Verification of ASTRA Code with PWR MOX/UO2 Transient Benchmark Problem

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1. Introduction

In recent, ASTRA (Advanced Static and Transient Reactor Analyzer) has been successfully developed by KNF (KEPCO Nuclear Fuel) as a nuclear design code for commercial reactor core. This code has the capability of the multi-group analysis because of the requirement of a neutron flux solver to simulate a core not only for UO2-fueled but also MOX-UO2 fueled. In addition, ASTRA has been designed to analyze the core characteristics under transient condition as control rod ejection accident.

In this paper, we have performed the benchmark analysis with the PWR MOX/UO2 control rod ejection problem provided by OECD/NEA and U.S. NRC [1] for the purpose of verifying these capabilities of ASTRA.

2. Methods and Results

In this section, we will introduce the methodologies used in the neutronic solver of ASTRA. Then the spec of the benchmark model will be described in detail and the numerical calculation results and their investigation will be followed.

2.1 Methodology

ASTRA employed SANM (Semi-Analytic Nodal Method) for the accurate and efficient analysis of two-group or multi-group diffusion problems. SANM has the advantages that the solution is not only similar in terms of computational accuracy but also superior in terms of the computational efficiency to that of ANM (Analytic Nodal Method) because fission source and scattering source are approximated by high order polynomial approximation [2]. Moreover, in order to accelerate the calculation of multi-group nodal method, two-level CMFD(Coarse Mesh Finite Difference) that accelerates multi-group CMFD with two-group CMFD is adopted.

2.2 Benchmark Model

OECD/NEA and U.S.NRC MOX/UO2 PWR core transient benchmark based on the characteristics of the NEACRP L-335 PWR benchmark proposed by Finnemann adds the complexity of modeling a control rod ejection in a core loaded partially with MOX fuel assemblies, which have the neutronic characteristics sufficiently different to UO2 fuel. Overall, the reactor core chosen for the simulation is based on four-loop Westinghouse PWR power plant. Fig.1 shows the quarter-core loading pattern of the benchmark problem.

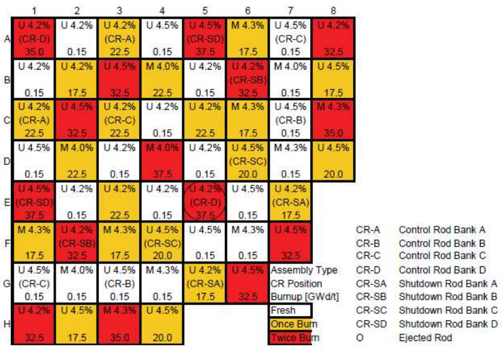

Fig. 1. Core configuration

Also, the calculations of the benchmark problems are divided into four parts:

- Part 1, 2D fixed Thermal Hydraulic conditions
- Part 2, 3D Hot Full Power (HFP) conditions
- Part 3, 3D Hot Zero Power (HZP) conditions
- Part 4, Control rod ejection accident with Part 3 conditions

The calculation results about the four parts above were compared to the results of PARCS [3] provided as reference in the final benchmark report [1] in the next section. In this study, we used the same spatial discretization and cross-sections interpolation for fair comparisons with PARCS.

2.3 Numerical Results

Table I shows the rod worth in terms of energy group and control rod on the 2D fixed Thermal-Hydraulic (T/H) conditions. Almost results are in a very good agreement, however, a slight difference is found out as increasing the number of energy groups on the Rod-In condition. Since the T/H conditions in Part 1 are fixed, the difference of the rod worth is caused by the different multi-group nodal method used in two codes, which are SANM in ASTRA and NEM (Nodal Expansion Method) in PARCS. Generally, it is known that SANM is more accurate than NEM.

Table I: Rod worth on the Part 1 conditions

Single Rod-in	2G (k_{eff})		4G (k_{eff})		8G (k_{eff})	
	ASTRA	*PARCS*	*ASTRA*	*PARCS*	*ASTRA*	*PARCS*
Rod(A,1)	*1.06191*	*1.06191*	*1.06191*	*1.06187*	*1.06169*	*1.06164*
Rod(B,6)	*1.06300*	*1.06299*	*1.06300*	*1.06297*	*1.06279*	*1.06275*
Rod(C,3)	*1.06240*	*1.06240*	*1.06240*	*1.06237*	*1.06220*	*1.06215*

Rod(D,6)	*1.06301*	*1.06301*	*1.06302*	*1.06299*	*1.06281*	*1.06277*
Rod(E,5)	*1.06306*	*1.06306*	*1.06306*	*1.06303*	*1.06285*	*1.06281*
Single Rod-out	2G (k_{eff})		4G (k_{eff})		8G (k_{eff})	
	ASTRA	*PARCS*	*ASTRA*	*PARCS*	*ASTRA*	*PARCS*
Rod(A,1)	*0.99986*	*0.99986*	*0.99977*	*0.99978*	*0.99964*	*0.99963*
Rod(B,6)	*0.99304*	*0.99304*	*0.99286*	*0.99287*	*0.99266*	*0.99265*
Rod(C,3)	*1.00274*	*1.00274*	*1.00263*	*1.00263*	*1.00247*	*1.00247*
Rod(D,6)	*0.99440*	*0.99440*	*0.99423*	*0.99422*	*0.99401*	*0.99400*
Rod(E,5)	*0.99399*	*0.99399*	*0.99383*	*0.99383*	*0.99364*	*0.99363*

The calculation results on the 3D HFP with T/H feedback are showed in Table II. Considering the difference of T/H feedback model between two codes, it shows that ASTRA can predict very similar to PARCS. As mentioned above, the slightly lower boron concentration predicted by ASTRA is attributable to the different nodal method.

Table II: Core parameters on the Part 2 conditions

	2G		4G		8G	
	ASTRA	*PARCS*	*ASTRA*	*PARCS*	*ASTRA*	*PARCS*
CBC(ppm)	*1675.6*	*1679.3*	*1672.0*	*1673.9*	*1670.3*	*1672.0*
T_f *(C)*	*564.2*	*562.9*	*564.3*	*563.0*	*564.3*	*563.1*
T_m *(C)*	*308.9*	*308.2*	*308.9*	*308.2*	*308.9*	*308.2*
D_m *(kg/m*^{3}*)*	*703.9*	*706.1*	*703.8*	*706.1*	*703.9*	*706.1*

The solutions of Part 3 on 3D HZP without T/H feedback were listed in Table III. As expected from the results of Part 1 and 2, the CBC results of ASTRA are almost same as those of PARCS.

Table III: CBC on the Part 3 conditions

	2G		4G		8G	
	ASTRA	*PARCS*	*ASTRA*	*PARCS*	*ASTRA*	*PARCS*
CBC(ppm)	*1340.4*	*1340.7*	*1337.7*	*1337.0*	*1334.9*	*1334.0*

Lastly, the results of Part 4, the transient response to the control rod ejection accident from the Part 3 condition as initial core state, are presented in Fig. 1, 2 and 3. Compared with PARCS results, ASTRA predicts higher by 10~20% in the peak power and slightly earlier in the peak time. This trend is consistent to the previous study [2] about the difference of transient behavior between SANM and NEM. Also, we can find out from Fig. 2 and 3 that the behaviors of fuel temperature and moderator temperature in ASTRA are very similar to the results of PARCS even though the difference of T/H feedback model.

3. Conclusions

In this work, we showed that ASTRA implemented as a neutronic solver for PWR has a good performance in MOX fueled core problem. Despite the difference of nodal method and T/H feedback, the solutions generated by ASTRA in the various core conditions given from the benchmark problems are in a very good agreement compared to reference solutions in terms of two-group and higher multi-groups.

Acknowledgments

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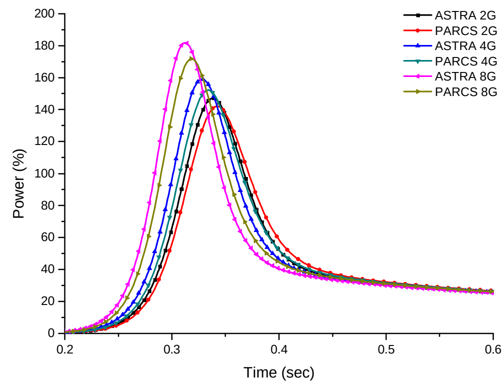

Fig. 2. Transient core power behavior in Part 4

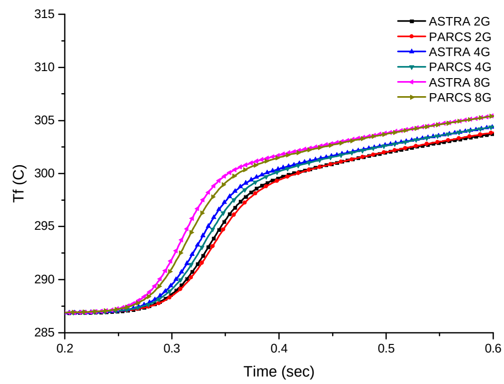

Fig. 3. Fuel temperature behavior in Part 4

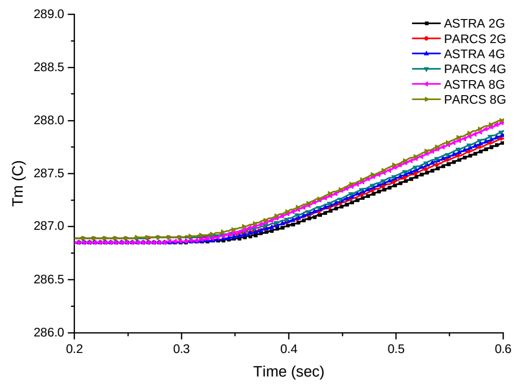

Fig. 4. Moderator temperature behavior in Part 4

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[3] T. Downar, et. al., PARCS: Purdue Advanced Reactor Core Simulator, PHYSOR2002, Seoul, Korea (2002).