A key challenge for full-core transport methods is reactor agnostic multi-group cross section (MGXS) generation. Monte Carlo (MC) presents the most accurate method for MGXS generation since it does not require any approximations to the neutron flux. This paper introduces a framework that replaces the traditional multi-level approach to MGXS generation with a single MC calculation to generate the fine-spatial mesh MGXS that are needed by high-fidelity transport codes. In addition, three pin-wise spatial homogenization schemes are introduced to model inter-pin spatial self-shielding spectral effects with varying levels of spatial fidelity. Most notably, the Local Neighbor Symmetry (LNS) scheme uses a nearest neighbor-like analysis of a reactor geometry to determine which fuel pins should be assigned the same MGXS. The single-step MGXS framework with the three spatial homogenization schemes are evaluated for two PWR benchmarks. The results demonstrate the potential for single-step MC simulations of the complete heterogeneous geometry as a means to generate reactor agnostic MGXS for deterministic transport codes.