TRANSACTIONS

OF THE
AMERICAN NUCLEAR SOCIETY

June 24–28, 2007
Boston Marriott Copley Place
Boston, Massachusetts

Volume 96
TANSAO 96 1–882 (2007)
ISSN: 0003-018X

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MCNP-5/ORIGEN-2.2/MCODE-2.2 versus CASMO-5 Depletion for A Heavily Gd-Poisoned BWR Fuel Assembly

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INTRODUCTION

Depletion analysis based on a Monte Carlo transport solution is attracting increased interest in recent years [1]. A heavily gadolinium-poisoned BWR fuel assembly is used here as a challenging problem to explore the coupled MCNP-5/ORIGEN-2.2 [2,3] (via utility program, MCODE-2.2 [4]) depletion analysis. A series of MCODE sensitivity runs are performed to investigate the effects of neutron histories, depletion step size, burnup corrector calculation, and a second end-of-step MCNP flux calculation. Eigenvalues and pin fission rates are compared between the converged MCODE solution and the deterministic lattice physics code, CASMO-5, result.

DESCRIPTION OF PROBLEM

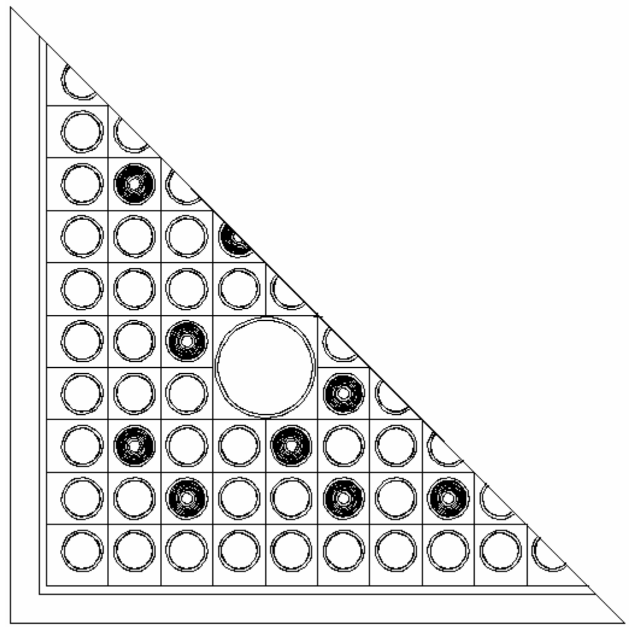
A 10×10-lattice BWR fuel assembly was chosen for this study. This design has 4 enrichment zones (2.4, 3.6, 4.4, 4.9 w/o) and 17 Gd-bearing (7.0 wt% Gd_2O_3 and 4.9 w/o UO_2) fuel rods. The heavy Gd loading, which holds down more than 30% of initial reactivity, makes this problem very challenging. The 2-D geometry of the BWR assembly is shown in Fig. 1. Note that the CASMO plot distinguishes fuel types with different colors. Mirror boundary conditions are used on all sides.

Hot, 0% void is assumed for the depletion calculations. For simplicity, the fuel temperature is 900 K, and all the other components, such as water, fuel cladding and box wall, are at 600 K.

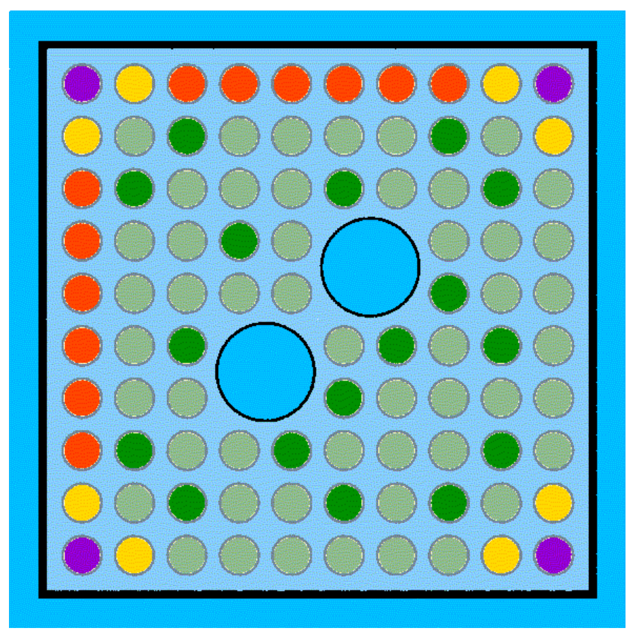
MCODE-2.2 DEPLETION CALCULATIONS

The MCNP continuous-energy cross sections are processed from ENDF/B-VI.8 data with NJOY99 [5]. Likewise, the CASMO-5 586-group library is based on ENDF/B-VI data (although difference release versions).

The MCODE-2.2 model input is an extension to a regular MCNP input by appending a fourth paragraph as depletion-related input. The first three paragraphs simply consist of a BOL MCNP model. A lower left-half BWR assembly is modeled (Fig. 1). For depletion analysis, each gadolinium-bearing fuel pellet is subdivided into 10 equal-volume regions to account for Gd self-shielding (but no azimuthal subdivision). The total number of depletion materials is 132 (same as CASMO-5).



(a) MCNP geometry model



(b) CASMO materials plot

Fig. 1. Two-dimensional geometry of the BWR assembly.

One unique characteristic of Monte Carlo-based depletion is the inherent randomness of the solution. Two cases of neutron histories were studied:

- 1 million (4000 neutrons/cycle ∗ 250 cycles)
- 4 million (10000 neutrons/cycle ∗ 400 cycles).

Typical reported standard statistical errors for the 1-million case are: 60 pcm of eigenvalue, 0.6% of pin fission rate, and 0.3% of neutron flux in a fuel rod.

For this heavily Gd-poisoned case, several other factors need to be considered. The first factor is the depletion step size. In the past, CASMO used fine sub-step depletion (0.125 MWd/kg) for Gd rods, which is nearly prohibitive for a typical MCNP depletion due to too many depletion points. Three time step sizes are examined here: 0.125, 0.25 and 0.5 MWd/kg. The second factor is the burnup corrector calculation. A second depletion calculation can be performed using the spectra at the end of the time step. Average nuclide number densities are taken as the final end-of-time-step values. This improves burnup depletion accuracy with a negligible computational cost. Hence, it is adopted in most modern lattice physics codes. The third factor is the second end-of-step MCNP flux calculation (also called the C-flux calculation). It is usually deemed as unnecessary since neutronic solutions are usually very close between the predictor number densities and the averaged number densities. However, for gadolinium depletion problems, the C-flux calculation might be needed because the predictor neutronic solution tends to slightly underestimate Gd rod power. The MCODE options of burnup corrector and C-flux calculation are performed in this study.

RESULTS

Eigenvalue results are plotted in Fig. 2. Five cases of MCODE runs are shown using different options. The most detailed case is the one with 4 million histories, a time step of 0.125 MWd/kg, burnup corrector and 2 MCNP calculations per time step, which takes about 40 days on a modern AMD 4800+ dual-core CPU. It agrees well with the CASMO-5 solution, i.e., within 250 pcm. As the gadolinium burns out after 22.5 MWd/kg, the agreement is well within statistics.

The neutron histories affect the magnitude of statistical variations. The 4-million case has a smoother line compared to other cases. In addition, the propagated depletion error in material number densities is also reduced when more histories used.

Fig. 2 also shows that the burnup corrector option is most important among all the sensitivities being investigated here. Turning off the burnup corrector option leads to an underestimation of eigenvalue of ~1300 pcm even with a fine time step size of 0.25 MWd/kg. The next important factor is the option of a second MCNP calculation. Omitting it can cause a sizable eigenvalue underestimation of ~400 pcm. Note that these differences occur during the gadolinium depletion. For normal poison-free UO_2 cases, both effects are expected to be much smaller. The time step size (between 0.5, 0.25, and 0.125 MWd/kg) has a small effect on eigenvalue (<200 pcm). And, the 0.25 MWd/kg is recommended for gadolinium depletion.

Next, the MCNP depletion case (1M, 0.25 MWd/kg with burnup corrector and C-flux calculation options) is taken as the reference case for a fission rate comparison. Fig. 3 shows the CASMO-5 vs. MCNP-5 fission rate distribution comparison at 0.1, 20, 40 MWd/kg. Note that the MCNP statistical pin fission root-mean-square (RMS) is 0.006, which is comparable to the CASMO-5 vs. MCNP-5 fission rate difference. Therefore, it can be concluded that the CASMO-5 fission rate distribution agrees with MCNP-5 within statistical deviations.

CONCLUSIONS

Sensitivities of the coupled MCNP-5/ORIGEN-2.2 depletion are investigated showing that the burnup corrector and the second MCNP calculation are important for LWR gadolinium depletion analysis. Furthermore, although there are numerous differences between CASMO and MCNP, it is demonstrated here that given appropriate options of calculations (e.g., consistent set of cross sections, burnup corrector option, etc.) reasonably satisfactory agreement can be achieved between the two codes.

REFERENCES

1. J. C. DAVIS, J. C. LEE, "Comparison of Monte Carlo and Deterministic Depletion Codes for LWR Fuel Cycle Analysis," *Trans. Am. Nucl. Soc.*, **92**, 651 (June 2005).
2. "MCNP — A General Monte Carlo N-Particle Transport Code, Version 5," LA-UR-03-1987, X-5 Monte Carlo Team, Los Alamos National Laboratory (April 2003).
3. A. G. CROFF, "A User's Manual for the ORIGEN2 Computer Code," ORNL/TM-7175, Oak Ridge National Laboratory (1980).
4. Z. XU, P. HEJZLAR, and M. S. KAZIMI, "MCODE, Version 2.2 — An MCNP-ORIGEN Depletion Program," Center for Advanced Nuclear Energy Systems, MIT (April 2006).
5. N. GHEORGHIU, Personal Communication, Studsvik Scandpower Inc. (August 2006).

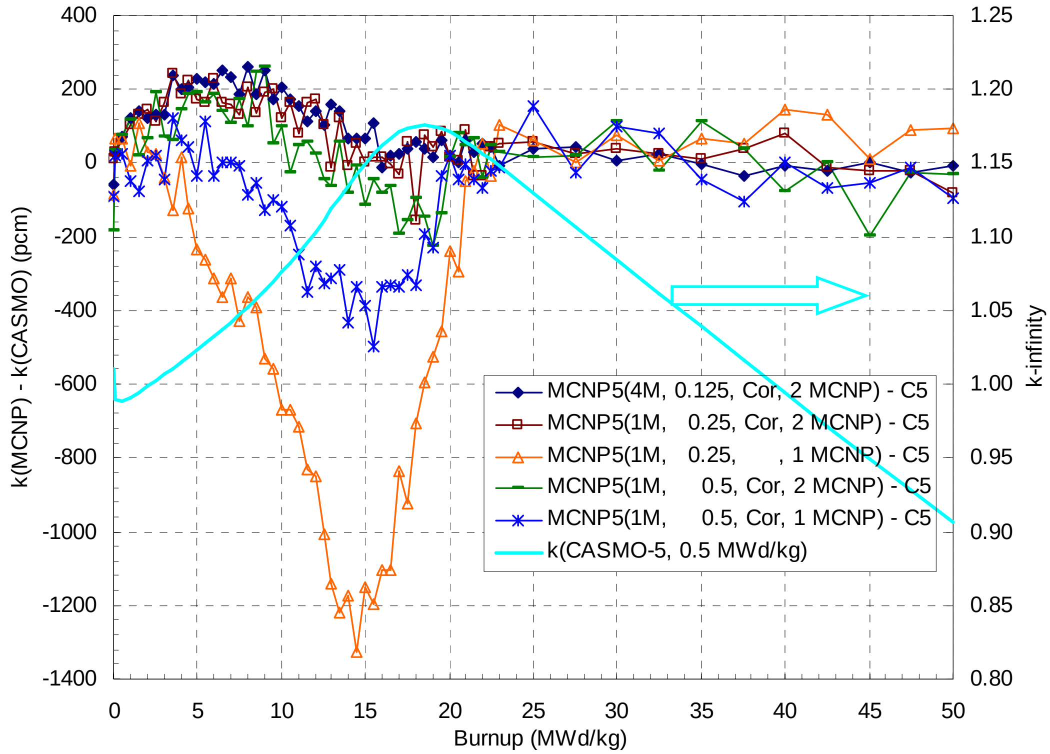


Fig. 2. Eigenvalue results as a function of burnup.

(CASMO - MCNP), 0.1 MWd/kg, RMS=0.008
(CASMO - MCNP), 20 MWd/kg, RMS=0.007
(CASMO - MCNP), 40 MWd/kg, RMS=0.005

-0.013 +0.005 -0.008									
-0.001 -0.002 -0.013	+0.000 +0.011 -0.005								
-0.011 -0.005 -0.005	-0.003 +0.001 +0.001	+0.003 -0.001 +0.008							
-0.007 +0.004 +0.001	-0.013 -0.006 -0.006	+0.003 +0.001 +0.008	+0.003 +0.018 +0.006						
-0.005 +0.001 +0.006	-0.005 +0.001 -0.003	+0.006 +0.012 +0.003	-0.004 -0.008 +0.003	+0.015 +0.009 -0.002					
+0.002 -0.003 -0.003	+0.003 +0.002 +0.000	+0.000 +0.002 +0.003	WATER ROD		+0.035 +0.009 -0.004				
-0.003 -0.001 +0.004	+0.010 +0.007 +0.000	+0.003 +0.014 +0.001			+0.003 +0.000 -0.001	-0.003 +0.004 +0.007			
-0.013 +0.002 +0.000	-0.003 +0.004 +0.009	+0.005 +0.005 +0.004	-0.005 +0.001 -0.007	-0.002 +0.005 +0.005	+0.005 +0.007 +0.002	+0.008 +0.002 +0.004	+0.002 -0.005 +0.000		
-0.013 -0.006 -0.009	+0.013 -0.008 -0.004	+0.000 -0.009 +0.004	-0.011 -0.007 +0.007	+0.004 +0.003 +0.006	-0.002 +0.001 +0.007	+0.008 -0.001 +0.007	-0.001 +0.000 -0.004	+0.003 -0.009 -0.003	
-0.002 -0.013 -0.008	-0.006 -0.009 -0.004	-0.005 -0.009 +0.004	+0.003 -0.001 -0.007	+0.009 +0.002 -0.005	+0.007 +0.009 -0.005	+0.001 +0.006 +0.001	-0.001 +0.004 +0.001	-0.008 -0.021 -0.002	+0.007 -0.011 -0.014

Fig. 3. Pin fission rate distribution comparison.