Depletion Calculation of Yonggwang Unit 3 Using ASTRA

Shane PARK[a], Jooil YOON[a] and Changkue LEE[a*]
[a]*Korea Nuclear Fuel, 1047 Daedukdaero, Yuseong-gu, Daejeon, Korea, 305-353*
[*]*Corresponding author: cklee@knfc.co.kr*

1. Introduction

As a part of the Development of the Reactor Core Design Codes for a Nuclear Power Plant Project, KNF is developing a three dimensional core analysis code, ASTRA (Advanced Static and Transient Reactor Analyzer). ASTRA adopts the two-node SANM (Semi-Analytic Nodal Method) as a neutron flux distribution solver, in which a set of trigonometric polynomial function is used for representing the intra-nodal flux [1].

To show the applicability of ASTRA's static burnup calculation, the depletion calculations of Yonggwang Unit 3 are compared with the NDR data and the measured data.

2. Code Descriptions and Characteristics

2.1 Functional Requirements

ASTRA is a three dimensional multigroup nodal code featured by microscopic depletion. Its capabilities are not completely developed yet. In Table I, a selective list of required functions [2] is presented and compared with the other commercial codes.

Table I: Functional Requirements of ASTRA

Functions	ASTRA	ROCS	ANC
Reactor type	OPR1000 APR1400 WH	OPR1000 APR1400	OPR1000 APR1400 WH
Nodal method	SANM	NEM	NEM
Energy group structure	Up to 7	2	2
Cross section type	Micro	Micro	Macro
3D transient simulation	O	X	X
Direct operation constant generation	O	X	X
MOX core modeling	O	△	△
Programming language	Fortran 90	Fortran 77	Fortran 77

2.2 Semi-Analytic Nodal Method

Although the recent trend in the development of the reactor analysis methods is to enhance the solution accuracy by increasing the level of modeling sophistication, there is still a need for an efficient multigroup nodal solver. This is caused by the fact that the sophisticated methods such as direct whole core transport calculation methods or simplified P_3 pin-by-pin calculation methods are not fast enough to be applied to routine design analyses and to simulations involving repeated transient calculations. Therefore, the multigroup nodal solver is practically the best choice and it can be distinguished by the methods used for solving the one-dimensional diffusion equations. The well-known and current widely-used variants are the nodal expansion method (NEM) [3] and the analytic nodal method (ANM) [4, 5, 6].

NEM, where one-dimensional fluxes are usually approximated by a 4th-order polynomial, gives very accurate results for UO_2-fueled LWRs over a wide range of reactor types, fuel loadings, and operating conditions. For MOX cores with high plutonium enrichment, however, the method is unable to describe the steep flux gradients near MOX-UO_2 assembly interfaces.

ANM solves the one-dimensional diffusion equation analytically and thus more accurately models the severe thermal flux gradients at assembly interfaces. But in ANM, there exists inherent complexity in the evaluation of energy group coupling terms. The solution algorithm of the method, however, includes an additional level of iteration involving moments of different orders.

As an effort to establish a fast, yet accurate multigroup nodal solution method which is crucial in repeated static and transient calculations for advanced reactors, the source expansion form of the SANM is applied within the framework of the coarse mesh finite difference (CMFD) formulation. The source expansion is to expand the analytic form of the source appearing in the groupwise neutron diffusion equation with a set of orthogonal polynomials in order to obtain a group decoupled analytic solution. For the acceleration, a two-level CMFD scheme is established for employing a multigroup and a one-level CMFD scheme for employing two-group CMFD [7, 8].

ASTRA employs SANM as a neutron flux solver and uses multi energy group cross section set to be able to accurately simulate the MOX loaded LWR cores and the large FBR cores as well as the UO_2-fueled cores.

3. Yonggwang Unit 3 depletion calculation

Yonggwang Unit 3 is selected to show the ASTRA's accuracy of depletion calculation. For the cycles from one through four, ASTRA's critical boron concentrations are compared with the measured data and NDR's.

Fig. 1~4 show the boron concentration change with cycle burnup for each cycle of Yonggwang Unit 3. In these figures, delta ppm means the difference between a measured concentration and each calculated value.

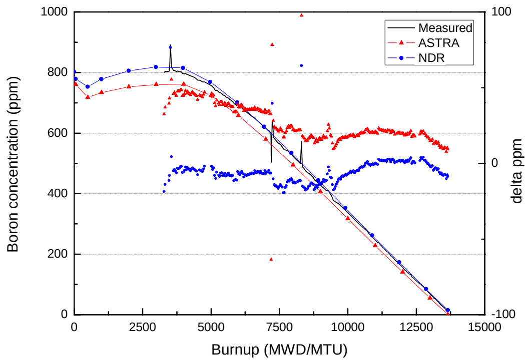

Fig. 1. Boron concentrations of Yonggwang Unit 3 cycle 1

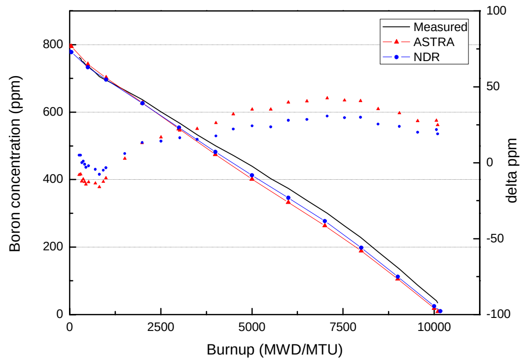

Fig. 2. Boron concentrations of Yonggwang Unit 3 cycle 2

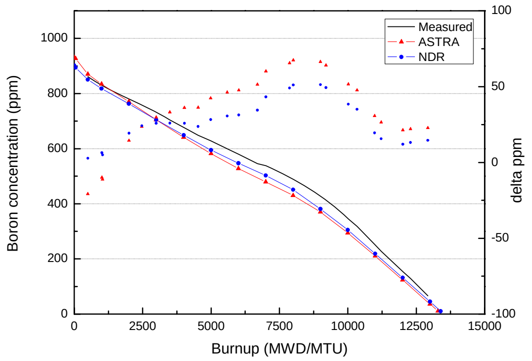

Fig. 3. Boron concentrations of Yonggwang Unit 3 cycle 3

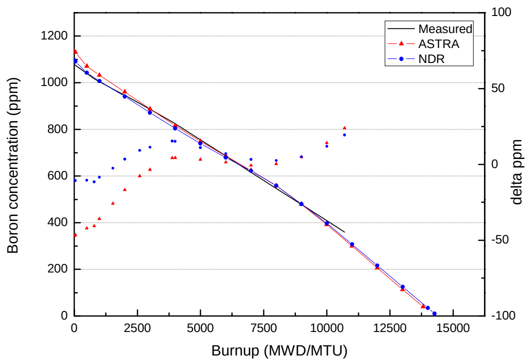

Fig. 4. Boron concentrations of Yonggwang Unit 3 cycle 4

The results of ASTRA show good agreement reasonably against those of the measured data as well as NDR's data. However, boron difference of ASTRA at BOC is relatively large, whereas NDR data shows less difference comparing with the measured data. It is suspected that the cross section set used in this calculation is not optimized to ASTRA. We expect that more accurate calculation results could be obtained after new cross section generation code for ASTRA is prepared.

4. Conclusions

ASTRA is a three dimensional nodal code with microscopic depletion capability and uses two-node SANM as a neutron flux distribution solver. Its functions of static flux distribution calculation and microscopic depletion were tested by simulating Yonggwang Unit 3 depletion of cycle 1 through 4.

Comparing the boron concentration with the measured data and NDR's, ASTRA simulates the initial and reload cores reasonably well and leaves room for improvement considering the cross section data.

ACKNOWLEGEMENTS

This work has been carried out under the project, "Development of Core Design Codes for Nuclear Power Plants", sponsored by the Ministry of Knowledge Economy.

REFERENCES

[1] P. D. Esser, K. S. Smith, "A Semianalytic Two-Group Nodal Model for SIMULATE-3," *Trans. Am. Nucl.* Soc., **68**, 220 (1993).
[2] B. J. Cho, "Code Require Document - ASTRA: Functional Requirements for 3 Dimensional Core Analysis Code," S06NX08-F-2-RD-02 Rev.0 (2007).
[3] H. Finnemann, F. Bennewitz, M. R. Wagner, "Interface Current Techniques for Multi-Dimensional Reactor Calculations," *Atomkernenergie*, **30**, 123 (1977).
[4] H. D. Fischer, H. Finnemann, "The Nodal Integration Method-a Diverse Solver for Neutron Diffusion Problems," *Atomkernenergie* **39**, 229 (1981).
[5] K. S. Smith, "An Analytic Nodal Method for Solving the Two-Group, Multidimensional, Static and Transient Neutron Diffusion Equations," nuclear engineering thesis, Massachusetts Institute of Technology, Cambridge, MA. (1979).
[6] D. L. Vogel, "An Alternative Theoretical and Numerical Approach to the Analytic Nodal Method," Topical Meeting on Advances in Mathematics, Computations and Reactor Physics, 28 April~2 May, Pittsburgh, Pa. (1991).
[7] Y. I. Kim, Y. J. Kim, S. J. Kim, and T. K. Kim, "A Semi-Analytic Multigroup Nodal Method," *Ann. Nucl. Energy*, **26** (1999).
[8] J. I. Yoon and H. G. Joo, "Two-Level Coarse Mesh Finite Difference Formulation with Multigroup Source Expansion Nodal Kernels," *J. Nucl. Sci. Technol.*, **45**, 668-682 (2008).