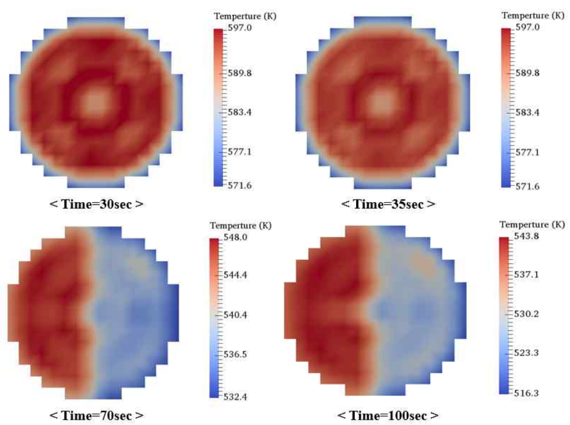


Fig.7. Contour map of moderator temperture (Case 2)

4. Conclusion

Multi-dimensional physics code system for a reactor core simulation was developed through the coupling of CUPID and ASTRA. The preliminary analysis of a SLB accident for OPR1000 was performed and the results show that asymmetric core power and moderator temperature behaviors during a SLB accident can be demonstrated by the coupled code. Therefore, the coupled code would be helpful to understand asymmetric core power behavior and thermal-hydraulic behaviors of a SLB accident in nuclear power plant.

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