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MONTEBURNS**

by

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Development of the Fully Automated Monte Carlo Burnup Code *Monteburns*

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Introduction

Several computer codes have been developed to perform nuclear burnup calculations over the past few decades.^{1,2} In addition, because of advances in computer technology, it recently has become more desirable to use Monte Carlo techniques for such problems. Monte Carlo techniques generally offer two distinct advantages over discrete ordinate methods: (1) the use of continuous energy cross sections and (2) the ability to model detailed, complex, three-dimensional (3-D) geometries. These advantages allow more accurate burnup results to be obtained, provided that the user possesses the required computing power (which is required for discrete ordinate methods as well).

Several linkage codes have been written that combine a Monte Carlo N-Particle transport code (such as MCNPTM)³ with a radioactive decay and burnup code.^{2,4} This paper describes one such code that was written at Los Alamos National Laboratory: *monteburns*.⁵ *Monteburns* links MCNP with the isotope generation and depletion code ORIGEN2.⁶ The basis for the development of *monteburns* was the need for a fully automated code that could perform accurate burnup (and other) calculations for any 3-D system (accelerator-driven or a full reactor core). Before the initial development of *monteburns*, a list of desired attributes was made and is given below.

- The code should be fully automated (that is, after the input is set up, no further user interaction is required).
- The code should allow for the irradiation of several materials concurrently (each material is evaluated collectively in MCNP and burned separately in ORIGEN2).
- The code should allow the transfer of materials (shuffling) between regions in MCNP.
- The code should allow any materials to be added or removed before, during, or after each step in an automated fashion.
- The code should not require the user to provide input for ORIGEN2 and should have minimal MCNP input file requirements (other than a working MCNP deck).
- The code should be relatively easy to use and not require several complicated input files.

the input system power. First, the recoverable energy per fission (Q_{fis}) is determined. The value of Q_{fis} can be input explicitly by the user, or it can be calculated according to the fission fraction of each actinide in the system. This allows Q_{fis} to vary as the isotopics of the system change; however, *monteburns* does not take into account the variance in capture gamma heating as the system changes. When Q_{fis} is determined, *monteburns* normalizes the flux in each region to correspond to the total system power. This is done differently depending on whether MCNP is run in criticality (KCODE) or source (SDEF) mode. After fluxes and one-group cross sections are calculated, *monteburns* modifies the ORIGEN2 input file and libraries to reflect these values. *Monteburns* also takes the necessary steps to add/remove materials before, during, and/or after the ORIGEN2 burn. Following the irradiation step in ORIGEN2, material isotopics are passed back to MCNP, and the process is repeated as necessary.

Conclusion

Monteburns has proven to be a very useful and versatile burnup code for both reactor and accelerator-driven systems. Initial benchmarking efforts have been successful, although additional work is still required. Several enhancements to *monteburns* currently are being considered, including expanding the number of platforms on which the code operates, modifying the burnup/depletion and fission product yield calculations currently performed by ORIGEN2, improving the calculation of Q_{fis} (particularly with respect to capture gamma heating), and expanding the output (material activities, neutrons/gamma sources, etc.).

References

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