A MONTE CARLO BURNUP CODE LINKING MCNP AND REBUS*

N. A. Hanan, A. P. Olson, R. B. Pond, and J. E. Matos

Argonne National Laboratory
Argonne, Illinois 60439-4841 USA

Presented at the 1998 International Meeting
on Reduced Enrichment for Research and Test Reactors

October 18 - 23, 1998
São Paulo, Brazil

*Work supported by the U. S. Department of Energy
Office of Nonproliferation and National Security
under Contract No. W-31-109-ENG-38.

A MONTE CARLO BURNUP CODE LINKING MCNP AND REBUS

N. A. Hanan, A. P. Olson, R. B. Pond, and J. E. Matos

Argonne National Laboratory
9700 S. Cass Avenue
Argonne, Illinois 60439-4841 USA

ABSTRACT

The REBUS-3[1] burnup code, used in the ANL RERTR Program, is a very general code that uses diffusion theory (DIF3D[2]) to obtain the fluxes required for reactor burnup analyses. Diffusion theory works well for most reactors. However, to include the effects of exact geometry and strong absorbers that are difficult to model using diffusion theory, a Monte Carlo method is required. MCNP[3], "a general-purpose, generalized-geometry, time-dependent, Monte Carlo transport code, is the most widely used Monte Carlo code. This paper presents a linking of the MCNP code and the REBUS burnup code to perform these difficult burnup analyses. The linked code will permit the use of the full capabilities of REBUS which include non-equilibrium and equilibrium burnup analyses. Results of burnup analyses using this new linked code are also presented.

INTRODUCTION

In the last few year, interest in burnup calculations using Monte Carlo methods has increased. Previous burnup codes have used diffusion theory for the neutronic portion of the codes. Diffusion theory works well for most reactors. However, diffusion theory does not produce accurate results in burnup problems that include strong absorbers or large voids. Also, diffusion theory codes are geometry-limited (rectangular, hexagonal, cylindrical, and spherical coordinates). Monte Carlo methods are ideal to solve very heterogeneous reactors and/or lattices/assemblies in which considerable burnable poisons are used. The key feature of this approach is that Monte Carlo methods permit essentially "exact" modeling of all geometrical detail, without resort to energy and spatial homogenization of neutron cross sections. Several codes, or combinations of codes have been developed[4-7] to perform Monte Carlo depletion analysis. Basically, these codes were developed with the idea of solving the burnup problem for unit cells and/or fuel assemblies/lattices. This paper presents the coupling of the REBUS-3 burnup code with the MCNP Monte Carlo code. This new linked code, called MC-REBUS, is intended for entire reactor cores as well as for unit cells and assemblies/lattices. Figure 1 gives an overview of the code structure.

Figure 1. MC-REBUS Overall Structure

REBUS side

- Input processor (geometry)
- Flux normalization (to a fixed power)
- Fuel cycle computational processor (fuel cycle details; fuel shuffling)
- Output processor

Interface between REBUS and MCNP

- Preparation of file containing burnup-dependent compositions for all burnable regions (performed inside REBUS code)
- Automatic input preparation for MCNP using burnup-dependent compositions from REBUS (REBUS post-processor)

Interface between MCNP and REBUS

- Preparation of ISOTXS-type file containing reaction cross sections for each burnable region (MCNP post-processor)
- Preparation of file containing region-wise fluxes for use in REBUS (MCNP post-processor)
- New routines in REBUS to use MCNP-generated cross sections file to determine the burnup matrix elements at each time node.

MCNP side

- The MCNP code, as a subroutine (with no changes)

CODE DESCRIPTIONS

REBUS

Reactor burnup calculations in the RERTR Program at Argonne National Laboratory (ANL) rely upon the REBUS code system, which presently uses DIF3D to perform both nodal and finite-difference neutronics solutions in rectangular, hexagonal, cylindrical, and spherical coordinate systems. The burnup and fuel shuffling aspects of REBUS are independent of the code(s) used to describe the reactor geometrical model and independent of the code used to obtain neutron flux solutions. This property permits REBUS to be used, with modifications, together with any other code that can provide the neutronic solution (fluxes) to REBUS.

The REBUS system solves either one of two classes of problems: the so-called equilibrium problem and the non-equilibrium problem. The equilibrium problem is defined as that solution which exists after infinite repetitions of a fixed fuel cycle strategy. The non-equilibrium problem is the cycle-by-cycle solution which ultimately may tend to the equilibrium solution, if its fuel cycle strategy is the same as that of the equilibrium problem. The equilibrium solution is helpful for design solutions while the non-equilibrium solution yields the actual reactor performance in search of that ultimate objective.

The reactor fuel cycle also includes the external fuel cycle. User-specified fuel shuffling capabilities can simulate real world situations. Discharged fuel can be re-introduced into the reactor at any subsequent cycle. Or it can be reprocessed and refabricated into fresh fuel. The

code has the capability to adjust the fuel enrichment, the burn time, and the control poison requirements in order to satisfy user-specified constraints on desired k-eff, and on desired average discharge burnup. Equilibrium solutions may search to a desired k-eff at beginning or end of the equilibrium cycle.

As for the representation of the fission products, REBUS can use all fission products explicitly, or it can use some fission products explicitly combined with one or more lumped fission products.

MCNP

MCNP is "a general-purpose Monte Carlo N-Particle code that can be used for neutron, photon, electron, or coupled neutron/photon/electron transport, including the capability to calculate eigenvalues for critical systems. The code treats an arbitrary three-dimensional configuration of materials in geometric cells bounded by first- and second-degree surfaces and fourth-degree elliptical tori."[3]

MC-REBUS

Essentially, REBUS was "wrapped around" the MCNP code. REBUS-3 was modified to accommodate MCNP as a subroutine. That is, any and all conflicts such as identically-named subroutines or identically-labeled common blocks were resolved by changing those in REBUS and keeping MCNP intact.

Communication between REBUS and MCNP is provided mainly by two post-processors (see Figure 1):

(a) MCNP Post-processor: It gets the MCNP tallies (fluxes and reaction rates) and prepares one-group zone-averaged fluxes and cross sections to be used by REBUS. These data are prepared following the Standard Interface Files Format (RZFLUX and ISOTXS). Note that only the isotopic transmutation cross sections such as (n, γ), (n, α), (n, p), (n, t), (n, d), (n, 2n), (n, f), and ν are required for burnup analyses.

(b) REBUS Post-processor: It gets the REBUS-provided burnup dependent compositions and writes the MCNP input for the next MCNP calculation without user interaction.

It was also necessary to create new logic (i.e. several new subroutines, called by the REBUS computational module) to retrieve the MCNP-generated reaction cross sections and update the matrix elements used to determine burnup and isotopic transmutation. It is important to note that these new subroutines do not add any input requirement. The post-processors are included as subroutines inside REBUS and require only a very small amount of input.

When the capabilities of MCNP are added to REBUS, either zone-averaged fluxes or both zone-averaged fluxes and zone-averaged reaction cross sections can be derived from MCNP for burnup calculations in REBUS. By combining the capabilities of REBUS for fuel cycle manipulations, independent of the source of the zone-averaged fluxes and reaction process cross sections, while obtaining those properties from MCNP, an extremely well-founded code for benchmark applications results.

For both cases, MCNP fluxes only, or MCNP fluxes and cross sections, the MCNP input remains basically unchanged; only the location of the burnable zones (cells in MCNP nomenclature) need to be positioned as the last cells. As for the REBUS input, a modification similar to that for MCNP is required for the case in which only MCNP fluxes are to be used. For the case in which both fluxes and cross sections are to be generated in MCNP, the REBUS geometry definition must be changed. The fuel zones need to be defined and their volumes should be preserved. Also, material compositions need to be defined explicitly for each burnable zone since each zone will use different cross sections generated by MCNP.

STATUS OF THE MC-REBUS CODE

Presently, MC-REBUS is operable for two classes of problems: a) Any depletion analysis for which only the fluxes from MCNP are required, and b) Reactivity rundown problems using MCNP-generated fluxes and cross-sections. Results for these types of problems are presented in the next section.

Work is in progress to complete modifications in REBUS to permit burnup analysis with any shuffling pattern (non-equilibrium problems; e.g. approach to equilibrium with rundown and shuffling performed automatically inside REBUS). Fuel replacement and shuffling analyses can be performed now if the user performs the input modifications required in REBUS after each cycle. This is not considered a good way to operate the code. In the near future the automated operating mode will be available. The same is true for equilibrium problems.

As in any code using Monte Carlo methods, uncertainty is a major factor in the results. Propagation of errors is not presently being addressed in the MC-REBUS. Based on the published information on the other Monte Carlo burnup codes,[4-7] propagation of errors is also not addressed.

ADVANTAGES AND DISADVANTAGES OF MC-REBUS

The linking of the MCNP code with the REBUS code for Monte Carlo depletion analysis of unit cells, fuel assemblies/lattices, and reactor cores presents the following advantages:

a) MCNP is the most widely used Monte Carlo code for reactor analysis.
b) REBUS is one of the most versatile burnup codes available.
c) In all MC depletion codes available, all fission products need to be modeled explicitly. In MC-REBUS, the representation based on lumped fission product can also be used[8].
d) The use of REBUS allows for a whole reactor depletion analysis, including shuffling of fuel.

The major disadvantage of this linked code is that MCNP is a slow code when compared to other Monte Carlo neutronics codes (e.g. VIM[9]). Also, the generation of reaction rates required for the burnup analysis slows MCNP dramatically.

RESULTS

The two modes of the MC-REBUS code that are currently operational were exercised for a reactivity rundown calculation. The Brookhaven Medical Research Reactor (BMRR) was used for this example because a diffusion theory burnup analysis[8] using REBUS/DIF3D was already available. The BMRR is a 3 MW research reactor that is cooled and moderated with light water and reflected with graphite. The present BMRR core uses 28 HEU (93% enriched) MTR-type fuel assemblies. However, in this paper, a potential alternative core containing 17 MTR-type fuel assemblies with LEU (19.75% enrichment) is used. Each fuel assembly contains 18 fuel plates. For this paper, a reactivity rundown for 15 full-power days was performed; this is equivalent to about one year of operation for the BMRR.

Figure 2 shows the reactivity rundown results obtained using three methods: a) REBUS/DIF3D code; b) MC-REBUS using only fluxes from MCNP; and c) MC-REBUS using both fluxes and cross sections generated by MCNP. For both MC-REBUS cases, only the mean reactivity values are shown because as discussed above propagation of errors (uncertainties) is not addressed in this paper.

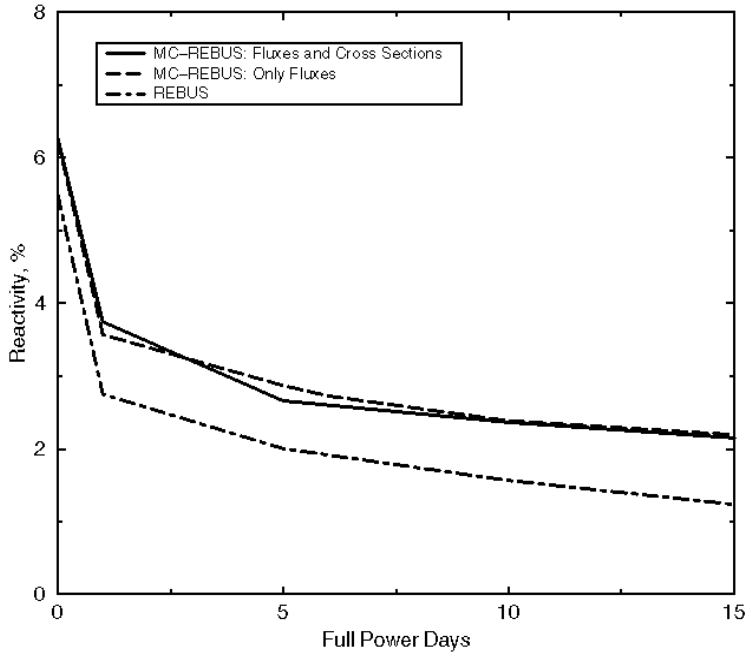

Figure 2. Reactivity Rundown for the BMRR: Three Different Methods

Results for the average burnup in the seventeen fuel assemblies are shown in Table 2. These results show that for this particular application there are only very small differences among the three methods. As expected, diffusion theory performs well for this reactor.

Table 1. Average Burnup (MWD/MT) for All 17 Fuel Assemblies: Three Different Methods

REBUS/DIF3D	MC-REBUS: ONLY FLUXES	MC-REBUS: FLUXES AND CROSS SECTIONS
2.73	2.78	2.77
3.14	3.17	3.20
3.26	3.29	3.26
3.04	3.12	3.06
2.88	2.91	2.92
3.92	3.78	3.79
4.14	4.08	4.07
3.34	3.32	3.24
2.36	2.45	2.39
3.29	3.28	3.27
4.00	3.85	3.81
3.96	3.91	3.84
3.02	3.04	2.98
2.73	2.77	2.75
3.16	3.18	3.14
3.29	3.32	3.26
2.92	2.95	2.88

In the future, validation of the code against experimental results will be performed.

CONCLUSIONS

Interest in burnup calculations using Monte Carlo methods has increased in the last few years. MC-REBUS, a code linking MCNP, the most widely used Monte Carlo code, with REBUS-3, one of the most versatile burnup codes, was developed. This code can operate in two different modes: a) REBUS uses only the MCNP-generated fluxes, or b) REBUS uses fluxes and cross sections generated by MCNP. Results for one whole core analysis, future plans for using the full capabilities of REBUS for fuel cycle analysis, and the advantages and disadvantages of the linked code were also discussed.

REFERENCES

1. B. J. Toppel, "A User's Guide for the REBUS-3 Fuel-Cycle-Analysis Capability," ANL-83-2, Argonne National Laboratory (1983).

2. K. L. Derstine, "*DIF3D, A Code to Solve One-, Two-, and Three-Dimensional Finite-Difference Diffusion Theory Problems"*, ANL-82-64, Argonne National Laboratory (April 1984).

3. J. F. Breismeister, ed., "MCNP - A General Monte Carlo N-Particle Transport Code, Version 4B," LA-12625-M. Los Alamos National Laboratory, (1997).

4. D. J. Kelly, "Depletion of a BWR Lattice Using the RACER Continuous Energy Monte Carlo Code," Proc. Int. Conf. Mathematics and Computations, Reactor Physics, and Environmental Analyses, Portland, Oregon, April 30-May 4, 1995, Vol. 2, p. 1011, American nuclear Society (1995).

5. R. L. Moore, et. al., "MOCUP: MCNP-ORIGEN2 Coupled Utility Program," INEL-95/0523, Idaho National Engineering Laboratory, (September 1995).

6. G. S. Chang, "Monte Carlo Depletion Analysis of UO_2 and MOX Fuels in a Checkerboard Assembly," Trans. Am. Nucl. Soc., **78**, 229 (1998).

7. G. S. Chang, "Monte Carlo Analysis of Burnup-Dependent Plutonium Concentration Profiles in UO_2 and MOX Fuel Pins," Trans. Am. Nucl. Soc., **78**, 246 (1998).

8. N. A. Hanan, R. B. Pond, J. E. Matos, and J. P. Hu, "A Neutronic Feasibility Study for LEU Conversion of the Brookhaven Medical Research Reactor (BMRR)," Proceedings of the XX International Meeting on Reduced Enrichment for Research and Test Reactors, 5-10 October 1997. Jackson Hole, Wyoming, USA.

9. N. A. Hanan, R. B. Pond, W. L. Woodruff, M. M. Bretscher, and J. E. Matos, "The Use of WIMS-ANL Lumped Fission Product Cross Sections for Burned Core Analysis with the MCNP Monte Carlo Code," (these proceedings).

10. R. N. Blomquist, "VIM Continuous Energy Monte Carlo Transport Code," Proc. Int. Conf. Mathematics and Computations, Reactor Physics, and Environmental Analyses, Portland, Oregon, April 30-May 4, 1995, Vol. 2, p. 1133, American nuclear Society (1995).