INVESTIGATION OF REHOMOGENIZATION IN THE FRAMEWORK OF NODAL CROSS SECTION CORRECTIONS

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ABSTRACT

Few-group cross sections used in nodal calculations derive from standard energy collapsing and spatial homogenization performed during preliminary lattice transport calculations, that implicitly assume an infinite array of identical fuel-assemblies. The infinite-medium neutron flux used for cross section weighting does not account for environmental effects arising in case of heterogeneous configurations, which can lead to considerable leakages out of or into the assembly and thus invalidate the reflective boundary conditions used for the lattice simulation. Core-environment effects can also cause variations, with respect to the infinite-lattice calculation, in the reference cross section distribution used for few-group constant collapsing. These sources of inaccuracy prevent from reproducing with high fidelity the best estimate of the reaction rates and multiplication factor coming from the reference transport global solution. Rehomogenization techniques are therefore needed. The purpose of the present paper, which builds upon previous work done at AREVA in the area of rehomogenization, is to formalize a mathematical model that encompasses the different kinds of homogenization errors. In order to investigate the accuracy of the corresponding cross section corrections, numerical tests of an assembly-configuration sample are presented.

Key Words: Nodal diffusion; Spatial and Spectral Rehomogenization; Cross section correction; Burn-up.

1. INTRODUCTION

Current deterministic methodologies for nuclear reactor core analysis make wide use of nodal diffusion approaches [1]. Cross sections used in nodal codes for 3-D core simulations are generally obtained by energy collapsing and spatial homogenization from single-assembly calculations with reflective boundary conditions (also referred to as infinite-medium conditions in the following) [2]. Their preparation involves a set of reference cross sections in a very fine number of energy groups and a reference condensation flux determined by neutron transport in the most detailed geometry. If the assembly is far away from the reflector and surrounded by assemblies of the same type in a large medium compared to the neutron mean free path, this assumption is clearly acceptable. However, when the assembly is in its real environment (i.e., the reactor core), the neutron flux distribution can differ significantly from the one calculated in the infinite medium due to different boundary conditions. Common examples are mixed MOX/UO2 fuel loading patterns, heterogeneous configurations with depletable strong local absorbers and controlled regions [3, 4]. Environmental conditions should be modeled with these layouts to reproduce more accurate nodal cross sections, still using the homogenization paradigm. Smith first

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