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Development of 3-D Transient Reactor Analyzer using ASTRA, THALES, and ROPER codes

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1. Introduction

Three dimensional core transient behaviors are spotlighted due to reactivity-initiated accident (RIA). Usually the calculation model for RIA includes the 3-D kinetics solutions with dynamic thermal-hydraulic and fuel temperature feedback. These models incorporate more detailed neutronic and thermal modeling to address more realistic response of core transient.

KNF has been developing the reactor core design codes for the nuclear power plants to establish the KNF design code system. KNF code system consists of Core Neutronics (ASTRA), Thermal Hydraulic design (THALES), and Fuel Rod design (ROPER) codes. These codes have the capability of steady state and transient calculation to deal with accidental transient case. This paper presents the descriptions on integration process of above 3 design codes and benchmark calculation for verification of transient capability.

2. Code System and Benchmark Calculation

Each code has been developed separately. In this section brief descriptions of the code system used in consolidation are described. And the consolidation method and first verification work are also presented.

2.1 ASTRA

The 3-D nuclear reactor core analysis code, ASTRA, has been developed for both core design and research tool. ASTRA code has a neutron kinetics solution method using multi-group nodal solution method [1][2], in which the source expansion form of the semi-analytic nodal method (SANM) is introduced within the framework of the coarse mesh finite difference (CMFD) formulation. Microscopic cross sections for transport, scattering, absorption and fission and their derivatives are provided by KARMA or CASMO. Here, two-group neutron model is used.

Thermal hydraulic (T-H) and fuel thermal feedbacks are crucial to dynamic core response on the transient condition. ASTRA code incorporates the T-H and fuel rod model for the transient core behavior.

2.2 THALES

The subchannel analysis code, THALES, has been developed for the core thermal hydraulic design of OPR1000, APR1400, and Westinghouse type nuclear power plants in Korea [3]. THALES code use 3 conservation equations of mass, momentum, and energy on the homogeneous flow field. The governing equations are discretized to the finite difference forms. The staggered mesh is used. THALES has the turbulent mixing, single/two-phase friction, several void and CHF correlations, and heat transfer model as thermal hydraulic models.

2.3 ROPER

A fuel rod performance analysis and design code, ROPER [4], has been developed adopting the recent findings and achievements through many researches and studies on the fuel rod in-pile behaviors. The ROPER code calculates the interrelated effects of temperature, pressure, cladding elastic and plastic behavior, fission gas release, and fuel densification and swelling under the time-varying irradiation conditions. Recently ROPER capability is extended for transient analysis. Transient version is incorporated into consolidated code.

2.4 Integration of ASTRA, THALES, and ROPER codes

The integration of ASTRA, THALES, and ROPER is processed following two-steps. First step is to extend the single fuel rod data structure to multi-rod one. Multi-rod data structure is designed to map the inter-related data of kinetics, T-H, and fuel rod model. Data communication is executed before and after each transient time-step shown in Figure 1. The multi-rod data structure is initialized as the steady-state value of ROPER steady-state calculation. For each time step, extracted data from multi-rod data are assigned to each code, and the converged values at the end of time step are restored to multi-rod data. This process is repeated to end of problem time.

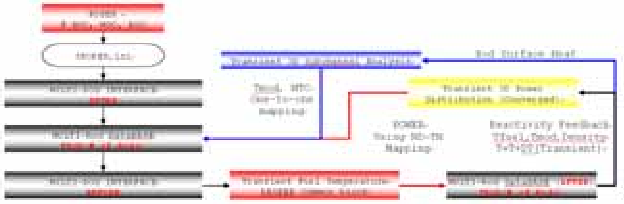


Fig.1 Multi-Rod Data Structure Interface

Second step is designed to calculate the rod heat flux as shown in Figure 2. Calculation is composed with 3-stages. First stage is the input and initialization. Second stage is steady-state calculation for initial condition of transient. Third one is the time-advance of transient.

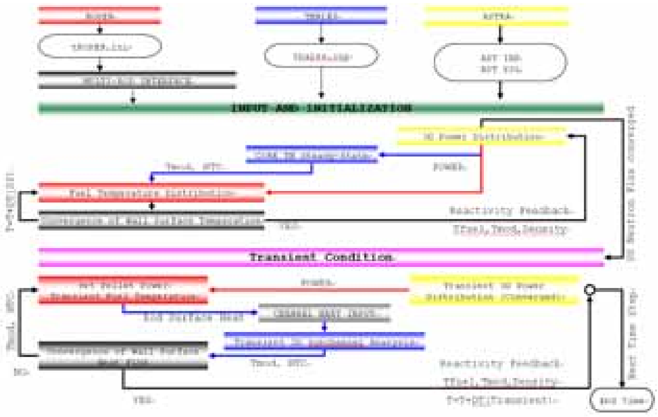


Fig.2 Computation Flow

Second and third stages are executed iteratively to converge the heat flux. Through convergence of heat flux, inter-related thermal condition between T-H and fuel rod can be established. The heat flux convergence is iteratively determined as following,

$$\tilde{T}_w^{m+1} = ROPER(POWER_{pellet}, \tilde{h}_w^m, \tilde{T}_f^m)$$
$$\tilde{q}_{rod}^{m+1} = \tilde{h}_w^m(\tilde{T}_w^{m+1} - \tilde{T}_f^m), \tilde{q}_{channel}^{m+1} \Leftarrow \tilde{q}_{rod}^{m+1}$$
$$\tilde{T}_f^{m+1} = THALES\left(\tilde{q}_{channel}^{m+1}\right), \tilde{h}_w^{m+1} = THALES\left(\tilde{T}_f^{m+1}, \tilde{T}_w^{m+1}, G''\right)$$
$$\tilde{q}_{channel}^{m+1} = \tilde{h}_w^{m+1}(\tilde{T}_w^{m+1} - \tilde{T}_f^{m+1})$$
$$if\,(convergence\_tested)\,next\_time\_step$$

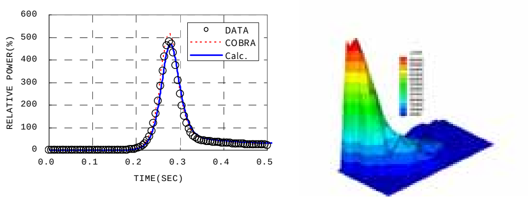
2.5 Benchmark Calculation

For verification of integrated code, NEACRP C1 and C2 cases are used. These benchmark tests are selected for full core with rotational symmetry, ejection of a peripheral control rod assembly at HZP and HFP. The results, shown in Table I, are compared with reference calculation [5] and COBRA-III-C/P [6].

Table I: Results on NEACRP Benchmark problem

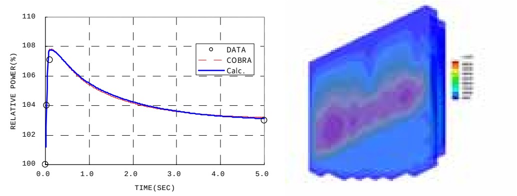
	NEACRP C1			NEACRP C2		
Max Power	Ref.	COB	Calc.	Ref.	COB	Calc.
Time (sec)	0.268	0.275	0.278	0.1	0.1	0.13
Relative (%)	477.3	515.3	469.5	107.1	107.8	107.8

For NEACRP C1 (HZP Rod Ejection) Case, calculated value of ASTRA, THALES, and ROPER code system shows the capability of transient behavior as shown in Figure 3, (a) core power and (b) power distribution in maximum power position at 0.275 sec.



(a) Core relative power (b) power distribution
Fig.3 Results of NEACRP C1 (HZP)

The 3-D transient reactor behavior in case of HFP case (NEACRP C2) is also well simulated as shown in Figure 4, (a) core power and (b) power distribution in core.



(a) Core relative power (b) power distribution
Fig.4 Results of NEACRP C2 (HFP)

3. Conclusions

For 3-D transient reactor behaviors, seamless integration of neutron kinetics (ASTRA), T-H (THALES), transient ROPER (tROPER) codes is performed. Multi-rod data structure is designed for data communication. For verification of integrated code, NEACRP transient cases are simulated. The results well represents the reactor transient behavior.

The integrated code can be served as the new tool to investigate the issues about RIA.

Acknowledgments

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