

MICROSCOPIC DEPLETION MODEL IN SIMULATE-4

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INTRODUCTION

Studsvik's advanced nodal code SIMULATE-3 [1] has been in use for LWR reactor analysis for nearly 20 years. Due to limitations on memory and concerns on run time when it was developed, the macroscopic cross section model was preferred over the microscopic depletion model. The macroscopic cross-section and depletion model employs history (i.e., exposure) weighted state parameters along with tabularized two-group macroscopic cross sections. Dependencies on the history variables are determined by performing separate single assembly CASMO-4 [2] depletion calculations for each of the history effects. The model has been proven to be accurate for typical LWR applications. The shortcomings of the model, such as modeling the shutdown cooling effects, have been resolved with ad-hoc models.

With today's modern desktop computers, memory and run time are no longer concerns. Also, today's more aggressive core designs with extended cycle lengths and operation strategies require more comprehensive reactor physics tools. As the industry leader in core analysis software, Studsvik has been working on developing the next generation nodal code, SIMULATE-4. As part of this development, the existing macroscopic depletion model has been replaced with a microscopic depletion model.

DESCRIPTION OF THE ACTUAL WORK

Ideally, the macroscopic cross section can be constructed using the isotopic number densities and microscopic cross sections:

$$\Sigma^{actual} = \sum_i \sigma_i(E, \rho_h, \rho, \dots) N_i^{actual} \quad (1)$$

However, even with today's computer resources, this type of calculation is not practical in accounting for all of the microscopic cross sections for every isotope (approximately two hundred to achieve the desired accuracy) that can be encountered during the fuel depletion of the entire core. Accordingly, a hybrid macro/micro model is employed in SIMULATE-4 in which the node-average cross sections for the given actual condition are estimated as:

$$\Sigma^{actual} = \Sigma^{SA}(E, \rho_h, \rho, \dots) + \sum_i \sigma_i(E, \rho_h, \rho, \dots) (N_i^{actual} - N_i^{SA}(E, \rho_h, \dots)) \quad (2)$$

where Σ represents Σ_a , Σ_f , and $\nu\Sigma_f$, and where the summation is over selected nuclides. The second term in Equation 2 provides a correction to the single-assembly macroscopic cross sections (Σ^{SA}) to make up for the fact that real life operation (represented by number density N^{actual}) differs from the idealized condition of the single assembly CASMO-4 evaluation (represented by N^{SA}). Note, that as long as real life operation does not differ from CASMO-4 conditions, the cross sections of SIMULATE-4 are as good as those of CASMO.

Approximately 50 isotopes (15 actinides, 30 fission products, Gd/B10 as burnable absorber) have been chosen according to their importance on reactivity during normal operation, depletion, transients and shutdowns (with possible long outages).

Both the macroscopic and microscopic multi-group cross sections appearing in Equation 2 are functionalized in two- and three-dimensional tables versus all important instantaneous effects (e.g., coolant density, control rod, fuel temperature, etc.) and historical effects (e.g., burnup, coolant density history, and control rod history).

The reference single-assembly number densities are functions of burnup and historical effects. The N^{SA} are not taken directly from CASMO-4 but calculated in the linking code by solving the SIMULATE-4 isotope chains using the single-assembly fluxes and microscopic cross sections. Hence, N^{SA} and N^{actual} are computed in consistent manners.

Due to its large spatial self shielding, the Gd isotopes require a special treatment. The chain of gadolinium isotopes has been replaced with an effective Gd. The microscopic absorber cross section of the Gd-eff has been functionalized versus the number density of effective Gd isotope.

The actual number densities appearing in Equation 2 are tracked for each node using the node-average fluxes available from the multi-group nodal solver. The burn-up chain calculations in SIMULATE-4 are carried out in two-steps, as predictor and corrector. In each depletion step as well as for each shutdown cooling period, linearized chain equations are solved.

RESULTS

The microscopic depletion model in SIMULATE-4 has been verified against single assembly CASMO-4 calculations and the micro model has been compared to traditional macroscopic depletion model.

Fig. 1 presents the eigenvalue error for a typical BWR assembly with Gd during an unrodded-rodded-unrodded depletion sequence. The deficiency of the macroscopic depletion model following the pulling of the control rod after 20 GWD/Mt has been significantly improved with the micro depletion model.

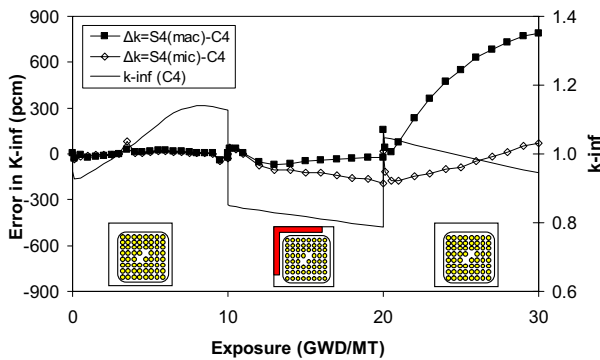


Fig. 1. A BWR assembly depletion: unrodded-rodded-unrodded

Fig. 2 illustrates a case with power varying over time (with relative powers of 1.0-0.5-2.0-0.5 and 1.0). Nuclide concentrations, especially those with short half-lives, are sensitive to the power density history. CASMO-4 runs assume a constant power level and, hence, Σ^{S4} does not catch this history effect. With no micro model correction, the maximum error is 460 pcm. This error is reduced to 110 pcm with the micro model.

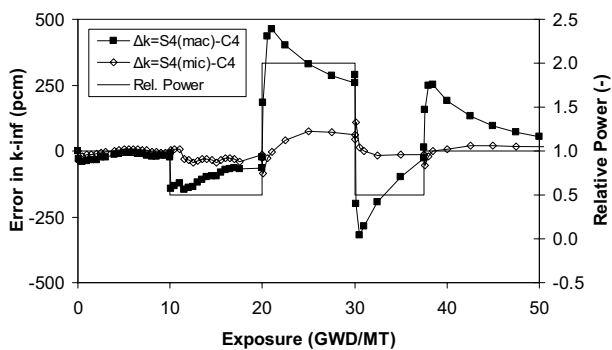


Fig. 2. A BWR assembly depletion – 5 step test case

CONCLUSION

A microscopic depletion model has been developed as part of the next generation nodal code SIMULATE-4. Single-assembly test problems have demonstrated significant improvements over the traditional macroscopic depletion model. Extension of testing for real core follow calculations is underway.

REFERENCES

1. K.S.SMITH et al., "SIMULATE-3 Methodology," Studsvik/SOA-95/18 (1995).
2. M. EDENIUS, et al., "CASMO-4, A Fuel Assembly Burnup Program, User's Manual," Studsvik/SOA-95/1 (1995).