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Coupling of SPACE and ASTRA for Transient Analysis based on 3-Dimensional Core Simulation

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1. Introduction

Thermal-hydraulics safety analysis codes, currently used for plant transient calculation, apply point-kinetics model to its core simulation. Because point-kinetics model has the fundamental limitation of asymmetry core simulation, additional conservative assumption makes up for its limitations. However it would predict the more adverse consequences than the phenomena in the real reactor system during transient. Thus, KNF is developing the multi-dimensional safety analysis methodology to predict more realistic consequence.

For this purpose, the best-estimate neutron kinetics code ASTRA[1] and the best-estimate nuclear system analysis code SPACE have been incorporated to provide a best-estimate coupled code system for performing plant transient, especially asymmetric core transient, calculation.

In this paper, ASTRA-SPACE coupling method and test analysis results are described.

2. Coupling Scheme

2.1 Coupling Variables

A-SPACE, external driver of coupled code execution, controls data exchange and time synchronization. Message Passing Interface (MPI) and Win Socket are applied to ASTRA and SPACE respectively.

Fig. 1. Shows the transferred variables to provide the interface between ASTRA and SPACE.

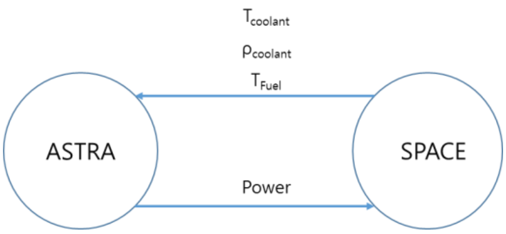

Fig. 1. Transferred variables between ASTRA and SPACE

The kinetics parameter, nuclear power, calculated by ASTRA, is transferred to SPACE via A-SPACE. SPACE calculates the thermal-hydraulic data including coolant temperature, coolant density and fuel temperature based on the nuclear power transferred from ASTRA.

Thermal-hydraulic data, i.e. effective fuel average temperature, reactor coolant temperature and density, related with the effect of reactivity feedback in core are passed to ASTRA, and then ASTRA calculates nuclear power considering Doppler and moderator feedbacks.

2.2 Core Node Grouping

Although ASTRA adopted coarse mesh calculation with one or 1/4 assembly size, SPACE channel modeling, equivalent to the number of node of ASTRA, is fundamentally limited and inefficient. Therefore it is necessary to group core node. One of the examples of core node grouping is shown in Fig. 2.

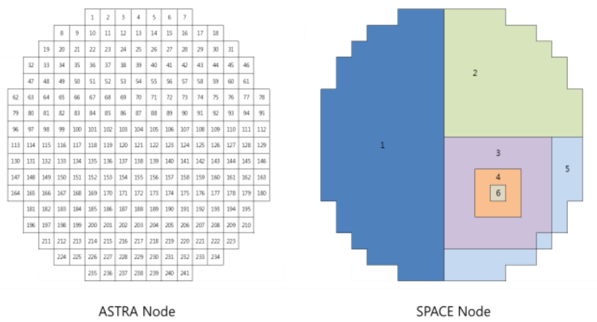

Fig. 2. Example of core node grouping

Example of SPACE core channel model, equivalent to core grouping, is shown in Fig. 3.

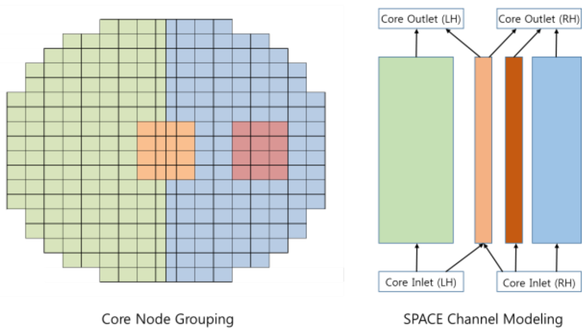

Fig. 3. Example of core channel modeling

3. Test Analysis Results

3.1 CEA Ejection Analysis (Node Grouping Sensitivity)

To confirm the effect of core node grouping, node grouping sensitivity study has been performed. CEA ejection, causes asymmetric core power distribution, is simulated. The selected core for the test analysis is APR1400-type nuclear power plant loaded with a 16x16 fuel assembly. The plant features and core node grouping information are shown in Table I and Fig. 4.

Table I: Main features considered in core node sensitivity

	Hot Full Power
Initial core power, MWt	4,062
Ejected rod worth, $	0.2772
β_{eff}, -	0.00412
Fuel type	16 X 16 lattice, 3.81 m (12.5 feet) active length
Core characteristics	241 assemblies with fresh and once burnt regions and 4.5% U235 average enrichment

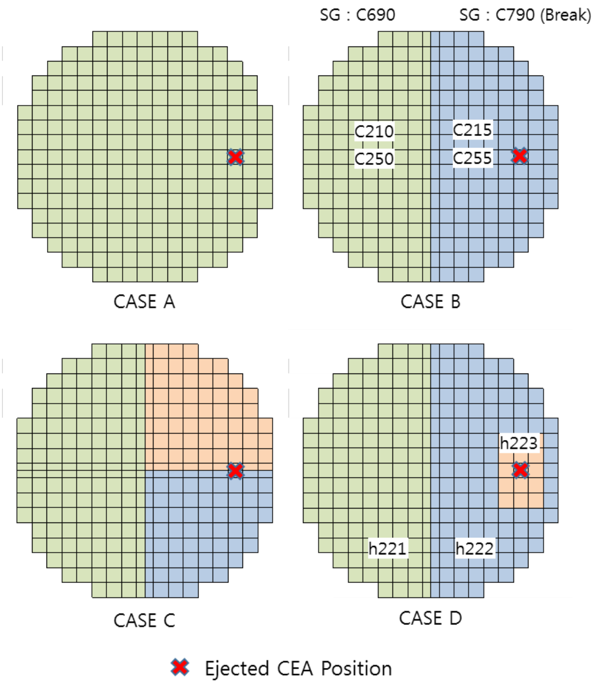

Fig. 4. Core node grouping for sensitivity

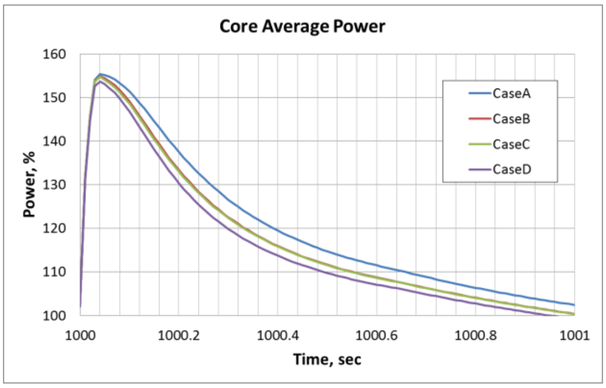

Fig. 5. Core average power (CEAE)

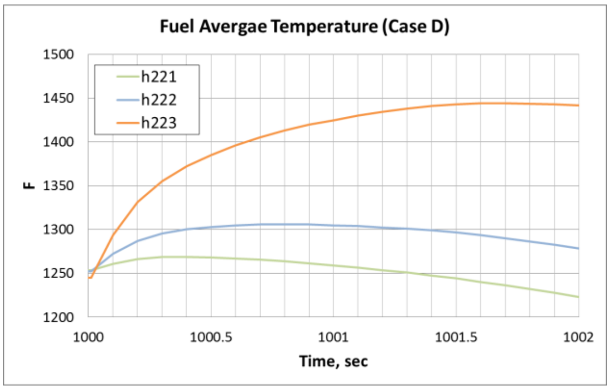

Fig. 6. Fuel temperature (CEAE, CASE D)

3.2 Bank CEA Withdrawal Analysis

Although Bank CEA Withdrawal induces symmetric core power increase, the system behaviors depending on 3D core power increase can be clearly observed. Plant type and features are same as those used in CEAE analysis and Case B grouping model in CEAE analysis is applied.

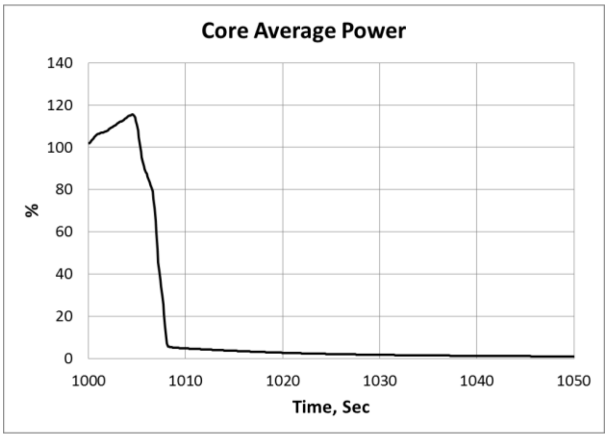

Fig. 7. Core average power (BCEAW, CASE B)

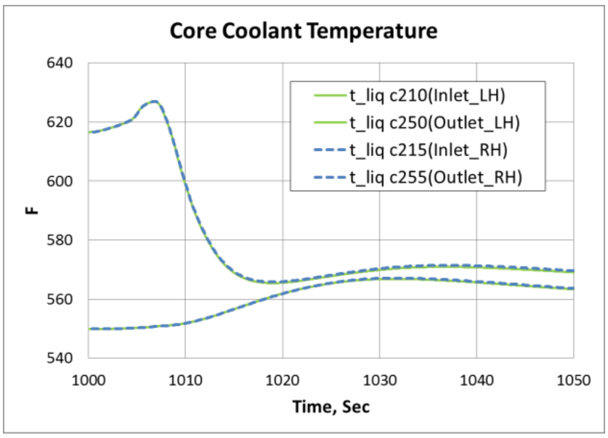

Fig. 8. Coolant temperature (BCEAW, CASE B)

3.3 Main Steam Line Break Analysis

Main steam line break (MSLB) is one of the representative accident inducing asymmetric core power behavior. Plant type and features are same as those used in CEAE analysis and Case B grouping model in CEAE analysis is applied.

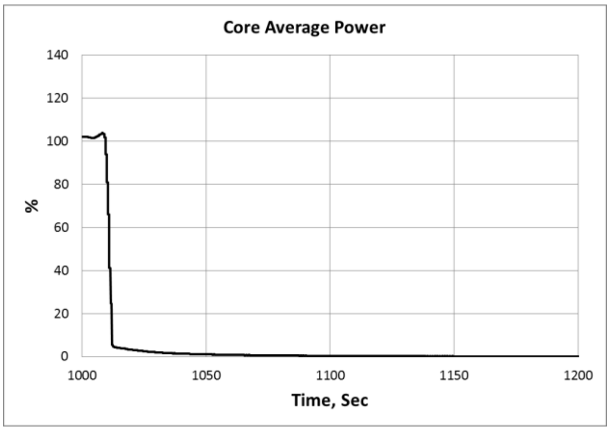

Fig. 9. Core average power (MSLB, CASE B)

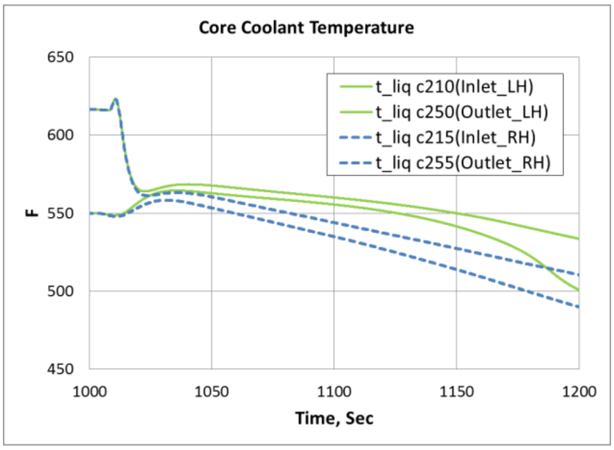

Fig. 10. Coolant temperature (MSLB, CASE B)

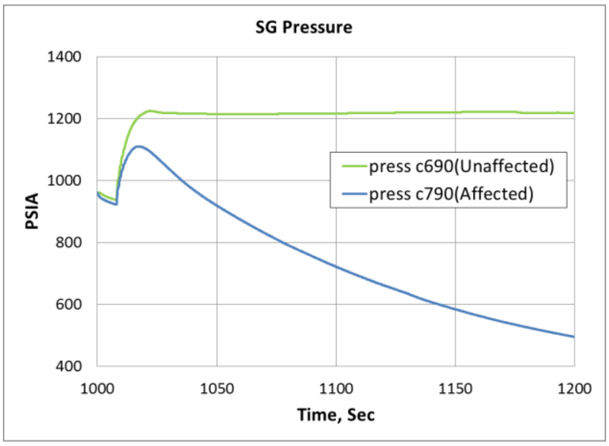

Fig. 11. Steam generator pressure (MSLB, CASE B)

3. Conclusions

The best-estimate neutron kinetics code ASTRA and the best-estimate nuclear system analysis code SPACE have been coupled for performing plant transient, especially asymmetric core transient, calculation. Because of the limitation of the number of channel model of SPACE and the efficiency of calculation, core node grouping has been introduced. CEAE including node grouping sensitivity, BCEAW and MSLB test analysis have been simulated with the coupled code. In conclusion, ASTRA-SPACE coupled code shows reasonable system and core behavior during transient simulation.

In near future, the validation of the coupled code using MSLB benchmark problem should be conducted and its analysis methodology also will be developed.

REFERENCES

[1]. J. I. Yoon, “Verification & Validation of KARMA/ ASTRA with Benchmark and Core-Follow Analyses,” ANS-2011, American Nuclear Society (2011).