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A Preliminary Analysis of SMART Reactor Core Using the COREDAX Code

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1. Introduction

The 3-D neutronics code COREDAX [1] has been developed based on AFEN (Analytic Function Expansion Nodal) method [2] for x-y-z geometry [3] and for hex-z geometry [4]. In this study, the COREDAX code, as a regulatory review tool independent of the designer's, was applied to the SMART reactor core that was designed by KAERI (Korea Atomic Energy Research Institute). For nuclear cross section generation, the HELIOS lattice code [5] was used in this study.

The preliminary results for steady state in various conditions are presented in this paper.

2. Methods and Results

2.1 AFEN Methodology

The AFEN formulation in the x-y-z system starts from the following multi-group diffusion equations in a homogenized node:

$$-\nabla^2 \vec{\phi}(x,y,z) + [\Lambda]\vec{\phi}(x,y,z) = 0, \tag{1}$$

where

$$[\Lambda] = [D]^{-1}\left([\Sigma] - \frac{1}{k_{eff}}[\chi][\nu\Sigma_f]\right),$$

and all the notations are standard.

A general solution to eq. (1) can be represented in terms of analytic basis functions that can be obtained using the method of separation of variables. For practical implementation, we choose the solution of a node expressed in a finite number of terms [3]:

$$\overrightarrow{\phi^n}(x,y,z) = \overrightarrow{E^n} + \overrightarrow{\varphi^n}(x,y,z) + \overrightarrow{\varphi^n}(y,x,z) + \overrightarrow{\varphi^n}(z,x,y), \tag{2}$$

where

$$\begin{aligned}\overrightarrow{\varphi^n}(x,y,z) &= \sinh(\sqrt{\Lambda^n}x)(\overrightarrow{A^n_{0x}}) + \cosh(\sqrt{\Lambda^n}x)(\overrightarrow{B^n_{0x}})\\ &\sinh(\frac{\sqrt{2\Lambda^n}}{2}(x+y))\overrightarrow{C^n_{00x}} + \cosh(\frac{\sqrt{2\Lambda^n}}{2}(x+y))\overrightarrow{C^n_{01x}}\\ &\sinh(\frac{\sqrt{2\Lambda^n}}{2}(x-y))\overrightarrow{C^n_{10x}} + \cosh(\frac{\sqrt{2\Lambda^n}}{2}(x-y))\overrightarrow{C^n_{11x}}\end{aligned} \tag{3}$$

Note that each term in eq. (2) is an analytic solution of eq. (1). The 19 coefficients in eq. (2) are made to correspond to the 19 nodal unknowns for a node: i) one node average flux, and ii) six interface fluxes, and iii) twelve edge fluxes.

To determine nodal unknowns, we build as many solvable nodal equations as the number of these nodal unknowns. These equations consist of a nodal balance equation and associated coupling equations. The coupling equations are obtained from six interface current continuity equations and twelve edge balance equations.

2.2 Verification of COREDAX Code

To verify the COREDAX code, LMW (Langenbuch, Maurer, and Werner) benchmark problem [6] was considered. This benchmark problem is 240cm x 240cm x 200cm size x-y-z problem, and assembly size in x-y direction is 20cm.

Fig.1 shows the configuration of LMW benchmark problem, and detailed information is in [6].

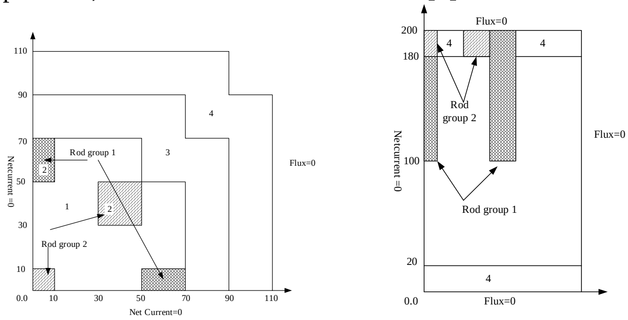


Fig.1 Configuration of LMW benchmark problem

Reference calculation was done by the VENTURE code [7] which is based on FDM (Finite Difference Method) neutron diffusion calculation.

Table 1: k_{eff} and relative error of LMW benchmark problem

	k_{eff}	Relative error
VENTURE*	0.99960	-
COREDAX	0.99964	4pcm

*Reference calculation

As shown in Table 1, estimation of k_{eff} value by the COREDAX code is very accurate. In addition, power distribution of this benchmark problem is shown in Fig.2. The calculation results of the COREDAX code are in excellent agreement with the reference calculation for the problem tested.

VENTURE
COREDAX error (%)

			0.8583 0.04	0.4329 -0.10
		1.1231 0.04	0.9802 0.06	0.6260 0.00
	1.5911 -0.06	1.3970 -0.03	1.0837 0.01	0.7072 0.08
1.5559 -0.05	1.6563 -0.07	1.4414 -0.05	0.9804 0.04	0.7257 0.10

Fig.2 Assembly level power distribution and relative error of LMW benchmark problem

2.3 Numerical Results

A nuclear reactor operates in various conditions (e.g., fuel/coolant temperature, boron concentration, etc…). Therefore, in this paper, six different cases were tested for the SMART reactor core as a preliminary analysis based on the reference design [8]. Detailed information of each case is described below in Table 2.

Table 2: Calculation conditions of six cases

	Control rod	Burnup (MWd/KgU)	Fuel temperature	Coolant temperature	Boron concentration
Case1	ARO*	0	600°C	300°C	500ppm
Case2	ARO	30	600°C	300°C	500ppm
Case3	ARI**	0	600°C	300°C	500ppm
Case4	ARI	30	600°C	300°C	500ppm
Case5	ARO	0	700°C	325°C	1000ppm
Case6	ARO	0	500°C	295°C	0ppm

*ARO: All control rods out from the reactor
**ARI: All control rods inserted into the reactor

The HELIOS lattice code generates assembly-level two group cross sections for given condition, and then the COREDAX code performs steady state whole-core analysis. Table 3 shows k_{eff} results for each case.

Table 3: k_{eff} values for each case

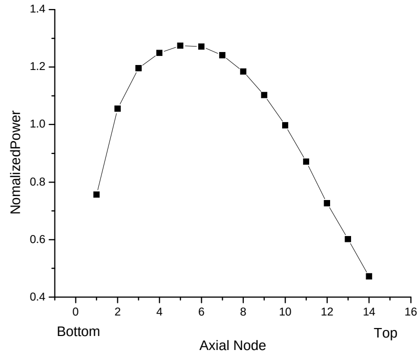
	k_{eff}
Case1	1.088338
Case2	0.930336
Case3	0.808174
Case4	0.736129
Case5	1.013029
Case6	1.113176

Additionally, from the COREDAX steady state analysis, assembly-level power distribution and power peaking factor were obtained. Figs.3 and 4 show these results for case 1.

			0.77044	0.91345	0.77044			
		0.796	1.05127	1.10517	1.05127	0.796		
	0.796	1.0745	1.05004	1.13649	1.05004	1.0745	0.796	
0.77044	1.05127	1.05004	1.1901	1.18329	1.1901	1.05004	1.05127	0.77044
0.91345	1.10517	1.13649	1.18329	1.24597	1.18329	1.13649	1.10517	0.91345
0.77044	1.05127	1.05004	1.1901	1.18329	1.1901	1.05004	1.05127	0.77044
	0.796	1.0745	1.05004	1.13649	1.05004	1.0745	0.796	
		0.796	1.05127	1.10517	1.05127	0.796		
			0.77044	0.91345	0.77044			

Radial power peaking factor = 1.24597

Fig.3 Normalized radial power distribution and power peaking factor (Case 1)



Axial power peaking factor = 1.27136

Fig.4 Normalized axial power distribution and power peaking factor (Case 1)

3. Conclusions

The COREDAX code based on the AFEN method was used to analyze, independently of the designer's tool, the SMART reactor core in various conditions. From these results, it is shown that whole-core (especially on parallelepiped reactor) steady state analysis can be done by the COREDAX code.

As ongoing works, reactor kinetics analysis including Xe/Sm variations and more realistic safety analysis based on neutronics / thermal-hydraulics coupled system codes are in progress.

REFERENCES

[1] N.Z Cho, J. Lee, and B. Cho, "COREDAX – A User's Manual for the Reactor Core Analysis Code Based on Analytic Function Expansion Nodal Method," prepared by Korea Advanced Institute of Science and Technology for Korea Institute of Nuclear Safety, August 17, (2010).

[2] N.Z. Cho, "Fundamentals and Recent Developments of Reactor Physics Methods," *Nuclear Engineering and Technology*, Vol.37, No.1, p25, (2005).

[3] D.S. Kim, N.Z. Cho, "Kinetics Calculation Under Space-Dependent Feedback in Analytic Function Expansion Nodal Method via Solution Decomposition and Galerkin Scheme," *Nuclear Science and Engineering*, Vol.140, p.267, (2002).

[4] S.W. Woo, N.Z. Cho, "The Analytic Function Expansion Nodal Method Refined with Transverse Gradient Basis Functions and Interface Flux Moments," *Nuclear Science and Engineering*, Vol.139, p.156, (2001).

[5] "HELIOS Methods," Studsvik™ Scandpower, (2000)

[6] S. Langenbuch, W. Maurer, W. Werner et al., "Coarse mesh flux expansion method for the analysis of space time effects in large light water reactor cores," *Nuclear Science and Engineering*, Vol.63, p437, (1977).

[7] D.R. Vondy, T.B. Fowler, G.W. Cunningham, "VENTURE: A Code Block for Solving Multigroup Neutronics Problems Applying the Finite-Difference Diffusion-Theory Approximation to Neutron Transport - Version III," ORNL-5062/R1, Oak Ridge National Laboratory, (1977).

[8] "SMART 노심 설계자료 (DL2)," design information sheets from KAERI to KINS, (2010).